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Plant Design Requirements

Advanced Neutron Source

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ORNL/TM-11625/R4
5/11/92

ADVANCED NEUTRON SOURCE PLANT DESIGN REQUIREMENTS

APPROVALS

REVISION 4

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LIST OF ACRONYMS

The acronyms listed below apply specifically to this document. A complete list of acronyms approved for use by the ANS Project is maintained by the office of the ANS editor.

AA	Aluminum Association
ACGIH	American Conference of Governmental Industrial Hygienists
ACI	American Concrete Institute
ACSM	American Congress of Surveying and Mapping
AFWS	auxiliary feed-water system
AIHA	American Industrial Hygiene Association
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as reasonably achievable
AMCA	Air Movement and Control Association
ANL	Argonne National Laboratory
ANS	Advanced Neutron Source
ANSI/ANS	American National Standards Institute/American Nuclear Society
AOO	anticipated operational occurrence
API	American Petroleum Institute
ARI	Air Conditioning and Refrigeration Institute
ASCE	American Society of Civil Engineers
ASHRAE	American Society of Heating, Refrigeration, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AWS	American Welding Society
AWWA	American Water Works Association
BDB	beyond design basis
BDBA	beyond design-basis accident
BNL	Brookhaven National Laboratory
BOP	balance of plant
Bq	becquerel
CB	core blockage
CDR	Conceptual Design Report
CFR	Code of Federal Regulations
CGA	Compressed Gas Association
CHF	critical heat flux
Ci	curie
CMAA	Construction Manufacturers Association of America
CMAA	Crane Manufacturers Association of America
CO	coincident occurrence
CPBT	core pressure boundary tube
CS	cold source

CSAO	Construction Safety Association of Ontario
CSAR	Conceptual Safety Analysis Report
DBE	design-basis earthquake
DBT	design-basis tornado
DOC	Department of Commerce
DOD	Department of Defense
DOE	Department of Energy
DOT	Department of Transportation
EAB	exclusion area boundary
EIA	Electronics Industries Association
EIS	environmental impact statement
EOF	emergency operations facility
EPA	Environmental Protection Agency
EPRI	Electrical Power Research Institute
EPZ	emergency planning zone
ER	environmental report
ERDA	Energy Research and Development Administration
ES&H	environment, safety, and health
FCC	Federal Construction Council
FCI	fuel-coolant interaction
FEMA	Federal Emergency Management Agency
FI	flow increase
FIPS	Federal Information Processing Standard
FR	Federal Register
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HEU	highly enriched uranium
HFIR	High Flux Isotope Reactor
HPCI	high-pressure coolant injection
IAEA	International Atomic Energy Agency
IAPMO	International Association of Plumbing and Mechanical Officials
IBL	incipient boiling level
ICBO	International Conference of Building Officials
ICRP	International Commission of Radiological Protection
ICRUM	International Commission on Radiation Units and Measurements
ICS	integrated control system
IEEE	Institute of Electrical and Electronic Engineers
IES	Illumination Engineering Society of North America
ILE	inspection level earthquake
ILL	Institute Laue-Langevin
INZ	immediate notification zone
IO	initiating occurrence
ISDD	Integrating System Design Description
ISI	inservice inspection
LANL	Los Alamos National Laboratory

LLNL	Lawrence Livermore National Laboratory
LOC	loss of coolant
LOCA	loss-of-coolant accident
LOEP	loss of electrical power
LOF	loss of flow
LOHS	loss of heat sink
LOPC	loss-of-coolant pressure control
LORC	loss of reflector coolant
LORF	loss of reflector flow
LOSC	loss of secondary coolant
LPZ	low-population zone
LWR	light water reactor
MBMA	Metal Building Manufacturers Association
MCCI	molten-core concrete interactions
NAVFAC	U.S. Naval Facilities Engineering Command
NBS	National Bureau of Standards
NCG	noncondensible gas
NCMA	National Concrete Masonry Association
NE	Office of Nuclear Energy within DOE
NEMA	National Electrical Manufacturers Association
NEPA	National Environmental Protection Act
NFPA	National Fire Protection Agency
NIOSH	National Institute for Occupational Safety and Health
NOA	notice of availability
NOI	notice of intent
NPR	new production reactor
NRC	Nuclear Regulatory Commission
NRS	nonreactor safety
NSCANS	National Steering Committee for the Advanced Neutron Source
OBE	operating-basis earthquake
OMB	Office of Management and Budget
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
OSC	Operational Support Center
OSHA	Occupational Safety and Health Act
PCA	Portland Cement Association
PCI	Prestressed Concrete Association
PDR	Plant Design Requirements
PMP	Project Management Plan
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSAR	Preliminary Safety Analysis Report
PTI	Post-Tensionary Institute
QA	Quality Assurance
RCIC	reactor core isolation cooling

RDT	reactor development technology
REDC	Radiochemical Engineering Development Center
ROD	record of decision
SAR	Safety Analysis Report
SCS	Soil Conservation, U.S. Department of Agriculture
SDC	Supplementary Design Criteria
SDD	System Design Description
SDI	Steel Deck Institute
SDI	Steel Door Institute
SEMP	Systems Engineering Management Plan
SER	safety evaluation report
SI	Le Systéme International d'Unités (International System of Units)
SJI	Steel Joist Institute
SMACNA	Sheet Metal and Air Conditioning Contractors National Association
SNL	Sandia National Laboratories
SNM	special nuclear materials
SRP	Standard Review Plan
SSE	safe-shutdown earthquake
SSST	site suitability source term
STA	shift technical adviser
Sv	sievert
TSC	Technical Support Center
UL	Underwriters' Laboratories
WBS	work breakdown structure
WPCF	Water Pollution Control Federation
WRC	Water Resources Council

LIST OF DEFINITIONS

Active Component: A component that has moving parts or that is designed to perform its functions by a change of configuration or properties.

Active System: A system that depends on major active components for operation. For example, active systems depend on pumps, motors, and ac power generators.

Anticipated Event: Event expected to occur one or more times during the operating life of a nuclear plant.

Availability: The ratio of the time that the unit or equipment is capable of operation to the total time in a given time period, usually a year.

Balance of Plant (BOP): Buildings, structures, and design tasks associated with work breakdown structure categories 1.5 and 1.6. Excludes the reactor core, refueling equipment, reflector vessel systems, reactor control systems, and experimental systems.

Becquerel (Bq): The unit for radioactivity equal to one nuclear transformation per second (s^{-1}). It is the unit used within the International System of Units (SI).

Beyond Design-Basis Accidents: Hypothesized accidents that bound the consequences of any design-basis events and that are used to test mitigating design features and safety margins.

CFR (Code of Federal Regulations): Written regulations of federal agencies. For example, Chapter 1 of Title 10 of the CFR (10 CFR) contains the regulations of the Nuclear Regulatory Commission.

Cold Source: A device used to modify neutron spectra to very low energy and thus provide substantial gain in the number of neutrons with energy less than 0.005 electron volts.

Common-Mode Failure: Multiple failures attributable to a common cause.

Containment: The structure or vessel that encloses, as a minimum, the components of the reactor coolant pressure boundary and serves as an essentially leaktight barrier against the uncontrolled release of radioactivity into the environment.

Control Rods: Rods or plates made of neutron-absorbing material and used to regulate or halt nuclear fission in a reactor.

Coolant: The fluid circulated through the reactor core and nearby systems in the reflector vessel to transfer the heat of the fission process to a secondary heat-transfer system.

Core: The central portion of the nuclear reactor containing the fuel elements. Nuclear fission takes place, and neutron flux and heat are generated within the core.

Core Damage: Damage to the fuel so that the core cannot be used for further neutron and heat production. Core damage is considered to start when the fuel temperature exceeds a specified limit.

Core Disruptive Accident: A reactor accident of sufficient magnitude to result in rapid relocation of fuel.

Core Pressure Boundary Tube (CPBT): That component in the Advanced Neutron Source (ANS) that provides the reactor coolant pressure boundary just outside the reactor core.

Criteria: Safety and licensing criteria as defined by licensing and regulatory bodies and as augmented by ANS-supplementary criteria. These safety and licensing criteria are design requirements, as defined in these Plant Design Requirements (PDR), unless stated otherwise.

Critical Heat Flux (CHF): The local heat-flux density (between a surface and a cooling liquid) that gives a maximum in the curve of heat-flux density against temperature difference; associated with the change from nucleate boiling to film boiling.

Curie (Ci): A radioactivity of 3.7×10^{10} disintegrations per second or 3.7×10^{10} Bq. By popular usage, curie also refers to the *quantity* of any radioactive material having an activity of 3.7×10^{10} Bq.

Decay Heat: The heat from a shutdown reactor or fuel element resulting from residual radioactivity and fission.

Defense-in-Depth: The concept of designing nuclear power plants to avoid equipment failure, human error, and severe natural events, and to provide redundant and back-up systems so that safety functions can be accomplished even in the event of the most unlikely malfunctions.

Design-Basis Events (DBEs): Events used in the design of a facility to establish the performance requirements of the structures, systems, and components. DBEs that the plant design must accommodate include normal operation, anticipated operational occurrences, design-basis accidents, external events, and natural phenomena. (See Regulatory Guide 1.135 and CFR 50.49).

Design-Basis Earthquake (DBE): That earthquake vibrational ground acceleration used for design of nonreactor safety structures, systems, and components in accordance with DOE 6430.1A and UCRL-15910.

Design Goals: General objectives that the design tries to achieve. These objectives are included to provide traceability of the final design requirements to the original goal established through interaction with the scientific community. When something is a goal, it will be stated as such in the design documentation. Achieving a design goal is desirable but not mandatory for the final

design and operation of the facility. To provide design margin, all design goals should be included in design and costing of the ANS conceptual design. Trade studies performed after conceptual design will lead to the final design requirements. Design goals that do not appear feasible at conceptual design will be placed on [HOLD].

Design Limit: The boundary value of a parameter for which a design analysis has been performed (e.g., pressure, temperature, flow).

Design Margin: Capability beyond that required by design requirements and regulations.

Design Requirements: Mandatory features and attributes of the ANS design that are specified in this PDR document, the System Design Descriptions, and other official design documents. Design requirements are more specific and, in general, less ambitious than design goals.

Detritiation: The process for removing tritium.

Dose: Quantity of radiation absorbed, in units of rem or sieverts, by the human body or by any portion of the human body.

Engineered Safety System: A hardware system for preventing or mitigating the consequences of an accident. In contrast to a passive safety feature, engineered safety systems often require external power and have moving parts.

Enrichment: The percent composition of a particular isotope in an element when it exceeds a natural composition.

Extremely Unlikely Events: Events of extremely low probability (10^{-6}/year to 10^{-4}/year) that are used to establish design bases, especially those related to reactor containment.

Faulted Events: Those combinations of conditions associated with extremely low probability of postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved.

Fission Product: A nuclide produced either by fission or by a subsequent radioactive decay of the nuclides thus formed.

Fuel Damage Limits: Those limits, such as cladding strain, amount of fuel melting, amount of cladding deformation or melting, and fractional fuel failure, beyond which the accident consequences are unacceptable.

Fuel Design Limits: Those limits, such as temperature, burn-up, fluence (neutron flux × time), and cladding strain, that are specified by the designer for normal operation and anticipated operational occurrences beyond which fuel-element failure may occur.

Heavy Water: Water in which hydrogen atoms (nuclear mass of one, due to one proton) are replaced by deuterium, an isotope of hydrogen with a nuclear mass of 2 (one neutron and one proton).

Hot Source: A device used to shift the neutron spectrum to a distribution consistent with a temperature in the range of about 1000°C to 3000°C.

[HOLD]: A term included in the text of this PDR and other project documents to indicate that a decision concerning accuracy or achievability of statements or values to which the [HOLD] applies is not final. [HOLD]s are numbered, like references, and controlled in an appendix of the document containing the indicated [HOLD]. The appearance of [HOLD] in this definition (non-bolded and unnumbered) is generic and non-technical.

Initiating Event: The first failure or action that, in the absence of adequate operator action and/or engineered safety systems, could lead to an accident.

Inspection-Level Earthquake (ILE): That earthquake vibratory ground acceleration for which the plant must be shut down. The plant may be restarted following an ILE only after a thorough inspection ensures that no functional damage has occurred.

Integrating System Design Description (ISDD): The top-level ANS design document for generic plant functions, like containment. Only the PDR, which defines plant-level requirements, has higher authority. The ISDDs augment the PDR by assigning functions to the system design descriptions (SDDs).

Interface: Functional, parametric, and physical requirements imposed by one system on another system. Primary interfaces for a system are those that it imposes on other systems. Secondary interfaces for a system are those imposed upon it by other systems.

Light Water: Ordinary water not enriched in any hydrogen isotopes.

Limiting Conditions for Operation: The lowest functional capability or performance levels of equipment required for safe operation of the facility. (See 10 CFR 50.36).

Limiting Safety System Settings: A variable setting point for automatic protection-device action that is closer to the normal operating range for the variable than is its safety limit. The difference between the safety limit and limiting safety systems setting indicates the margin of safety of the protection device. (See 10 CFR 50.36).

Nonsafety-related (nonsafety-class): All systems that are *not* safety related (see below).

Operating-Basis Earthquake (OBE): That earthquake that produces vibratory ground acceleration for which those structures, systems, and components important to reactor safety must remain functional. For the ANS, the OBE applies for seismic categories I and I-L only.

Passive Component: Generally, a component that has no mechanical moving parts and is designed to perform its functions without a change of configuration or properties.

Plant Design Requirements (PDRs): The top-level design control document of the ANS Project. This document contains the top-level design goals and design requirements.

Primary Interface: See Interface.

Reactor-Coolant Pressure Boundary (RCPB): Those components such as heat exchangers, piping, pumps, and valves that are part of the reactor coolant system or are connected to the reactor coolant system up to and including any and all of the following:

1. The second of two valves normally closed or automatically isolatable during normal reactor operation (a single valve that is normally shut is acceptable if failure does not prevent normal reactor inventory control).
2. The passive barrier between the reactor coolant and the working fluid of other portions of the heat transport system (e.g., that of a heat exchanger).

Reliability: The design attributes that ensure that equipment will operate for a given time period under stated operating conditions.

Reflector: Part of a nuclear reactor placed adjacent to the core to return some of the escaping neutrons back into the core.

Rem (Roentgen Equivalent Man): A unit of radiation dose. Frequently, radiation dose is measured in millirems for low-level radiation. The SI unit for radiation dose used by ANS is the sievert (Sv), which is equal to 100 rems.

Research Goals: Technical objectives for the ANS facility that originated from extensive interactions between the ANS Project and the research community that ANS will serve. Research goals, if sufficiently quantified, were included in the initial set of design goals and design requirements for this PDR.

Safe-Shutdown Earthquake (SSE): The earthquake that is based on an evaluation of the maximum earthquake potential considering regional and local geology and seismology and specific characteristics of local subsurface material. It is the earthquake that produces the maximum vibratory ground motion for which certain structures, systems, and components of a nuclear plant are designed to remain functional so that the plant can be brought to a safe shutdown. (See 10 CFR Part 100, Appendix A). For ANS, the SSE applies to seismic categories I and I-L only.

Safety Analysis Report (SAR): The part of an application for a construction permit (preliminary safety analysis report) or an operating license (final safety analysis report) that provides technical information concerning the proposed facility, including siting, design, engineered safety features, construction, quality assurance, operation, control, accident analysis, and technical specifications. (See 10 CFR 50.34).

Safety Evaluation Report (SER): A summation of the reviewing body's conclusions concerning action proposed by an applicant. The proposal is often in the form of a SAR.

Safety Limits: Limits on important process variables that are necessary to protect reasonably the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. (See 10 CFR 50.36).

Safety-related: Describes regulated systems, structures, and components relied upon to maintain the reactor-coolant boundary, to shut down the reactor, or to prevent or mitigate the consequences of accidents with off-site effects.

Scram: A rapid shutdown of the nuclear reactor accomplished by moving control rods into the core to halt fission.

Secondary Interface: See Interface.

Sievert (Sv): The SI unit of dose. (1 Sv = 100 rems).

Single Failure: An occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single credible failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions.

Single failure of passive components in electric systems should be assumed in designing against a single failure.

Site-Suitability-Source-Term (SSST): The quantity of fission products, activation products, and core fuel assumed or calculated to be released inside the containment structure to test the adequacy of the containment design. The SSST must be more severe than the release expected from any credible accident and is used to determine if a particular site is suitable for the proposed reactor/containment configuration.

Spent Fuel: Nuclear fuel that is removed from a reactor following irradiation because of depletion of fissile material.

Station Blackout: The complete loss of ac power except for power from station batteries through inverters.

System Design Descriptions (SDDs): The top-level ANS design document for a plant system. Only the PDR, which defines plant-level requirements, has higher authority. The SDDs are the principal means to establish, describe, and control individual system designs from conception and throughout the lifetime of the plant.

To Be Determined [TBD]: A term included in the text of this PDR and other project documents to indicate that the item or value for which the [TBD] applies has not been determined or calculated. [TBD]s are numbered, like references, and controlled in an appendix of the document containing the indicated TBDs. The appearance of [TBD] in this definition (non-bolded and unnumbered) is generic and non-technical.

Technical Specifications: Limits, controls, and surveillance requirements on process variables and equipment in an operating nuclear plant. These provisions cannot be changed without prior permission from the regulatory body.

Unlikely Events: Internal and external events of sufficiently low probability that none are expected to occur during the lifetime of the plant. No fuel damage should result from unlikely events even if conservatisms are used in assessing consequences.

SUMMARY OF CHANGES TO THE PDR RESULTING FROM EACH REVISION

Revisions to the Advanced Neutron Source (ANS) Plant Design Requirements (PDR) are indicated in the body of the document. Changes made by incorporating change-pages are indicated on each change-page by a broken vertical line in the margin adjacent to the change. Each page of the document has a header showing the revision number for the page and the date on which that revision was approved. Revisions where the entire document was reissued include only an updated header on each page.

Revision 1 (4/1/91): The original document (7/30/90) was modified to add material to Chap. 4, Safety Requirements; to change the definitions of neutron flux ranges indicated in Table 2.1-1; and to make several associated modifications. Four new sections (4.2–4.5) were added to Chap. 4, and other sections within the chapter were rearranged and renumbered.

Revision 2 (8/22/91): Two new sections were added to Chap. 4 (4.7.3, Plant Conditions; and 4.8, Design-Basis Events and Acceptance Criteria). Sections 6 and 6.1 were added to Chap. 6 to provide a tabulation of the functions and interfaces for each Integrating System Design Description (ISDD) and System Design Description (SDD). A list of definitions was added to provide a consistent set of terminology for key terms used within the project.

Revision 3 (1/22/92): Updates were made of the reactor safety classes and nonreactor safety classes used in the project and of the seismic and tornado requirements. Codes and standards that will be used for all of the defined classes were indicated, and editorial changes were made to maintain consistency throughout the document.

Revision 4 (5/11/92): The entire document has been updated, and new chapters and appendices have been added (Chap. 7 through App. D). A thorough editing of Chap. 1 through Chap. 11 has been conducted to achieve consistency of approach and style within the various chapters. The list of acronyms and the list of definitions have been increased to include items omitted in earlier versions of the document. Reference lists were moved to the end of each chapter and the number of references increased to improve traceability.

PREFACE

This document provides the plant-level requirements for the design, construction, and operation of the Advanced Neutron Source (ANS). It is a "living document," and will be revised throughout the life of the project to reflect the current configuration of ANS. The distribution of this document is being conducted in a controlled manner. Holders of controlled copies are to acknowledge receipt of the original issue and all revisions and are expected to keep the manual updated throughout the project. If a controlled copy is no longer needed, it shall be returned to the ANS Project office.

Because of the large number of revisions that will be issued for this document, the following numbering scheme is being used for tables, figures, [HOLD]s, and [TBD]s. For the [HOLD]s and [TBD]s, the first number is the chapter in which the item is mentioned, and the second number is the sequence of the item in that particular chapter. For tables and figures, the first number is that of the chapter in which the item appears; the second number is the section within the chapter; and the third number is a sequential ordering within that section. For example, Table 4.1-2 is the second table in Sect. 4.1. This arrangement will minimize the extent to which tables, figures, [HOLD]s, and [TBD]s are renumbered as the document is revised.

References are numbered sequentially within each chapter, and a list of references occurs at the end of each chapter. This PDR has two types of references: **Category A References**, identified in the text and reference lists by a **bold** font, are those written specifically to supplement this PDR, and Category B References are those cited within the text of the chapter but not written specifically for this PDR. Each Category A Reference is part of the technical requirements of this PDR because it is essential to defining the selected design. A Category B Reference is a part of this document only to the extent specified in the body of the PDR. Category B References also include supporting analyses, evaluations, and trade studies that provide background information.

Applicable revisions or revision dates are specified for selected references. Documents supporting this PDR shall specify the revision or revision date for those codes and standards and other nonproject documents utilized for system design. The equipment specifications, the construction specifications, and the test and operating specifications shall specify the applicable revisions or revision date for all codes and standards and other nonproject documents referenced in the specification.

When the requirements of this document are in conflict with documents referenced herein, with subsidiary documents, or with documents required by regulatory agencies, the requirements of this document shall take precedence until such conflicts are resolved and the conflicting document/documents is/are changed in accordance with established procedures.

ABSTRACT

The Advanced Neutron Source will be a new world-class facility for research using hot, thermal, cold, and ultra-cold neutrons. The heart of the facility will be a 330-MW (fission), heavy-water cooled and heavy-water moderated reactor. The reactor will be housed in a central reactor building, with supporting equipment located in an adjoining reactor support building. An array of cold neutron guides will fan out into a large guide hall, housing about 30 neutron research stations. Appropriate office, laboratory, and shop facilities will be included to provide a complete facility for users. The ANS is scheduled to begin operation at the Oak Ridge National Laboratory early in the next decade.

This PDR document defines the plant-level requirements for the design, construction, and operation of ANS. It also defines and provides input to the individual System Design Description (SDD) documents. Together, this PDR document and the set of SDD documents will define and control the baseline configuration of ANS.

1. SCOPE

1. SCOPE

1.1 INTRODUCTION

This Plant Design Requirements (PDR) document defines the plant-level requirements and goals* for the design, construction, and operation of the Advanced Neutron Source (ANS). This document also defines and provides input into the individual System Design Description (SDD) documents. Together, the PDR document and the SDD documents define and control the baseline configuration of ANS.

The top-level requirements specified in this PDR include those developed to meet the ANS user community needs as defined by the National Steering Committee for the Advanced Neutron Source (NSCANS); those defined in Department of Energy (DOE) orders applicable to the design and construction of DOE-owned reactors; other federal and state agency regulations, standards, and guidelines; national codes and standards that are applicable to ANS; and those specific requirements identified in the design, safety, and environmental studies conducted as part of the ANS Project. The safety and environmental studies will be documented in the Preliminary Safety Analysis Report (PSAR), the Final Safety Analysis Report (FSAR), a Probabilistic Risk Assessment (PRA), the Environmental Report (ER), and the Environmental Impact Statement (EIS).

This PDR follows the format outlined in NE F 1-2T.[1] The project organization and management structure is defined in the Project Management Plan (PMP);[2] the project quality assurance program is defined in the Quality Assurance (QA) Plan;[3] and other assessments and supporting documents are described in the Systems Engineering Management Plan (SEMP).[4]

1.2 PROJECT PURPOSE

ANS will meet the recognized national need for an intense, steady-state, broad-spectrum source of neutrons for research.[5–15] ANS will provide the American scientific community with a crucial tool for cross-disciplinary neutron beam research in physics, chemistry, biotechnology, pharmacology, medicine, and energy-related materials and structures. In addition, it will provide needed facilities for isotope production (including transuranic isotopes), materials irradiation testing, and analytical chemistry. The project provides the means for the United States to regain the world leadership that it previously held in neutron-based research.

The main scientific purpose for the ANS Project is expressed on pages 45–46 of the *Major Facilities for Materials Research and Related Disciplines*, a report from the National Research Council.[13] That report emphasizes the need for a world-class neutron scattering facility in the United States. Its essential research requirements are a very high thermal neutron flux in a region large enough for beam-tube accessibility and one or more cryogenic moderators sized to produce cold neutrons (those with very low kinetic energy). The report also suggests that the facility

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*Key terms used in this document are defined in the List of Definitions. Only *requirements* and *goals* define and control the design configurations.

include at least one large guide hall for cold neutron and thermal neutron research; a reactor hall for thermal neutron and high-energy (hot source) neutron research; a variety of advanced research instruments; and improved national facilities for isotope production, irradiation of materials, neutron activation analysis, and research in nuclear physics. The top-level research goals of the ANS Project are listed in Table 1.2-1.

Table 1.2-1. ANS Project research goals

To design and construct the world's highest-flux research reactor for neutron scattering
– Provide 5-to-10 times the flux of the best existing facilities
To provide isotope production facilities that are as good as, or better than, the HFIR
To provide materials irradiation facilities that are as good as, or better than, the HFIR

1.3 PROJECT SCOPE

The ANS Project includes all aspects of the design and construction of ANS defined in the Field Budget Request project data sheet 94-ORNL-KC(AF)-1[16] dated April 2, 1992. The project also includes (1) safety analyses and documentation to support funding requests and permitting; (2) environmental reports, assessments, and impact statements to support funding requests and permitting; and (3) research and development necessary to provide data for the safety analyses, environmental analyses, and design. All work elements within the project are defined by a work breakdown structure (WBS) (Fig. 1.3-1).

The scope of the project has been defined through comprehensive interaction with all of the relevant scientific communities whose purposes may be served by ANS. Contacts have been fostered by widespread discussion and dissemination of information about the project at national and international professional society meetings, by journal articles, by seminars, by both broad-based and focused newsletters, by mailed questionnaires, and by direct personal contacts within the neutron research community. The NSCANS has served as a clearinghouse for the information garnered and has acted as a review body, both directly and via special subcommittees.

The project will construct a neutron research laboratory based on a high-flux reactor. The reactor will also provide materials irradiation and transuranic isotope production capabilities that match or exceed the capabilities of the High Flux Isotope Reactor (HFIR). Facilities for radioisotope production and for analytical chemistry will be accommodated in the reflector. Safe operation of the reactor and efficient utilization of the experimental facilities will be provided by a suitable on-site infrastructure, appropriately interfaced with the facilities offered locally by the Oak Ridge Reservation (ORR) and by other DOE operations.

1.4 PROJECT COMPLETION

The line item funded portion of the ANS Project shall be considered complete upon achievement of the experiment systems benchmark measurement commissioning phase defined below.[2]

1.4.1 Neutron Source Intensity Verification

The technical objectives defined in Sect. 2.1 of ref. 2 are to be met by construction of a neutron research laboratory based on a high-flux reactor that has a peak thermal flux in the reflector exceeding the value quoted in ref. 13 (1×10^{19} $m^{-2} \cdot s^{-1}$) for similar, existing facilities by at least a factor of five (p. 13 of ref. 13). Verification of achieving this requirement shall be accomplished by an incident-source-brightness measurement performed at an appropriate thermal-neutron station of the ANS, e.g., the T-4 station on the HB-3 thermal guide. The measurement shall be on a standard sample approved by the Neutron Diffraction Commission of the International Union of Crystallography.[17,18]

1.4.2 Neutron Guide Transport Systems Verification

Neutron flux measurements will be accomplished at the beginning and end of each installed guide to provide verification that the neutron guide transport system's acceptance and transmission characteristics will exceed an effective reflectivity per bounce value of 0.95. These measurements will employ a standard, internationally accepted technique, e.g., foil activation. The beginning position is defined as an appropriate guide access point near the penetrations through the reactor biological shield or, if applicable, through the reactor building containment wall. The end position is defined as an appropriate guide access point planned for installation of a scientific research instrument.

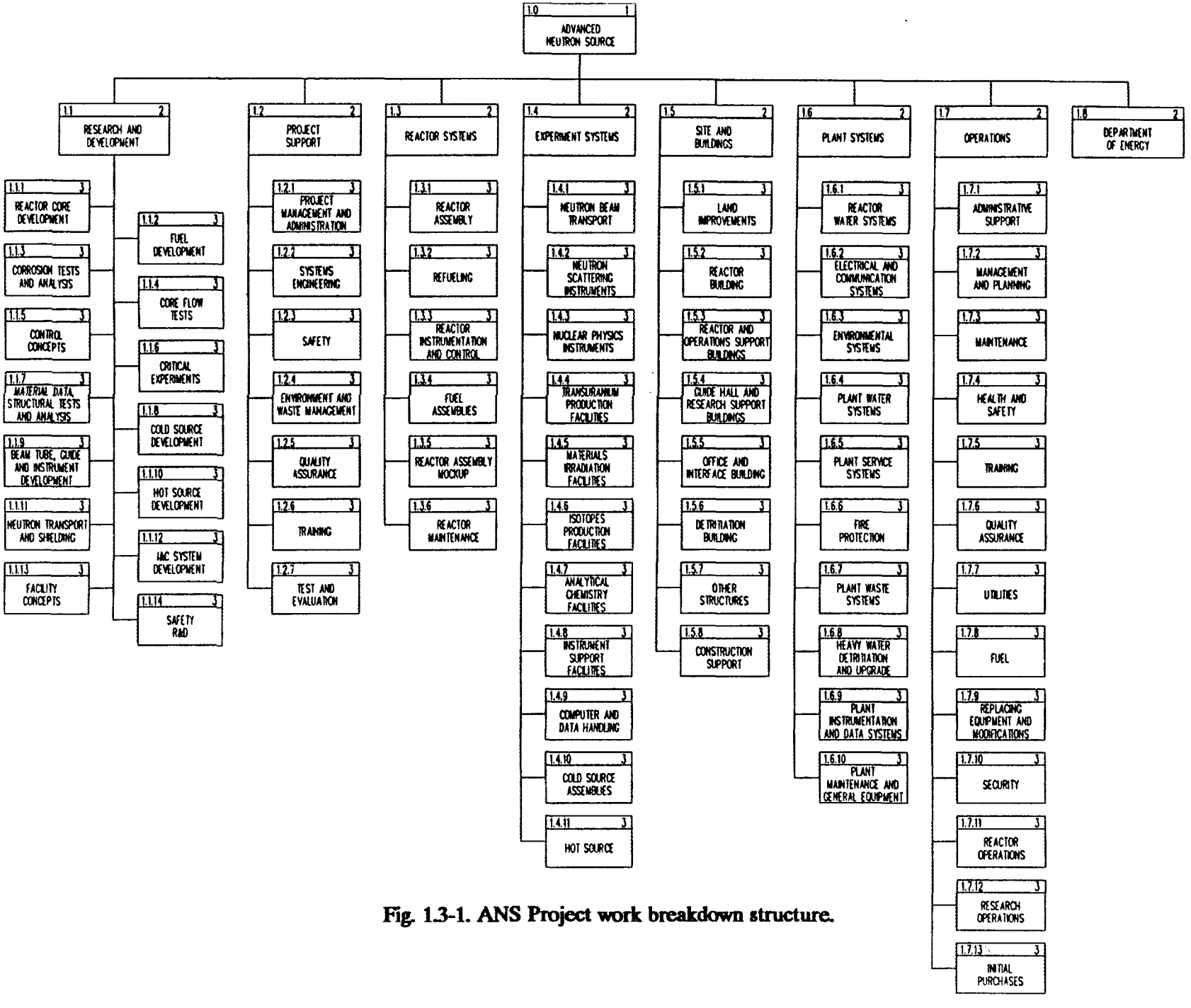

Fig. 1.3-1. ANS Project work breakdown structure.

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2. DESIGN GOALS

2. DESIGN GOALS

As discussed in Chap. 1, research goals for ANS were established with the research community that is to be served by the facility. Many of these goals were translated into design requirements during the initial design process because there was little likelihood that detailed design activities would reveal that these requirements could not be met. Achieving some of these research goals, however, may preclude or limit the extent to which other goals are fully achieved. In addition, achieving required regulatory, safety, certification, and environmental mandates and requirements, and following required design and engineering standards, may further limit the extent to which all research goals can be fully achieved. Also important are project policies to

1. achieve all research goals to the maximum extent possible and maintain visibility of the goals to the design team;
2. keep the research community informed of design progress and involved, as an advisory body, in decisions involving the balancing of goals in order to achieve the best research facilities; and
3. provide traceability of design features back to the research goals of the research community.

Thus, the research emphasis of ANS results in a need to expand the PDR beyond minimum design requirements.

To meet this need, while still controlling the design configuration, the ANS Project team used the following strategy in developing this PDR.

1. Any PDR objective that is not a design requirement in the traditional sense or according to the definitions included within the List of Definitions will be stated as a design goal. Design goals come from research goals, safety goals, or other sources. Design requirements come from research goals, regulatory requirements, standards, and DOE orders.
2. During conceptual design, all design goals will be considered additional design requirements, and the achievement of them will be incorporated into the conceptual design and its associated cost estimate.
3. If, at the conceptual design stage, a judgment is made and approved that a design goal is too ambitious (e.g., exorbitant cost or design complexity), then this is indicated in the PDR by one of two methods. The first method, used primarily for top-level research goals, is to state what was achieved and why the complete goal was not achieved (see note *d* of Table 2.1-1 for an example of this method). The second method is to place the design goal on [HOLD]. Costs for partial achievement of goals, within practical limits, are included in the conceptual design cost estimate.
4. Design goals not yet achieved and other items and issues on [HOLD] will be resolved beginning in the advanced conceptual design through cost and design trade-off studies. During this time, a systematic transition will occur in the PDR from goals to requirements.

2.1 RESEARCH FACILITY GOALS

The ANS will provide equipment and facilities for the following:

1. hot, thermal, cold, very cold, and ultracold neutron beam stations, with optimized neutron beam delivery systems;
2. state-of-the-art spectrometers for neutron scattering and nuclear and fundamental physics research;
3. irradiation of structural materials and nuclear fuels foreseen by fission, fusion, and other materials irradiation research programs; and
4. activation analysis and related materials analysis capabilities.

Specific NSCANS goals for the implementation of the project's research goals (Table 1.2-1) are given in Table 2.1-1. The NSCANS goals follow the recommendations of the ANS user community as defined by NSCANS and others (see refs. 1–43). The ANS conceptual design parameters that correspond to the NSCANS goals are also shown in Table 2.1-1 to facilitate an assessment of how well the goals have been achieved thus far by the design.

The ability to adapt varied experimental facilities to changing future priorities by design that allows maximum flexibility in the arrangement and use of space and operational adjustments is an additional requirement to be met by facility design.

2.2 PRODUCTION FACILITY GOALS

The ANS will provide production facilities for transuranium and other isotopes, as specified in Table 2.1-1.

2.3 REACTOR DESIGN GOALS

The source of neutrons (i.e., the reactor) shall meet certain performance goals for neutron flux, neutron spectrum, and experimental space. Consideration will also be given to performance beyond these goals to maximize the research value of the ANS by providing for optional research capability during initial plant design. If additional performance or facilities can be provided without significant penalties, they should be evaluated for incorporation into the plant design. If the impact is significant, the opportunity to retrofit such capability at a later date should be retained to the extent feasible. Table 2.1-1 lists the major flux and spectrum goals.

Necessary space shall be available to accommodate and cool the transuranium production and materials irradiation facilities (see Sects. 2.2, 2.1, respectively). Access shall be provided for electrical leads and gas lines for the instrumented irradiation capsules.

The reflector vessel shall accommodate the space and cooling requirements of the isotope production facilities, materials analysis facilities (rabbit tubes), cold sources, cold neutron guides, and neutron beam tubes.

As dictated by safety and neutronic requirements, the hot source can be placed outside or inside the reflector vessel.

Table 2.1-1. ANS conceptual design parameters

Parameter[a,b]	NSCANS goals	ANS conceptual design
Neutron scattering		
Hot neutrons		
Thermal flux at hot source	≥1	1
Number of hot sources	1	1
Number of hot beams	2	2
Thermal neutrons		
Peak thermal flux in reflector	5–10	7[c]
Thermal:fast ratio	≥80:1	170:1
Number of thermal tangential tubes	7	7
Cold neutrons		
Thermal flux at cold sources	2–4	4
Number of cold sources	2	2
Number of horizontal cold guides	14	14
Number of slant cold beams for scattering instruments	1	1
Nuclear and fundamental physics		
Number of thermal through tubes	1	1
Number of slant thermal beams	1	1
Number of very cold beams	2	2
Materials irradiation[d]		
Central irradiation facility		
Fast flux	≥1.4	2.0[e]
Fast:thermal ratio	≥1:2	2.9:1[e]
Total number of positions	10	10
Number of instrumented positions	2	5
Damage rate [displacement per atom per year (dpa/y) in stainless steel]	≥30	*[TBD]*[2-1]
Nuclear heating rate (W/g in stainless steel)	≤54	≤70
Axial flux gradient over 200 mm	≤30%	14%
Available diameter (mm)	≥17	48
Available length (mm)	≥500	500
Reflector vessel facility		
Fast flux	≥0.5	0.5
Fast:thermal ratio	≥1:3	1:13[f]
Number of instrumented positions	≥8	2[g]
Damage rate (dpa/y in stainless steel)	≥8	*[TBD]*[2-2]
Nuclear heating rate (W/g in stainless steel)	≤15	18
Axial flux gradient over 200 mm	≤30%	16%
Available diameter (mm)	≥48	48
Available length (mm)	≥500	300[h]

Table 2.1-1 (continued)

Parameter[a,b]	NSCANS goals	ANS conceptual design
Isotope production		
Transuranium production		
Epithermal flux	≥0.6	1.4
Epithermal:thermal ratio	≥1:4	1:3
Allowable peak heat flux (MW/m^2)	4	*[TBD]*[2-3]
Total annual production:		
^{252}Cf (g)	1.5	1.7[i]
^{254}Es (μg)	40	60[i]
Epithermal-hydraulic rabbit tube		
Epithermal flux	flux peak position	1.0[j]
Epithermal:thermal ratio	≥1:4	1:4.5[j]
Allowable peak heat flux (MW/m^2)	1.75	1.75
Other isotope production facilities		
Thermal flux	1	0.9
Number of reflector positions	≥4	4
Number of thermal–hydraulic rabbit tubes	3	3
Materials analysis		
Activation analysis pneumatic tubes		
40 cc^3 rabbits in reflector	4	4
2 cc^3 rabbits in reflector	1	1
Thermal flux at reflector rabbit positions	≥0.2	0.3 (tube 1) 0.06 (tubes 2,3,4 and 5)[k]
Heating rate:		
Temperature in a 40-cc high density polyethylene rabbit (°C)	≤120	*[TBD]*[2-4]
Rabbit tubes in light water pool	2	2
Thermal flux at light water rabbit positions	≥0.04	0.05; 0.02
Prompt-gamma activation analysis cold neutron stations		
Low-background (multiple beam) guide system	1	1
Neutron depth profiling		
Number of slant cold beams	1	1
Gamma irradiation facility in spent-fuel pool	≥1	1
Positron production position	1	1

Table 2.1-1 (continued)

Parameter[a,b]	NSCANS goals	ANS conceptual design

Notes:

[a]All fluxes in units of 10^{19} $m^{-2} \cdot s^{-1}$.

[b]Neutron spectra terms as used in this table are defined as follows:
thermal ≤ 0.625 eV 0.625 eV ≤ epithermal ≤ 100 eV fast > 0.1 MeV

Sources of neutrons for research are classified as follows:
ultracold < 25 μeV 25 μeV < very cold < 0.1 meV 0.1 meV < cold < 5 meV
5 meV < thermal < 0.625eV 0.625eV < hot < 1 eV

[c]End-of-cycle value.

[d]The materials irradiation facilities are intended to replace irradiation facilities in the HFIR. ANS can not meet all these goals since the simultaneous requirements of high fast:thermal flux-ratio, high-fast flux, and low heating rate are intrinsically incompatible with the physics of an undermoderated core (optimized for neutron scattering research goals).

[e]Values at r=160 mm and beginning-of-cycle. The fast flux increases during the fuel cycle. The fast:thermal ratio decreases to as low as 0.25:1 with decreasing radius and increasing burnup.

[f]Best available ratio in reflector vessel.

[g]Access restricted by other facilities in the reflector vessel. Partially offset by providing more instrumented positions than required in the central irradiation facility.

[h]Length restricted by outer shutdown rods.

[i]The production number quoted is a preliminary value obtained from a two-dimensional neutronic analysis. Possible thermal limitations on pellet loadings have not been evaluated.

[j]Value at 250 mm above midplane at midpoint of cycle.

[k]Optimized for lowest gamma heating, but will need to be located at a higher neutron flux level.

To avoid the coolant's flashing to steam in the event of depressurization, the bulk coolant outlet temperature will be below the boiling point at atmospheric pressure.

To minimize safety questions and technical risks, the reactor design shall be based, as far as possible, on known technology; in particular, the design should not rely on the invention of new technology to meet the minimum quantitative design goals. This goal, and the performance requirements, lead to the design choices shown in Table 2.3-1.

Detailed technical justifications for these design choices are often documented within ANS internal correspondence, in ANS presentations, and through the proceedings of numerous

workshops conducted by the ANS Project, and others, to address specific design issues. Many of the workshop documents are Oak Ridge National Laboratory (ORNL) internal reports that are maintained by the ANS Project office but are not available for general distribution. A list of these documents and the documents themselves are available upon request by qualified individuals.

Open-distribution references that provide traceability for design choices in Table 2.3-1 include ORNL internal correspondence from C. D. West on the topic of traceable documentation of the ANS core design and physical specifications,[44] an overview presentation by C. D. West during a 1991 ANS Reactor Systems Workshop,[45] and an ORNL letter from C. D. West to L. C. Ianniello dealing with fuel enrichment.[46] Additional references provide historical information.[47-51]

Table 2.3-1. Design choices for the ANS reactor

Type	Compact undermoderated core in a reflector region
Power level	≤330-MW fission power in the core
Coolant	Heavy water
Reflector	Heavy water
Fuel type	Highly enriched, aluminum-clad fuel formed into involute plates
Cold sources	Two, with liquid deuterium moderator
Cooling systems	Maximum use of passive or inherent safety features
Coolant gap	≥1.25 mm
Plate thickness	≥1.25 mm

2.4 USER AND SYSTEM SUPPORT

The ANS facility will be capable of handling at least 1000 short-term (one day to two weeks) scientific visitors per year, as well as providing support for permanent on-site staff. The environment for short-term users of standard beam facilities will be as similar as possible to that found in the best research laboratories [such as the Institue Laue Langevin (ILL)], including the availability of adequate in-house scientific and technical staff support. The facility's design and operation shall also accommodate and promote cooperative interactions among research groups. To facilitate achieving this goal, the ANS plant will be designed and constructed with secure physical barriers between the beam research and related support areas and the reactor operations areas. Users will be processed at an on-site reception area and given access to the research area, inside of which passage between the different beam rooms, shops, and laboratories will be as free as possible, consistent with keeping radiation exposures as low as reasonably achievable (ALARA) and with maintaining normal laboratory security requirements.

The ANS facility, in context with facilities available at ORNL and other DOE sites at Oak Ridge, will provide the necessary offices, shops, facilities, and maintenance and storage areas to

support the operation of the reactor and research facilities. Existing facilities and labor pools will be utilized to the greatest extent practical. The ANS facility will be integrated into the site infrastructure of ORR, including roadways, utilities, and monitoring systems. Security and fire protection beyond that required on site by regulatory requirements will be provided by ORNL Laboratory Protection. Interfaces will be defined between on-site operations support facilities and other facilities on the DOE reservation. The spare parts and supplies inventory needed to support operation will be defined; the appropriate storage environment will be provided at existing facilities, if possible, or at the ANS site, if necessary.

An appropriate interface will be provided to existing handling and separation facilities at ORNL, with particular reference to the transuranium facilities at the Radiochemical Engineering Development Center (REDC). Users will interface directly with reactor operations staff for research programs that involve interaction with the operation of the reactor or the handling of highly active materials (such as materials irradiation, pneumatic rabbit tubes, or isotope production).

Interfaces will be provided to reactor fuel storage and shipment facilities, covering receipt and storage of fresh fuel as well as shipment of spent fuel. Secure storage for unirradiated fuel will be identified; responsibilities for protection of the fuel will be assigned; and anticipated schedules for fuel handling will be established and integrated into the design of security systems at ANS.

Experimental procedures, equipment, and technology from other similar research facilities, including HFIR and ILL, will be used to the maximum extent possible in the operation of ANS.

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3. ECONOMIC REQUIREMENTS

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4. SAFETY AND CERTIFICATION GOALS AND REQUIREMENTS

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4.1 OVERALL SAFETY GOALS AND REQUIREMENTS

4.1.1 Licensability Goals

The ANS is to be designed, built, and operated under DOE ownership and is therefore, under 10 CFR 50.11, not subject to the Nuclear Regulatory Commission (NRC) licensing process. However, DOE orders require DOE reactors to meet the standards, codes, and guides that are applied to comparable licensed facilities. Therefore, standards, codes, and guides applied to the ANS design shall include those by which NRC would judge ANS.

Furthermore, no technical changes to the facility's structures, systems, equipment, hardware, or software is anticipated from a postulated NRC review. However, business systems or administrative procedures from NRC, which apply to commercial power stations, need not be applied to ANS.

Minimum actions to meet the project's requirements include:

1. complying with DOE's Orders on Mandatory Standards for New Reactors (see Appendix B, Tables B.1 through B.5 for a list of such standards);
2. applying the "principle of comparability," which is found in DOE 5480.6, in selecting the standards referenced in this PDR. Comparable facilities and designs include other research reactors throughout the world, new production reactor (NPR) designs, advanced commercial reactor designs sponsored by DOE, and design-requirements from the Electrical Power Research Institute (EPRI) for commercial power stations (see Appendix B, Table B.7 through B.9); and
3. considering unique features of ANS design that may require application of more demanding standards than are applied to comparable facilities. Such features include neutron flux intensity, power density, coolant flow rate, and advanced experimental facilities. This consideration has impacted system and component classification and has resulted in Supplementary Design Criteria (SDC) (Table B.6).

4.1.2 Risk Limitation Goals

ANS risk limitation goals are based on the NRC Policy Statement dated August 21, 1986,[1] and upon the similar draft DOE *Nuclear Safety Objectives Policy Statement*, dated February 15, 1989.[2] Those policies are directed at radiological risks associated with hypothetical severe accidents. Both short-term (prompt fatality) and long-term (latent fatality) risks are considered. Prompt fatality refers to an acute radiation dose of magnitude sufficient to cause death within a short period of time; latent fatality refers to an initially sublethal dose of radiation that may cause cancer, that in turn may cause death, perhaps years later.

The basic principle upon which both NRC and DOE policies are grounded is that radiological accident risks must be a small fraction of the risks to which individuals are normally exposed. The limit for radiological accident risk is, therefore, set by comparison with other risks.

Prompt fatality risk is compared with normal nonnuclear accident risk; latent fatality risk is compared with the normal background rate of cancer in the general population.

Two basically different populations must be considered: the off-site residents and the on-site workers and visitors. The risks attributable to nuclear accidents are allowed to cause only an insignificant increase in these pre-existing normal risks, as published by the U.S. Bureau of Census.[3] The latent (cancer) fatality risk is treated the same for both groups; that is, the basis for comparison is the background cancer death rate within the general population ($\sim 2 \times 10^{-3}$ per year per person). The prompt fatality risk limitation for the off-site residents is compared with accident fatality risk prevalent in the general population ($\sim 4 \times 10^{-4}$ per year per person). For on-site workers and visitors, the prompt fatality risk limitation is based on the average occupational death rate in the United States ($\sim 1 \times 10^{-4}$ per year per worker). The ANS risk goals are expressed in Table 4.1-1.

The risk limitation goals are conceptually simple, but achievement of the goals can be verified only after a detailed multistep process of calculations. It is possible to state simpler, more directly usable goals that, if met, ensure that the ultimate health risk limitation goals will be met. Toward this end, the following auxiliary goals are specified:

1. **Core-melt risk.** The median probability of severe core damage or meltdown because of internal events should not exceed 1×10^{-5} per year. The DOE *Safety Objectives Policy*[2] specified this goal for new production reactors. A major NRC risk study[4] of five representative existing power reactors has published core damage frequencies ranging from 8×10^{-6} to 1×10^{-4} per reactor year; 1×10^{-5} should therefore be a reasonable goal for a new plant. While not an inflexible limit, this goal shall be utilized to guide design decisions. Internal events are those initiated by equipment or operator failure.
2. **Large release risk.** The median probability of a large release should not exceed 1×10^{-6} per year, including both internal and external events. A large release is one that, considering reasonable emergency actions and realistic meteorological conditions, would be capable of causing prompt fatalities [i.e., exposure > 2 Sv (200 rem)] to workers outside the reactor containment or to the general public. This definition is consistent with that applied in NUREG-1150.[4]
3. **Inherent safety characteristics.** ANS safety-related systems (see Sect 4.7 for system classification requirements) will be designed with the maximum practicable degree of inherent or passive safety. The primary objective of passive cooling is to reduce dependence on operator actions and on active components. The capability to go into natural circulation cooling for decay heat removal shall be emphasized, and the dependence upon active components shall be minimized. The high priority on natural circulation decay heat removal extends to spent fuel cooling, reactor containment cooling, and containment or retention of fission products.

4.1.3 Pressure Boundary Integrity

The ANS primary-coolant pressure boundary shall be designed to minimize rupture probability. The reactor assembly and related systems shall be designed to maximize ability to withstand pressure-boundary ruptures. The following subgoals implement the general goal:

1. The primary coolant system piping and associated leak detection instrumentation shall be designed and analyzed in accordance with the leak-before-break evaluation procedures as detailed in Sect. 3.6.3 of the *Standard Review Plan*,[5] (NUREG-0800, as amended by the *Federal Register*, Vol. 52, No. 167, 1987).[6]

Table 4.1-1. Radiological accident risk goals for ANS

Population	Risk mode	Comparison basis for goal	Risk to average individual
Off-site residents within 1.6 km (1 mile) of reservation[a] boundary	Prompt	0.1% of all normal accident risk	4×10^{-7}/year
On-site workers and visitors within 1.6 km (1 mile) of the ANS facility security fence[b]	Prompt	0.1% of average U.S. occupational fatality risk	1×10^{-6}/year
Off-site residents within 16 km (10 miles) of the reservation boundary and on-site workers and guests	Latent	0.1% of U.S. average cancer death risk	2×10^{-6}/year

[a]This refers to the property boundary of the DOE ORR. The current preferred site for ANS is in eastern Melton Valley, about 2 km from the southern boundary of ORR, which winds along Melton Hill Lake.

[b]A security fence will surround the ANS facility to facilitate control of access to the facility buildings and immediate vicinity. Precise specification of the location of the security fence is not essential for this goal; the location of the fence will be set by security considerations.

2. With regard to a design-basis pipe break, the ANS design shall accommodate pipe break sizes up to the largest diameter of all the piping that comprises the four individual heat exchanger loops *[HOLD]*[4-1]. Risk of pipe break exceeding the design basis is minimized by the provisions of items 1 and 3 of this subsection.
3. The total mean probability of catastrophic rupture of the core pressure boundary tube (CPBT) or any primary-coolant pipe larger than the design-basis break (see item 2 above) shall be limited to $< 5 \times 10^{-7}$ per year. Catastrophic rupture is defined as any failure that initiates or results in fuel melting. This goal ensures that the risk contribution from unprotected pressure boundary failure is a small fraction of the total fuel damage risk. The current NRC screening criterion for through-wall reactor vessel crack probability is 5×10^{-6} per reactor year.[7] The goal stated in this paragraph is significantly lower for ANS because it deals with catastrophic failure, not just through-wall cracking.

4.1.4 Defense-in-Depth

The ANS reactor shall be designed in accordance with the defense-in-depth concept, in which succeeding layers of safety are built into the design and operations of the facility, and excessive reliance upon any one element is avoided.

1. The reactor shall be designed and built so that it will, with a high degree of reliability, operate without failures that could lead to accidents.

2. Protection devices and systems shall be provided to ensure that anticipated transients and off-normal conditions will be detected and either arrested or accommodated safely.
3. To provide additional margins of safety in the plant design in order to protect the public, the reactor shall be housed in a building capable of retaining radioactive nuclides that might be released in the event of a hypothetical severe fuel-damage accident.
4. To ensure public safety in the event of failure of all other levels of defense-in-depth, on-site and off-site emergency procedures shall be maintained. The on-site plans and equipment shall support prompt and accurate accident assessment and protective action decision-making for workers on the reservation, transients on or near the reservation, and residents off the reservation. Off-site plans and equipment shall support a range of protective actions (including evacuation) and the prompt communication to residents, civil authorities, and the press of any needed protective actions.

4.1.5 Respect for the Environment

The ANS reactor or facility shall not have a net harmful effect on the environment, as determined by the EIS. ANS systems will be designed to prevent pollution at all levels, utilizing benign products and processes to generate fewer potential pollution problems in the first place. This will be one of the criteria applied during design reviews of all systems.

4.1.6 Ensured Site Suitability

The ANS reactor and containment shall be designed so that the emergency planning zones for ANS are compatible with the existing ORNL planning zones,[8] including the 3.22-km (2-mile) Immediate Notification Zone (INZ) radius. Additionally, the radiation exposure criteria for the exclusion area boundary (EAB) and low population zone (LPZ) specified in the NRC 10 CFR 100, Reactor Site Criteria, must be satisfied. This requirement means that the ANS design-basis site suitability source term (SSST) accident (a fuel-damage accident more severe than any credible accident) must not cause radiation exposures exceeding the limits specified by Table 4.1-2. The goals specified in Table 4.1-2 are to be used in containment design to provide additional margin for personnel evacuation and dose avoidance in the event of a severe accident.

For calculations performed to assess compliance with the limits and goals of Table 4.1-2, an SSST has been formulated based on Regulatory Guide 1.4.[9] The recommended conditions postulate that severe fuel damage occurs and that 100% of the noble gas fission products, 25% of the iodine group nuclides, and 1% of all other fission product nuclides escape from the fuel to the containment atmosphere. Some of these radionuclides may escape from the primary containment, which is assumed to leak at its design leak rate. The noble gas nuclides pass unattenuated through the containment air filtration, but the other fission products are subject to removal by the absolute and charcoal filtration. Along with the fission product transport assumptions, Regulatory Guide 1.4 prescribes sufficiently conservative atmospheric dispersion conditions (e.g., Pasquill Type F conditions with 1 m/s wind speed for the first 8 h of the accident) that conditions more favorable to dispersion (to lower off-site radiation exposures) would prevail more than 95% of the time.

The fractional release of fission product nuclides, particularly iodine, from damaged fuel to containment atmosphere is subject to debate. The prescription in Regulatory Guide 1.4 of a 25% iodine release exceeds what would be expected for an ANS severe accident because of the large amount of water submersion employed in the ANS design. Nevertheless, to ensure a solid site

Table 4.1-2. ANS accident-related radiological exposure goals and limits

Zone	Radial distance	Time	Exposure limits[a] [Sv (rem)]	
			Thyroid	Whole body
Exclusion area boundary requirement[b]	1000 m	2 h	0.25 (25)	0.05 (5)
Exclusion area boundary goal	1000 m	4 h	0.25 (25)	0.05 (5)
Low population zone requirement[c]	2000 m	Duration of release	0.25 (25)	0.05 (5)
Immediate notification zone requirement[d]	3220 m (2 miles)	2 h	0.05 (5)	0.01 (1)
Immediate notification zone goal[e]	2500 m	24 h	0.05 (5)	0.01 (1)

[a]Maximum dose calculated at radial distance noted.

[b]Chosen to include the ANS site but to exclude workers at other major sites (e.g., HFIR and ORNL).

[c]Chosen to lie entirely within ORR. Excludes all of the general public except transients.

[d]Notification zone currently in use for ORR facilities (includes a limited number of private residences).

[e]ANS containment design goal to preclude the need for immediate notification at any private residence (i.e., inhabited location off ORR).

suitability posture, it shall be demonstrated that the containment air treatment has efficiency adequate to control iodine releases to acceptable levels, based on calculated severe accident off-site radiation exposures, even for the bounding case of 100% *[HOLD]*[4-2] escape of iodine fission product from fuel to containment atmosphere.

Criteria used to select the site shall include seismic response characteristics, wind characteristics, and flooding characteristics that are consistent with a design that meets the probability goals set out in this chapter for core damage and for release of radionuclides to the public.

4.2 SAFETY DOCUMENTATION AND PROCEDURES

4.2.1 Safety Analysis Reports

Two major documents that must be approved in order for ANS to proceed from the conceptual design stage into construction and subsequent operation are the Safety Analysis Report (SAR) and the EIS. The SAR demonstrates that worker and public radiological safety is not compromised by the proposed facility. Sufficient information shall be provided in the SAR to enable a technically competent outside reviewer to verify independently that DOE and applicable NRC regulations, standards, and guides have been followed. Per DOE Order 5480.6,[10] the ANS SARs will follow the format specified by NRC Regulatory Guide 1.70,[11] as supplemented by NUREG-0800.[5]

The SAR is written in three phases: the Conceptual SAR (CSAR), the PSAR, and the FSAR. The amount of detail in each SAR phase is consistent with the progress of the ANS design. The CSAR is written at about the same time as the Conceptual Design Report (CDR). Design detail in the CSAR will be consistent with that expected in the CDR. The PSAR is written during Title I design and is consistent, in degree of detail, with the PSAR utilized by the NRC to license nuclear power plants. The PSAR is submitted to DOE prior to the start of construction. The FSAR is submitted late in the construction phase, prior to the start of operations. The information in the FSAR is very detailed because it incorporates the final design and some as-built construction information.

The ANS is a DOE reactor and, per 10 CFR 50.11, is not subject to the NRC licensing process. The primary responsibility for the review process for SARs lies with the DOE line management groups within the Office of Nuclear Energy (NE). Primary review is provided by safety review groups within DOE, such as the Office of Self Assessment within NE and the Office of Nuclear Safety, reporting directly to the Office of the Secretary of Energy.

4.2.2 National Environmental Policy Act Process

The National Environmental Policy Act (NEPA) process consists of several essential actions. The first is publication of a notice of intent (NOI) to prepare an EIS in the *Federal Register*. Publication of the NOI formally begins the process. The NOI invites comments and suggestions on the scope of the EIS, including environmental issues and alternatives, and gives notice of planned scoping meetings. DOE requires preparation of an EIS implementation plan that should be made available to the public as soon as practical after the conclusion of the scoping period.

Prior to the preparation of the draft EIS, an ER has been prepared by the ANS Project. The ER is a document first developed for the NRC to facilitate preparation of an EIS in support of a nuclear power plant license. The ER is prepared by NRC license applicants following Regulatory Guide 4.2.[12] Data in the ER are also used by an independent organization preparing the EIS. Since the data needed to prepare the draft EIS are a subset of the requirements of Regulatory Guide 4.2, the ER for ANS will be prepared in two phases. Phase I will include those data and analyses that are needed to prepare the EIS. Phase II will include the additional data and analyses called for by Regulatory Guide 4.2.

Preparation of the draft EIS is the major effort of the NEPA process. The draft EIS will be prepared by an independent organization under contract to DOE.

When the draft EIS is complete, it is filed with the U.S. Environmental Protection Agency (EPA), which publishes a notice of availability (NOA) in the *Federal Register*. Publication of the

NOA begins a comment period during which public hearings may be held. The final EIS is published after revisions in response to public comments. The final EIS is filed with EPA, and once again an NOA is published in the *Federal Register*. The EIS process ends with publication (in the *Federal Register*) of a record of decision (ROD) no less than 30 days after publication of the final EIS NOA.

4.2.3 Other Permits

In addition to the SARs and the NEPA process discussed above, ANS will obtain all permits required under the Federal Facilities Agreement,[13] which includes discharge and emission permits required by the state and by the EPA.

4.3 DOE ORDERS

4.3.1 DOE Order 5480.6, Safety of Department-Owned Nuclear Reactors

DOE Order 5480.6[10] provides the primary guidance on safety requirements for DOE-owned reactors, including construction of new DOE-owned reactor facilities. The purpose of DOE 5480.6 is to ensure that (1) the safety of each DOE-owned reactor is properly analyzed, evaluated, documented, and approved by DOE, and that (2) reactors are sited, designed, constructed, modified, operated, maintained, and decommissioned in a manner that gives adequate protection for public health and safety and that will be in accordance with uniform standards, guides, and codes that are consistent with those applied to comparable NRC-licensed reactors.

The ANS will be classified as a Category A reactor (greater than 20-MW steady-state).

In Sect. 8.a, DOE 5480.6 directs that in the selection of the site for a new reactor, 10 CFR 100 shall be applied. This CFR (including Appendix A) shall be used to set the necessary site characteristics for ANS and shall be applied in establishing the seismic and geologic design bases for the design. The ANS Project plans to consider differences between the ANS research reactor and power reactors in determining the operating basis earthquake (OBE); however, all deviations from 10 CFR 100, Appendix A, shall be clearly noted and agreed to by the appropriate DOE regulatory groups.

In Sect. 8.b, DOE 5480.6 directs that the General Design Criteria (GDC) specified in 10 CFR 50, Appendix A, shall be applied to all new construction of DOE-owned reactor facilities; 10 CFR 50, Appendix A, shall therefore be applied to the design of ANS as outlined in Table B.5, Appendix B of this PDR.

In Sect. 8.c, DOE 5480.6 directs that the requirements of DOE 5481.1B[14] shall be applied. In addition, it directs that all new SARs shall follow NRC's guidelines on standard format and content. Some exceptions are given, but ANS shall follow the provisions of Sect. 8.c of DOE 5480.6.

In Sect. 8.d, DOE directs that each DOE-owned reactor shall have a technical specification document meeting the requirements of 10 CFR 50.36. Such a technical specification document shall be prepared for ANS.

Sect. 8.e provides considerable direction for operating personnel training and qualification. While not generally applicable to the design of ANS, these requirements will be followed in developing plans and procedures for operator training and certification. However, referenced documents (e.g., the American Nuclear Society Standard ANSI/ANS 3.1,[15] *Selection,*

Qualification, and Training of Personnel for Nuclear Power Plants) refer to the use of a simulator for operator training. A simulator shall be included as part of the ANS Project.

Other sections of DOE 5480.6 apply to organizational and programmatic issues associated with the design, operation, quality assurance, decommissioning, and responsibilities for DOE-owned reactors.

4.3.2 DOE Order 6430.1A, General Design Criteria

DOE Order 6430.1A[16] provides mandatory requirements for the design of DOE facilities. It applies to any building acquisition or new facility that is classified as real property under DOE 4300.1B.[17] As noted under the General Requirements for Special Facilities (1300-1.1, Coverage), the criteria apply to nonreactor facilities. Reactors and their safety systems shall be sited and designed in accordance with DOE 5480.6.

There are two key reasons to use DOE 6430.1A as well as DOE 5480.6 in the design of ANS.

1. DOE 6430.1A provides detailed criteria for design and construction standards for most traditional engineering disciplines. Such criteria are not explicitly provided in DOE 5480.6 or in its referenced NRC standards.
2. Many portions of the ANS Project can be viewed as nonreactor facilities included in a reactor project and are not directly associated with the reactor and its safety systems. An example is the detritiation plant, which is likely to be a stand-alone facility located on the ANS site. Many of the laboratory, office, service, waste, and other facilities are also not directly related to the reactor or its associated safety systems.

Therefore, the design of ANS shall in general follow the design criteria given in DOE 6430.1A. Application of DOE 6430.1A at DOE's ORR is governed by Y/EF-538, *Master Design Criteria*.[18] When specific requirements for the design and classification of reactor and reactor safety systems are given in DOE 5480.6 or its referenced documents, DOE 5480.6 shall take precedence over DOE 6430.1A.

The ANS safety classes and applicable industry codes are given in Sects. 4.7.4–4.7.9. ANS safety classes SC-1, SC-2, and SC-3 are intended to be equivalent to DOE SC-1 as defined in the DOE 6430.1A glossary. Also, ANS nonreactor safety (NRS) classes are intended to be equivalent to DOE SC-2 and DOE SC-3.

The ANS classifications and choices are consistent with DOE 6430.1A. For example, DOE 6430.1A, Sect. 1300-3.2, recommends application of American Society of Mechanical Engineers (ASME) Sect. III, Class 2, to all DOE safety class items, with the option of using other, more appropriate codes. The ANS Project invokes this option, thereby meeting the intent of DOE 6430.1A while using standard industry practices for meeting regulatory requirements. In addition, the seismic and tornado categories and loadings defined in Sects. 4.7.6, 4.7.7, and 6.2.3 meet both NRC and DOE 6430.1A, Sect. 0111-99.0, requirements.

Other DOE orders outline the health, safety, and environment programs that must be met in the design of ANS and provide a range of guidance applicable to the ANS Project. A list of key DOE orders relevant to the design of ANS is provided in Table 4.3-1. A complete list of such DOE orders is provided in Appendix B, Table B.1. To complete the list of DOE required and recommended standards, Table B.2 lists Oak Ridge Field Office Orders applicable to the ANS, and Table B.3 lists standards required or referenced by DOE.

Table 4.3-1. DOE orders relevant to the ANS conceptual design

Number	Date	Subject
Accidents/Emergencies		
DOE 5500.3, Change 1	07-02-90	Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for DOE Operations
Accountability		
DOE 5633.3	02-03-88	Control and Accountability of Nuclear Materials
Appraisals		
DOE 5481.1B	09-23-86	Safety Analysis and Review System
DOE 5482.1B	09-23-86	Environmental Protection, Safety, and Health Appraisal System
Design Criteria		
DOE 6430.1A	04-06-89	General Design Criteria
Environment		
DOE 5480.1B	09-23-86	Environmental Protection, Safety, and Health Protection Program for DOE Facilities
DOE 5480.4	05-15-84	Environmental Protection, Safety, and Health Standards
DOE 5440.1C	04-09-85	Implementation of the National Environmental Policy Act
Quality Assurance		
DOE 5700.6B	09-23-86	Quality Assurance
Safeguards and Security		
DOE 5632.2A	09-09-88	Physical Protection of Special Nuclear Material and Vital Equipment
DOE 5632.6	02-09-88	Physical Protection of DOE Property and Unclassified Facilities
Safety and Health		
DOE 5400.5	02-08-90	Radiation Protection of the Public and the Environment
DOE 5480.5	09-23-86	Safety of Nuclear Facilities
DOE 5480.6	09-23-86	Safety of DOE-owned Nuclear Reactors
DOE 5480.7	11-16-87	Fire Protection
DOE 5480.11	07-20-89	Radiation Protection for Occupational Workers
Waste Management		
DOE 5820.2A	09-26-88	Radioactive Waste Management

4.4 NRC REGULATIONS AND GUIDES

4.4.1 10 CFR 50

According to DOE Order 5480.6, the GDC given in Appendix A of 10 CFR 50 shall be applied to the design of ANS. Application of those GDC that address the design at a plant level is given in Table B.5 of Appendix B of this PDR. Application of those GDC that address the functions of specific systems is addressed in the appropriate SDD document.

According to DOE Order 5480.6, a set of technical specifications will be prepared for ANS in accordance with 10 CFR 50.36. Further guidance on technical specifications for research reactors is given in the American Nuclear Society Standard ANSI/ANS-15.1 (ANSI N378) 1990.[19]

The ANS QA program shall meet the intent of 10 CFR 50, Appendix B, as outlined in Chap. 7 of this PDR.

In addition to the specific paragraphs and appendices outlined above, all of 10 CFR 50 shall be reviewed for applicability to ANS. Paragraphs and appendices of 10 CFR 50 that address issues directed at power reactors, but which are also issues important to ANS, shall be followed in the design of ANS. Paragraphs and appendices of 10 CFR 50 for which application is directed by relevant DOE orders other than those described above shall be followed in the design of ANS. Those requirements that apply to ANS at the overall plant level are defined in Table B.4 of Appendix B of this document. Those requirements that apply at the system level will be defined in the SDD documents or in documents referenced by the SDDs. Those requirements that are deemed not to apply to ANS will also be defined in the PDR (or referenced documents) or in the individual SDDs, along with the rationale for nonapplication.

4.4.2 10 CFR 100

Consistent with DOE Order 5480.6, the reactor site criteria outlined in 10 CFR 100 shall be applied to the siting process used for ANS. The manner in which these criteria are applied shall be clearly outlined in the ANS site selection report.

Consistent with DOE Order 5480.6, the classifications for safe shutdown earthquake (SSE) and OBE shall be applied to ANS. Consideration of differences in the missions of power and research reactors may be taken into account in defining the rationale for designating an earthquake as OBE. Definitions of SSE and an appropriate equivalent to the OBE are referenced in Sect. 6.2.3.

4.4.3 Principle of Comparability

DOE 5480.6 directs that DOE-owned reactors be sited, designed, constructed, modified, operated, maintained, and decommissioned in accordance with uniform standards, guides, and codes that are consistent with those applied to comparable licensed reactors. This statement is referred to as the principle of comparability. To ensure that this principle is followed, all of 10 CFR 50 shall be reviewed for applicability to ANS.

In accordance with the principle of comparability established in DOE 5480.6, all other parts of Title 10 of the CFR shall be reviewed for applicability to the design of ANS. A list of the key parts that may be applicable is given in Table 4.4-1. Standards applicable to the overall ANS facility shall be implemented in Appendix B of this PDR, and those applicable to an individual ANS system shall be implemented in the relevant SDD. A detailed list of applicable parts of the

Title 10 CFR shall be prepared at the plant level and at the system level for each system. All parts of the Title 10 CFR considered fully applicable shall be indicated. All parts and paragraphs of the Title 10 CFR considered not applicable to ANS, or for which a modified approach toward the issue is proposed, shall be indicated; a rationale for the designation of not applicable or for the modified approach toward compliance shall be detailed. At a minimum, summary tables shall appear in the PDR and SDDs. A fully detailed report shall be provided as part of the SAR or the associated documentation supporting the DOE certification to construct and operate ANS.

4.4.4 NRC Regulatory Guides

Regulatory guides are issued to describe methods acceptable to the NRC staff for implementing specific parts of the NRC regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, and to provide guidance to license applicants. A notice in each regulatory guide states that the guides are not substitutes for regulations and that compliance with them is not required. It also states that methods and solutions different from those set out in the guides will be acceptable to the NRC if they provide a basis for the relevant licensing action.

Many of the regulatory guides address the specific implementation of the general design criteria given in 10 CFR 50, Appendix A. Others address safety and natural phenomena-resistance classification systems, the format and content of safety and environmental reports, and a wide variety of topical issues (some of wide applicability and some very specific to a given reactor type).

In accordance with the principle of comparability established in DOE 5480.6, all of the NRC regulatory guides shall be reviewed for applicability to the design of the ANS reactor. A detailed list of applicable regulatory guides shall be prepared at the plant level and at the system level for each system. All fully applicable regulatory guides shall be indicated. All nonapplicable regulatory guides, or those for which a modified approach is proposed, shall be indicated, and a rationale for such a designation shall be detailed. At a minimum, summary tables appear in the PDR, Appendix B, and shall appear in the SDDs (possibly as an appendix). A fully detailed report shall be provided as part of the SAR or associated documentation supporting the certification of ANS. The draft Reference Document List for ANS[20] served as a preliminary source for Appendix B of the PDR.

4.4.5 NRC *Standard Review Plan* and *Branch Technical Positions*

The NRC *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (Light Water Edition)*[5] provides guidance to NRC staff reviewers in performing safety reviews of applications to construct or to operate light-water-reactor (LWR) nuclear power plants. The principal purpose of the *Standard Review Plan* (SRP) is to ensure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the NRC staff review process by members of the public and by the nuclear power industry. SRPs are not substitutes for regulatory guides or NRC regulations, and compliance with them is not required. However, SRPs do provide an indication of the contents of the SARs that the NRC reviews as part of its licensing process. *Branch Technical Positions*, published along with the SRPs, further amplify the expectations of specific NRC review branches as to the contents of the SARs submitted with the license applications.

Table 4.4-1. NRC requirements applicable to ANS

Requirement compliance	Type	DOE 5480.6 mandate[a]	ANS
10 CFR 50, Domestic Licensing of Production and Utilization Facilities	Regulation	General	See note[b]
10 CFR 50.36, Technical Specifications	Regulation	Specific	Comply
10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants	Regulation	Specific	See note[c]
10 CFR 100, Reactor Site Criteria	Regulation	Specific	See note[d]
10 CFR 20, Standards for Protection Against Radiation	Regulation	General	See note[e]
10 CFR 73, Physical Protection of Plants and Material	Regulation	General	See note[e]
Other parts of Title 10	Regulation	General	See note[b]
Regulatory Guides: Division I – Power Reactors	Guides	General	See note[b]
Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	Guide	Specific	Comply
Regulatory Guides: Division II – Research and Test Reactors	Guides	General	See note[b]

[a]Those NRC requirements that are specifically mandated by DOE 5480.6 are designated as "specific." Other key requirements designated as "general" are those that should be considered as applicable to comparable licensed facilities.

[b]Applicability to the ANS reactor of all NRC requirements is to be determined as part of the technical work of the conceptual design phase. Nonapplicability or exceptions to NRC requirements will be justified in the CSAR. Exceptions pertinent to individual systems will be listed in the SDDs.

[c]Compliance with 10 CFR 50, Appendix A, General Design Criteria is required by DOE 5480.6; but specific exception to selected criteria may be justified in the CSAR.

[d]Compliance with 10 CFR 100 is intended, but exception to the 10 CFR 100, Appendix A, definition of OBE may be justified in the CSAR.

[e]DOE standards for radiation protection and security will be reviewed for consistency with NRC regulations. In cases where DOE standards fall short of requirements for comparable NRC-licensed facilities, the ANS Project team will recommend to DOE designs and operating procedures that meet the NRC requirements.

Although the SRPs address LWRs and the licensing procedures used by NRC rather than by DOE, they do provide insight into the contents of the SARs that will be generated for ANS. They also provide insight into the design requirements for the overall ANS plant and for specific ANS systems and components. Thus, the SRP and the *Branch Technical Positions* provide a checklist of requirements and information to be generated for ANS. Cases in which the plans and positions are deemed not to be applicable to ANS or where modified approaches are proposed shall be

documented. The draft Reference Document List for ANS is a preliminary source of such documentation. Appendix B, Table B.7, and Table B.8 provide additional guidance.

4.5 SUPPLEMENTAL DESIGN CRITERIA FOR ANS

Supplemental Design Criteria (SDC) have been generated for ANS because of particular features of its design. These SDCs are listed in Table B.6 of Appendix B.

4.6 OTHER APPLICABLE REGULATIONS AND STANDARDS

Table B.9 of Appendix B provides a listing of NRC regulations and publications that are of potential interest to ANS. Table B.10 provides a listing of Energy Systems' policy procedures that apply to the design, construction, and operation of ANS. Table B.11 lists standards and codes that are either required or recommended by these procedures.

4.7 SAFETY SYSTEM CLASSIFICATIONS

4.7.1 NRC Quality Group Classifications

NRC bases its standards and reviews on a four-level quality group system. The following quality group definitions are based on 10 CFR 50.55a, as further amplified in Regulatory Guide 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants* (February 1976 draft),[21] with a minor modification (indicated in italics in item 3 below) to reflect the systems that compose the ANS reactor plant.

4.7.1.1 Quality Group A

Quality Group A includes components of the reactor coolant pressure boundary. Quality Group A components must meet the requirements for Class 1 components, as indicated in Sect. III of the *ASME Boiler and Pressure Vessel Code.*

4.7.1.2 Quality Group B

Quality Group B includes components that are not part of the reactor coolant pressure boundary but that are part of the following:

1. systems or portions of systems important to safety and designed for (a) emergency core cooling, (b) postaccident containment heat removal, or (c) postaccident fission product removal;
2. systems or portions of systems important to safety and designed for (a) reactor shutdown or (b) residual heat removal;
3. those portions of *the secondary cooling system* extending from and including the secondary side of the primary-heat exchangers up to and including the outermost containment isolation

valves and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation; and

4. systems or portions of systems connected to the reactor-coolant pressure boundary and incapable of being isolated from the boundary during all modes of normal reactor operation. Such isolation is achieved by two valves, each of which is either normally closed or capable of automatic closure.

Quality Group B components must meet the requirements for Class 2 components in Sect. III of the *ASME Boiler and Pressure Vessel Code.*

4.7.1.3 Quality Group C

Quality Group C includes water-containing and radioactive-waste-containing pressure vessels, heat exchangers, storage tanks, piping, pumps, and valves not part of the reactor-coolant pressure boundary or included in Quality Group B. Instead, they are part of the following.

1. Cooling water systems or portions of systems important to safety and designed for (a) emergency core cooling, (b) postaccident containment heat removal, (c) postaccident containment atmosphere cleanup, or (d) residual heat removal from the reactor and from the spent-fuel storage pool (including primary- and secondary-cooling systems). Portions of these systems that are required for their safety functions and that (a) do not operate during any mode of normal reactor operations and that (b) cannot be tested adequately should be classified as Group B.
2. Cooling-water and seal-water systems or portions of those systems important to safety and designed to aid the functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and the control room.
3. Systems or portions of systems that are connected to the reactor-coolant pressure boundary and that are capable of being isolated from that boundary during all modes of normal reactor operation. Such isolation is achieved by two valves, each of which is either normally closed or capable of automatic closure.
4. Systems, other than radioactive waste-management systems, not covered by items within Quality Groups A, B, or C and containing or possibly containing radioactive material and whose postulated failure would result in conservatively calculated potential off-site doses exceeding 0.5 rem to the whole body or its equivalent to any part of the body. For those systems located in Seismic Category I structures, only single component failures need be assumed. (However, no credit for automatic isolation from other components in the system or for treatment of released material should be taken unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of off-site power and a single failure of an active component.)

Quality Group C components must meet the requirements for Class 3 components, as indicated in Sect. III of the *ASME Boiler and Pressure Vessel Code.*

4.7.1.4 Quality Group D

Quality Group D includes water-containing components that are not included in the reactor-coolant pressure boundary nor in Quality Groups B or C but that are part of systems or portions of systems that contain or may contain radioactive material.

4.7.2 American Nuclear Society Standards for Nuclear Safety Criteria

The American Nuclear Society has prepared a set of American National Standards directed toward developing a set of design requirements for nuclear and nonnuclear safety class items in terms of industry codes and standards for categories of plant conditions. These standards go beyond such guidance as the NRC Quality Groups in an attempt to establish a consistent set of classifications and requirements for LWR nuclear power plants. The American Nuclear Society standards are an attempt to be responsive to both the regulatory requirements of the NRC and to the design and technical requirements of industry codes and standards in a framework that allows the augmentation of criteria as additional standards are developed in the nuclear industry. The approach developed through these standards has gained a wide acceptance in the nuclear industry. The following sections adapt the classification systems proposed by the American Nuclear Society in ANSI/ANS-51.1-1983, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*,[22] to the needs of the modest-pressure, low-temperature, high-power-density heavy-water reactor of ANS.

4.7.3 Plant Conditions

A wide range of normal, off-normal, and accident events shall be considered in the design of ANS. To ensure that appropriate acceptance criteria are applied to the complete spectrum of events from mild to severe and to aid in presenting the results of safety analyses, events are grouped into categories determined by expected frequency of occurrence. The system of event categories adopted by the ANS Project is generally that outlined by ANSI/ANS 51.1-1983. In place of the ANSI/ANS PC-1 through PC-5 designators, ANS has retained the use of the descriptive names for each of the categories: normal operations, anticipated events, unlikely events, and extremely unlikely events. As explained in the following sections, the frequency limits for each category *are* in accordance with the ANSI/ANS standard. Design-basis events that fit into each event category and the accompanying acceptance criteria for ANS are given in Sect. 4.8.

4.7.3.1 Normal Operations

This category covers all scheduled plant operations and operational evolutions and is equivalent to the ANSI/ANS PC-1 category. There is no specific frequency limit, but these events are expected to occur numerous times each year. Acceptance criteria for normal operations are posed to maintain the reactor safety-related parameters within proven ranges for long-term operations.

4.7.3.2 Anticipated Events

This category combines ANSI/ANS categories PC-2 and PC-3 for all purposes except the ALARA analysis (see Sect. 4.8.1.3). Off-normal events and accidents with estimated mean

frequencies greater than 0.01 per year are included. The purpose of the anticipated-events category is equivalent to that of the current anticipated-operational-occurrences (AOOs) class of events defined in Appendix A of 10 CFR 50, *General Design Criteria for Nuclear Power Plants*. The frequency range for these events goes down to 10^{-2} per year, which corresponds to the frequency of events that may be expected to occur one or more times during the life of the plant.

The acceptance criteria for thermal-hydraulic and mechanical performance of the reactor fuel and other critical in-reactor components are devised to ensure that there is no damage and that there is no structural weakening or degradation of capacity to retain radioactivity. That is, the fuel and other critical components must remain fully within the design envelope as defined by the facility technical specifications. After anticipated events resolved by reactor shutdown, a subsequent restart (after appropriate administrative approvals and after any necessary xenon decay period) and continued safe operation would be possible.

4.7.3.3 Unlikely Events

This category corresponds to ANSI/ANS category PC-4 and includes events having the following characteristics:

1. Selected according to PRA methods, complemented by traditional engineering judgment, and including internal events with estimated mean frequency less than 10^{-2}/year, but more than 10^{-4} per year.
2. Including a comprehensive selection of external events (see Sect. 4.8.2 for a minimum list of such events).
3. Subjected to single failure criteria and other NRC-licensing conservatisms (no credit for nonsafety-grade equipment). Events within this category would require conservative analysis as presently done for LWRs.

The acceptance criteria for fuel performance have the objective of ensuring that no significant fuel damage occurs as a result of any unlikely event. Extensive inspections, assessments, and probable replacement of the reactor core would be necessary prior to restart following an unlikely event.

4.7.3.4 Extremely Unlikely Events

This category of events, the same as ANSI/ANS 51.1, Category PC-5, will be used by designers in establishing design bases, especially those related to the reactor containment. The events in this category will be selected PRAs complemented by engineering's best judgment. This approach is consistent with guidance provided in NRC Safety Goal and Severe Accident Policies,[23] which encourage the use of PRA methods to supplement engineering judgment and deterministic (nonmechanistic) analyses.[24] Specifically, extremely unlikely events would include internal events with estimated mean frequency less than 10^{-4} per year, but greater than a nominal lower cut-off of 10^{-6} per year. Fuel damage associated with events in the extremely unlikely category shall, to the extent practicable, be minimized by design, as guided by the ANS fuel damage and large release goals (see Sect. 4.1.2).

4.7.3.5 Beyond-Design-Basis Accidents

In order to ensure that ANS has an adequate degree of in-depth defense against severe accidents, a specific group of severe accidents has been defined: the beyond-design-basis accidents (BDBAs). BDBAs are conservatively hypothesized accidents that bound the consequences of any of the design-basis events (i.e., extremely unlikely events that may result in fuel damage). By definition, the frequency range for BDBAs is below the 10^{-6}/year cut-off for extremely unlikely events; and, because of the large degree of conservatism employed in formulating the BDBAs, their expected frequency is well below the 10^{-6}/year threshold. The primary use of the BDBAs will be for best-estimate severe-accident analyses (1) to guide the development of mitigative design features to prevent uncontrolled release of radioactivity and (2) to allow the assessment of safety margins against severe accidents. Appendix C describes the ANS BDBAs and outlines how they were derived.

ANSI/ANS-51.1 provides guidelines relevant to the use of the various event categories, and the ANS Project intends to follow these guidelines. For example, the guidelines for assessing the best-estimate frequency of occurrence of single events in combination with single failures or coincident occurrences are specified. These guidelines for event combination are especially helpful when complete and detailed PRA results are not yet available. Of course, the ANS Project reserves the prerogative to deviate from the 51.1 event combination and single failure rules when detailed PRA calculations based on the ANS Conceptual Design Report become available.

4.7.4 Safety Classifications for Systems, Structures, and Components Affecting Reactor Safety

Equipment that is necessary to accomplish a reactor safety function shall be assigned to one of three reactor safety classes, as identified in Table 4.7-1. These classifications correspond to the recommendations of ANSI/ANS-51.1-1983, modified to account for the differences between ANS and a pressurized LWR. Because of the small core, the high-power density, and the high coolant-flow rates of ANS, systems or portions of systems that maintain core geometry or provide core support and whose failure could initiate a core disruptive accident are assigned to SC-1.

Table 4.7-1. Classifications of components affecting reactor safety

	Class	Definition
1.	SC-1	Those systems or portions of systems that satisfy one or both of the following: • comprise part of the reactor coolant pressure boundary, except as noted under SC-2, below, or • maintain core geometry or provide core support and whose failure could initiate a core disruptive accident.*

Table 4.7-1 (continued)

	Class	Definition
2.	SC-2	Those systems or portions of systems not in SC-1 that satisfy one or more of the following: • are part of reactor coolant pressure boundary but are excluded from SC-1 by 10 CFR 50.55a(c); • are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary by two valves, each of which is normally closed or capable of automatic closure; • are required to maintain an adequate reactor coolant inventory following a reactor coolant boundary leak;* • are part of or an extension of the reactor containment boundary;* • are required for emergency core cooling, postaccident containment heat removal, or postaccident fission product removal; • are required for reactor shutdown or residual heat removal; • are required to remove residual heat from the reactor core or from the spent fuel storage and whose single failure following any plant condition constitutes a loss-of-safety function; or • are not normally operating or cannot be tested adequately during normal power operation.
3.	SC-3	Those systems or portions of systems not in SC-1 or SC-2 that satisfy one or more of the following: • are secondary coolant systems that are required for emergency core cooling, for postaccident containment heat removal, for postaccident containment atmosphere cleanup, or for removal of residual heat from the reactor core or from spent fuel storage; • are cooling water systems and seals that are required for functioning of reactor safety-related components and systems whose failure could result in the loss-of-safety function (e.g., loss of cooling to components that require cooling for accomplishing their safety function);

Table 4.7-1 (continued)

Class	Definition
	• are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal operation by two valves, each of which is either normally closed or capable of automatic closure; or • are systems whose failure could result in release to the environment of radioactivity and that would result in calculated potential exposures at the site boundary in excess of 5 mSv (0.5 rem) whole body (or its equivalent).

*This additional definition is not included in the NRC Quality Groups defined in Regulatory Guide 1.26.

The reactor safety classifications shown in Table 4.7-1 are compatible with the NRC Quality Groups defined in Sect. 4.7.1. SC-1 includes all components in Quality Group A. SC-2 includes all components in Quality Group B. SC-3 includes all components in Quality Group C. Items in Quality Group D do not directly affect reactor safety and are covered in the following section.

Electrical components of systems necessary for SC-1, SC-2, and SC-3 systems to accomplish their reactor safety function shall be classified 1E. In addition, electrical components of any system shall be classified 1E if they are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material into the environment. Pressure-retaining portions of electrical systems may be classified as SC-1, SC-2, or SC-3, as explained in Sects. 4.7.8 and 4.7.9.

Detailed lists of the safety classifications of all components that are important to reactor safety shall be included in the SDD that covers the design of the individual component. When a single item of equipment, or a portion thereof, provides two or more functions of different classes, it shall be classified to the more stringent class.

4.7.5 Safety Classifications for Systems, Structures, and Components Not Affecting Reactor Safety

Systems and components not affecting reactor safety shall be classified as nonreactor safety classes NRS-4 through NRS-0, as shown in Table 4.7-2. ANS classes NRS-4 through NRS-0 include all systems, components, and structures not in reactor safety classes SC-1, SC-2, or SC-3. Items in NRC Quality Group D are included in the NRS-3 class. This notation corresponds to the nonnuclear safety class of ANSI/ANS-51.1.–1983.

The five nonreactor safety classes shown in Table 4.7-2 are presented and discussed as hazard categories in UCRL-15910,[25] which defines categories one through four, and the Lawrence

Livermore National Laboratory (LLNL) draft report *Technical Basis, Supporting Information, and Strategy for Development and Implementation of DOE Policy for Natural Phenomena Hazards*,[26] which introduces category zero. UCRL-15910 supplements DOE 6430.1A by providing details of hazard categories and by providing requirements for seismic, wind, and flood design considerations and calculations. These categories apply to both existing and new DOE facilities and provide a graded approach to the requirements of DOE 6430.1A based upon function and performance requirements. Most ANS nonreactor safety components, systems, and structures will be classified as NRS-3, NRS-2, NRS-1, or NRS-0.

The classification of systems and components not affecting reactor safety shall be identified in the SDD that covers the design of the individual system or component. NRS systems, components, and structures needed for purposes of plant investment protection shall be classified as NRS-1 or higher. Additional guidance concerning application of NRS classes can be obtained from the report *Hazard Screening Application Guide*,[27] and from an ANS Project Letter addressing this issue.[28]

Table 4.7-2. Classifications of systems, structures, and components not affecting reactor safety

Class	UCRL-15910 hazard category[a]	Proposed performance categories from LLNL draft report[b]	UCRL-15910 performance goals[c]
NRS-4	High hazard	4	1×10^{-5}
NRS-3	Moderate hazard	3	1×10^{-4}
NRS-2	Important or low hazard	2	5×10^{-4}
NRS-1	General use	1	1×10^{-3}
NRS-0		0	N/A

[a]Category 5, Reactor Facilities, is included in reactor safety classes SC-1 through SC-3 of Table 4.7-1. Category 5 components are beyond the scope of UCRL-15910.

[b]*Technical Basis, Supporting Information, and Strategy for Development and Implementation of DOE Policy for Natural Phenomena Hazards, UCRL-ID-108242*, Lawrence Livermore Natl. Lab., September 1991.

[c]Goals for facility performance during natural phenomena hazards have been selected and expressed in UCRL-15910 as the annual probability of exceeding a particular level of damage. This allowed level of damage depends on the facility characteristics. For example, the performance goal for general-use facilities allows greater damage than that allowed for high-hazard facilities.

4.7.6 Seismic Classifications

4.7.6.1 Seismic Category I

Seismic Category I includes all ANS structures, systems, and components whose function is important to reactor safety. This classification includes all systems and components classified as

SC-1, SC-2, or SC-3 and all structures whose integrity is necessary to allow these systems to perform their safety function in the event of severe earthquakes. Seismic Category I-L is a subset of the Seismic Category I. This classification includes those NRS systems, components, and structures that must retain limited structural integrity because their failure could jeopardize to an unacceptable extent the achievement of a safety function or because they form an interface between Seismic Category I and nonseismic category plant features. The limited structural integrity requirements associated with these plant features are either position retention (remain in place) called 1-LB or pressure boundary and position retention called 1-LA. Detailed design and qualification requirements for Seismic Category I-L will be developed in Category A references to Sec. 6.2.3. The classification of Seismic Category I and Seismic Category I-L structures, systems, and components is to conform with the requirements of NRC Regulatory Guide 1.29,[29] *Seismic Design Classification*, Subsects. 1 and 2 respectively, and NRC Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems*, Subsect. C5.[30] Seismic Category I structures, systems, and components shall be designed to remain functional during an SSE (referenced in Sect. 6.2.3) without loss of capability to perform their safety function. Seismic Category I-L structures, systems, and components shall be designed to ensure continued functionality of Category I systems, components, and structures during an SSE. Seismic Category I structures, systems, and components shall also be capable of withstanding the effects of the OBE (referenced in Sect. 6.2.3) and remaining functional *[HOLD]*[4-3]. Seismic Category I systems and components shall not be located within Seismic Category I-L buildings unless it can be demonstrated that these Seismic Category I systems will remain functional for SSE and OBE events.

4.7.6.2 Seismic Category II

Seismic Category II includes those structures, systems, and components that are not included in either Seismic Category I or I-L but that are essential to maintaining support of normal operations or preventing the release of hazardous materials. For their designated hazard classification, Seismic Category II structures, systems, and components shall fulfill the requirements of DOE 6430.1A and UCRL-15910.

4.7.7 Tornado and Wind Classifications

4.7.7.1 Wind Category I

Wind Category I includes all ANS structures, systems, and components whose function is important to reactor safety. This classification includes all systems and components classified as SC-1, SC-2, or SC-3, and all structures whose integrity is necessary to allow these systems to perform their safety function in the event of severe winds. Wind Category I-L, a subset of the Wind Category I definition, includes all structures, systems, and components that are not in a reactor safety class but that require analysis to ensure continued functioning of Wind Category I components. This classification includes those NRS systems, components, and structures whose failure could prevent the functioning of any Wind Category I system, component, or structure. The classification of Wind Category I and Wind Category I-L structures, systems, and components is to conform with the requirements of NRC Regulatory Guide 1.117,[31] *Tornado Design Classification*. Wind Category I structures, systems, and components shall be designed to remain functional during a design basis tornado (DBT) (referenced in Sect. 6.2.3) without losing

the capability to perform their safety function. Wind Category I-L structures, systems, and components shall be designed to ensure continued functionality of Category I systems, components, and structures during a DBT. Analysis of tornado wind shall include the effect of translational and rotational wind velocity, including internal pressure changes associated with tornado winds for both Category I and I-L structures, systems, and components during a DBT.

Wind- or tornado-generated missiles shall not prevent the functioning of Wind Category I structures, systems, or components, either directly or indirectly. Wind Category I structures, systems, and components shall be evaluated for the DBT missile spectrum. Wind Category I-L structures, systems, and components shall be evaluated for the controlling load case of either the missile spectrum associated with their NRS classification or the Category I DBT missile spectrum. A Wind-Category-I-affected structure, system, or component may be protected from the failure of the Wind Category I-L structure, system, or component in lieu of designing Wind Category I-L structures, systems, and components for the controlling missiles.

4.7.7.2 Wind Category II

Wind Category II includes those structures, systems, and components that are not included in either Wind Category I or Wind Category I-L but that are essential to maintaining support of normal operations or to preventing the release of hazardous materials. Wind Category II structures, systems, and components shall fulfill the requirements of DOE 6430.1A and UCRL-15910 for their designated hazard classification.

4.7.8 Code and Industry Classifications of Systems and Components

ASME Sect. III, Class 1, shall be applied to all pressure vessels, piping, supports, pumps, and valves classified as SC-1 components. ASME Sect. III, Class CS, shall be applied to all core support components classified as SC-1 components. This code classification includes the structure or parts of a structure that are designed to provide direct support or restraint of the core within the CPBT.

ASME Sect. III, Class 2, shall be applied to all pressure vessels (except primary containment), piping, supports, pumps, and valves classified as SC-2 components. ASME Sect. III, Class MC, shall be applied to the metal primary containment vessel. The containment system includes the containment vessel; all penetration assemblies (such as electrical) or appurtenances attached to the containment vessel; and all piping, pumps, and valves attached to the containment vessel or to penetration assemblies out to and including any valves required to isolate the system and provide a pressure boundary for the containment function.

ASME Sect. III, Class 3, shall be applied to all pressure vessels, piping, and valves classified as SC-3 components.

ASME Sect. VIII shall be applied to all pressure vessels (as defined by the ASME Code) classified as NRS components. ANSI B31.1 (power plant piping); ANSI B31.3 (chemical plant piping); or ANSI B31.5 (refrigeration piping), as indicated in the SDD for that system, shall be applied to all piping systems classified as NRS-3. Codes and standards applicable to atmospheric storage tanks, 0-103 kPa(g) (0-15 psig) storage tanks, and supports are shown in Table 4.7-3 with their classification.

Institute of Electrical and Electronic Engineers (IEEE) Class 1E, as defined in IEEE Standard 308-1980,[32] shall be applied to electrical power systems necessary to allow SC-1, SC-2, and SC-3 components and systems to accomplish their safety function. The electrical portions of

components and systems classified as Class IE shall, as a minimum, comply with the applicable requirements of IEEE Standards 308 and 603.[33]

These code and industry classifications are to be used in design as a minimum. The designers shall verify the correctness of their classifications and shall augment or develop their own requirements, with appropriate justification, in preparing the SDD. PDR, Sect. 6.2, provides additional global design requirements for fluid system components, mechanical equipment and their supports, and structural systems.

4.7.9 Correlation of Safety, Quality Group, and Industry Classifications

A correlation of the reactor and nonreactor safety classifications, NRC quality groups, seismic and wind classifications, and key industry standards is given in Table 4.7-3. With regard to the pressure-retaining boundary of Safety Class SC-2, ASME Sect. III, Class MC design rules shall be applied to the metallic primary containment vessel where it is not adjacent to and connected with reinforced concrete that serves as a strength element for the containment design. Where the containment vessel is adjacent to and connected with reinforced concrete that serves as a strength element for containment loads, the design rules of ASME Sect. III, Division 2, apply. For ANS, it was necessary to expand the ANSI/ANS format to include the complete range of ANS NRS classes and to define requirements for electrical components more precisely. This expansion of requirements is presented in Table 4.7-4, which considers both pressure boundary requirements and electrical requirements of these components. The ANS QA levels that shall be applied to the safety classes of Table 4.7-3 are given in Chap. 7.

4.8 DESIGN-BASIS EVENTS AND ACCEPTANCE CRITERIA

Design-basis events (DBEs) provide a wide envelope of conditions and occurrences that are to be used in design of structures, systems, and components to ensure that the reactor can respond safely to any reasonable foreseeable circumstance. Design-basis events are chosen based on regulatory requirements (primarily 10 CFR 50, Appendix A, and Regulatory Guide 1.70) and on experience with power reactors and with high power research and test reactors. Design-basis events are posed in such a manner as to maximize demands on safety-related systems, structures, and components.

Section 4.8.1 gives the acceptance criteria by which the performance of the reactor in each design-basis event is judged, and Sect. 4.8.2 lists the design-basis events.

4.8.1 Design-Basis Events Acceptance Criteria/Limits

Acceptance criteria relating to fuel integrity, coolant-pressure boundary integrity, and radiation release from containment are necessary to address adequately the impact of the design-basis accidents. These areas represent the three major barriers of defense against the uncontrolled release of radioactivity. The integrity of the fuel itself prevents the escape of radioactivity, especially fission products, from fuel to coolant and is the primary barrier. The primary-coolant pressure boundary prevents radioactivity released to the coolant from escaping into the containment building and is the second barrier. The containment buildings and related engineered

Table 4.7-3. Correlation of reactor and nonreactor safety classifications and industry codes

Safety class	NRC quality group[a]	DOE performance category[b]	Seismic category[c]	Wind category[d]	Pressure retaining boundary[e]	Electrical equipment[f]
				Reactor safety		
SC-1	A	5	I	I	ASME III-1 or CS	Class 1E
SC-2	B	5	I	I	ASME III-2 or MC	Class 1E
SC-3	C	5	I	I	ASME III-3	Class 1E
				Nonreactor safety		
NRS-4		4	II or I-L	II or I-L	ASME III-3	Class-1E
NRS-3	D	3	II or I-L	II or I-L	ASME VIII/ANSI B31.1	Non-Class-1E
NRS-2		2	II or I-L	II or I-L	Commercial	Non-Class-1E
NRS-1		1	II or I-L	II or I-L	Commercial	Non-Class-1E
NRS-0		0	None or I-L	None or I-L	Commercial	None or I-L

[a]See NRC Regulatory Guide 1.26 for a discussion of quality groups. Correlations between safety classes and quality groups are included in this column only where there is a high degree of correspondence. The entries demonstrate that all NRC quality groups are included in the ANS classifications. However, some items in Quality Group D may be classified in NRS classes other than NRS-3.

[b]See DOE 6430.1A, UCRL-15910, and other supporting documentation for DOE 6430.1A.

[c]Definitions of the SSE and OBE are in 10 CFR, Part 100, Appendix A, and in NRC RG 1.29. See Category A reference for Sect. 6.2.3 of this PDR for a description of the SSE 2.94 m/s^2 (0.3g) and OBE 0.98 m/s^2 (0.1g) *[HOLD]*[4-3] earthquakes applied to SC-1 through SC-3. Site-specific data, to be justified in the ANS PSAR, provides the basis for selecting 0.98 m/s^2 (0.1g) for the OBE. Maximum accelerations for DBEs for the ANS NRS categories shall be taken from UCRL-15910. Values selected from review of UCRL-15910 at the time of this PDR revision were NRS-4, 2.94 m/s^2 (0.3g); NRS-3, 1.86 m/s^2 (0.19g); NRS-2, 1.86 m/s^2 (0.19g); NRS-1, 1.47 m/s^2 (0.15g); and NRS-0, none. Based upon site-specific information, lower values may be justified in the ANS PSAR. Seismic Category I-L applies to any NRS structures, systems, or components whose failure during a seismic event could damage safety class components. Specifically, in any mode of failure, NRS components that do not perform a nuclear safety function shall not cause loss of safety function of SC-1, -2, or -3 equipment. This objective shall be accomplished (1) by designing this equipment so that its probability of failure is acceptably low, (2) by designing to Seismic Category I SSE requirements as delineated in Regulatory Guide 1.29, Subsect. 2, or (3) by using the methods of *American National Standard Physical Protection for Systems and Components Important to Safety*, ANSI/ANS-58.3 1977,[34] to prevent its failure from affecting any SC-1, -2, or -3 equipment.

Table 4.7-3 (continued)

[d]See NRC Regulatory Guide 1.76[35] in connection with tornados for reactor safety classes SC-1 through SC-3. Maximum tornado and/or wind values for the ANS NRS classes shall be taken from UCRL-15910. Values selected from review of UCRL-15910 at the time of this PDR revision were NRS-4, 245 km/h (152 mph); NRS-3, 182 km/h (113 mph); NRS-2, 113 km/h (70 mph); NRS-1, 113 km/h (70 mph); and NRS-0, none. Lower values may be justified in the ANS PSAR based upon site-specific information.

Wind categories I and I-L [DBT = 579 km/h (360 mph)] *[HOLD]*[4,4] apply to any NRS structure, system, or component whose failure in the event of a DBT could cause loss of safety function. A lower, site-specific DBT of 418 km/h (260 mph) and associated missiles may be justified in the ANS PSAR.

[e]See Sect. 6.2 of this document.

[f]This column refers to the electrical portion of electrical, instrumentation, and control equipment. For details concerning design criteria for pressure-retaining boundaries and electrical portions, see Table 4.7-4.

Table 4.7-4. Electrical equipment classification

For the pressure retaining boundary component:

- If it meets criteria for SC-1, SC-2, OR SC-3, then design to applicable ASME code and qualify to applicable DBE/DBT.[a]
- If it meets criteria for NRS-3 OR NRS-4, then design to applicable ASME code and qualify to DBE/DBT used for the system in which the equipment is installed.
- If it meets criteria for NRS-1 OR NRS-2, then use commercially available components and qualify to NRS-2 DBE/DBT requirements.[a]
- If it meets criteria for NRS-0, then use commercially available pressure boundary components. No natural phenomena qualification is required.

For the electrical, instrumentation, or control component:

- If the control function meets criteria for SC-1, SC-2, SC-3, or NRS-4, then design and qualify component for Class 1E requirements.
- If the control function meets criteria for NRS-1, NRS-2, or NRS-3, then design and qualify component to NRS-3 requirements.[a]
- If the control function meets criteria for NRS-0, then design and qualify to performance requirements only. No natural phenomena qualification is required.

In all cases:

- Seismic and Wind Category I-L equipment will be designed to mitigate the threat on a case-by-case basis.
- Control functions to be considered are those that prevent undesirable control actions as well as cause protective action.

[a] These requirements are referenced in Table 4.7-3 and in Sect. 6.2.3.

safety features (e.g., isolation valves, water cells, and air-filtration units) comprise the third barrier against uncontrolled release of radioactivity into the environment.

The acceptance criteria are more stringent for the more frequent event categories, with those for normal operation being the most stringent. This graduated scale of acceptance criteria has the effect of concentrating design attention where it will yield the greatest risk reduction. Table 4.8-1 summarizes the acceptance criteria to be applied to ANS design-basis accidents, and the following subsections explain the corresponding bases.

Table 4.8-1. ANS event categories and acceptance criteria

Category	Fuel integrity[a] limits	Coolant-pressure boundary limits[b]	Radiation exposures at ORR boundary[c]
Normal Operation (>1/year)	No boiling Fuel temp. <400°C; temp. difference across Al_2O_3 < 119°C	A (Normal)	40 CFR 61, Subpart H 10 CFR 50, Appendix I (ALARA)
Anticipated (>0.01/year)	No CHF Fuel temp. <450°C *[HOLD]*[4-5]; temp. difference across Al_2O_3 < 119°C	B (Upset)	(a) Appendix I if frequency > 0.1/year (b) 10 CFR 20 if frequency < 10^{-1}/yr. Unrestricted area limit
Unlikely (10^{-4}< frequency < 10^{-2}/year)	No CHF *[HOLD]*[4-6d] Fuel temp <525°C *[HOLD]*[4-6d]	C (Emergency)	10% of 10 CFR 100 limits
Extremely Unlikely (10^{-6}< frequency < 10^{-4}/year)	Not applicable	D (Faulted)	20% of 10 CFR 100 limits[e]

[a]See Sect. 4.8.1.1 for detailed explanation of the fuel integrity limitations.
[b]In addition to not exceeding the stress limitations of the ASME Code, structures and components must be able to perform their required safety-related functions during design-basis events.
[c]The ORR boundary refers to the nearest border of the ORR.
[d]These limits will be replaced by specific transient calculations based on fuel models.
[e]See also Table 4.1-2 for exposure limitation goals and requirements referenced to locations other than the ORR boundary.

4.8.1.1 Fuel Integrity

Acceptance criteria for fuel performance are based on four determining phenomena: swelling, corrosion, structural response, and cooling. The required level of proof for calculations to determine compliance with acceptance criteria must be at least equivalent to a nonexceedance probability of 95% at a confidence level of 95% (see Sect. 4.4 of NUREG-0800).

Fission gas produced within the fuel will be retained without significant swelling if fuel temperature throughout the fuel cycle is limited to 400°C. Therefore, maximum fuel temperature

must be limited to 400°C during normal operation. Temperature excursions above 400°C are allowable if the durations and magnitudes of the excursions are limited so as to prevent unacceptable swelling. For anticipated events, a maximum temperature of 450°C is acceptable if the time above 400°C is limited to less than 8½ days (half of a complete irradiation cycle) *[HOLD]*[4-5]. Unacceptable swelling will not occur for unlikely events if the fuel centerline temperature does not exceed 525°C and the duration of the event does not exceed 1800 s *[HOLD]*[4-6]. In any case full temperature will not exceed the 582°C solidus temperature of the Al-6061 cladding.

Corrosion of the aluminum cladding by conversion to Al_2O_3 is known to occur in aluminum-fueled reactors. A primary concern for ANS is spallation of the oxide layer. The acceptance criterion that prevents oxide spallation[36] is that the computed temperature difference across the oxide layer must not exceed 119°C. Spallation is of concern because it would lead to accelerated corrosion and possible threat to cladding integrity and because uneven cooling after spallation could lead to uneven temperature profiles and possible uneven fuel plate thermal buckling. The 119°C limit is applicable to normal operation and anticipated events. Applicability to unlikely events is *[TBD]*[4-1].

Limiting the structural deflection of the fuel plates is essential because sufficient plate-to-plate separation is critical in maintaining adequate coolant flow past all parts of active fuel. Thermal expansion and change in mechanical-strength properties with temperature are primary agents of structural deflection, and the fuel plates will also respond to pressure and flow forces. Structural deflection during any design-basis event shall not be sufficient to cause violation of the other applicable acceptance criteria discussed in this section. The need for additional acceptance criteria related directly to fuel plate structural deflection is *[TBD]*[4-2].

The acceptance criteria for fuel cooling pertain to the mode of heat transfer between the surface of the fuel cladding and the coolant. For normal operations, boiling is undesirable and is prevented by limiting the heat flux to below the incipient boiling level (IBL) at the most unfavorable point in the core. While the prevention of boiling is a positive and desirable design goal, it is not critical to safety during transient events. Indeed, a small amount of void generation is possible and acceptable during anticipated events and probable during unlikely events. To ensure that this void generation does not lead to steam blanketing at any point and to subsequent fuel damage, the heat flux shall be limited to below the critical heat flux (CHF) at the most unfavorable point in the core. In accordance with the acceptance criteria in Sect. 4.4 of the *Standard Review Plan*,[5] the CHF limit for anticipated events is expressed as follows:

1. Throughout the core, heat flux at all points shall remain below that for critical heat flux (per the Gambill-Weatherhead[37] or other acceptable CHF correlation), and heat flux over all hot streaks shall remain below that for parallel channel-flow instabilities (per Costa[38] or other acceptable flow excursion correlation). For statistical evaluations of this limit during the most severe anticipated event, the more restrictive of the following statistical limits shall be met: minimum 95% nonexceedance probability at 95% confidence level or minimum 99.9% nonexceedance probability at 50% confidence level.
2. The CHF limitation is also important for unlikely events. NRC GDC 35 (see Table B.5 in Appendix B) requires that cladding metal-water reaction be limited to negligible amounts and that fuel and cladding damage that could interfere with continued effective core cooling be prevented.

4.8.1.2 Pressure Boundary and Structural Integrity

Acceptance criteria for the mechanical performance of pressure boundary and structural elements are specified by the appropriate ASME Code limitations for normal operations (Service Level A), anticipated events (Service Level B, upset condition), unlikely events (Service Level C, emergency condition), or extremely unlikely events (Service Level D, faulted condition). To ensure compliance with standard practice for ASME pressure vessel analysis, consideration shall be given to inclusion into the next highest category some events at the top of any frequency category. A very important overall criterion related to structural integrity is that safety-related components be capable of accomplishing their required safety-related functions in any design-basis event. Analyses of the mechanical response of safety-related components and structures shall always apply this very important acceptance criterion.

Combination of loads from certain design-basis events shall be considered in order to comply with standard ASME practice. For example, the SSE must be considered in conjunction with certain events. For a detailed discussion of the prescribed method of combining loads to compare with stress allowables, see the Category A reference to Sect. 6.2.3.

The thermal response during DBEs is a very significant consideration for components and structures that are within the intense radiation fields emanating from the reactor core. For example, cooling is of especial importance for components such as the CPBT and the control rods. It is essential that these components retain their structural integrity so that the core, control elements, and pressure boundary maintain their geometry and integrity. The acceptance criteria for the cooling of structures and components parallel the cooling criteria for the fuel: no boiling under normal conditions, no CHF for anticipated events, and very limited CHF for unlikely events. The stress evaluations for ASME-Code-based design of components shall carefully consider temperature transients that would be experienced during DBEs.

4.8.1.3 Radiation Dose Limits

Radioactivity releases under normal conditions and under frequently occurring off-normal events shall be maintained ALARA, per 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." Under the ALARA principle, release limits are below the 0.25 mSv (25 mrem) per year limit imposed by 40 CFR 61, Subpart H. ANSI/ANS 51.1 subdivides the anticipated event category in order to distinguish between those events that occur frequently enough to be considered under the ALARA guidelines and those that occur rarely enough to be considered under other limits. As provided by ANSI/ANS 51.1, two subcategories are used for consequences of potential radioactivity releases associated with anticipated events. Events expected to occur at a frequency greater than 0.1/year are analyzed in accordance with the ALARA principle. The analysis of events expected to occur less frequently than 0.1/year but more than 0.01/year is acceptable if the ORR site boundary exposure is below the limit specified in 10 CFR 20 for unrestricted areas [currently 5 mSv (0.5 rem) dose equivalent].

The site boundary radiation dose limits for accidents estimated to occur less frequently than 0.01/year are based on 10 CFR 100, "Reactor Site Criteria."

Also see Sect. 6.2.5, "Radiation Protection Criteria," which includes requirements for operating personnel exposures as a result of design-basis events.

4.8.2 Design-Basis Events

Table 4.8-2 lists the design-basis events that are applicable to ANS. The events are grouped by initiating-event cause or consequence and are classified as normal, anticipated, unlikely, or extremely unlikely events in accordance with the ANS 51.1-1983 classification scheme as outlined and discussed in Sect. 4.7.3 of this PDR.

The grouping of events into frequency categories is based on regulatory requirements as well as on available data from research and tests concerning power reactors. Data sources expressed in the ANS event category grouping include PRA studies conducted for ANS, HFIR Level I PRA, and applicable power reactor experience. The NRC GDC 10 CFR 50, Appendix A), provides a regulatory basis for classifying as anticipated any event that is initiated by a single failure of equipment or single operator error. Specifically, the GDC definition of anticipated operational occurrences mandates that loss of power to all main primary-coolant pump motors and the loss of all off-site power be classed as anticipated. Also GDC 25 requires that any event involving a single malfunction of the reactivity control systems must meet the acceptance criteria for anticipated events. The purpose of these dictates is that the reactor shall suffer no ill effect from any single failure. In Table 4.8-2, this philosophy has been applied to any event involving a well-defined single failure, such as the unintended closure of any one valve or the stoppage or failure of any one pump.

Table 4.8-2 is intended to be a compact, comprehensive listing of all the design-basis events that affect more than one plant system. The technical definition of each event is provided in Appendix C. Single failures that have to be considered with each event are also addressed in Appendix C.

A fifth event category, test conditions, is included to document the special plant-level test conditions specified for the reactor at predetermined but infrequent intervals in accordance with ASME Code rules or other requirements. The plant shall be designed to accommodate these test conditions.

Table 4.8-2. ANS design-basis conditions and events

Event	Frequency category*	Identification number
Normal reactor operations	Normal	
Fuel loading		N-1
Approach to criticality		N-2
Start-up to low power		N-3
Start-up to full power		N-4
Controlled shutdown to low power		N-5
Fast runback		N-6
Scram		N-7
Fuel unloading		N-8
Test conditions	Test	
Primary-coolant system hydrostatic pressure test		T-1
Secondary-coolant system hydrostatic pressure test		T-2
Reflector-coolant system hydrostatic pressure test		T-3
Containment building pressure-leak tests		
Integrated		T-4I
Type A		T-4A
Type B		T-4B
Type C		T-4C
Reactor natural circulation cooling test (primary- and secondary-cooling systems)		T-5
Reactivity events (RE)		
Negative reactivity (REN)		
Single control-element insertion or drop, partial or full	Anticipated	REN-1
Spurious actuation of one shutdown system	Anticipated	REN-2
Liquid poison injection *[HOLD]*[4-7]	*[TBD]*[4-3]	REN-3
Light-water injection into reflector vessel	Unlikely	REN-4
Positive reactivity (REP)		
Shim/safety withdrawal at normal speed from start-up or low-power conditions	Anticipated	REP-1
Shim/safety withdrawal at normal speed from full power	Anticipated	REP-2
All-rod withdrawal	Unlikely	REP-3

Table 4.8-2 (continued)

Event	Frequency category*	Identification number
Rapid expulsion of a single rod (may be precluded by design)	*TBD*[4-4]	REP-4
Single beam-tube flooding	Anticipated	REP-5
Maximum credible cold-source failure	Unlikely	REP-6
Multiple beam-tube flooding	Extremely unlikely	REP-7
Light-water injection via pressurizer pumps	Unlikely	REP-8
Light-water slug enters core following start of H_2O-contaminated spare loop (may be prevented by design)	Extremely unlikely	REP-9
Loss of coolant pressure control (LOPC)		
Pressure decrease		
One letdown valve goes fully open†	Anticipated	LOPC-1
All letdown valves go fully open	Anticipated	LOPC-2
Pressurizer pump shutdown	Anticipated	LOPC-3
Overpressure relief valve fails to open	Unlikely	LOPC-4
Pressure increase		
One letdown valve goes closed†	Anticipated	LOPC-5
All letdown valves go closed	Anticipated	LOPC-6
Inadvertent start of one or more pressurizer (charging) pumps	Anticipated	LOPC-7
Pressurizer (charging) pump overspeed (only possible if variable speed pump or speed-reduction coupling used)	Anticipated	LOPC-8
Primary coolant flow increase (FI)		
Inadvertent start of one or more primary-coolant pumps	Anticipated	FI-1
Failure of core bypass flow restrictor	Unlikely	FI-2
Inadvertent start of one primary coolant pump during refueling	Unlikely	FI-3
Loss or reduction of primary coolant flow (LOF)		
Loss of forced flow		
Single pump shutdown followed by loop transfer	Anticipated	LOF-1
All pumps coastdown to pony motor flow	Anticipated	LOF-2
Single pump shaft break or seizure	Unlikely	LOF-3
All pumps coastdown to natural circulation flow (all pony motors fail)	Extremely unlikely	LOF-4

Table 4.8-2 (continued)

Event	Frequency category*	Identification number
Loss of flow path		
Single isolation valve closed	Anticipated	LOF-5
Flow strainer in one loop blocked	Unlikely	LOF-6
Two isolation valves closed (Multiple isolation valve closure prevented by interlock)	Extremely unlikely	LOF-7
Loss-of-reflector coolant flow (LORF)		
Main pumps shutdown	Anticipated	LORF-1
One reflector coolant loop isolated	Anticipated	LORF-2
Core flow blockage (CB)		
Foreign object in coolant	Anticipated	CB-1
Experiment or transuranic target structural failure	Unlikely	CB-2
Core inlet strainer structural failure	Unlikely	CB-3
Major core inlet flow blockage	Extremely unlikely	CB-4
Blockage of CPBT cooling passage	Unlikely	CB-5
Blockage of inner control rod cooling passage	Unlikely	CB-6
Loss-of-coolant inventory		
See loss-of-coolant accidents, below		
Loss of heat sink (LOHS)		
Loss of one normal heat sink†	Anticipated	LOHS-1
Loss of all normal heat sinks outside containment	Unlikely	LOHS-2
Loss-of-coolant pD control (ACID)		
High pD (loss of DNO_3 addition)	Anticipated	ACID-1
Low pD (excessive DNO_3 addition)	Anticipated	ACID-2
Loss of primary coolant (LOC)		
Sizes:		
Small (Depressurization not sufficient to cause immediate scram)	Anticipated	SBLOC-i

Table 4.8-2 (continued)

Event		Frequency category*	Identification number
Medium	(Rapid depressurization to below scram set point, but pressure adequate for ac motor operation of one primary-coolant pump)	Unlikely	MBLOC-i
Large	(Immediate depressurization to ambient pressure)	Extremely unlikely	
Locations:			
Small, medium, and large leaks and breaks to be examined in variety of possible locations, including:			
Reactor to reflector coolant (CPBT)			LOC-1
Reactor to reactor pool			LOC-2
Reactor to water cell			LOC-3
Reactor to limited-volume air cell			LOC-4
Reactor to elevated air cell			LOC-5
Reactor main heat exchanger tube break			LOC-6
Reactor emergency heat exchanger tube break			LOC-7
Reactor to subpile room			LOC-8
Leak from accumulator gas space			LOC-9
Loss of reflector coolant (LORC)			
Sizes:			
Small	(Size insufficient to cause immediate degradation of safety-related reflector cooling or moderator functions)	Anticipated	SBLORC-i
Large	(immediate degradation of reflector cooling or moderator functions)	Unlikely	LBLORC-i
Locations:			
Small and large leaks and breaks to be examined in variety of possible locations, including:			
Reflector beam-tube break			LORC-1
Reflector to reactor pool			LORC-2
Reflector to water/air cell *[HOLD]*[4-8]			LORC-3
Reflector auxiliary heat exchanger tube break			LORC-4

Table 4.8-2 (continued)

Event		Frequency category*	Identification number
Loss of secondary coolant (LOSC)			
Sizes:			
Small	(insufficient to cause immediate degradation of secondary cooling)	Anticipated	SBLOSC-i
Large	(immediate degradation of secondary cooling)	Unlikely	LBLOSC-i
Locations:			
Small and large breaks to be examined in a variety of possible locations, including:			
Reactor support building			LOSC-1
Pipe chase			LOSC-2
Basin, pump section			LOSC-3
Basin, discharge			LOSC-4
Reactor building			LOSC-5
Noncondensible gas events (NCG)			
Coolant off-gas as a result of primary-coolant depressurization		Anticipated	NCG-1
Failure of gas-cooled irradiation experiment		Anticipated	NCG-2
Accumulator excess gas supply		Unlikely	NCG-3
Events with failure of scram system, anticipated transient without scram (ATWS)			
Anticipated event with failure of primary scram system		Unlikely	ATWS-1
REP-5 reactivity insertion with failure of primary scram system		Unlikely	ATWS-2
Fuel handling accidents (FHA)			
Spent fuel element dropped in D_2O		Extremely unlikely	FHA-1
Fresh fuel element criticality in D_2O		Extremely unlikely	FHA-2
Spent fuel element dropped inside containment		*[TBD]*[4-5]	FHA-3
Spent fuel element dropped outside containment		*[TBD]*[4-6]	FHA-4
Loss of spent-fuel cooling		*[TBD]*[4-7]	FHA-5
Fuel element stuck		*[TBD]*[4-8]	FHA-6

Table 4.8-2 (continued)

Event	Frequency category*	Identification number
Loss of electrical power (LOEP)		
Loss of all off-site power	Anticipated	LOEP-1
Station blackout	Unlikely	LOEP-2
Loss of all non-1E power	Unlikely	LOEP-3
Experiment accidents		
Cold source (CS)		
Loss of cooling	Anticipated	CS-1
Pressure boundary fracture	Unlikely	CS-2
Internal D_2-air oxidation	Extremely unlikely	CS-3
Leakage of D_2 to containment atmosphere (possible combustion)	Unlikely	CS-4
Hot source (HS)		
Loss-of-temperature control	Anticipated	HS-1
Pressure-boundary fracture	Unlikely	HS-2
Transuranic targets (TRU)		
Pinhole leak	Anticipated	TRU-1
Pinhole leak with waterlogging	Unlikely	TRU-2
Major perforation	Unlikely	TRU-3
Structural failure, target, or mounting hardware	Unlikely	TRU-4
Loading error (manufacturing, not detected before operation)	Extremely unlikely	TRU-5
Material irradiation (IRR)	Anticipated	IRR-1
Inadequate cooling		
Loss of primary-experiment containment boundary integrity	Anticipated	IRR-2
Loss of experiment containment primary- and secondary-boundary integrity	Unlikely	IRR-3
Major structural failure	Unlikely	IRR-4

Radiation release from components (RR)

Radioactivity contained in normal liquid or gaseous-process waste streams, not associated with severe fuel-damage accidents, shall be assumed to be released as a result of subsystem or component failure. Component and subsystem failures considered shall include but not be limited to the following:

Table 4.8-2 (continued)

Event	Frequency category*	Identification number
Radioactive waste system component failure, liquid release	Unlikely	RR-1
Radioactive waste system component failure, gaseous release	Unlikely	RR-2
Beam- or guide-tube rupture, resulting in tritium and D_2O release	Extremely unlikely	RR-3
Fuel damage events (FD)		
1% localized fuel damage	Unlikely	FD-1
100% fuel damage	Extremely unlikely	FD-2
Other internal events	*[TBD]*[4-9]	
Fires		
Equipment generated missiles		
Flooding		
Pools		
Water cells		
Secondary coolant		
Heavy object drop		
External events		
Tornado	*[TBD]*[4-10]	
Seismic		
OBE	Anticipated	
SSE	Unlikely/ Extremely unlikely	
Floods	Unlikely	
HFIR or transuranic facility accidents	Unlikely	

*The anticipated category includes events or mishaps at frequency greater than 10^{-2}/year. Unlikely includes accidents of frequency between 10^{-4}/year and 10^{-2}/year, and extremely unlikely includes accidents of frequency between 10^{-6}/year and 10^{-4}/year.

† These nonlimiting events are included for analysis to show that the plant control system is capable of controlling plant parameters in such a manner that the reactor does not scram as a result of the event and continues to operate at full power or some reduced power after the event.

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5. OPERATION AND MAINTENANCE GOALS AND REQUIREMENTS

5. OPERATION AND MAINTENANCE GOALS AND REQUIREMENTS

5.1 PLANT AVAILABILITY GOAL

Other than during scheduled major maintenance periods, the plant's availability should be at least 80%. Global availability (including major maintenance) shall be at least *[TBD]*[5-1] %.

5.2 PLANT PREDICTABILITY GOAL

Reactor cycles and maintenance schedules should be predictable and scheduled with a lead time of at least nine months once normal reactor operation is achieved. The reactor shall be in operation, at full power, at least *[TBD]*[5-2] % of the scheduled operating time.

5.3 PLANT LIFETIME REQUIREMENTS

The design lifetime of ANS shall be 40 years. This lifetime is consistent with the maximum NRC-license period allowed in 10 CFR 50. The design of ANS shall not knowingly preclude lifetime extensions beyond 40 years.

5.4 PLANT OPERABILITY

The ANS design shall provide features that ensure the ability to monitor and control the operation of the plant in a safe and effective manner and environment. The design shall:

1. Incorporate controls, instruments, displays, and alarms necessary (1) to allow prompt action to correct accident conditions, (2) to monitor plant conditions during normal and post-accident conditions, and (3) to carry on normal surveillance and control of plant systems during normal start-up, operation at power, shutdown, and refueling.
2. Meet applicable requirements for engineering safety features, per Regulatory Guide 1.62;[1] system safety status determination, per Regulatory Guide 1.47;[2] and post accident monitoring, per Regulatory Guide 1.97.[3]
3. Where possible, use commercially proven hardware and software.
4. Incorporate highly integrated man-machine interface equipment and highly automated process controls and business system functions that use human engineering principles to enhance operation performance capabilities.
5. Include a main control room from which actions can be taken to operate the plant under normal conditions and to maintain it in a safe condition under accident conditions. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 50 mSv (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident (10 CFR 50, Appendix A, Criterion 19). In addition, adequate protection shall be

provided to protect control-room occupants from the release of hazardous chemicals (i.e., chlorine and tritium) that may be discharged as a result of equipment failures, operator errors, or events and conditions outside the control of ANS (Regulatory Guide 1.78).[4]

6. Include a remote shutdown control room providing, in the event of a forced evacuation of the control room, alternate controls and instrumentation to bring the plant to, and maintain it in, a safe cold shutdown condition (10 CFR 50, Appendix A, Criterion 19).
7. Use local control areas and panels located in areas that limit the radiation exposure of operators during normal operations in accordance with Sect. 4.8.1.3. Local, accessible control panels used for mitigating an accident shall be qualified for that accident.
8. Use control functions designed so that it shall not be necessary to require operator actions within the first hour after a design-basis event. Control actions necessary to provide reactor shutdown, to actuate engineered safety features, to maintain the reactor in a safe shutdown condition, and to complete other safety-related functions within one hour of a single design-basis event shall be automatic.
9. Incorporate and control data acquisition and equipment arrangements that allow the obtaining of data needed for control room and archiving operations without using roving operators.
10. Use the number of plant operators and support personnel performing their duties as described in Sect. 5.4.1.

5.4.1 Operator, Supervisory, and Technical Support Personnel

ANS designs shall be based on an operating philosophy that incorporates the intent of 10 CFR 55, DOE Order 5480.20,[5] ANS Std. 3.1-1987[6] (required by DOE Order 5480.6[7]); associated NRC, DOE, and industry regulatory guides, codes, and standards; and other DOE Orders. DOE Order 5480.20 implements all of the above. Training, qualification, and certification of personnel shall be developed per ANS procedures; shall implement ANS requirements; and shall be documented in an easily auditable format.

A reactor operator, a senior reactor operator, a shift supervisor, a shift technical advisor (STA), and auxiliary operators designated to perform key system functions are the required personnel for efficiently and effectively operating and maintaining reactor and major plant systems. A brief description of the qualifications and functions associated with each position, along with that of operating engineers, is included below.

At least two of the following three personnel will be in the main control room (referred to as the control room) at all times: the reactor operator, the senior reactor operator, and the shift supervisor. During normal plant operations and during plant emergencies, either the senior reactor operator or the shift supervisor shall be in the control room at all times for oversight and management of reactor and plant operations and for various coordination activities being conducted on-site and off-site. Additionally, a shift technical advisor shall be in or near the control room on each shift at all times. Operating engineers shall be available on each shift as required. Each of the identified positions, along with offices, conference room(s), files, and other support space shall be provided within the control-room security envelope.

The following requirements are based on the documents referenced above:

1. **Reactor operator:** A reactor operator is a person trained, qualified, and certified via the ANS training procedures and plant simulator to operate (manipulate the controls of) the ANS reactor and associated and designated systems. One reactor operator shall be available

on site for assigned reactor and balance of plant-systems operations during normal and emergency operations. This operator shall be a fully trained, qualified, and certified reactor operator capable of operating reactor and associated systems and other key designated systems. These systems include both safety-related and nonsafety-related plant systems. The reactor operator shall have education and experience consistent with DOE Order 5480.20, as may be supplemented by ANS requirements and procedures. This operator shall remain in the control room for plant control and monitoring functions or may be dispatched to designated plant locations to assist in the control room operation. In the event this operator is dispatched from the main control room to another plant location, his/her assigned duties shall be performed by another reactor operator, the senior reactor operator, or the shift supervisor.

2. **Senior reactor operator:** A senior reactor operator is a person trained, qualified, and certified via the ANS training procedures and plant simulator to operate (manipulate the controls of) and to direct the operation of the ANS reactor. The senior reactor operator shall be responsible primarily for day-to-day reactor and major plant systems operations and for emergency operations. This operator is a fully trained, qualified, and certified senior reactor operator capable of operating or directing the operation of reactor and major plant systems. The senior reactor operator shall have education and experience consistent with DOE Order 5480.20, as may be supplemented by ANS requirements and procedures. One senior reactor operator or shift supervisor shall be available on-site and in the main control room at all times for oversight and management of reactor and plant operations and for various coordination activities being conducted on-site and off-site.
3. **Shift supervisor:** The shift supervisor is a person trained, qualified, and certified in the operating organization and designated by the facilities manager as the person to direct operations-related activities of personnel at ANS. The shift supervisor shall be a fully trained, qualified, and certified reactor and major-plant-systems operator. The shift supervisor shall hold and maintain senior-reactor-operator certification. At least one shift supervisor or senior reactor operator shall be available on-site and one shift supervisor or senior reactor operator shall be in the main control room at all times for oversight and management of reactor and plant operations and for various coordination activities being conducted on-site and off-site.
4. **Auxiliary operator:** Plant equipment and system operation and monitoring throughout the plant from the main control room or from local area panels or control stations shall be performed by auxiliary operator(s) fully trained, qualified, and certified on the equipment or systems to be operated or monitored. Auxiliary operator(s) shall be available in the control room when required to conduct designated plant systems/operations such as the following:

   A. Operations involving the transfer of heavy water from the reactor support building to the detritiation building. An auxiliary operator may also monitor operations and/or conditions of the detritiation (process for removing tritium) facility from the control room during detritiation facility operations and when there is no operator in the detritiation facility control room.
   B. Operations of power switchyard breakers. An ANS auxiliary operator shall coordinate switchyard breaker operations with an auxiliary operator of the K-25 Site Central Control Facility (CCF).
   C. Cold source and hot source monitoring and control.
   D. Experimental systems coordination between experimental areas (including the materials-irradiation control room) and the main control room. The auxiliary operator shall be

trained, qualified, and certified to perform designated control and monitoring functions. This operator does not need to be certified in reactor systems operations but will be required to work closely with the reactor operator.

E. Refueling operations. Refueling operations shall be performed by specially trained, qualified, and certified operators uniquely qualified to perform this specialty operation. This operation shall be performed from the control room.

F. Designated plant systems (as specified in each auxiliary operator's job description and certification).

Each auxiliary operator shall be trained, qualified, and certified in the operations and monitoring of the systems that he/she will be required to perform. The number of auxiliary operators required to be stationed concurrently in the main control room shall be determined by the number of concurrent operations to be performed from the main control room. Under normal conditions, a minimum of one auxiliary operator will be in the control room during each shift.

5. **Shift technical advisor:** A shift technical advisor is a person who has been assigned to provide on-shift advice and counsel to shift operating personnel to help determine cause and mitigation of facility accidents. One STA shall be in the control room area on each shift. The qualifications for this position shall be consistent with DOE Order 5480.20, as may be supplemented by ANS requirements and procedures.
6. **Crane operator:** Crane operators shall be specially trained, qualified, and certified for the crane configurations to be operated within ANS facilities.
7. **Manipulator operator:** Each manipulator operator for ANS shall be trained and certified for the specific mechanical or electromechanical (including servomechanical) manipulator to be operated.
8. **Detritiation facility operator:** Heavy-water upgrade and detritiation facility operations shall be done by a detritiation facility operator from the detritiation facility control room. Additionally, a facility shift supervisor shall also be present in or near the detritiation facility control room. Both persons shall be on shift at all times when the operation and monitoring of the facility require operator attention. Each shall be fully trained, qualified, and certified for the operation of this facility and shall have defined minimum operator performance experience. The shift supervisor, however, must have been a senior operator with more substantial operations experience. The qualifications of each will be determined later in an ANS operator position description. When the detritiation operations are possible without an operator in the detritiation control room, control and monitoring functions shall be transferred to the main control room in the reactor support building; the operation and monitoring functions shall be maintained by an auxiliary operator trained, qualified, and certified in the operation of the detritiation facility. A detritiation facility operator could fulfill the position of this auxiliary operator.
9. **Operating engineers:** There shall be several operating engineers available to provide technical support for control-room operations. These technical support personnel are typically degreed engineers with backgrounds of training and experience necessary to support plant operations. The detailed requirements for each position will be developed later. The number required to support each shift will be determined later, but at least four shall be required on the first shift and shall be located in offices in the immediate vicinity of the control room.

Additional and detailed training, qualification, and certification requirements will be defined later in the design phase of ANS.

5.4.2 Inspection Level Earthquake

The reactor and nonessential portions of the plant shall be shut down in the event of any earthquake that exceeds the inspection level earthquake (ILE).[8] The ground-level acceleration of the ILE shall be 0.98 m/s^2 (0.1g) *[HOLD]*[5-1]. The plant may be restarted and normal operation resumed for an earthquake greater than ILE only after assurance is provided by a thorough inspection that no functional damage has occurred to the features of the plant required for operation. Design and operation of the plant with regard to ILE shall meet the requirements of UCRL-15910[9] (required by DOE Order 6430.1A[10]) and 10 CFR 100, Appendix A (required by DOE Order 5480.4[11] and Order 5480.6).

5.5 PLANT INSERVICE INSPECTION

The plant inservice inspection (ISI) plan shall meet the requirements of 10 CFR 50.55a and ASME Code (Sect. III and XI,[12] as applicable to a low-pressure and low-temperature heavy-water reactor) and shall address the needs to assess possible damage to components because of severe internal accidents and/or severe external events. This plan shall also address the functional testing of pumps and valves as required by the ASME Code. In addition to the ISI program plan, the ISI requirements for each system shall be defined as applicable for the system's equipment and components. This information shall be included in the ISI section and appendix of the SDD for the system.

The design of systems and components for ANS shall incorporate features to implement required ISI with the plant on-line to the greatest degree practical. For those ISI activities that require the plant to be shut down, design features shall be provided to minimize the shutdown time required for these activities.

ANS shall be designed to provide adequate accessibility to components that require ISI, including functional testing of pumps and valves, whether required by code or employed as a prudent measure to ensure continuity of operations and plant availability and to achieve the 40-year design life of the plant.

The system boundaries subject to ISI include pressure vessels, tanks, system piping, pumps, valves, and components. ISI includes those tests, activities, and inspections required to ensure structure and component integrity.

Since remote inspection of some critical reactor pressure boundaries may be required, ANS shall be designed, fabricated, tested, and qualified for such inspection and testing. This design shall include appropriate nondestructive methods, special tools, sensors, and associated equipment necessary for ISI.

5.5.1 ISI Program Plan Elements

Areas to be addressed in the ISI program plan or in an SDD as appropriate for items within the scope of the SDD include but are not limited to the following:

1. Design and arrangement of Class 1 system components and clearances required for conducting examinations at the code-required inspection interval. Include listings of systems, components, equipment, and materials requiring inspections.
2. Scope of the examinations and inclusion of isometric drawings and component sketches showing weld locations in various piping systems and on components, diagrams to delineate system boundaries, and description(s) of examinations to be performed and extent of examination coverage for each system and component.
3. Schedule of examinations.
4. Examination techniques and procedures to be used, including detailed procedures for volumetric, surface penetrant, and visual examinations. Definition of tests to be performed and at what frequencies.
5. Equipment and components necessary for inspections, including any special test equipment.
6. Definition of data to be recorded for inspections.
7. Source of acceptance criteria or acceptable examination results (e.g., ASME Code, Sect. XI; Technical Specification; or ANS specifications).
8. Procedures for handling examination results.

5.5.2 ASME Code Class 1 Items Requiring ISI

The ISI for the reactor coolant pressure boundary's ASME Code Class I items in accordance with ASME Code, Sect. XI, includes the vessel, piping, pumps, valves, and components for the following typical systems and/or components:

1. reflector vessel,
2. reactor primary cooling system (through second isolation valve),
3. core pressure boundary tube,
4. primary rod drive mechanism and secondary rod drive mechanism,
5. primary and emergency heat exchangers, and
6. reactor pressure and inventory control and cleanup system.

The design and arrangements of Class 1 system components shall provide adequate clearances to conduct the required examinations at the code-required inspection interval.

The physical arrangement of ASME Code piping (including high-energy piping systems), pumps, valves, and components shall provide personnel access to each weld location. Working platforms shall be provided at appropriate locations to facilitate servicing of pumps and valves. Welds shall be located to permit examinations from at least one side and preferably from both sides. Weld joint configurations shall permit thorough approved-type(s) examination. Provisions shall be made for hydrostatic pressure tests to be conducted.

Provisions for space and access shall be made for both manual in-service examinations and for remote examinations where necessary. Requirements for remotely operated examination equipment and associated space and access requirements shall be identified in each SDD where remote ISI examinations are anticipated. Additionally, space and access shall be provided for performing repair procedures.

The ISI Program shall include provisions for examination of Class 2 and Class 3 systems as defined in each appropriate SDD.

For areas not covered by the ASME Code, the designs for ANS shall include inspection provisions consistent with the objective of maintaining safe operable conditions throughout the plant.

5.6 PLANT MAINTAINABILITY

ANS shall be designed to meet the maintainability criteria specified in this section. The plant designs shall minimize required maintenance and facilitate maintenance when needed. The designs shall minimize the manpower required for maintenance and minimize the skills required to keep the plant maintained. Design of the plant for maintainability shall not compromise or degrade plant performance, safety, or reliability. Maintenance devices and equipment to permit removal of components shall be engineered and provided as part of the plant design.

5.6.1 Maintenance

The ANS design shall incorporate the following maintenance requirements.

1. Maintenance shall be simplified by including features in the design and maintenance capabilities as included herein. Where possible, like components, materials, and parts within systems/equipment shall be interchangeable. Standardized "off-the-shelf" components shall be used, if possible; the number and variety of tools, accessories, and support equipment shall be limited through plant/equipment design. The ANS Project may use commercial grade equipment for some safety applications. This equipment requires qualification by the project for its intended use. The methods for qualifying dedicated commercial grade equipment are detailed in a procurement plan to be developed later. Where maintenance design results in undue complexity, the complexity shall be incorporated into the maintenance equipment rather than into the maintainable item.
2. Each SDD shall specify spare parts inventory required for supporting maintenance and shall include, but not be limited to, the following: manufacturer's recommendations, experience with similar items, delivery time, consequence of maintenance delay resulting from lack of parts, expected usage rate of parts, shelf life of parts, and degree of installation difficulty. The design shall, to the extent practical, use common parts that can be used interchangeably for common functions. However, the use of common parts shall not take precedence over reliability, operability, maintainability, or safety.
3. Design-dictated maintenance shall be reduced through application of fail-safe features, by designating components that require little or no preventative maintenance and by assigning tolerances that allow for use and wear throughout the equipment's useful life. The effect of redundancy requirements on maintenance of essential equipment shall be considered in the design selection process.
4. Progressive maintenance, where possible, is preferred for routine maintenance activities where maintenance action can be accomplished safely and efficiently without shutting down the plant. Other routine maintenance activities will be scheduled for performance during plant shutdown for refueling. ANS shall be designed to accommodate shutdown for inspection and maintenance of systems and components. The shutdown for planned maintenance shall occur during the refueling period.

5. ANS shall be designed to reduce maintenance outages to the level necessary to support the plant availability goal. Planning of component maintenance shall consider the following steps in a manner that minimizes plant unavailability:

   A. adjust or repair the component in place;
   B. replace component with a spare; and
   C. remove, repair, and reinstall the component.

6. Plant systems, equipment, and components shall be arranged and located in the plant so as to facilitate on-line maintenance. A labeling and color-coding system shall be used to assist in locating and tracing plant systems. The accessibility of individual components and equipment during normal plant operations and under accident conditions shall be considered when locating components and equipment. The operating range of inaccessible components shall be designed to minimize the probability of their failure.
7. Remote maintenance shall be minimized. Where remote maintenance/inspection is required, robotics will be considered. Viewing access shall be provided for optical or video inspection in high-radiation areas.
8. The plant systems, where possible, shall be designed to provide capability of hands-on inspection and maintenance without undue hazard to maintenance personnel.
9. Features shall be provided to facilitate the replacement of required major plant components other than the basic plant structures, consistent with the 40-year design life of the plant. The plant shall be designed so that equipment disassembly, cleaning, repair, reassembly, and return to service may be conducted in an expeditious manner.
10. All components shall be made readily accessible and maintainable with a logical removal path defined and documented.
11. Adequate space and work clearances shall be provided to access equipment for inspection and maintenance, and adequate equipment layout shall be provided for the easy replacement of components. Clearance shall be provided between adjacent components and structures for personnel access, installation and operation of tooling, and installation of temporary shielding. The designs shall provide the following:

    A. overhead room for equipment removal and replacement,
    B. space for pipe cutting and welding,
    C. access to specific pipes that are to be removed,
    D. lighting for all maintenance and inspection operations, and
    E. space for moving equipment and components from their permanent locations to shop facilities.

    Radioactive cells shall be designed to facilitate manned access and to minimize entry time. The design shall provide adequate clearance [610 mm (24 in.) or more] between components and structures where human access is required. This access includes provisions for personnel wearing protective garments.
12. Suitable ports or hatches shall be provided for personnel and maintenance equipment access as well as for component removal and replacement.
13. Component configurations and piping slopes shall be established to facilitate complete drainage. Liquid-containing systems and/or components shall be designed to facilitate complete drainage. For components that cannot be completely drained by normal means,

provisions shall be included in the design of the component to permit use of other liquid removal methods that utilize maintenance equipment.

14. Direct access shall be provided to each valve; valves shall not be located behind pipes, valves, cable trays, or equipment where they are not easily accessible.
15. Lay-down and equipment-handling space shall be provided for in-place repairs and/or dismantling of equipment for service.
16. Each crane of the appropriate type shall be provided and sized to lift the heaviest piece of equipment or component part that will have to be handled by the crane. The lifting capacities and safety margins for such cranes, as well as their deratings for critical lifts, shall be determined later.
17. Facilities shall be provided for performing maintenance on plant equipment, including facilities dedicated to contaminated equipment.
18. Shops shall be provided in appropriate areas of the facilities for the purpose of making repairs. Access to the shops shall also be considered.
19. The design of equipment and/or components containing heavy water shall minimize crevices and pockets that would make complete heavy-water removal difficult.
20. Equipment and facilities shall be provided for reactor refueling, maintenance, and repair.
21. The plant shall be designed to perform maintenance activities with minimal exposure to radiation.
22. Electrical, instrumentation, and gas or fluid connectors for radioactive components shall be designed to minimize personnel exposure to radiation. Design of instrumentation, controls, and associated circuits shall be for modular replacement.
23. Plant controls and instrument sensors for the reactor protection system shall be replaceable without disassembly of major components. Alternatively, adequate redundancy shall be provided.

Equipment designs and layouts shall incorporate capabilities to permit hands-on maintenance wherever possible. During Title I and Title II design, a maintenance plan shall be developed to address the maintenance requirements, tasks, methods, personnel skills, radiation dose, and anticipated man-hour requirements (on a system basis) for mechanical, electrical, and instrumentation and control maintenance.

5.6.2 Maintenance Personnel Safety

The following requirements shall be satisfied for all maintenance activities in order to provide for personnel safety:

1. Routine maintenance and operations shall not rely on life support systems.
2. The atmosphere in an area where maintenance is being performed shall consist of breathable air at a temperature no greater than 38°C (100°F).
3. The temperature of all surfaces with which personnel may come in contact shall be no greater than 52°C (125°F) for metallic surfaces and 60°C (140°F) for insulated surfaces.
4. Radiation from sources within an area to be habitable for maintenance shall either be removed or be limited by permanent or temporary local shielding as required prior to maintenance being performed.
5. Each component shall be isolated from its parent system to the extent necessary to allow safe performance of desired maintenance functions.

6. The system design and maintenance procedures shall provide assured isolation of electrical power to components in the maintenance area as necessary for personnel safety. An out-of-service tagging procedure shall be implemented to identify equipment that is to remain inoperable when tagged.
7. Provisions shall be made prior to a maintenance activity to minimize fire and explosion potential.
8. Design of the system and of the maintenance procedures shall provide for control and disposition of radioactive wastes generated or released during the maintenance activities.

5.7 EMERGENCY PLANNING AND PREPAREDNESS AND EMERGENCY RESPONSE FACILITIES

ANS shall include provisions for emergency planning and preparedness and shall prepare an emergency plan in accordance with 10 CFR 50, [specifically 50.34(b)(6)(v) and 50.54(q), and its 10 CFR 50, Appendix E, Subpart IV.E] and applicable DOE orders.

To cope with emergencies and to allow training, emergency planning, and preparedness exercises, emergency response facilities shall be provided for ANS. These facilities shall include a technical support center (TSC), operational support center (OSC), and emergency operations facility (EOF). These facilities shall have capabilities consistent with requirements in 10 CFR 50 and its Appendix E; DOE Orders 5500.1A,[13] 5500.2,[14] 5500.3,[15] and 5500.4;[16] and NUREG 0696.[17] SDD-69 (plant instrumentation, control, and data systems) shall define systems required and space requirements for these facilities. SDD-62 (electrical power and communications systems) shall provide cabling, communications, and electrical power capabilities as defined in SDD-69. SSDs for the appropriate buildings will provide space to house these facilities, with the possible exception of the EOF, which may be located in a facility away from ANS but on the ORNL reservation.

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8. H. A. Glovier, *High Flux Isotope Reactor Technical Specifications*, ORNL/TM-5711/R2, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., September 1990.

9. R. P. Kennedy et al., *Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards*, Report UCRL-15910, June 1990.

10. U.S. Department of Energy, Order 6430.1A, *General Design Criteria*, April 1989.

11. U.S. Department of Energy, Order 5480.4, *Environmental Protection, Safety, and Health Protection Standards*, May 1984.

12. American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code*, Sections I-XI, 1989.

13. U.S. Department of Energy, Order 5500.1A, *Emergency Management System*, February 1987, (Revised June 29, 1990).

14. U.S. Department of Energy, Order 5500.2, *Emergency Planning, Preparedness, and Response for Operations*.

15. U.S. Department of Energy, Order 5500.3, *Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for DOE Operations*, Aug. 13, 1981, (Revised July 2, 1990).

16. U.S. Department of Energy, Order 5500.4, *Public Affairs Policy and Planning Requirements for Emergencies*, Aug. 13, 1981.

17. *Functional Criteria for Emergency Response Facilities*, NUREG-0696, April 1981.

6. PLANT SYSTEM DESIGN REQUIREMENTS

6. PLANT SYSTEM DESIGN REQUIREMENTS

This chapter specifies how the ANS plant is divided into systems; assigns an SDD document to each system; and summarizes the major functions, responsibilities, and interfaces for each SDD.

This chapter also assigns design requirements to the SDDs. In some cases, the requirements are specified directly in this PDR. In other cases, the requirements are specified in Integrating System Design Documents (ISDDs), also defined here, which cover generic functional areas like containment. The requirements given here and in the ISDDs complement the safety and certification requirements found in Chap. 4. Requirements in the ISDDs carry the same authority as those in this PDR. Any inconsistencies in requirements shall be resolved at the PDR level.

Sect. 6.1 presents the ISDDs and SDDs in tabular form for easy reference. To enhance reactor safety, operability, testability, availability, and maintainability, maximized use of simplified concepts in the design of reactor and plant systems shall be used.

6.1 DEFINITION OF SYSTEMS

6.1.1 Integrating Systems

Table 6.1-1 identifies each ISDD by number and title, defines its responsibilities, and identifies the SDDs that implement its requirements. As shown in Table 6.1-1, ISDDs are numbered ISDD-21 through ISDD-25, consistent with the WBS shown in Fig. 1.3-1.

6.1.2 Individual Systems

Table 6.1-1 identifies each SDD by number and title, defines its functions and responsibilities, and identifies its primary interface systems. As done with the ISDDs, the SDDs are assigned numbers that follow the WBS (Fig. 1.3-1). Specifically, the general categories of SDDs include reactor systems (SDD-31 to SDD-35), experiment systems (SDD-41 to SDD-411), site and buildings (SDD-51 to SDD-58), and plant systems (SDD-61 to SDD-610).

Table 6.1-1. Responsibilities of the ISDDs and the SDDs that implement ISDD requirements

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
21	**Reactor containment systems** • fission-product control • heat removal and pressure control • combustible-gas control and detection • personnel protection • equipment protection	31, 32, 33, 41, 52, 61, 63, 69

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
22	**Reactor shutdown and cooling systems** • shutdown • coolant-flow control • coolant-pressure control • coolant-inventory control • coolant-heat sink • coolant-purity control *NOTE: Reactor = reactor assembly, including reactor primary and reflector cooling.*	31, 33, 34, 41, 52, 53, 61, 69
23	**Environmental compliance and monitoring** • federal, state, and local environmental regulations, guidance, recommendations, and best management practices. • air quality • water quality • waste and hazardous materials • monitoring and surveillance	47, 51, 57, 61, 63, 64, 65, 67, 68, 610
24	**Security functions** • threat definition • security-zone definitions • physical barriers • access control • intrusion detection • surveillance • security communications • security response	51, 52, 53, 54, 55, 56, 57, 58, 62, 69, site external services
25	**Instrumentation, control, computing, and telecommunications systems** • architecture • integration	33, 62, 69, and I/C portions of other systems
31	**Reactor assembly** • reactor core - neutron generation - heat generation	32, 33, 34, 35, 36, 41, 44, 45, 52, 61, 62, 64, 67, 68, 69, 410, 411

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	- fission-product barrier - heat transport to primary coolant • structural support - fuel elements - transuranic rods (1) - material irradiation samples (1) - control/shutdown mechanisms - cold source (1) - hot source (1) - reflector vessel - nuclear instrumentation (1) - reactor assembly coolant piping • heat transfer to reflector coolant • fission-product barriers - primary to reflector - primary to H_20 pool - reflector to H_20 pool • coolant inventory control - submerged operation - subpile room limited-volume enclosure *Note (1): Other systems provide these components.*	
32	**Refueling system** • new fuel management - nuclear criticality control and detection - receipt inspection - storage - reactor loading • spent fuel management - reactor unloading - nuclear criticality control and detection - cooling - shielding - storage - disposal	31, 33, 34, 35, 52, 53, 61, 62, 63, 64, 65, 66, 67, 69, site external facilities

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
33	**Reactor instrumentation and controls** • reactor and reflector monitoring requirements - shutdown rod positions - neutron flux - coolant-valve positions - coolant flow - coolant temperature - coolant pressure - coolant inventory - failed-fuel detection • reactor and reflector control requirements - reactivity - coolant temperature - coolant pressure - neutron flux - thermal power • reactor and reflector protection requirements - beam tube isolation - experimental system requirements for reactor shutdown - reactor reactivity shutdown - primary shutdown - secondary shutdown - fast secondary shutdown *[HOLD]*[6-1] - liquid poison injection *[HOLD]*[6-2] - reactor primary-coolant isolation - reactor secondary-coolant isolation - reactor main pump motor shutdown - experimental tube isolation • reactor protection system • experiment protection system • nuclear instrumentation	31, 32, 34, 35, 36, 41, 52, 52, 53, 61, 62, 63, 64, 65, 66, 67, 68, 69
34	**Fuel element assembly**	31, 32, 33, 35, 61
35	**Reactor assembly**	31, 32, 33, 34

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
36	**Reactor maintenance systems** • special maintenance equipment	31, 32
41	**Neutron beam transport** • neutron guide systems - cold/very cold neutron transport - cold-source interfaces - integrity monitoring - support and alignment - shielding - shutters - beam conditioning • neutron beam-tube systems - thermal and hot neutron transport - small sample insertion and viewing - reactor assembly interface - shielding - shutters - beam conditioning	31, 33, 42, 43, 47, 52, 54 61, 62, 67, 69, 410, 411, 610
42	**Neutron scattering instruments** • beam preparation - shutter, collimator, shielding, and drums - crystal monochromators - choppers - helical velocity selectors - optical devices - beam-geometry diaphragms • sample handling and environmental control - sample tables - goniometers - cryostats - furnaces - magnets • neutron beam monitoring • beam detection and analyses - energy analyzer - polarization analyzer	41, 48, 49, 52, 54, 62, 64, 65, 67, 69, 610

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	- detectors - multidetectors • instrument data collection and control • personnel safety	
43	**Nuclear and fundamental physics instruments** • beam or guide stations - beam preparation - shielding - personnel safety • specialized stations - liquid hydrogen facility - small fissile target handling - activated target handling • neutron beam monitoring	41, 48, 49, 52, 54, 62, 64, 65, 67, 10, 69, 610
44	**Transuranium production facilities** • transuranium target handling - loading and unloading - cooling - storage pool target handling - receiving/shipping • epithermal neutron hydraulic rabbit irradiations - hot-cell loading and unloading - cooling - transfer to/from irradiation stations - handling and storage	31, 32, 33, 52, 53, 61, 62, 67, 69
45	**Materials irradiation facilities** • in-core materials irradiation - loading/unloading - instrumentation connect/disconnect - cooling - monitoring and controls • reflector vessel materials irradiation - hot-cell loading and unloading - cooling - transfer to/from irradiation stations	31, 32, 33, 52, 53, 61, 62, 67, 69

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• handling - reactor pool - storage pool	
46	**Isotopes production facilities** • irradiation in reflector vessel - loading/unloading - cooling - handling and shipping - liquid or gas target handling • irradiation using hydraulic rabbit tubes - hot-cell loading and unloading - cooling - transfer to/from irradiation stations - handling and storage - controls	31, 33, 52, 53, 61, 62, 67, 69
47	**Analytical chemistry facilities** • activation analysis using pneumatic tubes - hot-cell loading and unloading - cooling - transfer to/from irradiation stations - handling and storage - controls - counting • cold-neutron beam activation analysis - beam splitting - handling and storage - counting • neutron depth profiling • gamma irradiation using spent fuel - sample handling - counting	31, 33, 52, 53, 54, 61, 62, 67, 69
48	**Instrument support facilities** • sample handling - laboratory support - specialized shop support	42, 43, 47, 52, 54, 62, 63, 64, 65, 67, 610

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	- cryogenic support - storage • instrument preparation - off-line assembly and testing - shop support - storage, maintenance, and calibration • crystal preparation - growth - characterization - mounting - storage	
49	**Experiment systems network** • data transfer - instrument to instrument - instrument to office - off-site • data analysis • data storage and retrieval	41, 42, 43, 44, 45, 46, 47, 48, 52, 54, 55, 62
410	**Cold source assemblies** • generate cold neutrons • cold source cooling • deuterium containment	31, 33, 41, 52, 62, 63, 64, 66, 67, 69
411	**Hot source assembly** • generate hot neutrons • hot source temperature control • hot source containment	31, 33, 41, 52, 62, 65, 67, 69
51	**Land improvements** • site characterization and monitoring • land erosion control • vehicle access and parking • personnel walkways • storm water management • site security control	24, 62, 64, 66, 67, 69

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
52	**Reactor building** • equipment space, arrangement, and structural support • personnel protection • reactor-pool structure and shielding • fuel-pool structure and shielding • hot-cell structure and shielding • equipment-cell structure and shielding - water-filled cells - air-filled limited volume cells - other air cells • reactor containment fission-product barriers - inner barriers - outer barriers • reactor containment penetration structural support • heat-removal and pressure control • combustible gas control • fuel-transfer facilities • personnel access/evacuation • personnel movement in building • personnel rest rooms • equipment access • security-control barriers • fire-control barriers • pool liner leak detection	51, 53, 54, 55, 61, 62, 63, 64, 65, 66, 69, 610, all systems with penetrations
53	**Reactor support building/operations support building** • equipment space, arrangement, and structural support • equipment protection • personnel protection • radiation shielding • waste storage • fuel transfer facilities • equipment cell structure and shielding • main and emergency control room space and shielding • fire-control barriers • security-control barriers • personnel access and egress	32, 51, 52, 55, 61, 62, 64, 65, 63, 66, 69, 610

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• personnel movement in building • personnel change rooms • personnel restrooms • pool liner leak detection	
54	**Guide hall and research support building** • equipment space, arrangement, and structural support • equipment protection • personnel protection • radiation shielding • fire-control barriers • security barriers • personnel access and egress • safety-equipment barriers • laboratory facilities • personnel movement in building • personnel change rooms • personnel restrooms	51, 52, 55, 62, 63, 64, 65, 66, 69, 610
55	**Office building/interface building** • office space • reception area • lounge, food service, and eating areas • auditorium • conference rooms • equipment space and arrangement • personnel protection • personnel access and egress • health physics monitoring area • fire-control barriers • security control • personnel movement in buildings • personnel restrooms	51, 52, 53, 54, 62, 63, 64, 65, 66, 69
56	**Detritiation building** • equipment space, arrangement, and structural support • equipment cell structures and shielding	51, 61, 62, 63, 64, 65, 66, 67, 68, 69, 610

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• equipment protection • tritium control • heavy-water control • personnel protection • personnel access and egress	
57	**Other structures** • equipment space, arrangement, and structural support • equipment protection • cooling-tower basins • pump house • waste storage areas • diesel-generator building • vent stack • security portals • electrical switch yards • reactor mockup building	51, 62, 63, 64, 65, 66, 67, 69
58	**Construction support** • construction material storage • construction personnel offices • construction shops • computer rooms plant simulator and data handling system equipment	51, 69
61	**Reactor water system** • heavy-water storage, management, and purity control • reactor primary cooling (2) (F,PR,I,PU,HS) • refueling water system • reflector vessel cooling (F,I,PU,HS) • fission-product control - submerged operation - leak-tight cells - isolation valves • heavy water leak detection	31, 33, 52, 53, 62, 63, 64, 65, 66, 67, 68, 69

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	Note (2): The following symbols are used to indicate what properties of the coolant are included in the responsibilities of the reactor-water system: F = flow, PR = pressure, I = inventory, PU = purity, HS = heat sinks.	
62	**Plant electrical power and communication systems** • off-site ac power supply • standby on-site ac power generation • ac power distribution - non-class 1E - class 1E • on-site dc power distribution - non-class 1E - class 1E • uninterruptible power supply - non-class 1E - class 1E • grounding • lightning protection • lighting • underground metal cathodic protection • plant communications • plant fire alarm • cable/raceway routing	51, 52, 53, 54, 56, 57, 61, 63, 64, 65, 66, 69, 610, site external services
63	**Environmental control systems** • reactor building - operating area atmosphere (3) (H,C,V,GT,CGC,P) - experimental area atmosphere (H,C,V,GT,CGC,P,HM) - primary-secondary annulus atmosphere (H,C,V,GT,P) - equipment-cell atmosphere (V,GT,CGC,P) - process-equipment vents (GT) - combustible gas control	51, 52, 53, 54, 55, 56, 57, 61, 62, 64, 67, 69

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• reactor and operations support building - operating area atmosphere (H,C,V,GT,P) - main and emergency control rooms (H,C,V,GT,P,HM) - equipment-cell atmosphere (V,GT,P,HM) - process-equipment vents (GT) • office and interface control building (H,C,V,HM) • guide hall and research support building (H,C,V,GT,HM) • detritiation building - operating areas (H,C,V,P) - equipment cells (V,GT,CGC,P) - process-equipment vents (GT) • diesel generator building (H,C,V)	
	Note (3): The following symbols are used to indicate what atmosphere control is required: H = heating, C = cooling, V = ventilation, GT = gas treatment, CGC = combustible gas control, P = pressure, HM = humidity.	
64	**Plant water systems** • light water pool cooling and cleaning • auxiliary safety-related cooling water • secondary cooling system • process water supply • nonsafety-related cooling water • potable water supply • sanitary sewer-water collection and disposal • equipment and floor drain • safety-related chilled water • nonsafety-related chilled water • building heating water	51, 52, 53, 57, 61, 62, 63, 65, 66, 67, 69, site external services

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• light water sampling system • cooling-tower makeup water • cooling-tower blowdown treatment and disposal • demineralized water management	
65	**Plant services systems** • instrument air storage and distribution • breathing air storage and distribution • vacuum • auxiliary gas storage and distribution • cryogenic storage and distribution • steam generation and distribution • steam condensate collection and disposal	52, 53, 54, 56, 57, 61, 62, 66, 67, 69
66	**Plant fire protection** • fire-protection program management • fire hazards analyses • fire detection and alarms • fire suppression, fire stops, and area separation • fire barriers • fire exits • fire water storage and distribution • fire brigade • control of combustibles	52, 53, 54, 55, 56, 57, 61, 62, 63, 64, 65, 69, site external services
67	**Plant waste systems** • overall waste management • waste collection, processing, disposal - tritium waste - other radioactive waste - chemical waste - laundry	52, 53, 57, 61, 64, 65, 66, 68, 69, 610, site external services
68	**Heavy-water upgrade and detritiation systems** • receipt, sampling, and analysis • protium removal and disposal • tritium removal and disposal • heavy-water recovery	52, 53, 56, 61, 62, 63, 64, 65, 66, 67, 69, 610, site external services

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• radioactivity confinement • heavy water leak detection	
69	**Plant instrumentation, control, and data systems** • plant monitoring - meteorological • emergency lighting and communication • electrical cable construction, cable trays, and cable penetrations • ventilation and smoke removal - radiation monitoring - security - intrusion detection and alarms - video monitoring - computers/communications - health physics - post-accident monitoring and sampling - loose-parts monitoring - seismic monitoring - combustible-gas detection • plant data management - acquisition - processing - records - storage and retrieval • plant safety-related systems controls • control and data acquisition systems for control of many safety systems • containment isolation, control, and monitoring • safety-parameter display system • plant remote control for testing and maintenance • plant control room simulator • main control room, main control consoles, and displays • main control room systems integration and human factors engineering • input/output interface equipment • remote shutdown consoles, displays, and communications	31, 32, 33, 52, 54, 55, 56, 57, 61, 62, 63, 64, 65, 66, 67, 68, 410, 411, 610

Table 6.1-1 (continued)

ISDD/SDD number	Title and implementing responsibilities	Interfacing systems (SDDs)
	• emergency operations center information systems and communications • technical support center information systems and communications • operational support center communications • set and maintain design and other criteria for all instrumentation and control systems	
610	**Plant maintenance and general purpose equipment** (4) • general maintenance shops and equipment • radioactive decontamination • major building cranes	32, 52, 53, 54, site external services
	Note (4): Excludes special tools provided by others.	

6.2 OTHER PLANT REQUIREMENTS

6.2.1 Operating Conditions

[TBD][6-1]

6.2.2 Site Characteristics

[TBD][6-2]

6.2.3 Structural Design

ANS Category I and I-L structures shall be designed to comply with 10 CFR 50, Sect. 50.55a, applicable criteria of Appendix A and Appendix B. NRC regulatory guides and industry standards shall be applicable to the design as provided in Chap. 3 of NUREG-0800. Category I and I-L structures shall be designed in accordance with their classification for appropriate design-basis events and combinations to allowable limits given in NUREG-0800, Sects. 3.8.1, 3.8.2, 3.8.4, and 3.8.5. Concrete and steel structure design shall be based on strength design and allowable stress design methods respectively. Category I-L structures shall not be designed for the OBE event.

ANS Category II structures shall be designed in accordance with DOE Order 6430.1A,[1] UCRL-15910,[2] and other applicable national codes and standards.

The design of all structures shall comply with applicable federal, state, and local codes.

Detailed structural design requirements are specified in the document **ANS Structural Design Criteria,**[3] which was written to supplement this section and which is an integral part (Category A Reference) of this PDR. Definitions of SSE and an appropriate equivalent to the OBE are presented in this reference as well.

6.2.4 Fluid System Components and Mechanical Equipment Design

The ANS fluid system components, mechanical equipment, and their supports shall be designed to comply with 10 CFR 50, Appendix A; DOE Order 6430.1A; and UCRL-15910. The design shall comply with all applicable federal, state, and local codes and standards. NRC Regulatory Guides shall be applicable to the design as specified in Chaps. 3 and 5 of NUREG-0800.[4] The system components, equipment, and their supports shall be designed in accordance with their classification and the appropriate design-basis events and event combinations to allowable service stress limits as shown in NUREG-0800, Sect. 3.9.3, Appendix A, Table I.

Detailed design requirements for fluid system components, mechanical equipment, and their supports are specified in the document **ANS Design Requirements for Fluid System Components and Mechanical Equipment,**[5] which was written to supplement this section and is an integral part (Category A Reference) of this PDR.

6.2.5 Radiation Protection Criteria

The principal design goal associated with radiation protection is to keep radiation doses to personnel ALARA. The controlling document for radiological protection at ORNL is DOE Order 5480.11, *Radiation Protection for Occupational Workers.*[6] That DOE Order sanctions the guidance of PNL-6577, *Health Physics Manual of Good Practices for Reducing Radiation Exposure to Levels That Are As Low As Reasonable Achievable (ALARA).*[7] NRC guidance is provided by Chap. 12 of the SRP[4] (see Table B.8), 10 CFR 20, and Regulatory Guides 8.8[8] and 8.10[9] (see Table B.7). Acceptable radiological practices at ORR facilities are given in the *ORNL Health Physics Division Procedures Manual.*[10] SDC-75 in Table B.6 also limits radiation exposure.

As a minimum, radiological protection shall address the following:

1. the responsibility of meeting dose limits primarily through physical controls (e.g., confinement, ventilation, remote handling, and shielding), as opposed to administrative controls;
2. appropriate monitoring of occupational workers and plant areas and systems to prevent exposure, to detect contamination, and to verify compliance with exposure limits of all persons on the site;
3. control of entry, appropriate to the hazard, into radiological areas;
4. appropriate training for all but casual visitors;
5. automatic isolation of facility effluent lines in the event that effluent radioactivity exceeds that specified in DOE 5400.5;[11] and
6. a system of fixed nuclear accident dosimeter units and monitors placed in the facility to aid in determining appropriate mitigation and recovery actions after postulated accidents.

Radiation exposures for personnel located within the plant shall not exceed those permitted by DOE 5480.11 or 10 CFR 20. Areas of the plant shall be assigned various radiation-zone classes

that specify personnel accessibility while the plant is operating and/or while maintenance is being performed. The radiation-zone classes shall include the following:

1. areas routinely occupied with no time restriction [< 2 μSv/h (0.2 mrem/h)];
2. areas not routinely occupied [2 to 30 μSv/h (0.2 to 3.0 mrem/h)]; and
3. three radiation areas defined by DOE 5480.11: radiation area [30 to 1000 μSv/h (3.0 to 100 mrem/h)], high radiation area [1 to 50 mSv/h (100 to 5000 mrem/h)], and very high radiation area [> 50 mSv/h (5000 mrem/h)].

Controls at least as restrictive as DOE Order 5480.11 shall apply to the three radiation areas included in the third group, and ALARA shall apply to the other two groups.

6.2.6 Building Arrangements and Separation Criteria

The ANS plant layout shall be developed giving maximum consideration for separation of all divisional and nondivisional electrical and mechanical equipment, including electrical cables. This requirement will prevent or reduce the possibility of damage to redundant safety-related equipment during design-basis events (e.g., fires, pipe breaks, missiles, or floods). If possible, redundant safety-related equipment shall be separated by barriers sufficient to provide protection from the design-basis event (e.g., fire-rated walls, missile shields, separate floors, conduits, or penetrations). Where redundant systems cannot be separated by barriers, as in containment and in the control room, other acceptable means shall be designed, evaluated, and installed to prevent the loss of function of safety-related systems during design-basis events.

The separation requirements outlined in 10 CFR 50, Appendix R; Branch Technical Position CMEB 9.5-1[12]; and Regulatory Guide 1.75[13] shall be met.

The protection system shall be separated from control systems so that failure of any single control-system component/channel or failure of/removal from service of any single protection system component/channel that is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to ensure that safety is not significantly impaired (10 CFR 50, Appendix A, Criterion 24).

6.2.7 Equipment Qualifications

All ANS electrical, mechanical, and instrumentation and control equipment performing a safety function shall be capable of performing that safety function under all normal, abnormal, and design-basis accident conditions as required by 10 CFR 50, Appendix A, Criterion 4. All Class 1E electrical, mechanical, and instrumentation and control equipment shall meet the requirements of IEEE Standard 323-1983.[14] See Sect. 4.7 for classifications of systems and components, seismic qualifications, and a minimum list of standards and requirements for these components.

6.2.8 Human Factors

A human-factors plan shall be prepared for ANS. This plan will satisfy the requirements set forth in DOE Order 6430.1A; DOE Draft Order HFE.1, Human Factors Engineering Program Requirements for Department of Energy Nuclear Facilities;[15] and DOE-Standard HFAC 1, Human Factors Engineering–Design Criteria.[16] The plan shall provide a strategy for conducting

the human-factors design activities in a time frame consistent with the overall ANS design effort. It shall also provide an awareness of required human-factors activities and a means for evaluating the contribution of such efforts. The plan shall comply with all human-factors concerns identified in NUREG 0800, Chapter 18, "Safety Analysis Review."[4]

Incorporating human factors into the design shall be the joint responsibility of the design team and human-factors experts. The designers' knowledge of the hardware, software, and application context and the expertise of the human-factors specialists concerning human abilities, characteristics, behavior, and motivation shall be applied cooperatively in the design of all systems.

Human factors shall be emphasized in the following areas:

1. the design and development of products, equipment, facilities, and environments that are used by people;
2. the development of procedures for performing human activities;
3. the provision of services to people; and
4. the evaluation of human-environment (or system) interfaces.

The objectives of these activities are to enhance overall system effectiveness and efficiency and to maintain a desirable level of human welfare and well-being. The approach is a systematic application of relevant information about human abilities, characteristics, behavior, and motivation in the execution of such functions.

If the design activity involves a human or impacts human performance for related activities, good design principles necessitate a total systems approach that explicitly and simultaneously considers the following characteristics:

1. activities/functions (interfaces, performance aids, procedures, and training);
2. the context of the activity/function (physical noise, lighting, and temperature; social crowding, isolation, and self-esteem); and
3. the human in the loop (knowledge, skills, abilities, and motivation).

Considerations such as designing for maintenance accessibility or designing panel layouts that minimize human data search requirements shall be inherent qualities of the design process.

6.2.9 Fire Protection

The ANS fire-protection program will have two objectives. The primary objective is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases into the environment (10 CFR 50, Appendix A, Criterion 3). The second objective is to provide assurance for life safety and protection of the investment in the plant.

To meet the first objective, the ANS structures, systems, and components important to safety shall be designed and located to minimize the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical and a fire-protection program for areas containing safety-related equipment shall include fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel as required by 10 CFR 50, Appendix R, and as implemented in the NRC Branch Technical Position Paper CMEB 9.5-1.[12]

To meet the second objective, the ANS structures and the fire-protection systems for areas not containing safety-related equipment shall be designed to comply with the requirements of DOE Order 6430.1A.

6.2.10 Safeguards and Security Requirements

6.2.10.1 Requirements

The safeguards and security policies and systems implemented for ANS shall be in compliance with DOE Orders 5480.6,[17] 5632.2A,[18] 5632.6,[19] and 6430.1A;[1] 10 CFR 50 and its appendices; 10 CFR 73; and 10 CFR 100.

A physical security assessment shall be concluded per 10 CFR 50.34(c).

A safeguards contingency assessment shall be conducted per 10 CFR 50.34(d).

DOE Order 5632.2A requires that the safeguards and security systems protect special nuclear material (SNM) and vital equipment, including nuclear reactors, from theft, sabotage, and other hostile acts that could adversely impact national security or the health and safety of the public. Specific hostile actions to be protected against include: theft of SNM, diversion of SNM, radiological sabotage of SNM or vital equipment, and industrial sabotage of SNM or vital equipment.

Security measures shall be implemented to provide protection against design-basis threats No. 1 and No. 2 below (required by 10 CFR 73.1 and DOE Order 5632.2A) and against design-basis threats No. 3 and No. 4 (based on implied requirements stated above and on good practices required by ORNL).

1. Radiological sabotage.
   A. A determined violent external assault; attack by stealth; or deceptive actions by several persons with attributes, assistance, and equipment as defined in 10 CFR 73.1.
   B. An internal threat by an insider, including an employee (in any position).
2. Theft or diversion of formula quantities of SNM.
   A. A determined, violent, external assault; attack by stealth; or deceptive actions by a small group with attributes, assistance, and equipment as defined in 10 CFR 73.1.
   B. An individual, including an employee (in any position).
   C. A conspiracy between individuals in any position who may have access to and/or detailed knowledge of facilities or items that could facilitate theft of SNM.
3. Attempts to disrupt the operations of the facility.
4. Spontaneous actions of a large group of people involved in acts of civil unrest.

6.2.10.2 Graded Approach to Security

ANS security requirements shall be developed so that ANS research zones remain accessible to users and so that operation zones are accessible only to operators and operations personnel. This graded approach to security will combine physical security measures at the plant site boundary, an enhanced level of security within sensitive internal areas, and a still further enhanced level of security for reactor operations and maintenance areas in and around the reactor and the reactor support building. The determination of the level of risk to ANS will be made via various vulnerability assessments that will identify weaknesses that could be exploited by possible adversaries and that will determine the levels and types of security systems to be

included. The safeguards and security program will protect ANS by integrating features designed to prevent forced entry by intruders and by using a composite of engineered security features and internal controls to provide passive protection of the reactor from insider sabotage.

In the event that results of a radiological sabotage assessment reflect a possible adverse impact upon public health and safety (10 CFR 100 limits exceeded), the secure environment shall then be designated and protected as a Vital Area.

6.2.11 Radioactive and Other Hazardous Waste Management Requirements

Radioactive and other hazardous waste produced from plant operations shall be managed to ensure that effluent and wastes are controlled and that adverse impacts on facility workers, the surrounding population, and the environment are minimized. ANS waste management requires detailed implementation of DOE Order 5820.2A, *Radioactive Waste Management*[20] and supporting implementation documents ORNL/TM-11433[21] and ORNL/TM-11166.[22] Other referenced requirements include 10 CFR 50, Appendix I; the SRP, Chapter 11;[4] 10 CFR 20; and GDC-60 and -64 from 10 CFR 50, Appendix A.

The waste management ISDDs and SDDs for ANS shall, as a minimum, indicate interfaces with appropriate organizations of the facility manager, such as the ORNL Environmental, Safety, and Health Compliance Directorate. Waste treatment and processing systems shall be designed, constructed, and operated to comply with regulatory requirements and limits and with good practices applicable to the following:

1. separation of dissimilar wastes and waste streams (e.g., radioactive and nonradioactive);
2. monitoring and discharge of effluent; and
3. waste forms for transport, storage, and disposal.

Design decisions, written procedures, training, and practices will be sufficient to minimize production of effluent and primary wastes; to contain the wastes safely; and, to the extent practicable, maximize recycling of wastes and minimize secondary waste generation.

REFERENCES FOR CHAPTER 6

1. U.S. Department of Energy, Order 6430.1A, *General Design Criteria*, April 1989.

2. R. P. Kennedy et al., *Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards*, Report UCRL-15910, June 1990.

3. **A. R. Perkins, R. J. Hunt, and N. E. Stone, ANS Structural Design Criteria, to be published at Oak Ridge Natl. Lab.**

4. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, June 1987.

5. **J. S. G. Williams, ANS Design Requirements for Fluid System Components and Mechanical Equipment, to be published at Oak Ridge Natl. Lab.**

6. U.S. Department of Energy, Order 5480.11, *Radiation Protection for Occupational Workers*, June 1990.

7. W. N. Harrington and D. P. Higby, *Health Physics Manual of Good Practices for Reducing Radiation Exposure to Levels that are As Low As Reasonably Achievable*, PNL-6577, June 1988.

8. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.8, *Information Relative to Ensuring that Occupational Radiation Exposures and Nuclear Power Stations will be As Low As Reasonably Achievable*, June 1978.

9. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable*, May 1977.

10. *ORNL Health Physics Manual, Procedures and Practices for Radiation Protection and Radiation Monitoring*, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., April 1992.

11. U.S. Department of Energy, Order 5400.5, *Radiation Protection and the Public and the Environment*, February 1990.

12. *Guidelines for Fire Protection for Nuclear Power Plants*, Branch Technical Position CMEB 9.5-1, USNRC Standard Review Plan, NUREG-0800, July 1981.

13. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.75, *Physical Independence of Electrical Systems*, September 1978.

14. *IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*, IEEE Std. 323-1983, Nuclear Power Engineering Committee for the IEEE Power Engineering Society.

15. U.S. Department of Energy Draft Order HFE.1, *Human Factors Engineering Program Requirements for Department of Energy (DOE) Nuclear Facilities*, May 1991.

16. *Human Factors Engineering–Design Criteria*, DOE-Standard HFAC 1, DOE/NE-LLNL/NSSP, February 1992.

17. U.S. Department of Energy, Order 5480.6, *Safety of Department of Energy Owned Nuclear Reactors*, September 1986.

18. U.S. Department of Energy, Order 5632.2A, *Physical Protection of Special Nuclear Material and Vital Equipment*, February 1988.

19. U.S. Department of Energy, Order 5632.6, *Physical Protection of DOE Property and Unclassified Facilities*, February 1988.

20. U.S. Department of Energy, Order 5820.2A, *Radioactive Waste Management*, September 1988.

21. J. S. Baldwin, J. D. Sease, and D. L. Jones, *Oak Ridge National Laboratory Waste Management Plan for Department of Energy Order 5480.2A*, ORNL/TM-11433, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1989, (Rev. 1, December 1991).

22. J. S. Baldwin et al., *Oak Ridge National Laboratory Implementation Plan for DOE Order 5480.2A*, ORNL/TM-11166, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., April 1989.

7. QA PROGRAM REQUIREMENTS

7. QUALITY ASSURANCE PROGRAM REQUIREMENTS

7.1 QUALITY ASSURANCE PROGRAM

The Quality Assurance (QA) program for the ANS Project shall meet the requirements of DOE Order 5700.6B, *Quality Assurance*,[1] and the intent of 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, as elaborated in Sect. 17 of the NRC SRP, NUREG-0800.[2] The ANS QA program, which is documented in the ANS QA Plan, ORNL/TM-11446,[3] addresses the basic, supplemental, and nonmandatory requirements indicated in the appendices of ASME NQA-1[4] and the provisions of ASME NQA-2.[5]

The quality standards in use for the project are:

1. ASME NQA-1, *Quality Assurance Program Requirements for Nuclear Facilities*, latest year and addenda:
   - basic requirements,
   - supplements,
   - appendices, and
   - figures.
2. ASME NQA-2, *Quality Assurance Requirements for Nuclear Facility Applications*, latest year and addenda:
   - codes, standards, and specifications referenced in the text;
   - introduction; and
   - individual parts.

The ANS QA Plan is to be reviewed annually. Updates or revisions to NQA-1 or NQA-2 are reviewed for applicability and are incorporated into the QA Plan as appropriate. The applicability of these standards to the project is contained in Table 7.2-1, ANS Project quality levels. The application of these standards to project activities is described further in this section. Additional grading of the application of QA elements is performed according to the work operations. This graded approach is further delineated in the ANS QA Plan.

Grading of QA program elements addresses qualitative factors such as:

1. **Safety**
   - minimum effect—no risk to the health and safety of on-site personnel
   - maximum effect—undue risk to the health and safety of on-site personnel and to the public
2. **Development of requirements**
   - minimum effect—minimal effort and simple
   - maximum effect—extensive effort and complex
3. **Activity complexity**
   - minimum effect—few simple processes required
   - maximum effect—state-of-the-art complex processes required

4. **Investigation maturity**
   - minimum effect—proven investigative method available for same use
   - maximum effect—completely new method for a new use
5. **Characteristics**
   - minimum effect—no restraining or interrelated dependence on other items
   - maximum effect—many restraining and/or interrelated attributes
6. **Economics**
   - minimum effect—negligible inconvenience, cost, or schedule impact
   - maximum effect—total loss of service capability and extreme cost or schedule impact
7. **Activity importance**
   - minimum effect—standard industrial or laboratory practice
   - maximum effect—critical to obtaining DOE authorization for facility design, construction, or operation.

The responsibility for the quality of ANS structures, systems, components, or activities rests with the line management of the project and the various supporting organizations and contractors. Subtier ANS QA plans showing commitment to and application of the aforementioned standards shall be required of contractor support organizations with extensive multidisciplinary scopes of work. Such plans shall be reviewed by the ANS Project organization before the contractors begin project work.

To ensure that the required quality of all systems within the overall plant is achieved, programs of plans and actions shall be implemented to control and verify quality attainment through management, design, procurement, manufacture, installation, testing, and operations. These programs shall conform, as applicable, to the requirements and shall provide the actions specified in ASME NQA-1, including supplements and nonmandatory appendices, and ASME NQA-2 as follows:

1. For management and planning, the program shall be in accordance with NQA-1, including supplements and nonmandatory appendices. The program shall address at a minimum:

   A. QA program, including
      - planning, and
      - QA Plans.
   B. organization, including
      - responsibilities and authorities;
      - personnel training and indoctrination; and
      - personnel, procedures, and equipment qualification.
   C. documentation, including
      - policies and procedures,
      - quality records,
      - trend analysis and reporting, and
      - quality-status reporting.
   D. management assessments
   E. corrective action
   F. engineering holds
   G. occurrence reporting

H. quality audits and assessments, including
   - planning;
   - evaluation of quality methods;
   - activity audits;
   - product audits;
   - self-assessments;
   - independent assessments;
   - nondestructive examination audits;
   - records audits; and
   - reporting, corrective action, and follow-up.

2. For system design and equipment procurement and manufacture, the program shall be in accordance with applicable portions of ASME NQA-1, including supplements and nonmandatory appendices, and ASME NQA-2, Sect. 2.7. The program shall address, at a minimum:

   A. design planning;
   B. design definition and control;
   C. design document review and control;
   D. design verification;
   E. development testing;
   F. quality records and reports;
   G. procurement planning;
   H. procurement requirements and procurement document review;
   I. evaluation and selection of suppliers, including
      - QA program documentation, and
      - procedures;
   J. configuration controls;
   K. source evaluation, surveillance, and inspection;
   L. receipt inspection and material controls;
   M. control of nonconforming items, deviations, and waivers;
   N. control of processes, including
      - special processes, and
      - manufacturing processes.

3. For systems installation and testing, the program shall be in accordance with applicable portions of ASME NQA-1 (including supplements and nonmandatory appendices) and ASME NQA-2 (except Sect. 2.7). The program shall address, at a minimum:

   A. organization,
   B. contract review,
   C. construction planning,
   D. configuration control,
   E. construction procurement,
   F. construction and installation control,
   G. construction inspections and tests,
   H. maintenance during construction,

I. QA records, and
J. audits and assessments.

4. For operations, the program shall be in accordance with applicable portions of ASME NQA-1, including supplements. The program shall address, at a minimum:

A. operation, maintenance, and modification;
B. planning;
C. organization;
D. preoperative control;
E. acceptance testing;
F. surveillance and surveillance testing;
G. periodic inspection and testing; and
H. inservice inspection.

Specifications for system components, materials, and services shall contain requirements for QA programs and special QA practices appropriate to the item or scope of work being defined.

The hierarchy of QA requirements is presented in Fig. 7.1-1.

The performance requirements and design features for ANS are defined in this PDR and are further detailed in the ISDDs and SDDs. These requirements establish the objectives of the facility design effort and serve as bench marks by which the success and acceptability of the design will be measured. They shall serve as standards of acceptance for the operability performance testing. The SDDs implement the QA requirements for management, planning, design, procurement, system installation, testing, and operations. The extent of application of QA requirements are related to the ANS classification and the ANS-assigned quality level (see Table 7.2-1).

A major component of the ANS QA program shall be a group of controls collectively referred to as the configuration management system. This system shall ensure that the performance requirements and design features of the facility are identified, are documented in a controlled reference source, and are distributed to project organizations involved in the design of the facility. Configuration management shall involve multiple aspects of the project's QA program, including personnel training, design, computer codes, drawings, safety analyses, procurement, procedures, documents, materials, tests, modifications, nonconformances, and quality records. Control of configuration shall be practiced throughout the life of the facility.

The configuration management system shall ensure that the facility performs as intended throughout its life cycle and that plant modifications do not compromise the defined function of the facility. It shall also ensure that the actual physical configuration of the facility is accurately reflected in the design documentation for the facility (e.g., drawings, reports, and analyses).

7.2 DEFINITION OF QUALITY LEVELS

The extent of management controls applied to an item or activity will vary as a function of the degree of confidence needed regarding the quality of an item or activity. Certain items or activities may require extensive QA efforts throughout all stages of development, while others may require only limited QA efforts in selected phases of development. QA measures will be selectively applied to an item or activity based upon grading/weighing factors such as

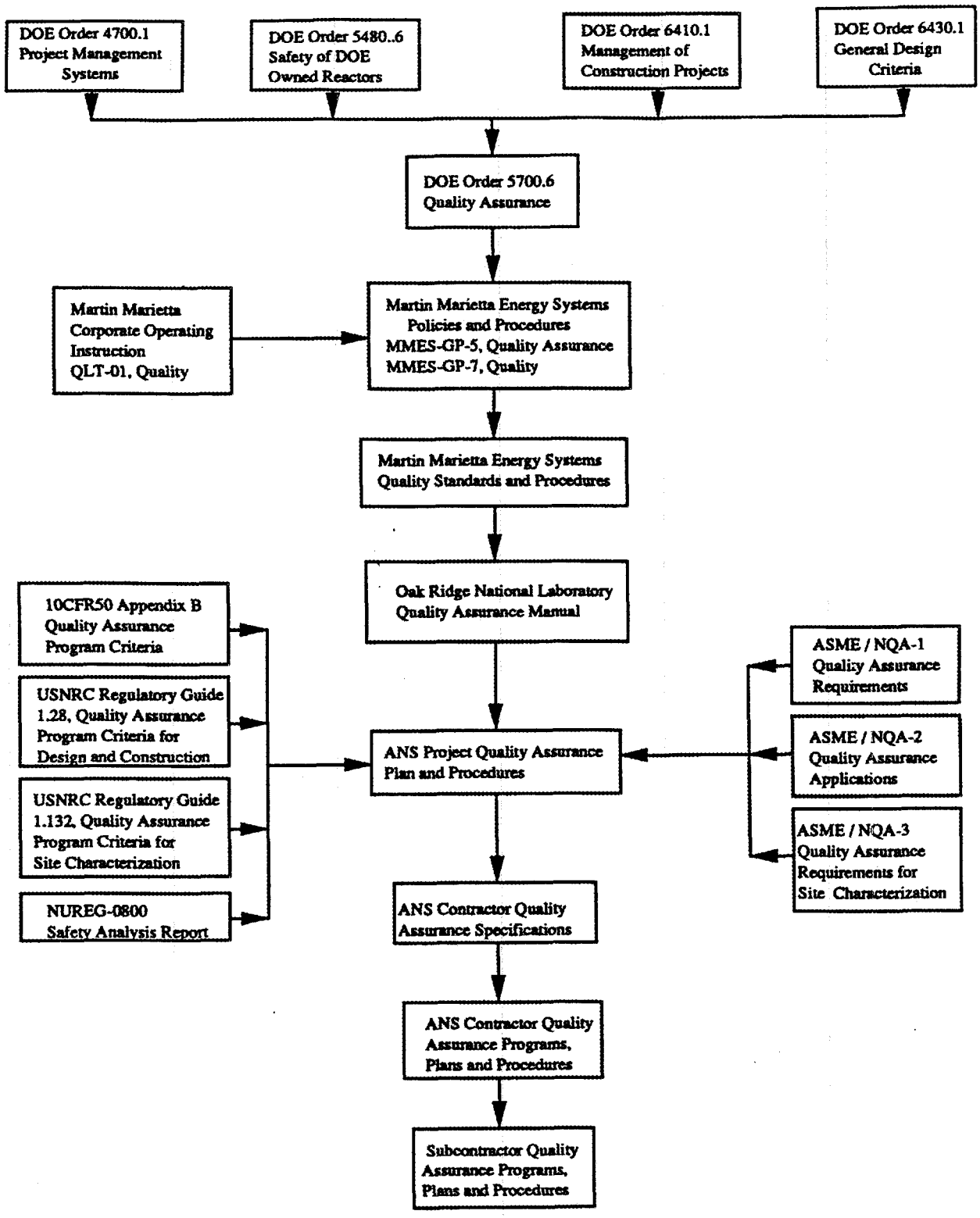

Fig. 7.1-1. Hierarchy of ANS Project QA program requirements.

Table 7.2-1. ANS Project quality levels

Quality level	Safety class[1]	Other considerations	Quality requirements[2]
QL-1	SC-1, -2, -3 NRS-4	Schedule delay > 6 months or Cost > $500,000	NQA-1 Basic requirements NQA-1 Supplemental requirements NQA-1 Nonmandatory appendices NQA-2 Requirements[3]
QL-2	NRS-3, -2	3 months < schedule delay < 6 months or $200,000 < Cost < $500,000	NQA-1 Basic requirements NQA-2 Requirements
QL-3	NRS-1, -0		Commercial practice per applicable codes

Notes:

[1] Safety classes are defined in Sect. 4.7 of this document.
[2] Quality assurance elements are selected based upon the item function or activity complexity and applicability of the quality program element to the item or activity.
[3] NQA-2 requirements as specified in Sect. 7.2.

1. consequences of failure;
2. data generation function (data importance);
3. complexity of design or fabrication uniqueness;
4. special controls needed;
5. functional compliance demonstrated by inspection, investigation, or test;
6. schedule impacts;
7. cost impacts; and
8. quality history or degree of standardization.

The provisions of the ANS QA program shall apply to ANS structures, systems, components, and activities in a graded manner commensurate with the quality-level designations defined herein, safety classifications designations defined in Sect. 4.7, and the grading criteria and methodology defined in the ANS QA Plan.

There are three levels under which ANS structures, systems, components, and activities are classified. These classifications serve to distinguish between the extent of application of ASME NQA-1 supplemental requirements, nonmandatory appendices, and ASME NQA-2 requirements. Further grading is described within the ANS QA Plan by describing which quality program elements are to be retained by the project and which elements are to be passed down to subcontractors. The final grading of quality program elements is based on work function and is contained in the working-level documents. These documents, tailored to the project's work scope, identify those appropriate QA elements that are to be implemented.

Quality Level 1 shall apply to ANS structures, systems, components, and activities

1. that impact reactor safety, or
2. that are identified as high-hazard facilities, or
3. whose failure could delay the project schedule by greater than six months, or
4. whose cost of repair or impact to the project is greater than $500,000, or
5. whose failure represents noncompliance with regulatory requirements.

This level includes ANS structures, systems, components, and activities assigned Safety Classes SC-1, -2, -3, and NRS-4 (refer to Sect. 4.7). The quality standards applicable to this safety-class level are ASME NQA-1 and NQA-2. As discussed in Sect. 7.1 and herein, further grading is based upon appropriate qualitative factors and work scope and is accomplished to determine the appropriate QA program needs and controls.

Quality Level 2 shall apply to ANS structures, systems, components, and activities

1. not affecting reactor safety but designated as
   - moderate hazard (NRS-3), or
   - important or low hazard (NRS-2), or
2. whose failure could delay the project schedule by more than three months (but less than six months), or
3. whose cost of repair or impact to the project is greater than $200,000 but less than $500,000.

The QA requirements applicable to this level are the basic requirements of ASME NQA-1 and the following sections of ASME NQA-2:

- Part 2.2, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storing and Handling of Items for Nuclear Power Plants,
- Part 2.3, Quality Assurance Requirements for Housekeeping for Nuclear Power Plants,
- Part 2.7, Quality Assurance Requirements for Computer Software for Nuclear Facility Applications, and
- Part 2.20, Quality Assurance Requirements for Subsurface Investigations for Nuclear Power Plants.

The remaining portions of ASME NQA-2 are not applicable since the basic requirements of ASME NQA-1 provide sufficient quality programmatic guidance.

Quality Level 3 shall apply to ANS structures, systems, components, and activities other than those assigned Quality Levels 1 or 2. QA requirements applicable to this level are those customary within standard or commercial use. Additional controls may be imposed at the discretion of the responsible design or construction organization to provide an appropriate level of confidence that items function in accordance with their design or that activities are performed as intended.

REFERENCES FOR CHAPTER 7

1. *Quality Assurance*, U.S. Department of Energy, Order 5700.6B, Sept. 23, 1986.

2. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, June 1987.

3. M. L. Gildner, *Advanced Neutron Source Quality Assurance Plan*, ORNL/TM-11446, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1989.

4. American Society of Mechanical Engineers, *Quality Assurance Program Requirements for Nuclear Facilities*, ASME NQA-1, September 1989.

5. American Society of Mechanical Engineers, *Quality Assurance Requirements for Nuclear Facility Applications*, ASME NQA-2, September 1989.

8. EXTERNAL INTERFACE REQUIREMENTS

8. EXTERNAL INTERFACE REQUIREMENTS

8.1 INTERFACES WITH ORNL

ANS will be fully integrated into the existing infrastructure of ORNL. Use will be made of existing capabilities at ORNL whenever practical. Under standard ORNL clearance rules, visitor badges will provide access to facilities such as the Central Research Library. ANS security and fire-protection personnel will coordinate activities with the ORNL Laboratory Protection Division. Operation of ANS will utilize existing roads, utilities, shops, and crafts organizations at ORNL to the extent commensurate with its mission as a fully integrated user facility.

ANS will be integrated into the existing infrastructure of ORNL with defined interfaces for the following:

1. transfer of shielded casks to other facilities handling transuranium, other isotope targets, materials irradiation targets, and other radioactive experiment components;
2. security and fire protection;
3. effluent monitoring, waste management, and waste disposal;
4. sample analyses at special laboratory facilities; and
5. telecommunications.

Several facilities at ORNL, including the Laboratory Emergency Response Center (Laboratory shift superintendent's office) and the guard and fire headquarters, will be provided with ANS alarm information.

The details of each of these interfaces will be defined in the appropriate SDD document(s).

8.2 INTERFACES WITH OTHERS

As required to meet the ANS operating schedule, interfaces will be established to cover the transfer of fresh fuel from the fabricator to secure storage locations and to the ANS site. Interfaces will also be established to cover the shipment of irradiated fuel off the ANS site.

Emergency response facilities, which are described in SDD 69, Plant Instrumentation, Control, and Data Systems, shall be provided for ANS. Information concerning the status of the plant and any emergency conditions and site monitoring will be provided for emergency planning and management per 10 CFR 50 and NUREG 0696.[1] System designs shall include the capability for this information to be available via nuclear data link(s) on a read-only basis for connection(s) to external state and federal agency(ies), including DOE, if and when determined necessary by DOE. The high-speed communications link(s) from ANS to the designated use point (DOE or other agency) and the associated computer terminal(s) for the user (DOE or other agency) to access this information will not be included in the ANS Project.

The details of each of these interfaces will be defined in the appropriate SDD document(s).

8.3 INTERFACE TO EXTERNAL USERS (EXPERIMENTERS)

A reception area shall be provided on site to clear visiting scientists into the facility. Tracking of personnel on site will be automatic by badge-operated control points, equipped as necessary with radiation monitors. Provision will be made for visitors to bring necessary specialized equipment from their home laboratories and to remove this equipment by using standard ORNL procedures, which shall include survey for radiation contamination. Provision shall be made to provide experimenters, when appropriate, with the following services on or near the site whenever experiments are to run for 24 consecutive hours or longer—access to food, short-term housing, transportation, and office and conference space.

Interfaces shall be designed to permit users to remove, in one or more industry-standard formats, data collected and/or analyzed at ANS. All instrument and control hardware provided for experimental use shall enable connection, by industry-standard interfaces, to user-supplied equipment, such as specialized temperature controllers.

Provisions shall be provided for electronic communications with remote locations using local and national networks.

The details of each of these interfaces will be defined in the appropriate SDD document(s).

REFERENCES FOR CHAPTER 8

1. *Functional Criteria for Emergency Response Facilities*, NUREG-0696, April 1981.

9. ACCEPTANCE TEST AND SURVEILLANCE TEST REQUIREMENTS

9. ACCEPTANCE TEST AND SURVEILLANCE TEST REQUIREMENTS

This section provides minimum requirements for inspections and tests performed during the plant acceptance test program and surveillance tests to be performed during the lifetime of the plant.

The scope, the depth, and the number of ANS acceptance tests shall meet the requirements in Regulatory Guide 1.68;[1] NUREG-0800, Sect. 14.2;[2] and the additional regulatory guides referenced by these documents. The acceptance test program shall be designed to demonstrate the performance of structures, systems, components, and design features that will be used during normal operations of the facility and to demonstrate the performance of standby systems and features that must function to maintain the plant in a safe condition in the event of malfunctions or accidents.

The scope, depth, and number of ANS surveillance tests shall be developed to provide assurance that equipment, systems, components, and process variables are within the limiting conditions and requirements for continued safe operation of the plant and for protection of the investment in the ANS plant.

9.1 PLANT ACCEPTANCE TEST PROGRAM

An ANS acceptance test program shall be developed to perform the following types of tests:

1. functional tests for all first-of-a-kind principal design features, qualification testing prior to installation, and acceptance tests of plant systems after installation;
2. construction tests on structures, equipment, hardware, and components prior to system turnover;
3. preoperational tests on structures, systems, and components prior to fuel load; and
4. start-up tests prior to operation at full power.

9.1.1 Functional Test Program

A functional test shall be conducted as required by NUREG-0800, Sect. 14.2, for all special, unique, or first-of-a-kind ANS principal design features. All such features shall be identified as early as possible during the conceptual design process, and a program shall be developed for each. The following features have been identified as possibly requiring a functional test:

1. reactor refueling machine/equipment and operations,
2. spent fuel transfer system,
3. control-rod assemblies,
4. shutdown-rod assemblies,
5. cold source,
6. hot source,
7. tritium leak-detection instruments,
8. guide and beam isolation system,

9. guide and beam replacement, and
10. core pressure boundary tube replacement.

9.1.2 Construction Test Program

Prior to system turnover, a construction test program shall be established to demonstrate that the construction and installation has been completed in accordance with the requirements specified to the contractor by the designer and equipment suppliers. The construction test program shall describe the scope of testing, provide detailed guidance for testing and controlling equipment or system status, and include methods for evaluating results from completed tests to determine conformance to design requirements.

9.1.3 Preoperational Test Program

A preoperational test program shall be established to demonstrate that structures, systems, and components will operate in accordance with the design's functional requirements in all operating modes and throughout the full operating range of the design. Testing will include, as appropriate, manual operation; operation of systems and components within systems; automatic operation; operation in all alternate or secondary modes of control; and operation and verification tests to demonstrate expected operation following loss-of-power sources and degraded modes for which the systems are designed to remain operational, including natural circulation.
Guidance from Appendix B includes 10 CFR 50, Appendix B, Sect. XI; 10 CFR 50, Appendix J; and 10 CFR 50.34(b)(6)(iii).

9.1.4 Start-Up Test Program

A start-up test program shall be established to conduct tests during and following fuel loading. These tests shall be divided into two phases:

1. **Low-power nuclear start-up tests:** These tests begin with the installation of fuel into the reactor and shall extend through initial criticality to power levels up to and including 5% power. These tests shall demonstrate satisfactory systems and overall plant operation at low-power levels. This phase of testing shall demonstrate that the reactor can be started up and shutdown in a safe manner and that the plant can be operated as designed in all the appropriate control modes from criticality to power levels up to and including 5% power.
2. **Power-ascension tests:** These tests shall demonstrate satisfactory steady-state system and overall plant operation at reactor power levels from 5% through 100% of full power, including safe shutdown and cool-down. Testing shall demonstrate that the plant can be operated in the appropriate control modes. Guidance from Appendix B includes 10 CFR 50.34.

9.2 SURVEILLANCE TEST PROGRAM

A surveillance test program and an acceptance test program shall be established to describe the required tests, calibrations, monitoring, or inspections needed to ensure that the necessary quality and operational status of systems and components are maintained or to ensure that

parameters and set points are periodically verified to be within the safety limits of the technical specifications and that the limiting conditions of operation can be met.

The design of systems and components shall incorporate features to implement required surveillance with the plant on-line to the greatest degree practical. For those surveillance activities that require the plant to be shut down, design features shall be provided to minimize the shutdown time required for these activities.

Plant surveillance includes the following:

1. tests, activities, and inspections required to ensure structure and component integrity as specifically identified in the applicable ASME code, national standards, and the ISI and surveillance program plan(s);
2. surveillance related to evaluating the amount of material property degradation resulting from the environmental effects to which it has been exposed under normal operating conditions;
3. those tests unrelated to technical specifications (comprised of operability checks, calibration, monitoring, and/or inspection), including materials surveillance;
4. periodic testing and inspection activities conducted in accordance with QA requirements to demonstrate that plant systems and components function in accordance with predetermined specifications; and
5. technical specification surveillance, including those tests, calibrations, monitoring, or inspections required to ensure that the necessary quality of systems relied upon in the safety analysis is maintained; that facility operation will be within the safety limits; and that the limiting conditions of operation will be met.

The technical specifications shall be developed as required by 10 CFR 50, Sects. 50.36 and 50.34(b)(6)(iii), and by NUREG 0800, Sect. 16.0, and its references. The testing program will meet the programmatic requirements of NQA-1, Basic Requirement 11 and Supplement Requirement 11S-1.

For areas not covered by the ASME Code, the designs for ANS shall include surveillance provisions consistent with the objective of maintaining safe operable conditions throughout the plant.

REFERENCES FOR CHAPTER 9

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.68, *Initial Test Programs for Water Cooled Nuclear Power Plants*, Rev. 2, August 1978.

2. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, Rev. 2, June 1987.

10. DECOMMISSIONING REQUIREMENTS

10. DECOMMISSIONING REQUIREMENTS

10.1 DECOMMISSIONING PLAN

A decommissioning plan will be prepared as part of the design of ANS. This plan will demonstrate the practicality of decommissioning the facility at the end of its useful lifetime and will identify any unique features of the facility that may be appropriate in aiding its ultimate decommissioning. The decommissioning plan will estimate the categories and volumes of wastes that will be generated during decommissioning, allowing this data to be integrated into waste management planning at ORNL. The decommissioning plan will also identify the levels of radioactivity, either induced or as contamination, that may be left at the ANS site after decommissioning. The plan will show how such residual activity is compatible with waste storage requirements and land use plans at ORR. The document, *Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors*, NUREG/CR-1756,[1] shall be used as a reference in developing this plan.

10.2 DESIGNING FOR DECOMMISSIONING

The design of ANS shall accommodate ultimate decommissioning, or major facility revisions, to the greatest extent practical. Systems and structures shall be designed for decontamination and removal. Design features to be considered include avoiding pockets in corners where frequent decontamination is likely and providing liners or suitable coverings on concrete walls to prevent activity from being absorbed deeply into the concrete and to allow removal during decontamination. Handling systems shall be provided for dismantling systems and structures, or it shall be shown that temporary systems can be provided when required.

Particular attention shall be given to dismantling or altering structures as may be required for major alterations or lifetime extension of the facility. Such considerations include the possibility of removal and replacement of the reactor assembly or the adding of additional guide halls to the facility.

REFERENCES FOR CHAPTER 10

1. G. J. Konzek, J. D. Ludwick, and W. E. Kennedy, Jr., *Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors*, NUREG/CR-1756, Pacific Northwest Lab., March 1982.

11. FUEL REPROCESSING, FABRICATION, AND STORAGE REQUIREMENTS

11. FUEL REPROCESSING, FABRICATION, AND STORAGE REQUIREMENTS

11.1 FUEL REPROCESSING REQUIREMENTS

At the end of a normal operating cycle, an ANS fuel assembly will contain at least 10 kg of uranium with a ^{235}U assay of nearly 80% along with approximately 7 kg fission products. The most effective utilization of resources suggests that these fuel elements be reprocessed to recover the residual uranium and to minimize the waste volume that must ultimately be placed in geologic storage. Currently, within the DOE system, spent research reactor fuels containing aluminum-clad U_3O_8 are shipped to the Savannah River Site for reprocessing, and continuation of this practice is assumed even though the ANS fuel form is aluminum-clad U_3Si_2. DOE has published in the *Federal Register*[1] that U_3Si_2 fuels with ^{235}U assays below 20% will be received and reprocessed. Examination of the supporting analyses[2,3] indicate that the 20% enrichment reference was included for safeguard reasons and not because of any problems associated with reprocessing. Therefore, design of ANS will be consistent with the plan of shipping spent fuel to the Savannah River Site after an initial cooling period. Shipping methods and administrative procedures used at HFIR for shipping spent fuel shall be used to the maximum extent possible.

Spent-fuel storage capacity shall be provided at ANS to accommodate the number of cores associated with normal full-power operation and with normal staging of fuel ahead of shipment. In addition, reserve storage capacity to accommodate uncertainties in fuel handling and shipments is required along with adequate space to afford later expansion of storage capacity should that ever become necessary. ANS shall provide spent-fuel storage for a minimum of 24 months in the reactor building. A goal of 60-month storage in the reactor building and in the reactor support building is imposed to provide additional safety associated with loading the spent fuel into its cask for off-site transport. Spent-fuel storage capacity at ANS is in addition to the complementary storage capacity associated with other reactor-related materials, such as irradiated core pressure boundary tubes and reactor control/shutdown elements.

11.2 FUEL FABRICATION REQUIREMENTS

Information provided in this section will be used for cost estimates during the conceptual design phase of the project. The number of fabricated reactor cores to be provided to support ANS developmental and operational needs is indicated in Table 11.2-1. This table provides the schedule for buildup of these reactor cores from initial development to a fully functioning operating facility. It also presents specific information on the amount of ^{235}U required as a function of this schedule.

The basic assumptions used in determining these data are as follows:

1. **Amount of ^{235}U**
   - Each ANS core will contain 15 kg of ^{235}U, with an additional 2 kg allowance for scrap during the manufacturing process, for a total of 17 kg of ^{235}U required for each core.

- For the first three cores, ^{235}U scrap shall include an increased allowance of 13 kg additional scrap for the first core (total of 15 kg scrap for this core) and 5 kg additional scrap for the second and third cores (total of 7 kg scrap for each of these cores).
- The amount of ^{235}U burn-up for a fully used core shall be about 7 kg.

2. **Durations**
   - The duration of each fuel cycle is 21 days, four of which are used to refuel.
   - The amount of time for storage of a spent core in the spent-fuel storage pool inside containment (the reactor building) shall be at least two years.
   - The amount of lead time needed for an ordered core shall be at least two years prior to its use (order in FY 1998 for use in FY 2000).
   - As presently approved by DOE, power operation is assumed to begin in 2002.
3. **Inventory**
   - Consistent with the operating schedule presently approved by DOE, an inventory buildup of an additional two cores per year (20 instead of the minimum 18) over the first five years of power operation (FY 2002 to FY 2006) shall also be provided.
   - After FY 2006, ten cores (150 kg of ^{235}U) shall be kept in reserve.

11.3 PREVENTION OF UNPLANNED NUCLEAR CRITICALITY

Fissile material shall be controlled in a manner to ensure prevention of an unplanned and uncontrolled criticality accident. The double-contingency principle in the facility manager's policy procedure ESH-22 shall apply to all operations.[4] Guidance in the implementation of ESH-22 is contained in ORNL Health Physics Procedures, RP 2.4.[5] This principle requires that process designs, operations, and material usage incorporate sufficient safety factors so that at least two unlikely, independent, and concurrent changes in physical conditions must occur before an accidental nuclear criticality is possible.

Accidental release of a fuel element during handling or movement shall be considered. In addition, a maximum of 10% of the core-melt prevention goal mentioned in Sect. 4.1.2 can be assigned to criticality events. A fuel-assembly drop-accident shall be included in the assessment of compliance with this goal.

To achieve these objectives, two redundant techniques shall be used for subcriticality control of fuel elements during all phases of new- and spent-fuel handling and storage. As a goal, acceptable subcriticality control shall be achieved by use of any two of the following five techniques:

1. Moving the fuel element from one position to another within a geometry (e.g., room, transport cell, or neutron-poisoned enclosure) so that subcriticality control is achieved by this geometry alone.
2. Securing the element in a transfer container that maintains subcriticality because of its size, geometry, and neutron absorption. Calculations shall account for any anticipated medium surrounding the container.
3. Removing sufficient moderating materials so that the energy spectrum of the neutrons maintains subcriticality.

Table 11.2-1. Reactor-core fabrication requirements planning schedule

Fiscal year	Number of cores ordered	Amount of ^{235}U kg			
		Fresh ^{235}U	Fabricated scrap to be returned	Irradiated ^{235}U returned for reprocessing	^{235}U burned
1992					
1993					
1994	1	30	5		
1995			10		
1996				15	
1997					
1998					
1999					
2000	2	44	14		
2001	12	204	24		
2002	20	340	40		7
2003	20	340	40		42
2004	20	340	40	23	126
2005	20	340	40	138	126
2006	20	340	40	144	126
2007	18	306	36	144	126
2008	18	306	36	144	126
2009	18	306	36	144	126

Note: 2009 and beyond are identical to the data in fiscal year 2007.

4. Securing to the fuel element, by two independent means, a neutron poison that maintains subcriticality through neutron absorption.
5. Securing to the fuel element, by two independent means, a structure that displaces heavy water. The system shall be subcritical because of increased neutron leakage and absorption caused by the structure alone.

The minimum requirement can be achieved by two applications of one technique (for example, poison addition) if sufficient diversity exists in these two applications.

Subcriticality control by each technique is defined as $K_{eff} < 0.95$, after including calculational uncertainties and biases. ANS 57.2-1983 (p. 12) and Appendix IV of IAEA Technical Reports Series No. 189 provide acceptable methods for including uncertainties.

11.4 MOVEMENT AND STORAGE OF NEW AND SPENT FUEL

The movement and storage of fuel elements within the ANS facility shall meet simultaneously the following requirements: subcriticality control; adequate cooling; radiation protection of personnel and sensitive structures and systems; inventory control; and safeguards, security, and sabotage protection. Standards and guides applied to these requirements include the following:

1. 10 CFR 50, Appendix A, General Design Criterion 61, Fuel Storage and Handling and Radioactivity Control; Criterion 62, Prevention of Criticality in Fuel Storage and Handling; and Criterion 63, Monitoring Fuel and Waste Storage.
2. NRC Standard Review Plan (SRP), Sect. 9.1.1, New Fuel Storage; Sect. 9.1.2, Spent Fuel Storage; Sect. 9.1.3, Spent Fuel Pool Cooling and Cleanup System; and, as appropriate, Sects. 9.1.4, 9.1.5, 15.7.4, and 15.7.5.3.[6]
3. NRC Regulatory Guides referenced in the SRP, including Regulatory Guide 1.13, *Design Objectives for Light-Water Reactor Spent Fuel Storage Facilities and Nuclear Power Stations.*[7]
4. DOE Order 6430.1A, particularly Sect. 1300-4, Nuclear Criticality Safety.[8]
5. American Nuclear Society Standards ANSI 57.1, *Design Requirements for Light Water Reactor Fuel Handling Systems;*[9] ANSI 57.2, *Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations;*[10] and ANSI 57.3, *Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants.*[11]
6. Facility manager's procedures governing nuclear criticality. See, for example, ORNL Health Physics Division Procedure RP-2.4, Fissile Material Safety, dated February 21, 1990.[5]

Additional useful information is available in International Atomic Energy Agency publications that deal with fuel transport and storage within nuclear facilities.[12,13]

Additional generic requirements also apply to fuel movement and storage. Inventory control shall, as a minimum, include assembly accountability of fresh-fuel assemblies entering the site and spent-fuel assemblies leaving the site. Safeguards and security requirements are stated in Sect. 6.2.10 of this PDR. Radiation protection of personnel shall meet the ALARA requirements in Sect. 4.8.1.3. Subcriticality requirements for ANS are presented in Sect. 11.3. Fuel assembly cooling and subcriticality control shall meet the requirements for inherent and passive safety, as stated in Sect. 4.1.2. This requirement is amplified by the SDC for passive safety found in Appendix B. Additional plant-level requirements specific to fuel movement and storage are:

1. Within containment, storage of full-burnup fuel shall be provided for a minimum of 24 months of normal operation.
2. Additional spent-fuel storage shall be provided in the reactor containment building or the reactor service building for 36 months *[HOLD]*[11-1] of normal operation.
3. Consistent with Appendix K of 10 CFR 50, heat from radioactive decay of fission products in the spent-fuel elements shall be 1.2 times the value associated with full-power operation for 17 days (see Table 11.5-1 for these values).
4. The design shall meet appropriate design criteria from Appendix B, including the spent-fuel facilities criterion (Table B.5, GDC 61); the nuclear criticality criterion (Table B.5, GDC 62; the single failure criterion (Table B.5); and the common mode failure criterion (Table B.6).
5. The design shall include a criticality alarm system that satisfies the requirements of DOE Order 5480.5, Subsect. 11c(2)(9) and Subsect. 11c(3)(h), respectively.[14]

The design shall accommodate the requirements listed above for fuel burnups from zero to the design value. This requirement means that new and spent fuel could follow the same path (in reverse direction) through the facility; however, a unique path for the loading of fresh-fuel assemblies is not excluded.

Time for radioactive decay of spent fuel shall be used to provide additional safety margins or to meet single-failure criteria associated with cooling of the fuel during transport within the facility. Examples of the application of this requirement include:

1. movement of full-burnup fuel within the core pressure boundary tube only after the time when natural circulation of the primary coolant is sufficient to meet cooling requirements,
2. movement of spent fuel in a container only after the time when passive cooling of the container by its surroundings is sufficient,
3. movement of spent fuel outside containment only after sufficient cooling of the bare fuel assembly is achieved by natural circulation of air alone, and
4. cooling of full-burnup spent fuel for a minimum of 24 months *[HOLD]*[11-2] before shipment off-site.

11.5 DECAY HEAT VALUES FOR SPENT FUEL ASSEMBLIES

Table 11.5-1 presents decay heat, power in megawatts from the ANS core versus time after completion of a full 17-d cycle at 330 MW (fission power). Assumptions used and the methodology applied to generate this table are from ANSI/ANS-5.1-1979.[15] Also shown are these power values multiplied by 1.2 as specified in Sect. 11.4 of this PDR. These larger values are to be used for performing calculations. The accuracy of this table is estimated as no more than three significant figures.

Table 11.5-1. Decay heat values

Time after shutdown (h)	Power fraction (%)	Power (MW)	Power • 1.2 (MW)
0.00	6.4703E-02	21.3522	25.6226
1.00	1.0860E-02	3.5840	4.3008
2.00	8.2765E-03	2.7312	3.2775
3.00	7.0104E-03	2.3134	2.7761
4.00	6.2078E-03	2.0486	2.4583
5.00	5.6414E-03	1.8616	2.2340
6.00	5.2128E-03	1.7202	2.0642
7.00	4.8714E-03	1.6075	1.9290
8.00	4.5889E-03	1.5143	1.8172
9.00	4.3488E-03	1.4351	1.7221
10.00	4.1408E-03	1.3664	1.6397
11.00	3.9582E-03	1.3062	1.5674
12.00	3.7962E-03	1.2527	1.5033
13.00	3.6514E-03	1.2049	1.4459
14.00	3.5210E-03	1.1619	1.3943
15.00	3.4030E-03	1.1229	1.3475
16.00	3.2955E-03	1.0875	1.3050
17.00	3.1973E-03	1.0551	1.2661
18.00	3.1071E-03	1.0253	1.2304
19.00	3.0239E-03	0.9979	1.1974
20.00	2.9469E-03	0.9724	1.1669
21.00	2.8753E-03	0.9488	1.1386
22.00	2.8086E-03	0.9268	1.1122
23.00	2.7463E-03	0.9062	1.0875
24.00	2.6878E-03	0.8869	1.0643
25.00	2.6329E-03	0.8688	1.0426
26.00	2.5811E-03	0.8517	1.0221

Table 11.5-1 (continued)

Time after shutdown (h)	Power fraction (%)	Power (MW)	Power • 1.2 (MW)
27.00	2.5322E-03	0.8356	1.0027
28.00	2.4860E-03	0.8203	0.9844
29.00	2.4422E-03	0.8059	0.9671
30.00	2.4005E-03	0.7921	0.9506
31.00	2.3610E-03	0.7791	0.9349
32.00	2.3233E-03	0.7666	0.9200
33.00	2.2873E-03	0.7548	0.9057
34.00	2.2529E-03	0.7434	0.8921
35.00	2.2201E-03	0.7326	0.8791
36.00	2.1886E-03	0.7222	0.8667
37.00	2.1585E-03	0.7123	0.8547
38.00	2.1296E-03	0.7027	0.8433
39.00	2.1018E-03	0.6936	0.8323
40.00	2.0751E-03	0.6848	0.8217
41.00	2.0494E-03	0.6763	0.8115
42.00	2.0247E-03	0.6681	0.8018
43.00	2.0009E-03	0.6603	0.7923
44.00	1.9779E-03	0.6527	0.7832
45.00	1.9557E-03	0.6453	0.7744
46.00	1.9343E-03	0.6383	0.7659
47.00	1.9136E-03	0.6314	0.7577
48.00	1.8935E-03	0.6248	0.7498
49.00	1.8741E-03	0.6184	0.7421
50.00	1.8553E-03	0.6122	0.7347
51.00	1.8371E-03	0.6062	0.7275
52.00	1.8195E-03	0.6004	0.7205
53.00	1.8024E-03	0.5947	0.7137
54.00	1.7587E-03	0.5893	0.7071

Table 11.5-1 (continued)

Time after shutdown (h)	Power fraction (%)	Power (MW)	Power • 1.2 (MW)
55.00	1.7696E-03	0.5839	0.7007
56.00	1.7539E-03	0.5787	0.6945
57.00	1.7386E-03	0.5737	0.6884
58.00	1.7237E-03	0.5688	0.6826
59.00	1.7029E-03	0.5640	0.6768
60.00	1.6951E-03	0.5594	0.6712
61.00	1.6814E-03	0.5548	0.6658
62.00	1.6680E-03	0.5504	0.6605
63.00	1.6549E-03	0.5461	0.6553
64.00	1.6422E-03	0.5419	0.6503
65.00	1.6297E-03	0.5378	0.6453
66.00	1.6175E-03	0.5337	0.6405
67.00	1.6056E-03	0.5298	0.6358
68.00	1.5940E-03	0.5260	0.6312
69.00	1.5826E-03	0.5222	0.6267
70.00	1.5715E-03	0.5186	0.6223
71.00	1.5606E-03	0.5150	0.6180
72.00	1.5499E-03	0.5114	0.6137
73.00	1.5395E-03	0.5080	0.6096
74.00	1.5292E-03	0.5046	0.6055
75.00	1.5192E-03	0.5013	0.6016
76.00	1.5093E-03	0.4980	0.5977
77.00	1.4997E-03	0.4949	0.5938
78.00	1.4902E-03	0.4917	0.5901
79.00	1.4809E-03	0.4887	0.5864
80.00	1.4717E-03	0.4856	0.5828
81.00	1.4628E-03	0.4827	0.5792
82.00	1.4540E-03	0.4798	0.5757

Table 11.5-1 (continued)

Time after shutdown (h)	Power fraction (%)	Power (MW)	Power • 1.2 (MW)
83.00	1.4453E-03	0.4769	0.5723
84.00	1.4368E-03	0.4741	0.5689
85.00	1.4284E-03	0.4713	0.5656
86.00	1.4201E-03	0.4686	0.5623
87.00	1.4120E-03	0.4659	0.5591
88.00	1.4040E-03	0.4633	0.5560
89.00	1.3962E-03	0.4607	0.5529
90.00	1.3884E-03	0.4582	0.5498
91.00	1.3808E-03	0.4556	0.5468
92.00	1.3733E-03	0.4532	0.5438
93.00	1.3659E-03	0.4507	0.5409
94.00	1.3586E-03	0.4483	0.5380
95.00	1.3514E-03	0.4459	0.5351
96.00	1.3444E-03	0.4436	0.5323
97.00	1.3374E-03	0.4413	0.5296
98.00	1.3305E-03	0.4390	0.5268
99.00	1.3237E-03	0.4368	0.5241

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Appendix A:

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Appendix B:

ADDITIONAL REQUIREMENTS FROM AUTHORITATIVE BODIES

Appendix B

ADDITIONAL REQUIREMENTS FROM AUTHORITATIVE BODIES

ANS will be a DOE facility. Therefore, it will be designed and operated in compliance with DOE requirements, including DOE and Oak Ridge (OR) Orders. DOE Orders incorporate, by reference, applicable requirements of the CFR, NRC regulatory guides, and national codes and standards. Tables B.1 and B.2 list DOE and OR orders applicable to the design, construction, and operation of ANS. Table B.3 lists the standards and codes that are either required, recommended, or published by the DOE. The last update of these three tables was based on the May 1990 issue of DOE Order 0000.2C.

DOE 5480.5 and DOE 5480.6 require that nuclear facilities (reactors and other designated facilities) are sited, designed, constructed, modified, operated, maintained, and decommissioned in accordance with generally uniform standards, guides, and codes that are consistent with those applied to comparable licensed nuclear facilities. DOE 5480.4 references Title 10 CFR 50, Licensing of Production and Utilization Facilities, and appendices, including regulatory guides. DOE 5480.6 requires that the GDC specified in the CFR, Title 10, Part 50, Appendix A, and that 10 CFR 100, Reactor Site Criteria, be applied to the construction of all new DOE-owned reactor facilities. In addition, DOE 5700.6B, Quality Assurance, endorses 10 CFR 50, Appendix B.

Tables B.4 and B.5 list the compliance commitment level for 10 CFR 50 and its appendices (including the GDC). Because of the design features of ANS, Supplemental Design Criteria (SDCs) were added (Table B.6) to provide a complete list of general design criteria in order to further ensure that the plant is designed, constructed, and operated safely and that the public health and safety are protected.

In addition to those NRC requirements specifically referenced in the DOE (and OR) Orders, ANS shall be designed as if it had to undergo the NRC-licensing process for a test facility at 330 MW (fission power). Tables B.7 and B.8 provide compliance commitment levels appropriate to ANS for all NRC regulatory guides, SRP sections, and Branch Technical Positions. Table B.9 provides a listing of other NRC regulations and publications that are of potential interest to ANS.

Finally, Table B.10 provides a listing of Martin Marietta Energy Systems policy procedures that are applicable to the design, construction, and operation of ANS. Table B.11 lists the standards and codes that are either required or recommended by these policy procedures. Table B.12 is an index of terms for Appendix B to facilitate use of the table.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
A review of the DOE Orders is given below. These orders are applicable to the design, construction, maintenance, operation, and decommissioning of ANS. Of primary importance, DOE 5480.6 provides the primary guidance on safety requirements for DOE-owned reactors, including construction of new DOE-owned facilities. DOE 6430.1A provides mandatory requirements for the design of DOE facilities and applies to any building acquisition or new facility that is classified as real property. As noted in the order, the criteria apply to non-reactor facilities. Reactors and their safety systems shall be sited and designed in accordance with DOE 5480.6. Other DOE Orders outline the Health, Safety, and Environment programs that must be met in the design of the ANS and provide a range of guidance applicable to the ANS project. This table provides guidance to those orders listed in Table 4.3.1, as well as guidance to orders that are of interest in the construction and operations phases of the project. In addition, several administrative orders are listed for the benefit (e.g., DOE 5500.7A) of designers and managers.		
DOE 0000.2C (05/04/90)	Directives Checklist and Cross-Reference Index	DOE 0000.2 provides a cross-reference subject index of DOE and HQ orders to assist users in locating directives alphabetically by subject. The order provides four attachments: orders, notices, cross-reference index table of contents, and cross-reference index.
DOE 1270.2A	Safeguards Agreement with the International Atomic Energy Agency	DOE 1270.2A prescribes policies and responsibilities of DOE for compliance with the agreement, including the associated protocol, between the government of the U.S. and the IAEA for the application of safeguards in the nuclear materials transaction, material balance, and inventory information shall be transmitted to the IAEA pursuant to the agreement or protocol and shall include national security information on a controlled basis, as necessary to comply with the agreement or protocol.
DOE 1324.2	Records Disposition	DOE 1324.2A assigns responsibilities and authorities and prescribes policies, procedures, standards, and guidelines for the orderly disposition of records. Objectives of this order are to establish a system for inventorying, maintaining, and obtaining disposition authority for all DOE records; identify and plan for the preservation of records documenting the organization, functions, policies, decisions, procedures, and essential transactions; develop a planned approach to retire or those records no longer required for current activities; transfer to lower-cost storage and actively pursue the disposition of records on a regular basis, including transfer to lower-cost storage, timely destruction of records in accordance with authorized dispositions, and the application of microfilming in the reduction of records holdings.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 1540.2	Hazardous Material Packaging for Transport-Administrative Procedures	DOE 1540.2 establishes administrative procedures for the certification and use of radioactive and other hazardous materials packaging by the DOE. DOE 1540.2 also standardizes the approval procedures to ensure that DOE packaging designs and transportation operations provide for public health and safety in accordance with regulations of DOT and in accordance with standards that are equivalent to the standards prescribed by NRC.
DOE 1540.3	Base Technology for Radioactive Material Transportation Packaging Systems	DOE 1540.3 establishes DOE policies and responsibilities for coordinating and planning base technology for radioactive material transportation packaging systems.
DOE 3791.2A	Federal Employee Motor Vehicle Safety Program	DOE 3791.2A establishes minimum licensing requirements for operating government-owned or leased motor vehicles and powered industrial equipment; establishes requirements for wearing of seat belts; and establishes operator training and safe driver award requirements.
DOE 5000.3	Unusual Occurrence Reporting System	DOE 5000.3 sets forth policy, assigns responsibility, and provides criteria and instructions for reporting unusual occurrences that have programmatic significance at DOE operations, analyzing the information reported, and disseminating the analysis results. It is DOE policy that unusual occurrences be promptly reported and documented; unusual or unplanned events be formally reported as unusual occurrences reports of unusual occurrences be reviewed, corrective action be taken to prevent or minimize the probability of recurrence, and information of generic significance be disseminated to other interested DOE organizational elements.
DOE 5400.5	Radiation Protection of the Public and the Environment	*[TBD]*[B-1]
DOE 5440.1C	Implementation of the National Environmental Policy Act	*[TBD]*[B-2]
4330.2C	In-house Energy Management	*[TBD]*[B-3]
4330.4A	Maintenance Management	*[TBD]*[B-4]

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5480.1B	Environmental Protection, Safety, and Health Protection Program for DOE Operations	DOE 5480.1B establishes the Environment, Safety, and Health (ES&H) Program for DOE operations. It is DOE policy to ensure the protection of the environment and the health and safety of the public; ensure safe and healthful workplaces and conditions of employment; protect government property against accidental loss and damage; ensure compliance with applicable statutory requirements affecting Federal facilities and operations: reduce where possible, identified environment, safety, and health risks, even though not mandated by specific requirements; and ensure that QA is pursued.
DOE 5480.3	Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances and Hazardous Wastes	DOE 5480.3 establishes the requirements for the packaging and transportation of hazardous materials, hazardous substances, and hazardous wastes. This order provides general standards for all packaging; structural standards for Type B packaging; criticality standards for fissile material packages; standards for normal and accident conditions for a single package; and quality assurance requirements for fabrication, testing, and assembly of shipping containers.
DOE 5480.4	Environmental Protection, Safety and Health Protection Standards	DOE 5480.4 specifies and provides requirements for the application of the mandatory ES&H standards, provides a listing of reference ES&H standards, and identifies the sources of the mandatory and reference ES&H standards. This order shall be followed during facility design, construction, operation, modification, and decommissioning.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5480.5	Safety of Nuclear Facilities	DOE 5480.5 requires that nuclear facilities are sited, designed, constructed, modified, operated, maintained, and decommissioned in accordance with generally uniform standards, guides, and codes that are consistent with those applied to comparable licensed nuclear facilities. This order requires a safety analysis and review; administrative and procedural controls; documented training program; QA program in accordance with DOE 5700.6B; application of ES&H codes, standards, and guides; reporting of occurrences; review of safety analyses design criteria; emergency plans; a nuclear criticality safety program; an independent safety review and appraisal system; and the safe storage of unirradiated fissionable material.
DOE 5480.6	Safety of Department of Energy-Owned Reactors	DOE 5480.6 establishes requirements for a reactor safety program to ensure that the safety of each reactor is properly analyzed, evaluated, documented, and approved by DOE and that reactors are sited, designed, constructed, modified, operated, maintained, and decommissioned in a manner that gives adequate protection for health and safety and that will be in accordance with uniform standards, guides, and codes that are consistent with those applied to comparable licensed reactors. This order endorses 10 CFR 100 and DOE 6430.1B for siting; 10 CFR 50, Appendix A (NRC General Design Criteria); a Safety Analysis Report to follow NRC guidelines (i.e., RG 1.70); and technical specifications to meet 10 CFR 50.36. This order also provides requirements on personnel training and qualification programs, QA, independent review and appraisal system, fissile material storage, standards, standby and decommissioning, reporting of occurrences, emergency planning, and record keeping. Only those sections of the order that pertain to Category A reactors is applicable to ANS (i.e., the section of Category B reactors is not applicable).

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5480.7	Fire Protection	DOE 5480.7 establishes requirements for an "improved risk" level of fire protection. A higher standard of protection may be justified in certain instances. The requirements in this order are to ensure that no threats to the public health or welfare will result from fire, that there are no undue hazards to employees from fire, that vital DOE programs will not suffer unacceptable delays as a result of fire, and that property damage will be held to manageable levels.
DOE 5480.8	Contractor Occupational Medical Program	DOE 5480.8 establishes minimum occupational medical program requirements. This order sets forth requirements for contractor occupational medical program, diagnosis and treatment of injury or disease, employee health maintenance and preventive medical activities, medical records, emergency and disaster preparedness; staffing for medical programs, and medical facilities and equipment.
DOE 5480.10	Contractor Industrial Hygiene Program	DOE 5480.10 establishes the requirements and guidelines for maintaining an effective industrial hygiene program to preserve employee health and well-being. This objective is to be accomplished by identifying, evaluating, and controlling environmental factors and stresses found in the work place. These environmental factors and stresses include: chemical (e.g., liquid, particulate, vapor, and gas); physical (e.g., electromagnetic radiation, noise, vibration, and magnetic fields); biological (e.g., agents of infectious diseases); and ergonomic (e.g., body position in relation to task, repetitive motion, and mental or physical fatigue). The industrial hygiene program must identify health hazards; evaluate the hazards; implement control measures; and, for OSHA-regulated carcinogens, implement a program that conforms to the requirements of applicable OSHA standards (refer to DOE 5483.1A).
DOE 5480.11	Radiation Protection for Occupational Workers	*[TBD]*[B-5]
DOE 5480.18	Accreditation of Performance-Based Training for Category A Reactors and Nuclear Facilities	

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5480.19	Conduct of Operations Requirements for DOE Facilities	*[TBD]*[B-4]
DOE 5480.20	Personnel Selection – Qualification, Training, and Staffing Requirements at DOE Reactors and Non-reactor Nuclear Facilities	*[TBD]*[B-7]
DOE 5481.1B	Safety Analysis and Review System	DOE 5481.1B establishes requirements for the preparation and review of safety analyses (including identification of hazards, their elimination or control, assessment of the risk, and documented management authorization of the operation). Detailed requirements for the review and authorization processes are addressed in DOE 5480.5 and DOE 5480.6. The safety analysis shall be initiated during the earliest phases; shall identify and demonstrate conformance with applicable guides, codes, and standards; shall demonstrate that risks to the public and employees are limited; and shall describe the features of the actual design and conduct of operation. The review of the safety analysis shall evaluate the adequacy of the preventative or mitigative design features, review the line organization, and be sufficiently documented to allow independent evaluation. All pertinent details of the analysis, review, and authorization shall be traceable from the initial identification of a hazard to its elimination or application of controls necessary to reduce the risk.
DOE 5482.1B	Environmental, Safety, and Health Appraisal Program	DOE 5482.1B establishes the ES&H Appraisal Program. The quality, frequency, and depth of appraisals shall be commensurate with the hazard attendant with the respective operating activities; comparable to similar regulatory programs; and consistent with DOE policy of protection of personnel, property, and the environment. The appraisal is performed to determine that ES&H policies and requirements are appropriately interpreted and implemented; to evaluate the effectiveness of ES&H policies, requirements, and standards and their implementation; to provide management with objective, timely, and reliable information on ES&H performance and recommendations for improvement; and to determine the adequacy of requirements for achieving DOE policy and Federal statutory ES&H requirements.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned Contractor-Operated Programs	DOE 5483.1A establishes requirements and procedures to ensure that occupational safety and health standards prescribed pursuant to the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974; and the Department of Energy Organization Act of 1977 provide occupational safety and health protection that is consistent with the protection afforded under the Occupational Safety and Health Act of 1970 (OSHA), Public Law 91-596. The provisions of this order apply only with respect to radiation hazards in the work place because the provisions of OSHA do not apply to DOE contractor employees working in government-owned contractor-operated facilities since DOE exercises statutory authority to prescribe and enforce safety and health standards at these facilities. This order enforces 29 CFR 1910, 29 CFR 1926, 29 CFR 1915, 29 CFR 1918, and 29 CFR 1928.
DOE 5484.1	Environmental Protection, Safety and Health Protection Information Reporting Requirements	DOE 5484.1 establishes the requirements and procedures for the reporting of information having ES&H significance. This order requires that all occurrences be investigated; that reports be submitted; that management take responsive action; that liquid and airborne effluents and environmental conditions be monitored, evaluated; that discharges be reported to assess the levels of radioactive and nonradioactive pollutants and their impact; and that there be consistency in the treatment of such occurrences.
DOE 5500.1A	Emergency Management System	DOE 5500.1A establishes the overall policy and requirements for an Emergency Management System that will provide for development, coordination, and direction of DOE planning, preparedness, and readiness assurance for response to operational, energy, and continuity of government emergencies. This order also requires compliance with the Federal Radiological Emergency Response Plan.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5500.2 (Canceled by 5500.1A, but Attachments 2 and 3 are still valid)	Emergency Planning, Preparedness, and Response for Operations	Attachment 2 to DOE 5500.2 requires emergency planning, preparedness, and response programs documents. An overall emergency organizational structure shall be established and included in the written emergency plan. Credible emergencies, as well as readiness actions, medical requirements, damage control, reentry planning, and damage assessment, will be identified and described. Emergency facility requirements and communications are *[TBD]*[B-2] based on this order. Attachment 3 to DOE 5500.2 establishes requirements for emergencies and notification systems.
DOE 5500.3	Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for DOE Operations	DOE 5500.3 establishes requirements for the development of site-specific emergency plans and procedures for radiological emergencies. It also requires that comprehensive emergency actions be planned, coordinated, and implemented to respond effectively to the on-site and off-site consequences of a radiological emergency; and it provides for appropriate coordination between DOE and off-site officials to ensure the protection of on-site personnel, public health and safety, and the environment. This order establishes four emergency response levels (unusual event, alert, site emergency, and general emergency) and two radiological emergency planning zones (plume exposure pathway and ingestion exposure pathway).
DOE 5500.4	Public Affairs Policy and Planning Requirements for Emergencies	DOE 5500.4 establishes requirements for emergency situations. This order covers emergency situations involving any actual, potential, or perceived threat to on-site personnel, public health and safety, and/or to the environment. These emergencies can range from natural disasters (tornados, floods, earthquakes, fires) to man-caused incidents (explosions, fires, releases of toxic or radioactive materials).
DOE 5500.7A	Vital Records Protection Program	DOE 5500.7A requires that vital records be protected. Records essential to the preservation of the legal rights and interests of individual citizens and their government must be protected, but storage points do not have to be at or in the vicinity of Emergency Operation Centers. Examples of rights and interest records are legal rights records, fiscal records, and valuable research records.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 5632.2A	Physical Protection of Special Nuclear Material and Vital Equipment	*[TBD]*[B-9]
DOE 5632.6	Physical Protection of DOE Property and Unclassified Facilities	*[TBD]*[B-10]
DE 5633.3	Control and Accountability of Nuclear Materials	*[TBD]*[B-11]
DOE 5700.6B	Quality Assurance	DOE 5700.6B sets forth requirements and assigns responsibilities for establishing, implementing, and maintaining plans and actions to ensure quality achievement. The QA program shall include, as a minimum the assignment of organization responsibility and authority for activities affecting quality and activities that ensure quality achievement, definition of quality and QA objectives and requirements, implementation of procedures, independent verification, and early detection and correction of deficiencies. The QA plan shall also be defined for each program, have resources provided, be implemented using written procedures, be defined in terms of measurable characteristics, confirm quality attainment, perform audits, and identify and correct deficiencies. This order states that ANSI/ASME NQA-1 is the preferred standard for QA in the nuclear area.
DOE 5820.2A	Radioactive Waste Management	DOE 5820.2 establishes policies and guidelines for managing radioactive waste, waste by-products, and radioactively contaminated surplus facilities. This order covers the management of high-level waste; transuranic waste, low-level waste, wastes contaminated with naturally occurring radionuclides and the decontamination and decommissioning of surplus facilities.

Table B.1. Department of Energy (DOE) Orders Applicable to ANS

DOE ORDER NO.	TITLE	GUIDANCE
DOE 6430.1A	General Design Criteria	DOE 6430.1A provides GDC and establishes the responsibilities and authorities for the development and maintenance of these criteria. It is DOE policy that professional architectural and engineering principles and practices be applied to the planning, design, construction, alterations, and/or acquisition of facilities; that all facilities will comply with the Federal and DOE regulations for energy conservation and use of renewable energy; that all applicable laws, regulations, and executive orders, whether Federal, state, or local, will be satisfied in the planning, design, and construction of the facility; and that all facilities are designed and constructed to be reasonable and adequate for their intended purpose and consistent with health, safety, and environmental protection requirements.

Table B.2. Oak Ridge Field Office Orders Applicable to ANS

ORO ORDER NO.	TITLE
The Oak Ridge Field Office Orders listed below supplement the DOE Orders listed in Table B.1.	
OR 1100.1	Organization and Functions - Oak Ridge Operations
OR 1270.2A	Safeguards Agreement with the International Atomic Energy Agency
OR 1540.1A	Materials Transportation and Traffic Management
OR 5000.3	Unusual Occurrence Reporting System
OR 5480.1A	Environmental Protection, Safety, and Health Protection Program for Oak Ridge Operations
OR 5480.4	Environmental Protection, Safety and Health Protection Standards
OR 5480.5	Safety of Nuclear Facilities
OR N 5480.6	Safety of Department of Energy-Owned Reactors
OR N 5480.8	Contractor Occupational Medical Program
OR N 5481.1B	Safety Analysis and Review System
OR N 5482.1B	Environmental Protection, Safety, and Health Protection Appraisal Program

Table B.2. Oak Ridge Field Office Orders Applicable to ANS

ORO ORDER NO.	TITLE
OR 5483.1	Occupational Safety and Health Program for Government-Owned Contractor-Operated Programs
OR 5484.1	Environmental Protection, Safety and Health Protection Information Reporting Requirements
OR N 5500.2A	Emergency Notification
OR 5500.2	Emergency Planning, Preparedness, and Response for Operations
OR 5500.3	Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for DOE Operations
OR 5500.4	Public Affairs Policy and Planning Requirements for Emergencies
OR N 5700.6B	Quality Assurance
OR 5820.2	Radioactive Waste Management

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE

A listing of standards and codes either required or recommended by DOE and OR orders, or published by DOE ares given below. The Justification and Guidance column indicates which order requires or recommends that particular standard or code. REQUIRED means that the order specifies that this standard "shall be" used in the design. RECOMMENDED means that the order states that this standard "shall be considered" in the design. Each standard in this table must be reviewed for the specific compliance level for ANS. Standards apply to varying degrees based on specific ANS design features and the particular statements in the orders. If the standard is required or recommended by an MMES Policy and Procedure, this is also noted. If an MMES Policy and Procedure is listed here, it is not relisted in Table B.11.

Some of the codes or standards listed below may contain statements that make it appear to be not-applicable to the design of ANS. This situation can occur because the code or standard may, in some cases, provide an example. Therefore, statements like "such as steam generators" simply provides guidance and is not part of the requirement it self.

The index and glossary in DOE 6430.1A will be useful to SDD task leaders for determining applicability on the SDD level.

Air Conditioning and Refrigeration Institute (ARI)

ARI 410	Forced Circulation Air Cooling and Air-Heating Coils	This standard is REQUIRED by DOE 6430.1A.
ARI 430	Central Station Air Handling Units	This standard is REQUIRED by DOE 6430.1A.
ARI 450	Water-Cooled Refrigerant Condensers, Remote Type	This standard is REQUIRED by DOE 6430.1A.
ARI 460	Remote Mechanical-Draft Air-Cooled Refrigerant Compressors	This standard is REQUIRED by DOE 6430.1A.
ARI 520	Positive Displacement Refrigerant Compressors and Condensing Units	This standard is REQUIRED by DOE 6430.1A.
ARI 550	Centrifugal Water Chilling Packages	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ARI 590	Reciprocating Water Chilling Packages	This standard is REQUIRED by DOE 6430.1A.
ARI 850	Commercial and Industrial Air Filter Equipment	This standard is REQUIRED by DOE 6430.1A.
ARI 1010	Drinking Fountains and Self-Contained Mechanically Refrigerated Drinking Water Coolers	This standard is REQUIRED by DOE 6430.1A.
Air Force (USAF)		
AFM 88-29	Engineering Weather Data	This standard is REQUIRED by DOE 6430.1A.
Air Movement and Control Association (AMCA)		
Publication 99	Standards Handbook	This standard is REQUIRED by DOE 6430.1A.
Publication 201	Fans and Systems	This standard is REQUIRED by DOE 6430.1A.
Publication 261	Directory of Products Licensed to Bear the AMCA Certified Rating Seal	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Standard 210	Laboratory Methods for Testing Fans for Ratings	This standard is REQUIRED by DOE 6430.1A.
Aluminum Association (AA)		
AA	Aluminum Finishes for Architecture	This standard is REQUIRED by DOE 6430.1A.
AA	Finishes for Aluminum in Building	This standard is REQUIRED by DOE 6430.1A.
American Concrete Institute (ACI)		
-	Guide to the Use of Waterproofing, Dampproofing, Protective, and Decorative Barrier Systems for Concrete	This standard is REQUIRED by DOE 6430.1A.
ACI SP-4	Formwork for Concrete	This standard is REQUIRED by DOE 6430.1A.
ACI SP-66	Detailing Manual	This standard is REQUIRED by DOE 6430.1A.
ACI 207.4R-80	Cooling and Insulating Systems for Mass Concrete	This standard is REQUIRED by DOE 6430.1A.
ACI 211.1	Standard Practice for Selecting Proportions for Normal, Heavyweight, and Mass Concrete	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ACI 211.2	Standard Practice for Selecting Proportions for Structural Lightweight Concrete	This standard is REQUIRED by DOE 6430.1A.
ACI 304-73 (R1983) (ANSI A186.1)	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete	This standard is REQUIRED by DOE 6430.1A.
ACI 305R-77 (R1982) (ANSI A170.1)	Recommended Practice for Hot Weather Concreting	This standard is REQUIRED by DOE 6430.1A.
ACI 306R-78 (ANSI A144.1)	Recommended Practice for Cold Weather Concreting	This standard is REQUIRED by DOE 6430.1A.
ACI-318-77 (ACI 318-83 [ANSI A89.1]) (COMMENTARY IS ACI 318R-1983)	Building Code Requirements for Reinforced Concrete	This standard is REQUIRED by DOE 6430.1A and supported by RG 1.142 and RG 1.143.
ACI 336.2R	Suggested Design Procedures for Combined Footing and Mats	This standard is REQUIRED by DOE 6430.1A.
ACI 336.3R	Suggested Design and Construction Procedures for Pier Foundations	This standard is REQUIRED by DOE 6430.1A.
ACI 347-78 (ANSI A145.1)	Recommended Practice for Concrete Formwork	This standard is REQUIRED by DOE 6430.1A.
ACI 349-80 (COMMENTARY IS IN ACI 349R-80)	Code Requirements for Nuclear Safety Related Concrete Structures	This standard is REQUIRED by DOE 6430.1A and supported by RG 1.142, SRP 3.8.3, SRP 3.8.4, and SRP 3.8.5.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ACI 350R	Concrete Sanitary Engineering Structures	This standard is REQUIRED by DOE 6430.1A.
ACI 352R	Recommendations for Design of Beam-Column Joints in Monolithic Reinforced Concrete Structures	This standard is REQUIRED by DOE 6430.1A.
ACI 503.4	Standard Specifications for Repairing Concrete with Epoxy Mortars	This standard is REQUIRED by DOE 6430.1A.
ACI 531-79 (R1983)	Building Code Requirements for Concrete Masonry Structures	This standard is REQUIRED by DOE 6430.1A and supported by SRP 3.8.4.
ACI 531.1	Specifications for Concrete Masonry Construction	This standard is REQUIRED by DOE 6430.1A.
ACI 543R	Recommendation for Design, Manufacture and Installation of Concrete Piles	This standard is REQUIRED by DOE 6430.1A.
<u>American Conference of Governmental Industrial Hygienists (ACGIH)</u>		
ACGIH	Industrial Ventilation: A Manual of Recommended Practice, Seventeenth Edition	This standard is REFERENCED by DOE 5480.4 and DOE 6430.1A.
ACGIH	TLVs Threshold Limit Values for Chemical Substances and Physical Agents in the Workroom Environment with Intended Changes	This standard is REQUIRED by DOE 5480.4, MMES ESH-7, and MMES ESH-16.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
American Congress of Surveying and Mapping (ACSM)		
ACSM	ASCM Horizontal Control as Applied to Local Surveying Needs	This standard is REQUIRED by DOE 6430.1A.
American Industrial Hygiene Association (AIHA)		
AIHA	Hygiene Guide Series, Volumes I and II	This standard is REFERENCED by DOE 5480.4.
AIHA	American Industrial Hygiene Association Workplace Environmental Exposure Level Guides	This standard is REFERENCED by DOE 5480.4.
AIHA	Heating and Cooling for Man in Industry	This standard is REFERENCED by DOE 5480.4.
AIHA	Industrial Noise Manual	This standard is REFERENCED by DOE 5480.4.
AIHA	Standards, Interpretation, and Audit Criteria for Performance of Occupational Health Programs, Occupational Health and Safety Programs, Accreditation Commission	This standard is REFERENCED by DOE 5480.4.
American Institute of Steel Construction (AISC)		
AISC M011	Manual of Steel Construction-8th Edition	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
AISC N690-1980 (R1984) (ANSI)	Nuclear Facilities: Steel Safety-Related Structures for Design, Fabrication and Erection	This standard is REQUIRED by DOE 6430.1A.
AISC S326	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (included in AISC Manual of Steel Construction)	This standard is REQUIRED by DOE 6430.1A.
<u>American Iron and Steel Institute (AISI)</u>		
AISI	Manual for Structural Applications of Steel Cables for Buildings	This standard is REQUIRED by DOE 6430.1A.
AISI	Specifications for the Design of Light Gauge Cold-Formed Steel Structural Members	This standard is REQUIRED by DOE 6430.1A.
<u>American Nuclear Society (ANS)</u>		
ANS 2.2-1978 (R1988) (ANSI N18.5)	Earthquake Instrumentation Criteria for Nuclear Power Plants (Revision 1 of N18.5-1974)	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.12.
ANS 2.17-1980	Evaluation of Radionuclide Transport in Ground Water for Nuclear Power Sites	This standard is REFERENCED by DOE 5480.4.
ANS 3.1-1981 (R1987) (ANSI N18.1)	Qualification and Training of Personnel for Nuclear Power Plants	This standard is REQUIRED by DOE 5480.6 and OR 5480.6. The standard is also supported by RG 1.8, SRP 12.5, SRP 13.2.1, and SRP 13.2.2.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANS 3.4-1988 (ANSI N546)	Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants	This standard is REQUIRED by DOE 5480.6 and supported by RG 1.134. ANS 3.4 supersedes ANSI N546-1976.
ANS 3.5-1979 (R1985) (ANSI N675)	Nuclear Power Plant Simulators for Use in Operator Training	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.149.
ANS 3.7.2-1979	Emergency Control Centers for Nuclear Power Plants	This standard is REFERENCED by DOE 5480.4.
ANS 4.1-1978	Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations	This standard is REFERENCED by DOE 5480.4.
ANS 4.5-1980 (R1986)	Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors	This standard is REFERENCED by DOE 5480.4 and supported by SRP 7.5.
ANS 6.4-1977 (R1985) (ANSI N403)	Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants	This standard is REQUIRED by DOE 6430.1A and 5480.4. The standard is also supported by RG 1.142. (This standard should cover those areas voided by the withdrawal of ANSI N101.6-1972.)
ANS 6.4.2-1985	Specification for Radiation Shielding Materials	This standard is REFERENCED by DOE 6430.1A.
ANS 8.1-1983		This standard is REQUIRED by DOE 5480.5.
ANS 8.3-1986 (ANSI N16.2)	Criticality Accident Alarm System	This standard is REQUIRED by DOE 6430.1A and DOE 5480.5. This standard is also supported by SRP 12.3.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANS 8.5-1979	Use of Borosicilicate-Glass Raching Rings as a Neutron Absorber in Solutions of Fissile Materials	This standard is REQUIRED by DOE 5480.4.
ANS 8.6-1975 (ANSI N16.3)	Safety in Conducting Subcritical Neutron Multiplication Measurements in Situ	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ANS 8.7-1975 (ANSI N16.5)	Guide for Nuclear Criticality Safety in the Storage of Fissile Materials	This standard is REFERENCED by DOE 5480.4.
ANS 8.10-1975 (ANSI N16.8)	Criteria for Nuclear Criticality Safety Controls in Operations Where Shielding Protects Personnel	This standard is REFERENCED by DOE 5480.4.
ANS 8.11-1975 (ANSI N16.9)	Validation of Calculational Methods for Nuclear Criticality Safety	This standard is REFERENCED by DOE 5480.4.
ANS 8.15-1981	American National Standard for Nuclear Criticality Control of Special Actinide Elements	This standard is REQUIRED by DOE 5480.5.
ANS 11.13-1972 (ANSI N101.6)	Concrete Radiation Shields	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.69, RG 3.9, and SRP 12.3.
ANS 14.1-1975 (R1982) (ANSI N394)	Operation of Fast Pulse Reactors	This standard is REFERENCED by DOE 5480.4.
ANS 15.1-1983 (ANSI N378)	Development of Technical Specifications for Research Reactors" (American Nuclear Society).	This standard is REQUIRED by DOE 5480.4.
ANS 15.2-1974 (ANSI N398)	Quality Control for Plate-Type Uranium-Aluminum Fuel Elements	This standard is REFERENCED by DOE 5480.4.
ANS 15.3-1974 (ANSI N399)	Records and Reports for Research Reactors	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANS 15.8-1976 (R1986) (ANSI N402)	Quality Assurance Program Requirements for Research Reactors	This standard is REFERENCED by DOE 5480.4.
ANS 15.11-1977 (ANSI 628)	Radiological Control at Research Reactor Facilities	This standard is REFERENCED by DOE 5480.4.
ANS 15.12-1977	Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents	This standard is REFERENCED by DOE 5480.4.
ANS 15.15-1978 (R1985) (ANSI N701)	Criteria for the Reactor Safety System of Research Reactors	This standard is REFERENCED by DOE 5480.4.
ANS 15.16-1978 (R1988) (ANSI N17.2)	Emergency Planning for Research Reactors	This standard is REFERENCED by DOE 5480.4.
ANS 15.18-1979	Administrative Controls for Research Reactors	This standard is REFERENCED by DOE 5480.4.
ANS 57.7-1982	Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)	This standard is REFERENCED by DOE 5480.4.
ANS 58.3-1977	Physical Protection for Systems and Components Important to Safety	This standard is REFERENCED by DOE 5480.4.
ANS 59.1-1979	Safety-Related Cooling Water Systems in Nuclear Power Plants	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANS 59.3-1977	Safety-Related Control Air Systems	This standard is REFERENCED by DOE 5480.4.
ANS 59.4-1979 (WITHDRAWN?)	Generic Requirements for Light Water Nuclear Power Plant Fire Protection	This standard is REFERENCED by DOE 5480.4.
ANS 59.51-1976 (ANSI N195)	Fuel Oil Systems for Standby Diesel-Generators	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.137 and SRP 9.5.4.
American National Standards Institute (ANSI)		
ANSI 3.19	Method for the Measurement of Real-Ear Protection of Hearing Protectors and Physical Attenuation of Earmuffs	This standard is REQUIRED by OR 5480.4.
ANSI A58.1-1982	Minimum Design Loads for Buildings and Other Structures	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 2.3.1, SRP 3.3.1, and SRP 3.3.2.
ANSI A115 Series	Door and Frame Preparation	This standard is REQUIRED by DOE 6430.1A.
ANSI A156 Series	Hardware	This standard is REQUIRED by DOE 6430.1A.
ANSI A216.1	Sectional Overhead Type Doors	This standard is REQUIRED by DOE 6430.1A.
ANSI B30 Series		This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANSI C2-1981 (R1984)	National Electrical Safety Code	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ANSI C84.1	Electric Power Systems and Equipment - Voltage Ratings (60 Hz)	This standard is REQUIRED by DOE 6430.1A.
ANSI C95.1-1982	Safety Levels with Respect to Human Exposure to Radio Frequency Electromagnetic Fields, 300 kHz to 100 GHz	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI D7.1-1973	Inspection Requirements for Motor Vehicles, Trailers, and Semi-trailers Operated on Public Highways	This standard is REQUIRED by DOE 5480.4.
ANSI N2.1-1969	Radiation Symbol	This standard is REQUIRED by OR 5480.4 and MMES ESH-7.
ANSI N2.3	Immediate Evacuation Signal for Use in Industrial Installations	This standard is REQUIRED by DOE 6430.1A.
ANSI N13.1-1969	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities	This standard is REQUIRED by DOE 6430.1A, OR 5480.4, and DOE 5480.4. This standard is also supported by SRP 9.3.1, SRP 11.5, and SRP 12.3.
ANSI N13.3-1969	Dosimetry for Criticality Accidents	This standard is REQUIRED by OR 5480.4 and MMES ESH-7.
ANSI N13.4-1971 (R1983)	Specification for Portable X- or Gamma Radiation Survey Instruments	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANSI N13.5-1972	Performance Specifications for Direct Reading and Indirect Reading Pocket Dosimeters for X- and Gamma Radiation	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7. This standard is also supported by SRP 12.5.
ANSI N13.6-1966 (R1972)	Practice for Occupational Radiation Exposure Records Systems	This standard is REQUIRED by OR 5480.4 and supported by SRP 12.5.
ANSI N13.7-1983	Standard for Radiation Protection Photographic Film Dosimeters - Criteria for Performance	This standard is REFERENCED by DOE 5480.4 and supported by SRP 12.5.
ANSI N13.10-1974	Specifications and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents	This standard is REQUIRED by OR 5480.4 and MMES ESH-7. This standard is also supported by SRP 11.5.
ANSI N16.1-1975	Safety Standards for Operations with Fissionable Materials Outside Reactors	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ANSI N42.18-1980 (R1985)	Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactive Effluents	This standard is REQUIRED by DOE 6430.1A, DOE 5480.4, and MMES ESH-7.
ANSI N43.1-1978	Radiological Safety in the Design and Operation of Particle Accelerators	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N43.2-1977 (Same as NBS Handbook 114)	Radiation Safety for X-Ray Diffraction and Fluorescence Analysis Equipment	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N317-1980 (R1985)	Performance Criteria for Instrumentation Used for Inplant Plutonium Monitoring	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N319-1976 (R1984)	Personnel Neutron Dosimeters	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANSI N320-1979 (R1985)	Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N322-1977 (R1983)	Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N323-1978 (R1983)	Radiation Protection Instrumentation Test and Calibration	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N343-1978 (R1984)	Internal Dosimetry for Mixed Fission and Activation Products	This standard is REFERENCED by DOE 5480.4.
ANSI N510-1980	Testing of Nuclear Air Cleaning Systems	This standard is REFERENCED by DOE 5480.4 and supported by SRP 6.5.1.
ANSI N512-1974	Protective Coatings (Paints) for the Nuclear Industry	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ANSI N537-1976	Radiological Safety Standard for the Design of Radiographic and Fluoroscopic Industrial X-Ray Equipment	This standard is REFERENCED by DOE 5480.4.
ANSI N542-1977	Sealed Radioactive Sources	This standard is REFERENCED by DOE 5480.4.
ANSI N543-1974	General Safety Standard for Installations Using Nonmedical X-ray and Sealed Gamma-Ray Sources	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI N545-1975	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry (Environmental Applications)	This standard is REQUIRED by DOE 5480.4 and OR 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANSI S1.4	Specifications for Sound Level Meters	This standard is REQUIRED by OR 5480.4.
ANSI S1.13	Methods for the Measurement of Sound Pressure Levels	This standard is REQUIRED by OR 5480.4.
ANSI S1.25	Specifications for Personnel Noise Dosimeters	This standard is REQUIRED by OR 5480.4.
ANSI S1.36	Survey Methods for the Determination of Sound Power Levels of Noise Sources	This standard is REQUIRED by OR 5480.4.
ANSI S3.18	Guide for Evaluation of Human Exposure to Whole-Body Vibration	This standard is REQUIRED by OR 5480.4.
ANSI Z9.2	Fundamentals Governing the Design and Operation of Local Exhaust Systems	This standard is REQUIRED by OR 5480.4.
ANSI Z9.4	Ventilation and Safe Practices of Abrasive Blasting Operations	This standard is REQUIRED by OR 5480.4.
ANSI Z88.2-1980	Practices for Respiratory Protection	This standard is REQUIRED by DOE 6430.1A, DOE 5480.4, and MMES ESH-7.
ANSI Z117.1-1977	Safety Requirements for Working in Tanks and Other Confined Spaces	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI Z129.1	Precautionary Labeling of Hazardous Industrial Chemicals	This standard is REQUIRED by OR 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ANSI Z136.1-1980	Safe Use of Lasers	This standard is REQUIRED by DOE 5480.4 and MMES ESH-7.
ANSI Z358.1	Eyewash and Shower Equipment, Emergency	This standard is REFERENCED by DOE 6430.1A.
ANSI/ANS 3.1-1981	Nuclear Power Plant Simulators for Use in Operator Training	This standard is REFERENCED by DOE 5480.6.
ANSI/ASHRAE 62	Natural and Mechanical Ventilation	This standard is REQUIRED by OR 5480.4.
ANSI/IESA 11.1, RP7-1979	Practice for Industrial Lighting	This standard is REQUIRED by OR 5480.4.
American Petroleum Institute (API)		
API-650	Welded Steel Tanks for Oil Storage	This standard is REQUIRED by DOE 6430.1A and supported by SRP 3.2.2.
American Society of Civil Engineers (ASCE)		
ASCE Manual 63	Structural Plastic Design Manual	This standard is REQUIRED by DOE 6430.1A.
ASCE MOP No. 37	Design and Construction of Sanitary and Storm Sewers	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ASCE 52-71	Guide for Design of Steel Transmission Towers	This standard is REQUIRED by DOE 6430.1A.
ASCE 1978-1	Design of Steel Transmission Pole Structures	This standard is REQUIRED by DOE 6430.1A.
American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE)		
ASHRAE	ASHRAE Applications Handbook	This standard is REQUIRED by DOE 6430.1A.
ASHRAE	ASHRAE Equipment Handbook	This standard is REQUIRED by DOE 6430.1A.
ASHRAE	ASHRAE Fundamentals Handbook	This standard is REQUIRED by DOE 6430.1A.
ASHRAE	ASHRAE Manual for Design of Smoke Control Systems for Buildings	This standard is REQUIRED by DOE 6430.1A.
ASHRAE	ASHRAE Refrigeration Handbook	This standard is REQUIRED by DOE 6430.1A.
ASHRAE	ASHRAE Systems Handbook	This standard is REQUIRED by DOE 6430.1A.
ASHRAE GRP-158	Heating and Cooling Load Calculation Manual	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ASHRAE Standard 15	Safety Code for Mechanical Refrigeration	This standard is REQUIRED by DOE 6430.1A.
ASHRAE Standard 20	Methods of Testing for Rating Remote Mechanical Draft Air Cooled Refrigerant Condensers	This standard is REQUIRED by DOE 6430.1A.
ASHRAE Standard 24	Methods of Testing for Rating Liquid Coolers	This standard is REQUIRED by DOE 6430.1A.
ASHRAE Standard 51	Methods of Testing Fans for Rating	This standard is REQUIRED by DOE 6430.1A.
ASHRAE Standard 55-74 (ANSI B193.1)	Thermal Environmental Conditions for Human Occupancy	This standard is REQUIRED by DOE 6430.1A.
ASHRAE Standard 62	Ventilation for Acceptable Indoor Air Quality	This standard is REQUIRED by DOE 6430.1A.
ASHRAE Standard 90	Energy Conservation in New Building Design	This standard is REQUIRED by DOE 6430.1A.
<u>American Society of Mechanical Engineers (ASME)</u>		
ASME	ASME Boiler and Pressure Vessel Code, Sections I-XI	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ASME	ASME Boiler and Pressure Vessel Code Section II, Materials Specifications	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by SRP 5.2.3.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ASME	ASME Boiler and Pressure Vessel Code Section III, Division 1, Nuclear Power Plant Components	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by RG 1.34, RG 1.50, RG 1.84, and RG 1.85.
ASME	ASME Boiler and Pressure Vessel Code Section III, Division 2, Code for Concrete Reactor Vessels and Containments	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ASME	ASME Boiler and Pressure Vessel Code Section V, Nondestructive Examination	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ASME	ASME Boiler and Pressure Vessel Code Section VIII, Division 1, Pressure Vessels	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ASME	ASME Boiler and Pressure Vessel Code Section IX, Welding and Brazing Qualifications	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by RG 1.34 and RG 1.50.
ASME	ASME Boiler and Pressure Vessel Code Section XI, Division 3, Section XI - Rules for In-Service Inspection of Nuclear Power Plant Components and Section XI, Division 1 - Rules for Inspection and Testing of Components of Light-Water Cooled Plants	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
ASME A17.1	Elevators and Escalators	This standard is REQUIRED by DOE 6430.1A.
ASME B16 Series	Fittings, Flanges and Valves	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ASME B31.1	Power Piping	This standard is REQUIRED by DOE 643 0.1A. This standard is also supported by S RP 3.2.2 and SRP 3.9.4.
ASME N509-80	Nuclear Power Plant Air Cleaning Units and Components	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by RG 1.52 and RG 1.140.
ASME N510-80	Testing of Nuclear Air Cleaning Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by RG 1.52 and RG 1.140.
ASME NQA-1-1983	Quality Assurance Program Requirements for Nuclear Facilities	This standard is REQUIRED by DOE 5700.6A, OR 5480.6, DOE 5700.6B, and MMES GP-5. This standard is also supported by RG 1.28.
ASME NQA-2-1986	Quality Assurance Requirements for Nuclear Power Plants	
<u>American Society for Testing and Materials (ASTM)</u>		
ASTM	Testing Materials Standards (those applicable to safety)	This standard is REFERENCED by DOE 5480.4.
ASTM A312	Specification for Seamless and Austenitic Stainless Steel Pipe	This standard is REQUIRED by DOE 6430.1A.
ASTM B1	Specification for Hard-Drawn Copper Wire	This standard is REQUIRED by DOE 6430.1A.
ASTM C55	Concrete Building Brick	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ASTM C62	Building Brick	This standard is REQUIRED by DOE 6430.1A.
ASTM C90	Hollow Load-Bearing Concrete Masonry Units	This standard is REQUIRED by DOE 6430.1A.
ASTM C145	Solid Load-Bearing Concrete Masonry Units	This standard is REQUIRED by DOE 6430.1A.
ASTM C270	Mortar for Unit Masonry	This standard is REQUIRED by DOE 6430.1A.
ASTM D1357-57	Practice for Planning the Sampling of the Atmosphere	This standard is REQUIRED by OR 5480.4.
ASTM D1586	Penetration Test and Split-Barrel Sampling of Soils	This standard is REQUIRED by DOE 6430.1A.
ASTM D1587	Thin-Walled Tube Sampling of Soils	This standard is REQUIRED by DOE 6430.1A.
ASTM D 1605-60	Practices for Sampling Atmospheres for Analysis of Gases and Vapors	This standard is REQUIRED by OR 5480.4.
ASTM D2113	Diamond Core Drilling for Site Investigation	This standard is REQUIRED by DOE 6430.1A.
ASTM D2488	Description and Identification of Soils (Visual-Manual Procedure)	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
ASTM D4256	Test Method for Determination of the Decontaminability of Coatings Used in Light-Water Nuclear Power Plants	This standard is REQUIRED by DOE 6430.1A.
ASTM D4546	Test Methods for One-Dimensional Swell or Settlement Potential of Cohesive Soils	This standard is REQUIRED by DOE 6430.1A.
ASTM G46	Recommended Practice for Examination and Evaluation of Pitting Corrosion	This standard is REQUIRED by DOE 6430.1A.
American Welding Society (AWS)		
AWS D1.1-83	Structural Welding Code - Steel	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 6.1.1.
AWS D1.2	Structural Welding Code - Aluminum	This standard is REQUIRED by DOE 6430.1A.
AWS D1.3-81	Specification, Structural Welding Code--Sheet Steel	This standard is REQUIRED by DOE 6430.1A.
AWS F1.1	Method for Sampling Airborne Particulated Generated by Welding and Allied Processes	This standard is REQUIRED by OR 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
American Water Works Association (AWWA)		
AWWA	Standard Methods for the Examination of Water and Waste Water	This standard is REFERENCED by DOE 5480.4.
AWWA	Manual of Water Quality and Treatment	This standard is REFERENCED by DOE 5480.4.
AWWA	Water Treatment Plant Design	This standard is REQUIRED by DOE 6430.1A.
AWWA D100-79 (AWS D5.2)	Welded Steel Tanks for Water Storage	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 3.2.2.
Argonne National Laboratory (ANL)		
ANL/EES TM-264, Rev. 1	Environmental Protection Appraisals: A Suggested Guide for U.S. Department of Energy Field Organizations	This standard is REQUIRED by DOE 6430.1A.
Army		
AMCR/DARCOM 385-100	Safety Manual	This standard is REQUIRED by DOE 6430.1A.
TM 5-809-10	Seismic Design for Buildings	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
TM 5-810-7/AFM 88-12	Joint Department of the Army and Air Force, USA Technical Manual, High Pressure Gas and Cryogenic Systems	This standard is REQUIRED by DOE 6430.1A.
TM 5-814-1	Sanitary and Industrial Wastewater Collection-Gravity Sewers and Appurtenances	This standard is REQUIRED by DOE 6430.1A.
TM 5-814-2	Sanitary and Industrial Wastewater Collection-Pumping Stations and Force Mains	This standard is REQUIRED by DOE 6430.1A.
TM 5-818-1	Procedures for Foundation Design of Buildings and Other Structures (Except Hydraulic Structures)	This standard is REQUIRED by DOE 6430.1A.
TM 5-818-5	Dewatering and Groundwater Control	This standard is REQUIRED by DOE 6430.1A.
TM 5-830-3	Dust Control	This standard is REQUIRED by DOE 6430.1A.
TM 5-1300	Structures to Resist the Effects of Accidental Explosions	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by SRP 2.2.1.
TM 11-486-5	Electrical Communications Systems Engineering: Outside Plant, Wire	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Brick Institute of America (BIA)		
BIA	Dampproofing and Waterproofing Masonry Walls	This standard is REQUIRED by DOE 6430.1A.
BIA	Building Code Requirements for Engineered Brick Masonry	This standard is REQUIRED by DOE 6430.1A.
Brookhaven National Laboratory (BNL)		
BNL-50831-I	Design Guide; Critical Facilities" (BNL)	This standard is REFERENCED by DOE 5480.4.
BNL-50831-II	Design Guide; Light and Heavy Water Cooled Reactor" (BNL)	This standard is REFERENCED by DOE 5480.4.
BNL-50831-III	Design Guide; Pool Type Reactors" (BNL)	This standard is REFERENCED by DOE 5480.4.
Code of Federal Regulations (CFR)		
10 CFR 20	Permissible dose levels and activity concentrations in restricted and unrestricted areas	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
10 CFR 30	Rules of General Applicability to Licensing of Byproduct Material	This standard is REFERENCED by DOE 5480.4.
10 CFR 40	Licensing of Source Material	This standard is REFERENCED by DOE 5480.4.
10 CFR 50	Licensing of Production and Utilization Facilities," and appendices,	This standard is REFERENCED by DOE

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
	including regulatory guides issued to describe a method of implementing these regulations (NRC).	5480.4. Compliance with each 10 CFR 50 section is given in Table B.4 in this appendix.
10 CFR 50.36	Technical specifications	This standard is REQUIRED by DOE 5480.6, Sect. 8.d. The technical specifications shall be defined in the SDDs and in the SAR in accordance with the requirements set forth in this section of 10 CFR 50 (see Table B.4 in this appendix).
10 CFR 50, Appendix A	General design criteria	This standard is REQUIRED by DOE 5480.4 and DOE 5480.6. Compliance with each General Design Criteria listed in Appendix A is given in Table B.5.
10 CFR 50, Appendix I	Numerical dose guidelines for meeting the criterion Appendix I "as low as reasonably achievable" (ALARA) for routine power reactor effluents	This standard is REQUIRED by DOE 5480.4 and DOE 5480.6.
10 CFR 55	Operator's Licenses	This standard is REFERENCED by DOE 5480.4.
10 CFR 70	Special Nuclear Material	This standard is REFERENCED by DOE 5480.4.
10 CFR 71	Packaging of Radioactive Material for Transport	This standard is REQUIRED by DOE 5480.3.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
10 CFR 100	Reactor Site Criteria	This standard is REQUIRED by DOE 5480.4 and 5480.6. This standard is also supported by SRP 2.1.1.
10 CFR 435	DOE (Energy Conservation)	This standard is REQUIRED by DOE 6430.1A.
10 CFR 436	Federal ENERGY Management and Planning Programs	This standard is REQUIRED by DOE 6430.1A.
10 CFR 712	Grand Junction Remedial Action Criteria (DOE)	This standard is REQUIRED by DOE 5480.4.
10 CFR 1022	Compliance with Floodplains/Wetlands Environmental Review Requirements	This standard is REQUIRED by DOE 6430.1A.
15 CFR 921	NOAA Guidelines on Estuarine Sanctuaries (DOC)	This standard is REQUIRED by DOE 5480.4.
15 CFR 923	NOAA Coastal Zone Management Program Approval Regulations (DOC)	This standard is REQUIRED by DOE 5480.4.
15 CFR 930	NOAA Regulations on Federal Consistency with Approved Coastal Management Program (DOC)	This standard is REQUIRED by DOE 5480.4.
15 CFR 931	NOAA Regulations on Coastal Energy Impact Program (DOC)	This standard is REQUIRED by DOE 5480.4.
16 CFR 1630	Standards for Surface Flammability of Carpets and Rugs (FF 1)	This standard is REQUIRED by DOE 6430.1A.
29 CFR 1904	Recording and Reporting Occupational Injury and Illnesses	This standard is REQUIRED by DOE 5481.1 and MMES ESH-7.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
29 CFR 1910	Occupational Safety and Health Standards	This standard is REQUIRED by DOE 6430.1A, DOE 5483.1A, and OR 5480.4.
29 CFR 1926	Safety and Health Regulations for Construction	This standard is REQUIRED by DOE 6430.1A, DOE 5480.4, DOE 5480.9, and DOE 5483.1A.
29 CFR 1960	Basic Program Elements for Federal Employee Occupational Safety and Health Programs and Related Matters, Subpart I, Recordkeeping and Reporting Requirements	This standard is REQUIRED by DOE 5484.1.
33 CFR 153-157	Coast Guard Regulations on Control of Pollution by Oil and Hazardous Substances (DOT)	This standard is REQUIRED by DOE 5480.4.
33 CFR 159	Coast Guard Regulations on Marine Sanitation Devices (DOT)	This standard is REQUIRED by DOE 5480.4.
33 CFR 320, 322-329	Army Corps of Engineers Permit Programs Regulations (DOD)	This standard is REQUIRED by DOE 5480.4.
36 CFR 800	Protection of Historic and Cultural Properties (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 50	EPA Regulations on National Primary and Secondary Ambient Air Quality Standards (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 51.24	EPA Regulations on Prevention of Significant Deterioration of Air Quality	This standard is REQUIRED by OR 5480.4.
40 CFR 53	EPA Regulations on Ambient Air Monitoring Reference and Equivalent Methods (EPA)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 55	EPA Regulations on Energy Related Authority (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 58	EPA Ambient Air Quality Surveillance Regulations (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 60	EPA Regulations on Standard of Performance for New Stationary Sources (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 61	EPA Regulations on National Emission Standards for Hazardous Air Pollutants (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 61, Subpart H	National emission standards for radionuclide emissions, Subpart H from DOE facilities	This standard is REQUIRED by DOE 6430.1A.
40 CFR 65	EPA Regulations on Delayed Compliance Order Under the Clean Air Act (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 81	EPA Regulations Designating Areas for Air Quality Planning Purposes (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 110	EPA Regulations on Discharge of Oil (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 112	EPA Regulations on Oil Pollution Prevention (EPA)	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 113	EPA Regulations on Liability Limits for Small Onshore Storage Facilities (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 114	EPA Interim Regulations on Penalties; Oil Pollution Prevention Regulations (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 116	EPA Designation of Hazardous Substances (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 117	EPA Regulations on Determination of Reportable Quantities for Hazardous Substances (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 120	EPA Regulations on Water Quality Standards (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 121	EPA Regulations on State Certification of Activities Requiring a Federal License or Permit (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 122	EPA Regulations on Consolidated Permit Program (EPA)	This standard is REQUIRED by DOE 6430.1A, DOE 5480.4, and OR 5480.4.
40 CFR 125	EPA Regulations on Criteria and Standards for the Pollutant Discharge Elimination System (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 129	EPA Regulations on Toxic Pollutant Effluent Standards (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 133	EPA Regulations on Secondary Treatment Information (EPA)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 136	EPA Regulations on Test Procedures for the Analysis of Pollutants (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 140	EPA Regulations on Marine Sanitation Device Standard (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 141	EPA National Interim Primary Drinking Water Regulations (EPA)	This standard is REQUIRED by DOE 5480.4 and DOE 6430.1A.
40 CFR 142	EPA National Interim Primary Drinking Water Regulations Implementation (EPA)	This standard is REQUIRED by DOE 5480.4 and DOE 6430.1A.
40 CFR 143	EPA National Secondary Drinking Water Regulations (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 144	EPA Regulations on Underground Injection Control Program	This standard is REQUIRED by OR 5480.4.
40 CFR 146	EPA Criteria and Standards for Underground Injection Control Program (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 149	Protection of Sole Source Aquifers (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 162	EPA Regulations for the Enforcement of the Federal Insecticide Fungicide and Rodenticide Act (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 165	EPA Regulations for the Acceptance of Certain Pesticides and Recommended Procedures for the Disposal and Storage of Pesticides and Pesticides Containers (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 166	EPA Regulations for Exemptions of Federal and State Agencies for Use of Pesticides Under Emergency Conditions (EPA)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 170	EPA Regulations for Worker Protection Standards for Agricultural Pesticides (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 171	EPA Regulations for Certification of Pesticides Applicators (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 191	Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High Level and Transuranic Radioactive Wastes	This standard is REQUIRED by DOE 6430.1A.
40 CFR 192	EPA Standards for Protection Against Uranium Mill Tailings (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 220-225, 227-229	EPA Ocean Dumping Regulations and Criteria (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 230	EPA Guideline on Discharge of Dredged or Fill Material into Navigable Waters (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 231	EPA Regulations on Disposal Site Determinations Under Clean Water Act (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 240 and 241	EPA Guidelines for the Thermal Processing of Solid Waste and for Land Disposal of Solid Wastes (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 243	EPA Guidelines for So|id Waste Storage and Collection (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 244	EPA Guidelines for Beverage Containers (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 245	EPA Guidelines for Resource Recovery Facilities (EPA)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 246	EPA Guidelines for Resource Separation for Material Recovery (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 247	EPA Guidelines for Procurement of Products that Contain Recycled Material (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 256	EPA Guidelines for Identification of Site Solid Waste Management Plans (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 257	EPA Regulations on Criteria for Classification of Solid Waste Disposal Facility (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 260	EPA General Regulations for Hazardous Waste Management (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.
40 CFR 261	EPA General Regulations for Identifying Hazardous Waste (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.
40 CFR 262	EPA Regulations for Hazardous Waste Generators (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.
40 CFR 263	EPA Regulations for Hazardous Waste Transporters (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 264	EPA Regulations for Owners and Operators of Permitted Hazardous Waste Facilities (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.
40 CFR 265	EPA Interim Standards for Owners and Operators of Permitted Hazardous Waste Facilities (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.
40 CFR 267	EPA Interim Standards for Owners and Operators of New Hazardous Waste Land Disposal Facilities (EPA)	This standard is REQUIRED by DOE 5480.4, DOE 6430.1A, and OR 5480.4.
40 CFR 270	EPA Administered Permit Programs: The Hazardous Waste Permit Program	This standard is REQUIRED by OR 5480.4 and DOE 6430.1A.
40 CFR 271	Requirements for Authorization of State Hazardous Waste Programs	This standard is REQUIRED by DOE 6430.1A.
40 CFR 280	Underground Storage Tanks	This standard is REQUIRED by DOE 6430.1A.
40 CFR 300	Hazardous Substances; National Oil and Hazardous Substances Pollution Contingency Plan (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 401	EPA Regulations on General Provisions for Effluent Guidelines and Standards (Note: 40 CFR 401.14, "Cooling Water Intake Structures" (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 403	EPA Pretreatment Regulations for Existing and New Sources of Pollution (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 413	EPA Regulations for Electroplating Point Source Category (EPA)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 423	EPA Regulations - Guidelines and Standards for Steam Electric Power Generation (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 457	Effluent Guidelines and Standards for Explosive Manufacturing (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 459	Effluent Guidelines for Photographic Processing (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 542	Financial Responsibilities for Water Pollution (EPA)	This standard is REQUIRED by DOE 5480.4.
40 CFR 761	EPA Regulations on Polychlorinated Biphenyls Manufacturing, Processing, Distribution in Commerce, and Use Prohibitions (EPA)	This standard is REQUIRED by DOE 5480.4 and DOE 6430.1A.
40 CFR 1500	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
40 CFR 1501	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 1502	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
40 CFR 1503	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
40 CFR 1504	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
40 CFR 1505	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
40 CFR 1506	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
40 CFR 1507	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
40 CFR 1508	Council on Environmental Quality Regulations	This CFR section establishes regulations for implementing the procedural provisions of the National Environmental Policy Act and is REQUIRED by DOE 5440.1C.
41 CFR 101	Federal Property Management Regulations	This standard is REQUIRED by DOE 6430.1A.
41 CFR 109	DOE Property Management Regulations	This standard is REFERENCED by DOE 5480.13.
48 CFR Chapter 9	DOE Acquisition Regulation (DEAR)	This standard is REFERENCED by DOE 5480.9.
49 CFR 397	Hazardous Materials Regulations (DOT)	This standard is REQUIRED by DOE 5480.4.
50 CFR 17	Fish and Wildlife Service List of Endangered and Threatened Wildlife and Plants (DOI)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Compressed Gas Association (CGA)		
CGA Pamphlet G-4.1	Cleaning Equipment for Oxygen	This standard is REQUIRED by DOE 6430.1A.
CGA Pamphlet G-4.4	Industrial Practices for Gaseous Oxygen Transportation and Distribution Piping Systems	This standard is REQUIRED by DOE 6430.1A.
CGA Pamphlet P-1	Safe Handling of Compressed Gases in Containers	This standard is REQUIRED by DOE 6430.1A.
CGA Pamphlet S-1.1	Pressure Relief Device Standards, Part 1 - Cylinders for Compressed Gases	This standard is REQUIRED by DOE 6430.1A.
CGA Pamphlet S-1.2	Pressure Relief Device Standards, Part 2 - Cargo and Portable Tanks for Compressed Gases	This standard is REQUIRED by DOE 6430.1A.
CGA Pamphlet S-1.3	Pressure Relief Device Standards, Part 3 - Compressed Gas Storage Containers	This standard is REQUIRED by DOE 6430.1A.
Congressional Acts		
The National Historic Preservation Act of 1966	The National Historic Preservation Act of 1966	This standard is REQUIRED by DOE 5480.4.
7 U.S.C. 136, Et Seq.	Federal Insecticide Fungicide and Rodenticide Act	This standard is REQUIRED by DOE 5480.4.
15 U.S.C. 2601, Et Seq.	Toxic Substances Control Act	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
16 U.S.C. 1531, Et Seq.	Endangered Species Act of 1973	This standard is REQUIRED by DOE 5480.4.
33 U.S.C. 1251, Et Seq.	Clean Water Act	This standard is REQUIRED by DOE 5480.4.
42 U.S.C. 300f, Et Seq.	Safe Drinking Water Act	This standard is REQUIRED by DOE 5480.4.
42 U.S.C. 4901, Et Seq.	Noise Control Act of 1972 (EPA)	This standard is REQUIRED by DOE 5480.4.
42 U.S.C. 6901, Et Seq.	Resource Conservation and Recovery Act of 1976 (RCRA)	This standard is REQUIRED by DOE 5480.4.
42 U.S.C. 7401, Et Seq.	Clean Air Act	This standard is REQUIRED by DOE 5480.4.
42 U.S.C. 9615 (Superfund)	Comprehensive Environmental Response, Compensation, and Liability Act of 1980	This standard is REQUIRED by DOE 5480.4.
<u>Construction Safety Association of Ontario (CSAO)</u>		
CSAO	Crane Handbook	This standard is REFERENCED by DOE 5480.4.
CSAO	Rigging Manual	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Crane Manufacturers Association of America (CMAA)		
CMAA-70	Specification for Electric Overhead Traveling Cranes	This standard is REQUIRED by DOE 5480.4.
Department of Commerce (DOC)		
National Fire Prevention and Control Administration, RP-1 (8/78) (DOC)	Standard Practice for the Fire Protection of Essential Electronic Equipment Operations	This standard is REFERENCED by DOE 5480.4.
Department of Defense (DOD)		
MIL-HDBK-419	Grounding, Bonding, and Shielding for Electronic Equipment and Facilities	This standard is REQUIRED by DOE 6430.1A.
MIL-P-43951	Padlocks and Padlock Sets, Key Operated, Medium Security, Regular Shackle	This standard is REQUIRED by DOE 6430.1A.
MIL-STD-882B	Safety System Program Requirements	This standard is REFERENCED by DOE 5481.1B.
MIL-STD-1472C	Human Engineering Design Criteria for Military Systems, Equipment, and Facilities	This standard is REQUIRED by DOE 6430.1A.
Van Cott and Kincade	Human Engineering Guide to Equipment Design (from the Joint Army-Navy-Air Force Steering Committee, 1972, U.S. Government Printing Office)	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Department of Energy (DOE)		
DOE	Reactor operations, design, construction, modification, modifications, etc. will be conducted in accordance with established standards.	This standard is REQUIRED by DOE 5480.5 and DOE 5480.6.
DOE	DOE Hoisting and Rigging Manual	
DOE Environmental Safety and Health Information Bulletin	Filter Unit Inspection and Testing Service	This standard is supported by RG 1.52.
DOE Design Guide	Graphic Design Standard 8	This standard is REQUIRED by DOE 6430.1A.
DOE Radiation Standards	DOE Radiation Standards of Protection of the Public in the Vicinity of DOE Facilities (William A. Vaughn Memorandum, August 5, 1985)	This standard is REQUIRED by DOE 6430.1A.
DOE	DOE Safeguards and Security Standards and Criteria	This standard is REQUIRED by DOE 6430.1A.
NE (RDT) Standards	Applicable Nuclear Energy Reactor Development Technology (RDT) Standards" [see index of NE (RDT) standards] (DOE).	This standard is REFERENCED by DOE 5480.4.
DOE/TIC-11268	A Manual for the Prediction of Blast and Fragment Loading on Structures	This standard is REQUIRED by DOE 6430.1A.
DOE/TIC-11603	Non-Reactor Nuclear Facilities Standards and Guidelines	This standard is REQUIRED by MMES ESH-8 and MMES ESH-20
DOE EH-35	Safety Objectives/Policy Statement	

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
DOE/EP-0023	A Guide for Environmental Radiological Surveillance of U.S. Department of Energy Installations	This standard is REFERENCED by DOE 5480.4.
DOE/EP-0096	A Guide for Effluent Radiological Measurement at DOE Installations	This standard is REFERENCED by DOE 5480.4.
DOE/EP-0100	A Guide for Radiological Characterization and Measurements for Decommissioning of U.S. Department of Energy Surplus Facilities	This standard is REFERENCED by DOE 5480.4.
DOE/EP-0108 (WASH-1245-1)	Standard for Fire Protection of DOE Electronic Computer/Data Processing Systems	This standard is REQUIRED by DOE 6430.1A and MMES ESH-21.
DOE/EV-0043	Standard on Fire Protection for Portable Structures	This standard is REQUIRED by DOE 5480.4 and MMES ESH-21.
DOE/EV-0051-1	Electrical Safety Criteria for Research and Development Activities	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
DOE/EV 0132	Environmental Compliance Guide, Volumes I and II	This standard is REQUIRED by DOE 6430.1A and OR 5440.1B.
DOE/EV-1830-TS	A Guide for Reducing Radiation Exposure to As Low As Reasonably Achievable	This standard is REFERENCED by DOE 5480.4.
DOE/EV 06194-1	U.S. Department of Energy Explosives Safety Manual (NTIS)	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
DOE/FAR PART 10	Specification Standards and Other Purchase Descriptions	This standard is REQUIRED by DOE 6430.1A.
DOE/MA 0129	Site Development Planning for Energy Management	This standard is REQUIRED by DOE 6430.1A.
DOE/OR 860	Evaluation of Potential for Incidents Having Health or Safety Impact	This standard is REQUIRED by DOE 6430.1A.
DOE/P 3	Site Planning for Energy Management	This standard is REQUIRED by DOE 6430.1A.
NE C 1-1T	Instrumentation and Control Equipment Grounding and Shielding Practices	
NE C 7-15T	Thermocouple Connectors and Thermocouple Connector Panels	
NE C 7-16T	Thermocouple Assemblies, Magnesium Oxide Insulated, Stainless Steel Sheathed	
NE C 7-17T	Platinum Resistance Thermometer	
NE C 10-1T	Thermocouple Signal Transmitter	
NE C 10-3T	Multiple Input Preamplifier/Discriminator for Use With Neutron Counters	
NE C 15-3T	Current Pulse Preamplifiers for Use With Fission Counters	
NE C 15-11T	BF_3 Gamma-Tolerant Neutron Detector Tubes	
NE C 16-1T	Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems	This standard is REFERENCED by DOE 5481.1A and supported by RG 1.22, RG 1.49, RG 1.62, RG 1.97,and RG 1.118.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NE C 17-5T	Metal-Sheathed, Mineral-Insulated Cable Bulk Material	
NE C 18-1T	Ceramic Electrical Insulators	
NE E 4-3	Heat Transfer Equipment Design	
NE E 7-9T	Mechanical and Hydraulic Snubbers for Nuclear Applications	
NE E 12-47	Shielded Shipping Cask for Spent Reactor Fuel Elements	
NE E 12-5T	Operating Manuals for Fuel Shipping Containers	
NE E 12-7T	Inspection and Preventive Maintenance of Fuel Shipping Containers	
NE F 1-2T	Preparation of Plant and System Design Description Documents	
NE F 2-7T	Acceptance Sampling Plans	
NE F 2-9T	Reliability Assurance	
NE F 3-2T	Calibration Program Requirements	
NE F 3-41T	In-Place Testing of HEPA Filter Systems by the Single-Particle, Particle-Size Spectrometer Method	
NE F 3-42	Operating Policy of DOE Filter Test Program	
NE F 3-43	Quality Assurance Testing of HEPA Filters	

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NE F 3-44	DOE Filter Test Facilities Quality Program Plan	
NE F 3-45	Specifications for HEPA Filters Used by DOE Contractors	
NE F 6-2T	Welding of Reactor Core Components and Test Assemblies	
NE F 7-2T	Packaging, Packing, and Marking of Components for Shipment and Storage	
NE F 7-3T	Marking of Components and Parts	
NE F 8-1T	Preloaded Threaded Fasteners and Closures	
NE F 8-6T	Hoisting and Rigging of Critical Components and Related Equipment	
NE F 8-11T	Fuel Shipping Container Tiedown for Truck Transport	
NE F 9-2T	Seismic Requirements for Design of Nuclear Power Plants and Nuclear Test Facilities	
NE F 11-2T	Analytical Chemistry Methods for Boron Carbide Absorber Material	
NE F 11-3T	Analytic Chemistry Methods for Metallic Core Components	
NE M 12-1T	Test Requirements for Thermal Insulating Materials for Use on Austenitic Stainless Steel	This standard is supported by RG 1.36.
NE M 16-1T	Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds	This standard is supported by RG 1.140.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NE M 17-1T	Sodium Carbonate, Low-Chloride Fire Extinguishing Agent	
Department of Transportation (DOT)		
Federal Highway Administration (DOT)	Motor Carrier Safety Regulations (DOT)	This standard is REQUIRED by DOE 5480.4.
Bureau of Public Roads (DOT)	Manual on Uniform Traffic Control Devices for Streets and Highways	This standard is REQUIRED by DOE 5480.4.
U.S. Coast Guard, CG-193 (DOT)	Miscellaneous Electrical Equipment List	This standard is REFERENCED by DOE 5480.4.
U.S. Coast Guard, CG-259 (DOT)	Electrical Engineering Regulations	This standard is REFERENCED by DOE 5480.4.
Drew Chemical		
-	Principles of Industrial Water Treatment (1977)	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
EG&G Idaho (EG&G)		
EG&G (5/80)	DOE Hoisting and Rigging Manual	This standard is REFERENCED by DOE 5480.4.
EG&G	DOE Drilling Safety Manual	This standard is REFERENCED by DOE 5480.4.
Electronics Industries Association (EIA)		
EIA-RS-222-C	Structural Standards for Steel Antenna Towers and Antenna Supporting Structures	This standard is REQUIRED by DOE 6430.1A.
Electric Power Research Institute (EPRI)		
EPRI NP-3659	Human Factors Guide for Nuclear Power Plant Control Room	This standard is REQUIRED by DOE 6430.1A.
Energy Research and Development Administration (ERDA)		
ERDA	Energy Research and Development Administration, Division of Safety, Standards, and Compliance Respirator Manual	This standard is REFERENCED by DOE 5480.4.
ERDA 76-21 (ORNL-NSIC-65-1)	Nuclear Air Cleaning Handbook	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by RG 1.140 and SRP 6.5.1.
ERDA 77-24	A Guide for Environmental Radiological Surveillance at ERDA Installations	This standard is REFERENCED by DOE 5484.1.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Executive Orders		
Executive Order 11490	Assigning Preparedness Functions to Federal Departments and Agencies	This standard is REQUIRED by DOE 6430.1A.
Executive Order 11514 (as amended by Executive Order 11991)		This standard is REFERENCED by DOE 5440.1C.
Executive Order 11593	Protection and Enhancement of the Cultural Environment	This standard is REQUIRED by DOE 6430.1A.
Executive Order 11644	Floodplain Management (EPA)	This standard is REQUIRED by DOE 5480.4.
Executive Order 11987	Exotic Organisms (EPA)	This standard is REQUIRED by DOE 5480.4.
Executive Order 11988	Off-Road Vehicles on Public Lands (EPA)	This standard is REQUIRED by DOE 5480.4.
Executive Order 11989	Floodplain Management (EPA)	This standard is REQUIRED by DOE 6430.1A.
Executive Order 11990	Protection of Wetlands (EPA)	This standard is REQUIRED by DOE 6430.1A.
Executive Order 11514 and Executive Order 11991	Protection and Enhancement of Environmental Quality (EPA)	This standard is REQUIRED by DOE 5480.4.
Executive Order 12088	Federal Compliance with Pollution Control Standards (EPA)	This standard is REQUIRED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Executive Order 12286	Response to Environmental Damage (EPA)	This standard is REQUIRED by DOE 5480.4.
Executive Order 12342	Environmental Safeguards on Activities for Animal Damage Control on Federal Lands (EPA)	This standard is REQUIRED by DOE 5480.4.
Federal Construction Council (FCC)		
FCC Technical Report No. 69	Stationary Diesel Engines for Use with Generators to Supply Electric Power	This standard is REQUIRED by DOE 6430.1A.
Federal Information Processing Standards (FIPS)		
FIPS PUB 94	Guidelines on Electrical Power for ADP Installations	This standard is REQUIRED by DOE 6430.1A.
Federal Radiation Council		
Federal Radiation Council	Applicable Federal Radiation Council Reports (NRC)	This standard is REFERENCED by DOE 5480.4.
Federal Radiation Council Report No. 1-1960	Background Material for the Development of Radiation Protection Standards	This standard is REFERENCED by DOE 5480.4.
Federal Radiation Council Report No. 2-1961	Background Material for the Development of Radiation Protection Standards	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Federal Radiation Council Report No. 5-1964	Background Material for the Development of Radiation Protection Standards	This standard is REFERENCED by DOE 5480.4.
Federal Radiation Council Report No. 7-1965	Background Material for the Development of Radiation Protection Standards; Protection Action Guides for Strontium-89, Strontium-90, and Cesium-137	This standard is REFERENCED by DOE 5480.4.
Federal Radiation Council Report No. 8-1967	Guidance for the Control of Radiation Hazards in Uranium Mining	This standard is REFERENCED by DOE 5480.4.
Federal Register (FR)		
43 FR 55978	Council on Environmental Quality Regulations	This standard is REFERENCED by DOE 5440.1C.
45 FR 12746	Preliminary Notification of Hazardous Waste Activity	This standard is REQUIRED by DOE 6430.1A.
49 FR 25565	DOE Property Management Regulations	This standard is REFERENCED by DOE 548-13.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
54 FR 20694	DOE Guidelines for Compliance With the National Environmental Policy Act	This standard is REQUIRED by DOE 6430.1A and DOE 5440.1C.
FEMA CPG2-17	Electromagnetic Pulse Protection Guidance	This standard is REQUIRED by DOE 6430.1A.
FEMA TR-83A	Interim Guidelines for Building Occupant Protection From Tornadoes and Extreme Winds	This standard is REQUIRED by DOE 6430.1A.
FEMA TR-83B	Tornado Protection - Selecting and Designing Safe Areas in Buildings	This standard is REQUIRED by DOE 6430.1A.
FEMA TR-87	Standards for Fallout Shelters	This standard is REQUIRED by DOE 6430.1A.
Fink and Beaty		
-	Standard Handbook for Electrical Engineers, 12th Edition	This standard is REQUIRED by DOE 6430.1A.
Illuminating Engineering Society of North America (IES)		
IES	Lighting Handbook	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by SRP 9.5.3.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Institute of Electrical and Electronic Engineers (IEEE)		
IEEE	Institute of Electrical and Electronic Engineers Standards	This standard is REFERENCED by DOE 5480.4.
IEEE 80-1986	Guide for Safety in AC Substation Grounding	This standard is REFERENCED by DOE 5480.4.
IEEE 141-1976 (R1986)	IEEE Recommended Practice for Electrical Power Distribution for Industrial Plants (IEEE Red Book)	This standard is REQUIRED by DOE 6430.1A.
IEEE 142-1982 (ANSI C114.1)	IEEE Recommended Practice for Grounding of Industrial and Commercial Power Systems (IEEE Green Book)	This standard is REFERENCED by DOE 5480.4.
IEEE 242-1986	IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)	This standard is REFERENCED by DOE 5480.4.
IEEE 279-1971 (ANSI N42.7-1972) [This std. was officially withdrawn June 14, 1984 and is now included in IEEE 603-1980]	Criteria for Protection Systems for Nuclear Power Generating Stations	This standard is REQUIRED by DOE 5480.4. This standard is also supported by RG 1.47, RG 1.52, RG 1.53, RG 1.62, RG 1.75, RG 1.95, RG 1.106, RG 1.118, RG 1.151, RG 1.153, SRP 7.1-7.3, and SRP 3.11.
IEEE 308-1980 (ANSI N41.12)	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by RG 1.32, RG 1.41, RG 1.118, SRP 8.2, and SRP 8.3.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
IEEE 317-1976 (R1983) (ANSI N41.51)	Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations" (IEEE)	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.63, SRP 3.11, and SRP 8.3.
IEEE 323-1974 (R1983) (ANSI N41.5)	Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (IEEE).	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.89, SRP 3.10, and SRP 3.11.
IEEE 336-1980 (R1985) (ANSI N45.2.4)	Installation, Inspection, and Testing Requirement for Class 1E Instrumentation and Electric Equipment at Nuclear Power Generating Stations	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.30.
IEEE 338-1977 (ANSI N41.3)	Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.118, SRP 7.1, and SRP 7.3.
IEEE 344-1974 (R1987) (ANSI N41.7)	Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	This standard is REFERENCED by DOE 5480.4. This standard is also supported by RG 1.100, SRP 3.10, and SRP 5.4.12.
IEEE 352-1975 (R1987) (ANSI N41.4)	Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems" (ANSI/IEEE).	This standard is REFERENCED by DOE 5480.4.
IEEE 379-1977 (R1988) (ANSI N41.2)	Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems" (ANSI/IEEE)	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by RG 1.53, SRP 7.1-7.3, and SRP 8.3.
IEEE 382-1980 (R1985) (ANSI N41.6)	IEEE Standard for Qualification of Safety-Related Valve Actuators	This standard is REFERENCED by DOE 5480.4. This standard is also supported by RG 1.73, SRP 3.10, and SRP 3.11.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
IEEE 384-1977 (R1981) (ANSI N41.14)	Standard Criteria for Independence of Class 1E Equipment and Circuits" (IEEE)	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by RG 1.75, SRP 7.1-7.2, and SRP 8.3.
IEEE 387-1977 (ANSI N41.13)	Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations (ANSI\IEEE)	This standard is REFERENCED by DOE 5480.4. This standard is also supported by RG 1.9, SRP 8.3, and SRP 9.5.4 through SRP 9.5.8.
IEEE 399-1980	IEEE Recommended Practice for Power System Analysis (IEEE Brown Book)	This standard is REQUIRED by DOE 6430.1A.
IEEE 446-1980 (R1987)	Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications	This standard is REQUIRED by DOE 6430.1A.
IEEE 450-1980 (ANSI N41.15)	IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.129.
IEEE 467-1980	Standard Quality Assurance Program Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations" (IEEE)	This standard is REFERENCED by DOE 5480.4.
IEEE 484-1981 (R1987) (ANSI N41.24)	IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations	This standard is REFERENCED by DOE 5480.4 and supported by RG 1.128.
IEEE 493-1980	IEEE Recommended Practice for the Design of Reliable Industrial and Commercial Power Systems (IEEE Gold Book)	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
IEEE 498-1980	Standard Requirements for the Calibration and Control of Measuring and Test Equipment Used in the Construction and Maintenance of Nuclear Power Generating Stations" (IEEE)	This standard is REFERENCED by DOE 5480.4.
IEEE 567-1980 (ANSI N41.1)	Trial-Use Standard Criteria for the Design of the Control Room Complex for a Nuclear Power Generating Station" (ANSI/IEEE)	This standard is REFERENCED by DOE 5480.4.
IEEE 603-1980	IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations	This standard is REQUIRED by DOE 5480.4. This standard is also supported by RG 1.152.
IEEE 739-1984	Recommended Practice for Energy Conservation and Cost-effective Planning in Industrial Facilities (IEEE Bronze Book)	This standard is REQUIRED by DOE 6430.1A.
IEEE 837-1984	Qualifying Permanent Connections Used in Substation Grounding	This standard is REQUIRED by DOE 6430.1A.
IEEE C37 Series	Circuit Breakers, Switchgear, Relays, Substations, and Fuses	This standard is REQUIRED by DOE 6430.1A.
IEEE C37.91	Guide for Protective Relay Applications to Power Transformers	This standard is REQUIRED by DOE 6430.1A.
<u>International Association of Plumbing and Mechanical Officials (IAMPO)</u>		
IAPMO	Uniform Plumbing Code	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
International Atomic Energy Agency (IAEA)		
IAEA Safety Series (in general)	IAEA Safety Series	This standard is REFERENCED by DOE 5480.4.
IAEA Safety Series No. 6	International Atomic Energy Agency Regulations for the Safe Transport of Radioactive Materials (International Shipments)	This standard is REFERENCED by DOE 5480.4 and REQUIRED by MMES ESH-17.
IAEA Safety Series No. 30	Manual on Safety Aspects of the Design and Equipment of Hot Laboratories	This standard is REQUIRED by DOE 6430.1A.
IAEA Safety Series No. 31	Safe Operation of Nuclear Power Plants (IAEA).	This standard is REFERENCED by DOE 5480.4.
IAEA Safety Series No. 35	Safe Operation of Critical Assemblies and Research Reactors," 1971 edition (IAEA).	This standard is REFERENCED by DOE 5480.4.
International Commission on Radiation Units and Measurements (ICRU)		
ICRU Report 14, 1969	Radiation Dosimetry	This standard is REFERENCED by DOE 5480.4.
ICRU Report 20, 1971	Radiation Protection Instrumentation and Its Application	This standard is REFERENCED by DOE 5480.4.
ICRU Report 33	Radiation Quantities and Units	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
International Conference of Building Officials (ICBO)		
UBC	Uniform Building Code	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 3.8.4.
International Commission of Radiological Protection (ICRP)		
ICRP	Radiation Protection Standards Reports	This standard is REFERENCED by DOE 5480.4.
ICRP Report No. 26	Recommendations of the ICRP	This standard is REQUIRED by DOE 6430.1A.
Lawrence Berkeley Laboratory (LBL)		
LBL-9143	Seismic Safety Guide	This standard is REQUIRED by DOE 6430.1A.
Lawrence Livermore National Laboratory (LLNL)		
UCRL 15673	Human Factors Design Guidelines for Maintainability of DOE Nuclear Facilities	This standard is REQUIRED by DOE 6430.1A.
UCRL 15910	Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
UCRL 53526	Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites	This standard is REQUIRED by DOE 6430.1A.
UCRL 53582	Natural Phenomena Hazards Modeling Project Seismic Hazard Models for Department of Energy Sites	This standard is REQUIRED by DOE 6430.1A.
Los Alamos National Laboratory (LANL)		
LA-10294-MS	A Guide to Radiological Accident Considerations for Siting and Design of DOE Nonreactor Nuclear Facilities	This standard is REQUIRED by DOE 6430.1A, MMES ESH-8, and MMES ESH-20.
Metal Building Manufacturers Association (MBMA)		
MBMA	Metal Building Systems Manual	This standard is REQUIRED by DOE 6430.1A.
Meister and Rabideau		
Meister and Rabideau	Human Factors Evaluation in System Development	This standard is REQUIRED by DOE 6430.1A.
National Bureau of Standards (NBS)		
NBS Handbook	National Council on Radiation Protection and Measurement Recommendation	This standard is REFERENCED by DOE 5480.4.
NBS Handbook 81	Safety Rules for the Installation and Maintenance of Electric Supply and Communication Lines	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NBS Handbook 135	Life Cycle Cost Manual for Federal Energy Management Programs	This standard is REQUIRED by DOE 6430.1A.
National Concrete Masonry Association (NCMA)		
NCMA TR 75B	Specifications for the Design and Construction of Load-Bearing Concrete Masonry	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 3.8.4.
NCMA	Waterproof Coatings for Concrete Masonry	This standard is REQUIRED by DOE 6430.1A.
NCMA	Waterproofing Concrete Masonry Basements and Earth-Sheltered Structures	This standard is REQUIRED by DOE 6430.1A.
National Electrical Manufacturers Association (NEMA)		
NEMA ICS-1970 (R1975)	NEMA Standards Publication, Industrial Controls and Systems	This standard is REQUIRED by DOE 6430.1A.
NEMA MG-1-1977	Motors and Generators	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
National Fire Protection Agency (NFPA)		
NFPA	Fire Protection Handbook - 16th Edition	This standard is REFERENCED by DOE 5480.4. This standard is also supported by SRP 9.5.1.
NFPA	Applicable NFPA Fire Codes	This standard is REFERENCED by DOE 5480.4 and REQUIRED by MMES ESH-21.
NFPA 10 (ANSI Z112.1)	Installation, Maintenance, and Use of Portable Fire Extinguishers	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 11 (ANSI Z286.1)	Low Expansion Foam and Combined Agent Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 11A (ANSI Z286.2)	Medium and High Expansion Foam Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 12 (ANSI A54.1)	Carbon Dioxide Extinguishing Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 12A (ANSI A54.4)	Halon 1301 Fire Extinguishing Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NFPA 12B (ANSI A54.5)	Halon 1211 Fire Extinguishing Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 13 (ANSI A54.3)	Installation of Sprinkler Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 14 (ANSI Z273.1)	Standpipe and Hose Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 15 (ANSI A54.2)	Water Spray Fixed Systems for Fire Protection	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 16 (ANSI Z274.1)	Deluge Foam-Water Sprinkler and Spray Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 16A	Installation of Closed-Head Foam-Water Sprinkler Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 17 (ANSI Z275.1)	Dry Chemical Extinguishing Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 20	Installation of Centrifugal Fire Pumps	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NFPA 22	Water Tanks for Private Fire Protection	This standard is REQUIRED by DOE 6430.1A.
NFPA 24	Installation of Private Fire Service Mains	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 30	Flammable and Combustible Liquids Code	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 31	Standard for the Installation of Oil Burning Equipment	This standard is REQUIRED by DOE 6430.1A.
NFPA 37	Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines	This standard is REQUIRED by DOE 6430.1A.
NFPA 45	Fire Protection for Laboratories Using Chemicals	This standard is REQUIRED by DOE 6430.1A.
NFPA 54 (ANSI Z 233.1)	National Fuel Gas Code	This standard is REQUIRED by DOE 6430.1A.
NFPA 58	Standard for the Storage and Handling of Liquified Petroleum Gases	This standard is REQUIRED by DOE 6430.1A.
NFPA 68	Guide for Explosion Venting	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NFPA 70-1981 (ANSI C1)	National Electrical Safety Code	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4. This standard is also supported by SRP 9.5.1.
NFPA 71	Installation, Maintenance and Use of Central Station Signalling Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 72A	Standard for the Installation, Maintenance, and Use of Local Protective Signaling Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 72B	Standard for the Installation, Maintenance and Use of Auxiliary Protective Signaling Systems for Fire Alarm Service	This standard is REQUIRED by DOE 6430.1A.
NFPA 72C	Installation, Maintenance, and Use of Remote Station Protective Signaling Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 72D	Standard for the Installation, Maintenance, and Use of Proprietary Protective Signaling Systems	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 72E	Automatic Fire Detectors	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
NFPA 72F	Installation, Maintenance, and Use of Emergency Voice/Alarm Communication Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 72G	Installation, Maintenance, and Use of Notification Appliances for Protective Signaling Systems	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NFPA 72H	Testing Procedures for Local, Auxiliary, Remote Station, and Proprietary Protective Signaling Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 75 (WASH 1245.1)	Protection of Electronic Computer/Data Processing Equipment	This standard is REQUIRED by DOE 6430.1A and DOE 5480.4.
NFPA 78-1975	Lightning Protection Code	This standard is REQUIRED by DOE 6430.1A.
NFPA 80	Fire Doors and Windows	This standard is REQUIRED by DOE 6430.1A and supported by SRP 9.5.1.
NFPA 80A	Recommended Practice for Protection of Buildings from Exterior Fire Exposure	This standard is REQUIRED by DOE 6430.1A.
NFPA 90A	Installation of Air Conditioning and Ventilating Systems	This standard is REQUIRED by DOE 6430.1A.
NFPA 91	Blower and Exhaust Systems for Dust, Stock and Vapor Removal or Conveying	This standard is REQUIRED by DOE 6430.1A.
NFPA 99	Standard for Health Care Facilities	This standard is REQUIRED by DOE 6430.1A.
NFPA 101 (ANSI A9.1)	Life Safety Code	This standard is REQUIRED by DOE 6430.1A.
NFPA 110	Emergency and Standby Power Systems	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
NFPA 232	Protection of Records	This standard is REQUIRED by DOE 6430.1A.
NFPA 255 (ASTM E84)	Method of Test of Surface Burning Characteristics of Building Materials	This standard is REQUIRED by DOE 6430.1A and supported by SRP 9.5.1.
NFPA 496	Purged and Pressurized Enclosures for Electrical Equipment	This standard is REQUIRED by DOE 6430.1A.
NFPA 1410	Training Standard on Initial Fire Attack	This standard is REQUIRED by DOE 6430.1A.
<u>National Institute for Occupational Safety and Health (NIOSH)</u>		
Department of Health and Human Services (NIOSH)	NIOSH Criteria Documents	This standard is REFERENCED by DOE 5480.4.
Department of Health and Human Services (NIOSH) No. 77-157A	NIOSH Manual of Analytical Methods, Vol. 1	This standard is REFERENCED by DOE 5480.4.
Department of Health and Human Services (NIOSH) No. 77-157B	NIOSH Manual of Analytical Methods, Vol. 2	This standard is REFERENCED by DOE 5480.4.
Department of Health and Human Services (NIOSH) No. 77-157C	NIOSH Manual of Analytical Methods, Vol. 3	This standard is REFERENCED by DOE 5480.4.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Department of Health and Human Services (NIOSH) No. 78-175	NIOSH Manual of Analytical Methods, Vol. 4	This standard is REFERENCED by DOE 5480.4.
Department of Health and Human Services (NIOSH) No. 79-141	NIOSH Manual of Analytical Methods, Vol. 5	This standard is REFERENCED by DOE 5480.4.
Nuclear Regulatory Commission (NRC)		
Form NRC 396	Certificate of Medical Examination	This standard is REQUIRED by DOE 5480.6.
NUREG-0360	Qualification Criteria to Certify a Package for Air Transport of Plutonium	This standard is REFERENCED by DOE 5480.3.
NUREG-0700	Guidelines for Control Room Design Reviews	This standard is REQUIRED by DOE 6430.1A.
NUREG/CR-2496	Human Engineering Design Considerations for Cathode Ray Tube Generated Displays	This standard is REQUIRED by DOE 6430.1A.
NUREG/CR-3331	A Methodology for Allocating Nuclear Power Plant Control Functions to Human or Automatic Control	This standard is REQUIRED by DOE 6430.1A.
Regulatory Guides	NRC Regulatory Guides (Divisions 1-10)	This standard is REFERENCED by DOE 5480.4. Table B.7 provides the compliance level for each regulatory guide.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Oak Ridge National Laboratory (ORNL)		
NE F 8-6	Hoisting and Rigging of Critical Components and Related Equipment	This standard is REFERENCED by DOE 5480.4.
Office of Management and Budget (OMB)		
OMB Circular A-130	Management of Federal Information Resources	This standard is REQUIRED by DOE 6430.1A.
Portland Cement Association (PCA)		
PCA	Clear Coatings for Exposed Architectural Concrete	This standard is REQUIRED by DOE 6430.1A.
PCA	Painting Concrete	This standard is REQUIRED by DOE 6430.1A.
PCA	Effect of Substances on Concrete and Guide to Protective Treatment	This standard is REQUIRED by DOE 6430.1A.
Prestressed Concrete Institute (PCI)		
PCI MNL-116	Manual for Quality Control for Plants and Production of Precast and Prestressed Concrete Products	This standard is REQUIRED by DOE 6430.1A.
PCI MNL-120	Design Handbook	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Post-Tensioning Institute (PTI)		
PTI	Post-Tensioning Manual	This standard is REQUIRED by DOE 6430.1A.
Rossnagel, W. E.		
Rossnagel, W. E.	Handbook of Rigging	This standard is REFERENCED by DOE 5480.4.
Sandia National Laboratories (SNL)		
SAND 77-0554	Intrusion Detection System Handbook, Vol. I & II	This standard is REQUIRED by DOE 6430.1A.
SAND 77-0777	Barrier Technology Handbook	This standard is REQUIRED by DOE 6430.1A.
SAND 77-1033	Entry Control Systems Handbook	This standard is REQUIRED by DOE 6430.1A.
Sheet Metal and Air Conditioning Contractors National Association (SMACNA)		
SMACNA	Architectural Sheet Metal Manual	This standard is REQUIRED by DOE 6430.1A.
SMACNA	Energy Recovery Equipment and Systems Manual	This standard is REQUIRED by DOE 6430.1A.
SMACNA	High, Medium, and Low Pressure Duct Construction Standards	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
SMACNA	HVAC Air Duct Leakage Test Manual	This standard is REQUIRED by DOE 6430.1A.
SMACNA	HVAC Duct Construction Standards - Metal and Flexible	This standard is REQUIRED by DOE 6430.1A.
SMACNA	HVAC Duct Design Manual	This standard is REQUIRED by DOE 6430.1A.
SMACNA	Round Industrial Duct Construction Standards	This standard is REQUIRED by DOE 6430.1A.
Soil Conservation Service, U.S. Department of Agriculture (SCS)		
SCS	SCS National Engineering Handbook	This standard is REQUIRED by DOE 6430.1A.
Steel Deck Institute (SDI)		
SCI-Publ. 25	Design Manual for Composite Decks, Form Decks and Roof Decks	This standard is REQUIRED by DOE 6430.1A.
SDI-Publ. DDM01	Diaphragm Design Manual	This standard is REQUIRED by DOE 6430.1A.
Steel Door Institute (SDI)		
SDI 100	Standard Steel Doors and Frames	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
SDI 108	Selection and Usage Guide for Standard Steel Doors	This standard is REQUIRED by DOE 6430.1A.
Steel Joist Institute (SJI)		
SJI	Standard Specifications Load Tables and Weight Tables for Steel Joists and Joist Girders	This standard is REQUIRED by DOE 6430.1A.
U.S. Naval Facilities Engineering Command (NAVFAC)		
DM-4.04	Electric Utilization Systems	This standard is REQUIRED by DOE 6430.1A.
DM-4.09	Energy Monitoring and Control Systems	This standard is REQUIRED by DOE 6430.1A.
DM-7.03	Soil Dynamics, Deep Stabilization, and Special Geotechnical Construction	This standard is REQUIRED by DOE 6430.1A.
DM-13.01	Physical Security	This standard is REQUIRED by DOE 6430.1A.

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Underwriters' Laboratories (UL)		
UL	Applicable UL Directories	This standard is REFERENCED by DOE 5480.4 and REQUIRED by MMES ESH-21.
UL	Building Material Directory	This standard is REQUIRED by DOE 6430.1A. This standard is also supported by SRP 9.5.1.
UL 10A	Tin Clad Fire Doors	This standard is REQUIRED by DOE 6430.1A.
UL 155	Tests for Fire Resistance for Vault and File Room Doors	This standard is REQUIRED by DOE 6430.1A.
UL 207	Refrigerant-Containing Components and Accessories, Nonelectrical	This standard is REQUIRED by DOE 6430.1A.
UL 325	Door, Drapery, Louver, and Window Operators and Systems	This standard is REQUIRED by DOE 6430.1A.
UL 508-1983	Industrial Control Equipment	This standard is REQUIRED by DOE 6430.1A.
UL 586-1977	Test Performance of High Efficiency Particulate Air Filter Units	This standard is REQUIRED by DOE 6430.1A.
UL 900-1982 (Also designated as ANSI B124.1-1971)	Test Performance of Air Filter Units	This standard is REQUIRED by DOE 6430.1A and supported by RG 1.52.
UL 1479	Fire Test of Through-Penetration Firestops	This standard is REQUIRED by DOE

Table B.3. Standards Required or Referenced by the Department of Energy

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
UL 1479	Fire Test of Through-Penetration Firestops	This standard is REQUIRED by DOE 6430.1A.
Water Pollution Control Federation (WPCF)		
MOP/8 CTG-77	Wastewater Treatment Plant Design	This standard is REQUIRED by DOE 6430.1A.
Water Resources Council, Hydrology Committee (WRC)		
Bulletin No. 17A	Guidelines for Determining Flood Flow Frequency	This standard is REQUIRED by DOE 6430.1A.
Winterkorn and Fang		
Winterkorn and Fang	Foundation Engineering Handbook (1975)	This standard is REQUIRED by DOE 6430.1A.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE

A review of Title 10, Part 50 of CFR, "Domestic Licensing of Production and Utilization Facilities," (Revised as of January 1, 1991) is given below. Section numbers are read as follows: 50.1 means 10 CFR 50, §50.1. Justification and Guidance is given for each requirement to aid in determining applicability at the SDD level. The guidance given is, in essence, a requirement. Any requests for exceptions to the guidance must be submitted to the Chairman of the Plant Design Requirements Working Group. Each exception will be reviewed on a case-by-case basis.

Definitions required to interpret this table are:

TOTAL — *All technical requirements are applicable as written.*

PARTIAL — *Due to specific requirements in the regulations or design features in ANS, not all of the technical requirements apply as written. The PDR shall define that part of the regulation that is and is not applicable at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

INTENT — *Only the purpose (or intent) of the technical parts of the regulation are applicable. The PDR shall define the intent of the regulation and how it applies to ANS at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

NONE — *Due to specific requirements in the regulations or design features in ANS, none of the technical requirements are applicable. This appendix provides justification why the regulation is not applicable.*

N/A-L — *The regulation sets forth only licensing procedural requirements and does not contain any technical requirements.*

Parts of 10 CFR 50 listed below may contain statements that make it appear to be not-applicable to the design of ANS; yet the compliance commitment level given is TOTAL, PARTIAL, or INTENT. This situation can occur because the regulation defines the criterion and then, in some cases, provides an example. Therefore, statements like "such as steam generators" simply provide guidance and are not part of the criterion.

GENERAL PROVISIONS

50.1	Basis, purpose, and procedures applicable	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.2	Definitions As used in this part, "Act" means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	"Alternate ac source" means an alternating current (ac) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements: (1) Is connectable to but not normally connected to the offsite or onsite emergency ac power systems; (2) Has minimum potential for common mode failure with offsite power or the onsite emergency ac power sources; (3) Is available in a timely manner after the onset of station blackout; and (4) Has sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).		
	"Atomic energy" means all forms of energy released in the course of nuclear fission or nuclear transformation.		
	"Atomic weapon" means any device utilizing atomic energy, exclusive of the means for transporting or propelling the device (where such means is a separable and divisible part of the device), the principal purpose of which is for use as, or for development of, a weapon, a weapon prototype, or a weapon test device.		
	"By-product material" means any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.		
	"Commission" means the Nuclear Regulatory Commission or its duly authorized representatives.		
	"Common defense and security" means the common defense and security of the United States.		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	"Controls" when used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor. "Controls" when used with respect to any other facility means apparatus and mechanisms, the manipulation of which could affect the chemical, physical, metallurgical, or nuclear process of the facility in such a manner as to affect the protection of health and safety against radiation. "Decommission" means to remove (as a facility) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license. "Department" and "Department of Energy" means the Department of Energy established by the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565, 42 U.S.C. 7101 et seq.), to the extent that the department, or its duly authorized representatives, exercises functions formerly vested in the Atomic Energy Commission, its Chairman, members, officers and components and transferred to the U.S. Energy Research and Development Administration and to the Administrator thereof pursuant to sections 104(b), (c) and (d) of the Energy Reorganization Act of 1974 (Pub.L.93-438, 88 Stat. 1233 at 1237, 42 U.S.C. 5814) and retransferred to the Secretary of Energy pursuant to section 301(a) of the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565 at 577-578, 42 U.S.C. 7151). "Design bases" means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. "Electric utility" means any entity that generates or distributes electricity and which recovers the coat of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority. Investor-owned utilities, including generation or distribution subsidiaries, public utility districts, municipalities, rural electric cooperatives, and State and Federal agencies, including associations of any of the foregoing, are included within the meaning of "electric utility." "Federal Government funding for conversion" means funds appropriated to the Department of Energy or to any other Federal Agency to pay directly to or to reimburse non--power reactor licensees for costs attendant to conversion. "Fuel acceptable to the Commission" means that the fuel replacing the existing HEU fuel in a specific non-power reactor (1) meets the operating requirements of the existing license or, through appropriate NRC safety review and approval, can be used in a manner which protects public health and safety and promotes the common defense and security; and (2) meets the Commission's policy of limiting, to the maximum extent possible, the use of HEU fuel in that reactor. "Government agency" means any executive department, commission, independent establishment, corporation,		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	wholly or partly owned by the United States of America which is an instrumentality of the United States, or any board, bureau, division, service, office, officer, authority, administration, or other establishment in the executive branch of the Government. "Highly enriched uranium (HEU) fuel" means fuel in which the weight percent of U-235 in the uranium is 20% or greater. Target material, special instrumentation, or experimental devices using HEU are not included. "Low enriched uranium (LEU) fuel" means fuel in which the weight percent of U-235 in the uranium is less than 20%. "Non-power reactor" means a research or test reactor licensed under 50.21(c) or 50.22 of this part for research and development. "Nuclear reactor" means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self- supporting chain reaction. "Person" means (1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the Commission or the Department, except that the Department shall be considered a person to the extent that its facilities are subject to the licensing and related regulatory authority of the Commission pursuant to section 202 of the Energy Reorganization Act of 1974,1* any State or any political subdivision of, or any political entity within a State, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and (2) any legal successor, representative, agent, or agency of the foregoing.		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	"Produce," when used in relation to special nuclear material, means (1) to manufacture, make, produce, or refine special nuclear material; (2) to separate special nuclear material from other substances in which such material may be contained; or (3) to make or to produce new special nuclear material. "Production facility" means: (1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; or (2) Any facility designed or used for the separation of the isotopes of uranium or the isotopes of plutonium, except laboratory scale facilities designed or used for experimental or analytical purposes only; or (3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, except (i) laboratory scale facilities designed or used for experimental or analytical purposes, (ii) facilities in which the only special nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U-235 and plutonium produced by the irradiation, if the material processed contains not more than 10E-6 grams of plutonium per gram of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235, and (iii) facilities in which processing is conducted pursuant to a license issued under Parts 30 and 70 of this chapter, or equivalent regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear material, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to not more than 100 grams of uranium enriched in the isotope 235 and not more than 15 grams of any other special nuclear material. "Reactor coolant pressure boundary" means all those		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	pressure- containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are: (1) Part of the reactor coolant system, or (2) Connected to the reactor coolant system, up to and including any and all of the following: (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment, (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment, (iii) The reactor coolant system safety and relief valves. For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping. "Research and development" means (1) theoretical analysis, exploration, or experimentation; or (2) the extension of investigative findings and theories of a scientific or technical nature into practical application for experimental and demonstration purposes, including the experimental production and testing of models, devices, equipment, materials, and processes. "Restricted Data" means all data concerning (1) design, manufacture, or utilization of atomic weapons; (2) the production of special nuclear material; or (3) the use of special nuclear material in the production of energy, but shall not include data declassified or removed from the Restricted Data category pursuant to section 142 of the Act. "Safe shutdown (non-design basis accident (non-DBA))" for station blackout means bringing the plant to those shutdown conditions specified in plant technical specifica-		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	tions as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures). "Source material" means source material as defined in subsection 11z. of the Act and in the regulations contained in Part 40 of this chapter. "Special nuclear material" means (1) plutonium, uranium--233, uranium enriched in the isotope-233 or in the isotope--235, and any other material which the Commission, pursuant to the provisions of section 51 of the act, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material. "Station blackout" means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident. At single unit sites, any emergency ac power source(s) in excess of the number required to meet minimum redundancy requirements (i.e., single failure) for safe shutdown (non-DBA) is assumed to be available and may be designated as an alternate power source(s) provided the applicable requirements are met. At multi-unit sites, where the combination of emergency ac power sources exceeds the minimum redundancy requirements for safe shutdown (non-DBA) of all units, the remaining emergency ac power sources may be used as alternate ac power sources provided they meet the applica-		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.2 continued	ble requirements. If these criteria are not met, station blackout must be assumed on all the units. "Testing facility" means a nuclear reactor which is of a type described in 50.21(c) of this part and for which an application has been filed for a license authorizing operation at: (1) A thermal power level in excess of 10 megawatts; or (2) A thermal power level in excess of 1 megawatt, if the reactor is to contain: (i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments; or (ii) A liquid fuel loading; or (iii) An experimental facility in the core in excess of 16 square inches in cross-section. "Unique purpose" means a project, program, or commercial activity which cannot reasonably be accomplished without the use of HEU fuel, and may include: (1) A specific experiment, program, or commercial activity (typically long-term) that significantly serves the U.S. national interest and cannot be accomplished without the use of HEU fuel; (2) Reactor physics or reactor development based explicitly on the use of HEU fuel; (3) Research projects based on neutron flux levels or spectra attainable only with HEU fuel; or (4) A reactor core of special design that could not perform its intended function without using HEU fuel. "United States", when used in a geographical sense, includes Puerto Rico and all territories and possessions of the United States.		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
	"Utilization facility" means any nuclear reactor other than one designed or used primarily for the formation of plutonium or U-233.		
50.3	Interpretations.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.4	Written communications.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.7	Employee protection	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.8	Information collection requirements: OMB approval	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.9	Completeness and accuracy of information	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
REQUIREMENT OF LICENSE, EXCEPTIONS			
50.10	License required	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.11	Exceptions and exemptions from licensing requirements	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.12	Specific exemptions	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.13	Attacks and destructive acts by enemies of the United States; and defense activities	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
CLASSIFICATION AND DESCRIPTION OF LICENSES			
50.20	Two classes of licenses	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.21	Class 104 licenses; for medical therapy and research and development facilities	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.22	Class 103 licenses; for commercial and industrial facilities	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.23	Construction permits	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
APPLICATIONS FOR LICENSES, FORM, CONTENTS, INELIGIBILITY OF CERTAIN APPLICANTS			
50.30	Filing of applications for licenses; oath or affirmation (f) Environmental report. An application for a construction permit or an operating license for a nuclear power reactor, testing facility, fuel reprocessing plant, or such other production or utilization facility whose	INTENT	Only the INTENT of 30(f) is applicable. The ANS will produce an Environmental Report.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.30 continued	construction or operation may be determined by the Commission to have a significant impact on the environment shall be accompanied by any Environmental Report required pursuant to Subpart A of Part 51 of this chapter.		
50.31	Combining applications	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.32	Elimination of repetition	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.33	Contents of applications; general information (g) If the application is for an operating license for a nuclear power reactor, the applicant shall submit radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within the plume exposure pathway Emergency Planning Zone (EPZ)*, as well as the plans of State governments wholly or partially within the ingestion pathway EPZ. Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to the local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such	PARTIAL	Only 33(g) is applicable. Emergency Planning Zones (EPZs) are discussed in NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants," December 1978.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.33 continued	actions as are appropriate to protect the food ingestion pathway.		
50.33a	Information requested by the Attorney General for antitrust review	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.34	Contents of applications; technical information		
50.34(a)	Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:	PARTIAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70. Each subsection of 50.34(a) is addressed below. Information required by this paragraph may be in the form of a discussion, with specific references to similarities to and differences from facilities of similar design for which applications have previously been filed.
50.34(a)(1)	A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in Part 100 of this chapter. Such assessment shall contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in Part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(a)(2)	A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(3)	The preliminary design of the facility including: (i) The principal design criteria for the facility. Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units; (ii) The design bases and the relation of the design bases to the principal design criteria; (iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(4)	A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of (i) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (ii) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 of this part for facilities for which construction permits may be issued after December 28, 1974.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(a)(5)	An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design: Provided, however, That this requirement is not applicable to an application for a construction permit filed prior to January 16, 1969.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(6)	A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(7)	A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for quality assurance programs for nuclear power plants and fuel reprocessing plants. The description of the quality assurance program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(8)	An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(a)(9)	The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(10)	A discussion of the applicant's preliminary plans for coping with emergencies. Appendix E sets forth items which shall be included in these plans.	TOTAL	Requirements from 50.34(a) apply to the PSAR and may impact the design. The CSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(a)(11)	On or after February 5, 1979, applicants who apply for construction permits for nuclear powerplants to be built on multiunit sites shall identify potential hazards to the structures, systems and components important to safety of operating nuclear facilities from construction activities. A discussion shall also be included of any managerial and administrative controls that will be used during construction to assure the safety of the operating unit.	NONE	ANS will be the only facility on the site.
50.34(b)	Final safety analysis report. Each application for a license to operate a facility shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:	PARTIAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70. Each subsection of 50.34(b) is addressed below.
50.34(b)(1)	All current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issuance of the construction permit, relating to site evaluation factors identified in Part 100 of this chapter.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(2)	A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
	be sufficient to permit understanding of the system designs and their relationship to safety evaluations.		
50.34(b)(2)(i)	For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(2)(ii)	For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(3)	The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(4)	A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of 50.46 for facilities for which a license to operate may be issued after December 28, 1974.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(b)(5)	A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(6)(i)	The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(6)(ii)	Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(6)(iii)	Plans for preoperational testing and initial operations.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(6)(iv)	Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(6)(v)	Plans for coping with emergencies, which shall include the items specified in Appendix E.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(6)(vi)	Proposed technical specifications prepared in accordance with the requirements of 50.36.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(b)(6)(vii)	On or after February 5, 1979, applicants who apply for operating licenses for nuclear power plants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites.	NONE	The ANS will be the only facility on the site.
50.34(b)(7)	The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(8)	A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in 55.59 of Part 55 of this chapter.	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(b)(9)	A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 50.61 (b)(1) and (b)(2).	TOTAL	Requirements from 50.34(b) apply to the FSAR and may impact the design. The CSAR and PSAR will comply if possible. DOE 5480.6 requires the SAR format to follow RG 1.70.
50.34(c)	Physical security plan. Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 (and Part 11 of this chapter, if applicable, including the identification and description of jobs as required by 11.11(a), at the proposed facility). Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(d)	Safeguards contingency plan. Each application for a license to operate a production or utilization facility that will be subject to 73.50, 73.55, or 73.60 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information as set forth in Appendix C to 10 CFR Part 73 are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)	TOTAL	A physical security plan that contains all the information required in both § 73.55 and Appendix C to Part 73 satisfies the requirement for a contingency plan.
50.34(e)	Each applicant for a license to operate a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of 73.21 of this chapter, as appropriate.	TOTAL	The physical security plans, safeguards contingency plans, or guard qualification plans will be protected.
50.34(f)	Additional TMI-related requirements. In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982 shall meet the requirements in paragraphs (f) (1) through (3) of this section. This rule applies only to the pending applications by Duke Power Company (Perkins Nuclear Station Units 1, 2 and 3), Houston Lighting & Power Company (Allens Creek Nuclear Generating Station, Unit 1), Portland General Electric Company (Pebble Springs Nuclear Plant, Units 1	PARTIAL	Each subsection of 50.34(f) is addressed below. Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
	and 2), Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Puget Sound Power & Light Company (Skagit/Hanford Nuclear Power Project, Units 1 and 2), and Offshore Power Systems (License to Manufacture Floating Nuclear Plants). The number of units that will be specified in the manufacturing license, if issued, will be that number whose start of manufacture, as defined in the license application, can practically begin within a ten-year period commencing on the date of issuance of the manufacturing license, but in no event will that number be in excess of ten. The manufacturing license will require the plant design to be updated no later than five years after its approval. Paragraphs (f)(1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this rule. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and Criterion 50 of Appendix A to 10 CFR Part 50, is sufficient for issuance of the manufacturing license may be considered in the manufacturing license proceeding.		
50.34(f)(1)	To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license.	TOTAL	Alphanumeric designations correspond to the related action plan items in NUREG 0718 and NUREG 0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." They are provided herein for information only.
50.34(f)(1)(i)	Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only) (II.E.1.1): (A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques. (B) A design review of AFWS. (C) An evaluation of AFWS flow design bases and criteria.	INTENT	The INTENT of this subsection will be applied to the Essential Cooling Water System.
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)	INTENT	Seal damage following small-break loss-of-coolant accident with loss of off-site power will be evaluated for all primary coolant pumps.
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)	INTENT	Even though ANS does not have a HPCI, RCIC, or comparable system, the effects of a stuck-open relief valve will be analyzed.
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.13)	NONE	ANS does not have a HPCI, RCIC, or comparable system.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)	TOTAL	
50.34(f)(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)	NONE	ANS does not have an ADS or comparable system.
50.34(f)(1)(viii)	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)	NONE	ANS does not have a core spray, LPCI, or comparable system.
50.34(f)(1)(ix)	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.24)	NONE	ANS does not have a RCIC, HPCI, or comparable system.
50.34(f)(1)(x)	Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)	NONE	ANS does not have an ADS or comparable system.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(1)(xi)	Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)	PARTIAL	ANS does not have an ADS or comparable system. Nevertheless, depressurization methods will be evaluated.
50.34(f)(1)(xii)	Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include: (A) A comparison of costs and benefits of the alternative systems considered. (B) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section. (C) For the selected system, preliminary design descriptions of equipment, function, and layout.	TOTAL	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.
50.34(f)(2)(i)	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's (I.A.4.2.)	TOTAL	An operator training simulator shall provide real-time interactive simulation of normal operations and accidents. The control consoles shall be identical to those in the main control room.
50.34(f)(2)(ii)	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts (I.C.9)	TOTAL	
50.34(f)(2)(iii)	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)	INTENT	Human factors shall be considered in the design of ANS systems and components that involve interface with plant personnel, including operator, engineers, and management, during all activities including operations, maintenance, testing, and inservice inspection. The local and remote control panels and displays shall be designed to enhance man-machine interfaces.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(2)(iv)	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)	TOTAL	
50.34(f)(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)	TOTAL	
50.34(f)(2)(vi)	Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)	TOTAL	A system to collect and/or vent noncondensible gases from the high points of the reactor cooling and reflector cooling loops shall be provided. The safety function of this system is to assure that adequate reactor forced and natural circulation cooling is maintained in event of release of non-condensible gases into the reactor coolant. Sources of noncondensible gases include those resulting from gases dissolved in coolant water, fuel-water reactions, and failed experiments. Any active vent systems required to achieve this safety function shall be capable of being operated from the control room. Operation of any required vent valves shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. See also Table B.6, SRP 5.4.12. The vent system design shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of the vent.
50.34(f)(2)(vii)	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)	INTENT	Because the source term for ANS is much smaller than that for a power reactor, the intent of TID-14844 will be followed. The source terms for ANS are given in Appendix C of the PDR.
50.34(f)(2)(viii)	Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the	INTENT	The methodology for evaluating the design basis loss of coolant accidents is based on TID-14844 to determine compliance with the siting requirements of 10CFR100. Because the source term for ANS is much smaller than that for a power reactor, the intent of TID-14844 will be followed. Source terms for ANS are given in Appendix C of the PDR.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
	extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)		
50.34(f)(2)(ix)	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8) (A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion. (B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features. (C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.	TOTAL	Combustible gas-control systems must consider all combustible gases (hydrogen, deuterium, CO, etc.) and should be passive systems. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, and the cold source inventory.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
	(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.		
50.34(f)(2)(x)	Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)	TOTAL	
50.34(f)(2)(xi)	Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)	TOTAL	
50.34(f)(2)(xii)	Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)	NONE	ANS does not have an AFW or comparable system.
50.34(f)(2)(xiii)	Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)	NONE	ANS does not have a pressurizer or comparable equipment.
50.34(f)(2)(xiv)	Provide containment isolation systems that: (II.E.4.2)	TOTAL	
	(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,		
	(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(2)(xiv) continued	(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.		
50.34(f)(2)(xv)	Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)	NONE	This requirement concerns operating plants that purge containment during operation.
50.34(f)(2)(xvi)	Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)	*[TBD]*[B-12]	
50.34(f)(2)(xvii)	Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)	TOTAL	Instrumentation should be provided to measure, record, and read out in the control room the combustible gas concentration in containment. Combustible gases include hydrogen, deuterium, CO, etc.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(2)(xviii)	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)	TOTAL	
50.34(f)(2)(xix)	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)	TOTAL	
50.34(f)(2)(xx)	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)	NONE	ANS does not have a pressurizer, motor controls on block valves in the primary system, level indicators, or comparable system.
50.34(f)(2)(xxi)	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)	NONE	ANS does not have an AFW or comparable system.
50.34(f)(2)(xxii)	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (II.K.2.9)	INTENT	A Failure Modes and Effects Analysis will be performed on the control system.
50.34(f)(2)(xxiii)	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (II.K.2.10)	NONE	ANS does not have a feed-water system, turbine, or comparable system.

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SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(2)(xxiv)	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)	PARTIAL	ANS should have water-level indicators in the primary accumulators, pools, cells, etc.
50.34(f)(2)(xxv)	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2).	TOTAL	NUREG-0696 provides guidance on these response facilities.
50.34(f)(2)(xxvi)	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)	INTENT	Because the source term for ANS is much smaller than that for a power reactor, the intent of TID-14844 will be followed. The source terms for ANS are given in Appendix C of the PDR.
50.34(f)(2)(xxvii)	Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)	TOTAL	
50.34(f)(2)(xxviii)	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. (III.D.3.4)	INTENT	Because the source term for ANS is much smaller than that for a power reactor, the intent of TID-14844 will be followed. The source terms for ANS are given in Appendix C of the PDR.
50.34(f)(3)(i)	Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(3)(ii)	Ensure that the quality assurance (QA) list required by Criterion II, App. B, 10 CFR Part 50 includes all structures, systems, and components important to safety. (I.F.1)	TOTAL	
50.34(f)(3)(iii)	Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)	TOTAL	
50.34(f)(3)(iv)	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)	TOTAL	
50.34(f)(3)(v)	Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8) (A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required,	PARTIAL	The internal design pressure of ANS is 69 kPa(g) (10 psig) rather than 310 kPa(g) (45 psig) as specified. In addition, ANS does not have an inerting system. Accidents to be evaluated should include combustible gases from fuel-metal reactions, core-concrete reactions, and cold source inventory.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(3)(v) continued	considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions. [A](2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. (B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(f)(3)(v) continued	liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category, [B](2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.		
50.34(f)(3)(vi)	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)	TOTAL	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible-gas" control.
50.34(f)(3)(vii)	Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)	TOTAL	
50.34(g)	Conformance with the Standard Review Plan (SRP)		
50.34(g)(1)(i)	Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34(g)(1)(ii)	Applications for light water cooled nuclear power plant construction permits, manufacturing licenses, and preliminary or final design approvals for standard plants docketed after May 17, 1982 shall include an evaluation of the facility against the SRP in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.	TOTAL	
50.34(g)(2)	The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.	TOTAL	
50.34(g)(3)	The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.	TOTAL	
50.34a	Design objectives for equipment to control releases of radioactive material in effluents-nuclear power reactors	TOTAL	
50.34a(a)	An application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34a(a) continued	application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term 'X' as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest. The guides set out in Appendix I to this part provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirements that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.		
50.34a(b)	Each application for a permit to construct a nuclear power reactor shall include: (1) A description of the preliminary design of equipment to be installed pursuant to paragraph (a) of this section; (2) An estimate of: (i) The quantity of each of the principal radio-nuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations; and (ii) The quantity of each of the principal radio-nuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal reactor operations. (3) A general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.34a(c)	Each application for a license to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, pursuant to paragraph (a) of this section; an (2) a revised estimate of the information required in paragraph (b)(2) of this section if the expected releases and exposures differ significantly from the estimates submitted in the application for a construction permit.	TOTAL	
50.35	Issuance of construction permits	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.36	Technical specifications		Required by DOE 5480.6. The technical specifications shall be defined in the SDDs and in the SAR in accordance with the requirements set forth in this Part of 10 CFR 50.
50.36(a)	Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.	TOTAL	
50.36(b)	Each license authorizing operation of a production or utilization facility of a type described in 50.21 or 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36(c)	Technical specifications will include items in the following categories:		
50.36(c)(1)	Safety limits, limiting safety system settings, and limiting control settings.	TOTAL	
50.36(c)(1)(i)(A)	Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under Sec. 50.21(b) or Sec. 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by Sec. 50.72 and submit a Licensee Event Report to the Commission as required by Sec. 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.	TOTAL	If any safety limit is exceeded, the reactor will be shut down.
50.36(c)(1)(i)(B)	Safety limits for fuel reprocessing plants are those bounds within which the process variables must be maintained for adequate control of the operation and that must not be exceeded in order to protect the integrity of the physical system that is designed to guard against the uncontrolled release or radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action must be taken as stated in the technical specification or the affected part of the process, or the entire process if required, must be shut down, unless this action would further reduce the margin of safety. The licensee shall	NONE	These limits apply to fuel reprocessing plants.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36(c)(1)(i)(B) continued	notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. If a portion of the process or the entire process has been shutdown, operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.		
50.36(c)(1)(ii)(A)	Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor except for nuclear power reactors licensed under Sec. 50.21(b) or Sec. 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by Sec. 50.72 and submit a Licensee Event Report to the Commission as required by Sec. 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.	TOTAL	
50.36(c)(1)(ii)(B)	Limiting control settings for fuel reprocessing plants are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting must	NONE	These limits apply to fuel reprocessing plants.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
	be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic alarm or protective devices do not function as required, the licensee shall take appropriate action to maintain the variables within the limiting control-setting values and to repair promptly the automatic devices or to shut down the affected part of the process and, if required, to shut down the entire process for repair of automatic devices. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.		
50.36(c)(1)(ii)(2)	Limiting conditions for operation. Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met. In the case of a nuclear reactor not licensed under Sec. 50.21(b) or Sec. 50.22 of this part or fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of nuclear power reactors licensed under Sec. 50.21(b) or	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36(c)(1)(ii)(2) continued	Sec. 50.22, the licensee shall notify the Commission if required by Sec. 50.72 and shall submit a Licensee Event Report to the Commission as required by Sec. 50.73. In this case, licensees shall retain records associated with preparation of a Licensee Event Report for a period of three years following issuance of the report. For events which do not require a Licensee Event Report, the licensee shall retain each record as required by the technical specifications.		
50.36(c)(3)	Surveillance requirements. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met.	TOTAL	
50.36(c)(4)	Design features. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.	TOTAL	
50.36(c)(5)	Administrative controls. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.	TOTAL	
50.36(c)(6)	Initial notification. Reports made to the Commission by licensees in response to the requirements of this section must be made as follows: (i) Licensees that have an installed Emergency Notification System shall make the initial notification to the NRC Operations Center in accordance with §50.72 of this part.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36(c)(6)	(ii) All other licensees shall make the initial notification by telephone to the Administrator of the appropriate NRC Regional Office listed in Appendix D, Part 20, of this chapter.		
50.36(c)(7)	Written Reports. Licensees for nuclear power reactors licensed under Sec.50.21(b) and Sec. 50.22 of this part shall submit written reports to the Commission in accordance with Sec. 50.73 of this part for events described in paragraphs (c)(1) and (c)(2) of this section. For all licensees, the Commission may require Special Reports as appropriate.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.36(d)(1)	This section shall not be deemed to modify the technical specifications included in any license issued prior to January 16, 1969. A license in which technical specifications have not been designated shall be deemed to include the entire safety analysis report as technical specifications.	TOTAL	
50.36(d)(2)	An applicant for a license authorizing operation of a production or utilization facility to whom a construction permit has been issued prior to January 16, 1969, may submit technical specifications in accordance with this section, or in accordance with the requirements of this part in effect prior to January 16, 1969.	NONE	This subsection applies to facilities that received a construction permit prior to January 16, 1969.
50.36(3)	At the initiative of the Commission or the licensee, any license may be amended to include technical specifications of the scope and content which would be required if a new license were being issued.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.36a	Technical specifications on effluents from nuclear power reactors		The Technical Specifications shall be defined in the SDD's and in the SAR in accordance with the requirements set forth in this Part of 10 CFR 50.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36a(a)	In order to keep releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable, each license authorizing operation of a nuclear power reactor will include technical specifications that, in addition to requiring compliance with applicable provisions of § 20.106 of this chapter, require: (1) That operating procedures developed pursuant to Sec. 50.34a(c) for the control of effluents be established and followed and that equipment installed in the radioactive waste system, pursuant to Sec. 50.34(a), be maintained and used. The licensee shall retain the operating procedures in effect as a record until the Commission terminates the reactor license and shall retain each superseded revision of the procedures for three years from the date it was superseded. (2) The submission of a report to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter within sixty (60) days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous six (6) months of operation, and such other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. Copies of such report shall be sent to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36a(b)	In establishing and implementing the operating procedures described in paragraph (a) of this section, the licensee shall be guided by the following considerations: Experience with the design, construction and operation of nuclear power reactors indicates that compliance with the technical specifications described in this section will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in § 20.106 of this chapter and in the operating license. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in § 20.106 of this chapter and the operating license. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable. The guides set out in Appendix I provide numerical guidance on limiting conditions for operation for light-water-cooled nuclear power reactors to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.	PARTIAL	ANS will not produce electric power.
50.36b	Environmental conditions. Each license authorizing operation of a production or utilization facility which is of a type described in § 50.21(b) (2) or (3) or § 50.22 or is a testing facility may include conditions to protect the environment to be set out in an attachment to the license which is incorporated in and made a part of the license. These conditions will be derived from information contained in the environmental report and the supplement to the environmental report submitted pursuant to § 51.50 and 51.53 of this chapter as analyzed and evaluated in the NRC record of decision, and will identify the obligations of the licensee in the environmental area, including, as	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.36b continued	appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirements for the protection of the nonaquatic environment.		
50.37	Agreement limiting access to Restricted Data. As part of his application and in any event prior to the receipt of Restricted Data or the issuance of a license or construction permit, the applicant shall agree in writing that he will not permit any individual to have access to Restricted Data until the Civil Service Commission shall have made an investigation and report to the Commission on the character, associations, and loyalty of such individual, and the Commission shall have determined that permitting such person to have access to Restricted Data will not endanger the common defense and security. The agreement of the applicant in this regard shall be deemed part of the license or construction permit, whether so stated therein or not.	TOTAL	
50.38	Ineligibility of certain applicants	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.39	Public inspection of applications	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
STANDARDS FOR LICENSES AND CONSTRUCTION PERMITS			
50.40	Common standards	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.41	Additional standards for class 104 licenses	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.42	Additional standards for class 103 licenses	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.43	Additional standards and provisions affecting class 103 licenses for commercial power	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.44	Standards for combustible gas control system in light-water-cooled power reactors	PARTIAL	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc.
50.44(a)	Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, shall, as provided in paragraphs (b) through (d) of this section, include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA), by (1) Metal-water reaction involving the fuel cladding and the reactor coolant, (2) Radiolytic decomposition of the reactor coolant, and (3) Corrosion of metals.	INTENT	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.
50.44(b)	Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding shall be provided with the capability for (1) Measuring the hydrogen concentration in the containment, (2) Insuring a mixed atmosphere in the containment, and (3) Controlling combustible gas concentrations in the containment following a postulated LOCA.	INTENT	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.
50.44(c)(1)	For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either: (i) An uncontrolled hydrogen-oxygen recombination	INTENT	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(1) continued	would not take place in the containment; or (ii) The plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function.		
50.44(c)(2)	If the conditions set out in paragraph (c)(1) of this section cannot be shown, the containment shall be provided with an inerted or an oxygen deficient atmosphere in order to provide protection against hydrogen burning and explosions during the time period specified in paragraph (c)(1) of this section.	TOTAL	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.
50.44(c)(3)	Notwithstanding paragraphs (c)(1) and (c)(2) of this section:		
50.44(c)(3)(i)	Effective May 4, 1982 or 6 months after initial criticality, whichever is later, an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment; and	NONE	Because of the access requirements of ANS, containment cannot be inerted.
50.44(c)(3)(ii)	By the end of the first scheduled outage beginning after July 5, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor that relies upon a purge/repressurization system as the primary means for controlling combustible gases following a LOCA shall be provided with either an internal recombiner or the capability to install an external recombiner following the start of an accident. The internal or external recombiners must meet the combustible gas control requirements in paragraph (d) of this section. The containment penetrations used for external recombiners must either be: (A) Dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, be designed against postulated single failures for containment isolation purposes, and be sized to satisfy the flow requirements of the external recombiners, or	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(ii) continued	(B) Of a combined design for use by either external recombiners or purge/repressurization systems and other systems, conform to the requirements of criteria 54 and 56 of Appendix A of this part, be designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/repressurization systems, and be sized to satisfy the flow requirements of the external recombiners or purge repressurization systems.		
50.44(c)(3)(iii)	To provide improved operational capability to maintain adequate core cooling following an accident, by the end of the first scheduled outage beginning after July 1, 1982 and of sufficient duration to permit required modifications, each light- water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. (High point vents are not required, however, for the tubes in U-tube steam generators.) The high point vents must be remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B of this part. In particular, the vent system shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident.	TOTAL	
50.44(c)(3)(iv)(A)	Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(iv)(A) continued	construction permit before March 28, 1979, shall provide its nuclear power reactor with a hydrogen control system justified by a suitable program of experiment and analysis. The hydrogen control system must be capable of handling without loss of containment structural integrity an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).		
50.44(c)(3)(iv)(B)	Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on. Another method could include a showing that the following specific criteria of the ASME Boiler and Pressure Vessel Code are met: (1) That steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in § 50.55a(b)(1) of this part), specifically in Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required); and (2) That concrete containments meet the requirements of the ASME Boiler and Pressure vessel Code, Section III, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(iv)(C)	Subsubarticle NE-3220, Division 1, and Subsubarticle CC-3720, Division 2, of Section III of the ASME Boiler and Pressure Vessel Code, referenced in paragraphs (c)(3)(iv)(B)(1) and (c)(3)(iv)(B)(2) of this section, have been approved for incorporation by reference by the	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(iv)(C) continued	Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L Street NW., Washington, D.C.		
50.44(c)(3)(iv)(D)	If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the increased pressure: (1) During the accident, where it is acceptable to show that it does not exceed Service Level C Limits or the Factored load Category (as described in paragraph (c)(3)(iv)(B) of this section; and (2) Following inadvertent full inerting during normal plant operations, where it is acceptable to show that it does not exceed either the Service Level A Limits of Subsubarticle NE-3220 (for a steel containment) or the Service Load Category of Subsubarticle CC-3720 (for a concrete containment.) (3) Modest deviations from the criteria in paragraph (c)(3)(iv)(D) of this section will be considered by the Commission if good cause is shown.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(iv)(E)	If the hydrogen control system relies on post-accident inerting, the systems and components required to establish and maintain safe shutdown and containment integrity must be designed and qualified for the environment caused by such inerting. Furthermore, inadvertent full inerting during normal plant operations must not adversely affect systems and components needed for safe operation of the plant.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(v)(A)	Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, for a reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment, shall provide its nuclear power reactor with systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity. These systems and components must be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(v)(B)	The amount of hydrogen to be considered is equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(vi)(A)	Each applicant for or holder of an operating license for a boiling light-water nuclear power reactor with a Mark III type of containment or for a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall submit an analysis to the Director of the Office of Nuclear Reactor Regulation.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(vi)(B)	The analysis required by paragraph (c)(3)(vi)(A) of this section must:	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(vi)(B)(1)	Provide an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75% of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include consideration of hydrogen control measures as appropriate;	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(vi)(B)(2)	Include the period of recovery from the degraded condition;	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(vi)(B)(3)	Use accident scenarios that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(vi)(B)(4)	Support the design of the hydrogen control system selected under paragraph (c)(3)(iv) of this section; and,	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.
50.44(c)(3)(vi)(B)(5)	Show that for those reactors described in paragraph (c)(3)(iv) of this section that do not rely upon an inerted atmosphere to control hydrogen inside the containment: (i) The containment structural integrity as described in paragraph (c)(3)(iv) of this section will be maintained; and (ii) Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including the effect of local detonations, unless such detonations can be shown unlikely to occur.	NONE	50.44(c)(3) (iv), (v), and (vi) apply to BWRs with a Mark III containment and PWRs with an ice-condenser containment. ANS will comply with 50.44(b), (d), (e), and (h) for evaluating combustible gases, fuel-clad interactions, etc.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(vii)(A)	By June 25, 1985, each applicant for or holder of an operating license subject to the requirements of paragraphs (c)(3) (iv), (v) and (vi) of this section shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a proposed schedule for meeting these requirements. The schedule may be developed using integrated scheduling systems previously approved for the facility by the NRC.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.44(c)(3)(vii)(B)	For each applicant for an operating license as of February 25, 1985, the schedule shall provide for compliance with the requirements of paragraph (c)(3)(iv)(A) of this section prior to operation of the reactor in excess of 5 percent power. Completed final analyses are not necessary for a staff determination that a plant is safe to operate at full power provided that prior to such operation an applicant has provided a preliminary analysis which the staff has determined provides a satisfactory basis for a decision to support interim operation at full power until the final analysis has been completed. However, the record in this rulemaking shows that such preliminary analyses are not necessary for a staff determination that a plant is safe to operate at full power if the staff has determined for similar plants, referenced in this notice of rulemaking, that similar systems provide a satisfactory basis for a decision to support operation at full power until the preliminary analyses have been completed.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.44(c)(3)(vii)(C)	For those holders of operating licenses containing license conditions on Hydrogen Control Measures covered by this section, the schedule shall be consistent with those license conditions, or approved amendments thereto.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.44(c)(3)(vii)(D)	For those facilities not having an NRC approved integrated scheduling system, a final schedule for meeting the requirements of paragraphs (c)(3) (iv), (v) and (vi) of this section shall be established by the NRC staff within 90 days of receipt of a proposed schedule from the licensee	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(c)(3)(vii)(D) continued	or applicant, taking into account the current status of efforts at the facility to comply with the requirements; analyses that may be provided by applicants or licensees regarding the impacts of these requirements on other scheduled plant modifications, including any NRC-mandated safety modifications, and their relative importance to safety; and the Commission's objective that these requirements be complied with without undue delay.		
50.44(d)(1)	For facilities that are in compliance with § 50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with § 50.46(b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch (0.0058 mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs.	TOTAL	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.
50.44(d)(2)	For facilities as to which no evaluation of compliance in accordance with § 50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reaction of 5 percent of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.	NONE	ANS will comply with the INTENT of 50.46(b).
50.44(e)	For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970, purging and/or repressurization shall not be the primary means for controlling combustible gases following a LOCA. However, the capability for	TOTAL	Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(e) continued	controlled purging shall be provided. For these facilities, the primary means for controlling combustible gases following a LOCA shall consist of a combustible gas control system, such as recombiners, that does not result in a significant release from containment.		
50.44(f)	For facilities with respect to which the notice of hearing on the application for a construction permit was published between December 22, 1968, and November 5, 1970, if the incremental radiation dose from purging (and repressurization if a repressurization system is provided) occurring at all points beyond the exclusion area boundary after a postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 2.5 rem to the whole body and less than 30 rem to the thyroid, and if the combined radiation dose at the low population zone outer boundary from purging and the postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, the purge system shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.	NONE	This subsection is applicable to facilities with an application for a construction permit published between December 22, 1968, and November 5, 1970.
50.44(g)	For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary	NONE	This subsection is applicable to facilities with an application for a construction permit published on or before December 22, 1968.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(g) continued	from purging (and repressurization if a repressurization system is provided) and the postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise, the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.		
50.44(h)	As used in this section: (1) Degradation, but not total failure, of emergency core cooling functioning means that the performance of the emergency core cooling system is postulated, for purposes of design of the combustible gas control system, not to meet the acceptance criteria in § 50.46 and that there could be localized clad melting and metal-water reaction to the extent postulated in paragraph (d) of this section. The degree of performance degradation is not postulated to be sufficient to cause core meltdown. (2) A combustible gas control system is a system that operates after a LOCA to maintain the concentrations of combustible gases within the containment, such as hydrogen, below flammability limits. Combustible gas control systems are of two types: (i) Systems that allow controlled release from containment, through filters if necessary, such as purging systems and repressurization systems, and (ii) systems that do not result in a significant release from containment such as recombiners.	INTENT	The vent system design should be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of the vent. The combustible gas control system shall provide capability for measuring the combustible concentration in containment, insuring a mixed atmosphere in containment, and controlling combustible gas concentrations in the containment.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.44(h) continued	(3) A purging system is a system for the controlled release of the containment atmosphere to the environment through filters if needed. (4) A repressurization system is a system used to dilute the concentration of combustible gas within containment by adding inert gas or air to the containment. Dilution of the combustible gas results in a delay in time until a flammable concentration is reached and permits fission product decay. Operation is limited to a containment repressurization to 50 percent of the containment design pressure. A purging system is normally part of the repressurization system.		
50.45	Standards for construction permits. An applicant for a license or an amendment of a license who proposes to construct or alter a production or utilization facility will be initially granted a construction permit, if the application is in conformity with and acceptable under the criteria of § 50.31 through 50.38 and the standards of § 50.40 through 50.43.	TOTAL	See § 50.31 through 50.38 and 50.40 through 50.43.
50.46	Acceptance criteria for emergency core cooling systems for light water nuclear power reactors.	INTENT	At a minimum, the INTENT of 50.46 will be satisfied although ANS fuel limits are different since ANS uses silicide fuel clad in aluminum rather than uranium oxide pellets clad in zircaloy.
50.46(a)(1)(i)	Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding must be provided with an emergency core cooling system (ECCS) that must be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided	*[TBD]*[B-14]	ANS may not be able to comply with the criteria set forth in 50.46(b).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.46(a)(1)(i) continued	in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II, Required Documentation, sets forth the documentation requirements for each evaluation model.		
50.46(a)(1)(ii)	Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models.	TOTAL	
50.46(a)(2)	The Director of Nuclear Reactor Regulations may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1) (i) and (ii) of this section.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.46(a)(3)(i)	Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50 deg. F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.46(a)(3)(i) continued	magnitudes of the respective temperature changes is greater than 50 deg. F.		
50.46(a)(3)(ii)	For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in Sec. 50.4. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with Sec. 50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in Secs. 50.55(e), 50.72 and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with Sec. 50.46 requirements.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.46(b)(1)	Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200x F.	*[TBD]*[B-15]	
50.46(b)(2)	Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide.	*[TBD]*[B-16]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.46(b)(2) continued	If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.		
50.46(b)(3)	Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.	*[TBD]*[B-17]	
50.46(b)(4)	Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.	TOTAL	
50.46(b)(5)	Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.	TOTAL	
50.46(c)	As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.46(c) continued	capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system; (2) An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.		
50.46(d)	The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A.	TOTAL	
50.47	Emergency plans.	PARTIAL	ANS will comply with the requirements for on-site and off-site emergency preparedness.
50.47(a)(1)	Except as provided in paragraph (d) of this section, no operating license for a nuclear power reactor will be issued unless a finding is made by NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.	TOTAL	ANS will provide reasonable assurance to DOE that adequate protective measures can and will be taken in the event of a radiological emergency.
50.47(a)(2)	The NRC will base its finding on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(a)(2) continued	plans are adequate and whether there is reasonable assurance that they can be implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented. A FEMA finding will primarily be based on a review of the plans. Any other information already available to FEMA may be considered in assessing whether there is reasonable assurance that the plans can be implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on questions of adequacy and implementation capability.		
50.47(b)	The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:		
50.47(b)(1)	Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.	TOTAL	
50.47(b)(2)	On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.	TOTAL	
50.47(b)(3)	Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made,	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(b)(3) continued	and other organizations capable of augmenting the planned response have been identified.		
50.47(b)(4)	A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.	TOTAL	
50.47(b)(5)	Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.	TOTAL	
50.47(b)(6)	Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.	TOTAL	
50.47(b)(7)	Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.	*[TBD]*[B-18]	
50.47(b)(8)	Adequate emergency facilities and equipment to support the emergency response are provided and maintained.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(b)(9)	Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.	TOTAL	
50.47(b)(10)	A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.	TOTAL	
50.47(b)(11)	Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.	TOTAL	
50.47(b)(12)	Arrangements are made for medical services for contaminated injured individuals.	TOTAL	
50.47(b)(13)	General plans for recovery and reentry are developed.	TOTAL	
50.47(b)(14)	Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.	TOTAL	
50.47(b)(15)	Radiological emergency response training is provided to those who may be called on to assist in an emergency.	TOTAL	
50.47(b)(16)	Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(c)(1)	Failure to meet the applicable standards set forth in paragraph (b) of this section may result in the Commission declining to issue an operating license; however, the applicant will have an opportunity to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operations. Where an applicant for an operating license asserts that its inability to demonstrate compliance with the requirements of paragraph (b) of this section results wholly or substantially from the decision of state and/or local governments not to participate further in emergency planning, an operating license may be issued if the applicant demonstrates to the Commission's satisfaction that:	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.47(c)(1)(i)	The applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of the non-participation of state and/or local governments.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.47(c)(1)(ii)	The applicant has made a sustained, good faith effort to secure and retain the participation of the pertinent state and/or local governmental authorities, including the furnishing of copies of its emergency plan.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.47(c)(1)(iii)	The applicant's emergency plan provides reasonable assurance that public health and safety is not endangered by operation of the facility concerned. To make that finding, the applicant must demonstrate that, as outlined below, adequate protective measures can and will be taken in the event of an emergency. A utility plan will be evaluated against the same planning standards applicable to a state or local plan, as listed in paragraph (b) of this section, with due allowance made both for- (A) Those elements for which state and/or local non-participation makes compliance infeasible and	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(c)(1)(iii) continued	(B) The utility's measures designed to compensate for any deficiencies resulting from state and/or local non-participation. In making its determination on the adequacy of a utility plan, the NRC will recognize the reality that in an actual emergency, state and local government officials will exercise their best efforts to protect the health and safety of the public. The NRC will determine the adequacy of that expected response, in combination with the utility's compensating measures, on a case-by-case basis, subject to the following guidance. In addressing the circumstance where applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of non-participation of state and/or local governments, it may be presumed that in the event of an actual radiological emergency state and local officials would generally follow the utility plan. However, this presumption may be rebutted by, for example, a good faith and timely proffer of an adequate and feasible state and/or local radiological emergency plan that would in fact be relied upon in a radiological emergency.		
50.47(c)(2)	Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas- cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.	*[TBD]*[B-19]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(d)	Notwithstanding the requirements of paragraphs (a) and (b) of this section, and except as specified by this paragraph, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and capability to implement State and local or utility offsite emergency plans are required prior to issuance of an operating license authorizing only fuel loading or low power testing and training (up to 5 percent of the rated power). Insofar as emergency planning and preparedness requirements are concerned, a license authorizing fuel loading and/or low power testing and training may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC will base this finding on its assessment of the applicant's onsite emergency plans against the pertinent standards in paragraph (b) of this section and Appendix E. Review of applicant's emergency plans will include the following standards with offsite aspects:	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.47(d)(1)	Arrangements for requesting and effectively using offsite assistance on site have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned onsite response have been identified.	TOTAL	
50.47(d)(2)	Procedures have been established for licensee communications with State and local response organizations, including initial notification of the declaration of emergency and periodic provision of plant and response status reports.	TOTAL	
50.47(d)(3)	Provisions exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.47(d)(4)	Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained.	TOTAL	
50.47(d)(5)	Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use onsite.	TOTAL	
50.47(d)(6)	Arrangements are made for medical services for contaminated and injured onsite individuals.	TOTAL	
50.47(d)(7)	Radiological emergency response training has been made available to those offsite who may be called to assist in an emergency onsite.	TOTAL	
50.48	Fire protection	PARTIAL	ANS will comply with those subsections pertaining to new facilities.
50.48(a)	Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of Appendix A of this part. This fire protection plan must describe the overall fire protection program for the facility, identify the various positions within the licensee's organization that are responsible for the program, state the authorities that are delegated to each of these positions to implement those responsibilities, and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan must also describe specific features necessary to implement the program described above, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured. The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license and shall retain each	TOTAL	Basic fire-protection guidance for nuclear power plants is contained in two NRC documents: ** Branch Technical Position Auxiliary Power Conversion System Branch BTP APCSB 9.5- 1, "Guidelines for Fire Protection for Nuclear Power Plants," for new plants docketed after July 1, 1976, dated May 1976. ** Appendix A to BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976," for plants that were operating or under various stages of design or construction before July 1, 1976, dated August 23, 1976. Also see the note in 50.48(b).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.48(a) continued	superseded revision of the procedures for three years from the date it was superseded.		
50.48(b)	Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate prior to January 1, 1979. Except for the requirements of Sections III.G, III.J, and III.O, the provisions of Appendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-14 reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule, or to the extent that fire protection features were accepted by the staff in comprehensive fire protection safety evaluation reports issued before Appendix A to Branch Technical Position BTP APCSB 9.5-1 was published in August 1976. With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate prior to January 1, 1979 shall satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.	*[TBD]*[B-30]	Clarification and guidance with respect to permissible alternatives to satisfy Appendix A to BTP APCSB 9.5-1 have been provided in four other NRC documents. "Supplementary Guidance on Information Needed for Fire Protection Evaluation," dated October 21, 1976. "Sample Technical Specification," dated May 12, 1977. "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance," dated June 14, 1977. "Manpower Requirements for Operating Reactors," dated May 11, 1978. A Fire-Protection Safety Evaluation Report that has been issued for each operating plant states how these guidelines were applied to each facility and identifies open fire protection issues that will be resolved when the facility satisfies the appropriate requirements of Appendix R to this part.
50.48(c)	All fire protection modifications required to satisfy the provisions of Appendix R to this part or directly affected by such requirements shall be completed on the following schedule:	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(c)(1)	Those fire protection features that involve revisions of administrative controls, manpower changes, and training, shall be implemented within 30 days after the effective date of this section and Appendix R to this part.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.48(c)(2)	Those fire protection features that involve installation of modifications that do not require prior NRC approval or plant shutdown shall be implemented within 9 months after the effective date of this section and Appendix R to this part.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(c)(3)	Those fire protection features, except for those requiring prior NRC approval by paragraph (c)(5) of this section, that involve installation of modifications that do require plant shutdown, the need for which is justified in the plans and schedules required by the provisions of paragraph (c)(5) of this section, shall be implemented before startup after the earliest of the following events commencing 180 days or more after the effective date of this section and Appendix R to this part: (i) The first refueling outage; (ii) Another planned outage that lasts for at least 60 days; or (iii) An unplanned outage that lasts for at least 120 days.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(c)(4)	Those fire protection features that require prior NRC approval by paragraph (c)(5) of this section, shall be implemented within the following schedule: Dedicated shutdown systems-30 months after NRC approval; modifications requiring plant shutdown-before startup after the earliest of the events given in paragraph (c)(3) commencing 180 days after NRC approval; modifications not requiring plant shutdown-6 months after NRC approval.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(c)(5)	Licensees shall make any modifications necessary to comply with these requirements in accordance with the above schedule without prior review and approval by NRC except for modifications required by Section III.G.3 of Appendix R to this part. Licensees shall submit plans and schedules for meeting the provisions of paragraphs (c)(2), (c)(3), and (c)(4) within 30 days after the effective date of this section and Appendix R to this part. Licensees shall	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.48(c)(5) continued	submit design descriptions of modifications needed to satisfy Section III.G.3 of Appendix R to this part within 30 days after the effective date of this section and Appendix R to this part.		
50.48(c)(6)	In the event that a request for exemption from a requirement to comply with one or more of the provisions of Appendix R filed within 30 days of the effective date of this rule is based on an assertion by the licensee that such required modifications would not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety, the schedule requirements of paragraph (c) shall be tolled until final Commission action on the exemption request upon a determination by the Director of Nuclear Reactor Regulation that the licensee has provided a sound technical basis for such assertion that warrants further staff review of the request.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(d)	Fire protection features accepted by the NRC staff in Fire Protection Safety Evaluation Reports referred to in paragraph (b) of this section and supplements to such reports, other than features covered by paragraph (c), shall be completed as soon as practicable but no later than the completion date currently specified in license conditions or technical specifications for such facility, or the date determined by paragraphs (d)(1) through (d)(4) of this section, whichever is sooner, unless the Director of Nuclear Reactor Regulation determines, upon a showing by the licensee, that there is good cause for extending such date and that the public health and safety is not adversely affected by such extension. Extensions of such date shall not exceed the dates determined by paragraphs (c)(1) through (c)(4) of this section.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.48(d)(1)	Those fire protection features that involve revisions of administrative controls, manpower changes, and training shall be implemented within 4 months after the date of the NRC staff Fire Protection Evaluation Report accepting or requiring such features.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(d)(2)	Those fire protection features involving installation of modifications not requiring prior approval or plant shutdown shall be implemented within 12 months after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(d)(3)	Those fire protection features, including alternative shutdown capability, involving installation of modifications requiring plant shutdown shall be implemented before the startup after the earliest of the following events commencing 9 months or more after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features: (i) The first refueling outage; (ii) Another planned outage that lasts for at least 60 days; or (iii) An unplanned outage that lasts for at least 120 days.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(d)(4)	Those fire protection features involving dedicated shutdown capability requiring new buildings and systems shall be implemented within 30 months of NRC approval. Other modifications requiring NRC approval prior to installation shall be implemented within 6 months after NRC approval.	NONE	This subsection applies to plants licensed to operate prior to January 1, 1979, and is not applicable to new facilities.
50.48(e)	Nuclear power plants licensed to operate after January 1, 1979, shall complete all fire protection modifications needed to satisfy Criterion 3 of Appendix A to this part in accordance with the provisions of their licenses.	TOTAL	
50.49	Environmental qualification of electric equipment important to safety for nuclear power plants	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(a)	Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.	INTENT	
50.49(b)	Electric equipment important to safety covered by this section is:	PARTIAL	
50.49(b)(1)	Safety-related electric equipment: This equipment is that relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.	TOTAL	Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE 323-1974. Design-basis events are defined in Appendix C of the PDR.
50.49(b)(2)	Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety- related equipment.	TOTAL	
50.49(b)(3)	Certain post-accident monitoring equipment.	PARTIAL	Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."
50.49(c)	Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(c) continued	important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.		
50.49(d)	The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safety meets the requirements of paragraph (j) of this section.	TOTAL	
50.49(d)(1)	The performance specifications under conditions existing during and following design basis accidents.	TOTAL	
50.49(d)(2)	The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.	TOTAL	
50.49(d)(3)	The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.	TOTAL	
50.49(e)	The electric equipment qualification program must include and be based on the following: (1) Temperature and pressure. The time-dependent temperature and pressure at the location of the electric	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(e) continued	equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional. (2) Humidity. Humidity during design basis accidents must be considered. (3) Chemical effects. The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed. (4) Radiation. The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects. (5) Aging. Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of- installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life. (6) Submergence (if subject to being submerged). (7) Synergistic effects. Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.		

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(e) continued	(8) Margins. Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.		
50.49(f)	Each item of electric equipment important to safety must be qualified by one of the following methods: (1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. (2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable. (3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. (4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.	TOTAL	
50.49(g)	Each holder of an operating license issued prior to February 22, 1983, shall, by May 20, 1983, identify the electric equipment important to safety within the scope of this section already qualified and submit a schedule for either the qualification to the provisions of this section or for the replacement of the remaining electric equipment important to safety within the scope of this section. This schedule must establish a goal of final environmental qualification of the electric equipment within the scope of this section by the end of the second refueling outage after March 31, 1982 or by March 31, 1985, whichever is earlier. The Director of the Office of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if these requests are filed on a timely basis and	NONE	50.49(g) applies to plants licensed to operate prior to February 22, 1983.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(g) continued	demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification. The schedule in this paragraph supersedes the June 30, 1982, deadline, or any other previously imposed date, for environmental qualification of electric equipment contained in certain nuclear power operating licenses.		
50.49(h)	Each license shall notify the Commission as specified in 50.4 of any significant equipment qualification problem that may require extension of the completion date provided in accordance with paragraph (g) of this section within 60 days of its discovery.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.49(i)	Applicants for operating licenses granted after February 22, 1983, but prior to November 30, 1985, shall perform an analysis to ensure that the plant can be safely operated pending completion of equipment qualification required by this section. This analysis must be submitted, as specified in § 50.4, for consideration prior to the granting of an operating license and must include, where appropriate, consideration of: (1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified. (2) The validity of partial test data in support of the original qualification. (3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified. (4) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator.	NONE	This subsection applies to plants licensed to operate prior to November 30, 1985, and is not applicable to new facilities.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(i) continued	(5) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.		
50.49(j)	A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section: (1) Is qualified for its application; and (2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.	TOTAL	
50.49(k)	Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section if the Commission has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 (DOR Guidelines), or NUREG-0588 (For Comment version), "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.49(l)	Replacement equipment must be qualified in accordance with the provisions of this section unless there are sound reasons to the contrary.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
ISSUANCE, LIMITATIONS, AND CONDITIONS OF LICENSES AND CONSTRUCTION PERMITS			
50.50	Issuance of licenses and construction permits	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.51	Duration of license, renewal	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.52	Combining licenses	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.53	Jurisdictional limitations	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.54	Conditions of licenses	*[TBD]*[B-21]	To be determined.
50.55	Conditions of construction permits	*[TBD]*[B-22]	To be determined.
50.55a	Codes and standards. Each operating license for a boiling or pressurized water-cooled nuclear power facility shall be subject to the conditions in paragraph (g) of this section and each construction permit for a utilization facility shall be subject to the following conditions in addition to those specified in § 50.55.		
50.55a(a)(1)	Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.	TOTAL	
50.55a(a)(2)	Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), and (g) of	*[TBD]*[B-23]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(a)(2) continued	this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section.		
50.55a(a)(3)	Proposed alternatives to the requirements of paragraphs (c), (d), (e), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant must demonstrate that (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.	*[TBD]*[B-24]	
50.55a(b)	The ASME Boiler and Pressure Vessel Code, which is referenced in the following paragraphs, was approved for incorporation by reference by the Director of the Federal Register on January 1, 1981. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L Street NW., Washington, D.C.	*[TBD]*[B-25]	
50.55a(b)(1)	As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions through the 1983 Edition and Addenda through the Winter 1985 Addenda, and the 1986 Edition.	*[TBD]*[B-26]	
50.55a(b)(2)	As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1, and include editions through the 1983 Edition and Addenda through the Winter 1985 Addenda, and the 1986 Edition, subject to the following limitations and modifications:	*[TBD]*[B-27]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(b)(2)(i)	Limitations on specific editions and addenda. When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (b)(2) of this section are not affected by these limitations.	*[TBD]*[B-28]	
50.55a(b)(2)(ii)	Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB-2500 and IWB-2500-1 and Category B-J). If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB-2500 and Table IWB-2600 Category B-J of Section XI of the ASME Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.	*[TBD]*[B-29]	
50.55a(b)(2)(iii)	Steam generator tubing (modifies Article IWB-2000). If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications.	*[TBD]*[B-30]	
50.55a(b)(2)(iv)	Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC-2520 or IWC-2520-1, Category C-F).	*[TBD]*[B-31]	
50.55a(b)(2)(iv)(A)	Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems and Containment Heat Removal Systems, shall be examined.	*[TBD]*[B-32]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(b)(2)(iv)(A) continued	When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of Section XI of the ASME Code, the extent of examination for these systems shall be determined by the requirements of paragraph IWC- 1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC- 2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda.		
50.55a(b)(2)(iv)(B)	For a nuclear power plant whose application for a construction permit was docketed prior to July 1, 1978, when applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of Section XI of the ASME Code, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and CG and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.	*[TBD]*[B-33]	
50.55a(b)(2)(v)	Evaluation procedures and acceptance criteria for austenitic piping (applies to IWB-3640). When applying the Winter 1983 Addenda and Winter 1984 Addenda, the rules of paragraph IWB- 3640 may be used for all applications permitted in that paragraph, except those associated with submerged arc welds (SAW) or shielded metal arc welds (SMAW). For SAW or SMAW, use paragraph IWB3640, as modified by the Winter 1985 Addenda.	*[TBD]*[B-34]	
50.55a(c)	Reactor coolant pressure boundary.		
50.55a(c)(1)	Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.	*[TBD]*[B-35]	USAS and ASME Code addenda issued prior to the Winter 1977 Addenda are considered to be "in effect" or "effective" six months after their date of issuance and after they are incorporated by reference in paragraph (b) of this section. Addenda to the ASME Code issued after the Summer 1977 Addenda are considered to be "in effect" or "effective" after the date of publication of the addenda and after they are incorporated by reference in paragraph (b)

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(c)(1) continued			of this section. For ASME Code Editions and Addenda issued prior to the Winter 1977 Addenda, the Code Edition and Addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the Winter 1977 Addenda and subsequent editions and addenda, the method for determining the applicable code editions and addenda is contained in Paragraph NCA 1140 of Section III of the ASME Code.
50.55a(c)(2)	Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, Provided: (i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or (ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.	*[TBD]*[B-36]	
50.55a(c)(3)	The Code Edition, Addenda, and optional Code Cases to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but (i) the edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions	*[TBD]*[B-37]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to Sec. 50.55a(a)(3).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(c)(3) continued	applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition, (iii) the ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 edition, and (iv) ASME Code Cases must have been determined suitable for use by the NRC.		
50.55a(c)(4)	For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit.	*[TBD]*[B-38]	
50.55a(d)	Quality Group B components.		
50.55a(d)(1)	For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B 9 must meet the requirements for Class 2 Components in Section III of the ASME Boiler and Pressure Vessel Code.	*[TBD]*[B-39]	
50.55a(d)(2)	The Code Edition, Addenda, and optional Code Cases to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler Vessel and Pressure Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases must have been determined suitable for use by the NRC.	*[TBD]*[B-40]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1"; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to Sec. 50.55a(a)(3).
50.55a(e)	Quality Group C components.		
		[TBD][B-41]	Guidance for quality group classifications of components which are to be

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(e)(1)	For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group must meet the requirements for Class 3 components in Section III of the ASME Boiler and Pressure Vessel Code.		included in the safety analysis reports pursuant to § 50.34(a) and § 50.34(b) may be found in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants," and in Section 3.2.2 of NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants."
50.55a(e)(2)	The Code Edition, Addenda, and optional Code Cases to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA-1140, subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases must have been determined suitable for use by the NRC.	*[TBD]*[B-42]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1"; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to Sec. 50.55a(a)(3).
50.55a(f)	[Reserved]		
50.55a(g)	Inservice inspection requirements.		
50.55a(g)(1)	For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) shall meet the requirements of paragraphs (g) (4) and (5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports shall meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves shall meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.	*[TBD]*[B-43]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(g)(2)	For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 shall be designed and be provided with access to enable the performance of (i) inservice examination of such components (including supports) and (ii) tests for operational readiness of pumps and valves, and shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda6 in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.	*[TBD]*[B-44]	
50.55a(g)(3)	For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:	*[TBD]*[B-45]	
50.55a(g)(3)(i)	Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component.	*[TBD]*[B-46]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1"; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to Sec. 50.55a(a)(3).
50.55a(g)(3)(ii)	Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the	*[TBD]*[B-47]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1"; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(g)(3)(ii) continued	ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component.		Regulation upon request pursuant to Sec. 50.55a(a)(3).
50.55a(g)(3)(iii)	Pumps and valves which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.	*[TBD]*[B-48]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1"; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to Sec. 50.55a(a)(3).
50.55a(g)(3)(iv)	Pumps and valves which are classified as ASME Code Class 2 and Class 3 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.	*[TBD]*[B-49]	ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability--ASME Section III Division 1"; NRC Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1"; and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI Division 1." The use of other code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to Sec. 50.55a(a)(3).
50.55a(g)(3)(v)	All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.	*[TBD]*[B-50]	
50.55a(g)(4)	Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 shall meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design,	*[TBD]*[B-51]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(g)(4) continued	geometry and materials of construction of the components.		
50.55a(g)(4)(i)	Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, conducted during the initial 120-month inspection interval shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.	*[TBD]*[B-52]	
50.55a(g)(4)(ii)	Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, conducted during successive 120-month inspection intervals shall comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.	*[TBD]*[B-53]	
50.55a(g)(4)(iii)	[Reserved]		
50.55a(g)(4)(iv)	Inservice examinations of components, tests of pumps and valves, and system pressure tests, may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.	*[TBD]*[B-54]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(g)(5)(i)	The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility shall be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.	*[TBD]*[B-55]	
50.55a(g)(5)(ii)	If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. This application shall be submitted at least 6 months before the start of the period during which the provisions become applicable as determined by paragraph (g)(4) of this section.	*[TBD]*[B-56]	
50.55a(g)(5)(iii)	If the licensee has determined that conformance with certain code requirements is impractical for his facility, the licensee shall notify the Commission and submit information to support his determinations.	*[TBD]*[B-57]	
50.55a(g)(5)(iv)	Where an examination or test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination shall be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination or test is determined to be impractical.	*[TBD]*[B-58]	
50.55a(g)(6)(i)	The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden	*[TBD]*[B-59]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.55a(g)(6)(i) continued	upon the licensee that could result if the requirements were imposed on the facility.		
50.55a(g)(6)(ii)	The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.	*[TBD]*[B-40]	
50.55a(h)	Protection systems. For construction permits issued after January 1, 1971, protection systems shall meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," (IEEE-279) in effect on the formal docket date of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE-279 which become effective.	*[TBD]*[B-41]	For purposes of this regulation, the proposed IEEE 279 became "in effect" on August 30, 1968; and the revised issue IEEE 279-1971 became "in effect" on June 3, 1971. Where an application for a construction permit is submitted in four parts pursuant to the provisions of § 2.101(a-1) and Subpart F of Part 2 of this chapter, "the formal docket date of the application for a construction permit" for purposes of this section shall be the date of docketing of the information required by § 2.101(a-1) (2) or (3), whichever is later.
50.56	Conversion of construction permit to license; or amendment of license.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.57	Issuance of operating license	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.58	Hearings and report of the Advisory Committee on Reactor Safeguards	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.59	Changes, tests and experiments.		
50.59(a)(1)	The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.59(a)(1) continued	described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.		
50.59(a)(2)	A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.	TOTAL	
50.59(b)(1)	The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that such changes constitute changes in the facility as described in the safety analysis report or constitute changes in procedures as described in the safety analysis report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records shall include a written safety evaluation which provides the bases for the determination that the change, test or experiment does not involve an unreviewed safety question. The licensee shall furnish to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made a part of the public record of the licensing proceeding. In addition	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.49(b)(1) continued	to a signed original, 39 copies of each report of changes in a facility of the type described in § 50.21(b) or § 50.22 or a testing facility, and 12 copies of each report of changes in any other facility, shall be filed. The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.		
50.59(b)(2)	The licensee shall submit, as specified in § 50.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each. The report must be submitted annually or at such shorter intervals as may be specified in the license.	TOTAL	Any changes, tests, and experiments shall be evaluated for safety.
50.59(b)(3)	The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.	TOTAL	
50.59(c)	The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to 50.90.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.60	Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation	INTENT	See PDR Chapter 6.
50.60(a)	Except as provided in paragraph (b) of this section, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.	*TBD*[B-42]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.60(b)	Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.	N/A-L	
50.61	Fracture toughness requirements for protection against pressurized thermal shock events	INTENT	See PDR Section 6.
50.61(a)	Definitions. For the purposes of this section: (1) "ASME Code" means the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Rules for the Construction of Nuclear Power Plant Components," edition and addenda as specified by § 50.55a, Codes and Standards. (2) "Pressurized Thermal Shock Event" means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. (3) "Reactor Vessel Beltline" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. (4) "Initial RTNDT" means the reference temperature for a reactor vessel material as defined in the ASME Code, Paragraph NB-2331. RTNDT means the reference temperature as adjusted for the effects of neutron radiation for the period of service in question. (5) "RTPTS" means the reference temperature calculated by the method given in paragraph (b)(2) of this section for use as a screening criterion.	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.61(b)(1)	Requirements. (1) For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RTPTS (at the inner vessel surface) of reactor vessel beltline materials by giving values from the time of submittal to the expiration date of the operating license. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. This assessment must be submitted by January 23, 1986, and must be updated whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values.	*[TBD]*[B-43]	
50.61(b)(2)	The pressurized thermal shock (PTS) screening criterion is 270 xF for plates, forgings, and axial weld materials, or 300 xF for circumferential weld materials. For the purpose of comparison with this criterion, the value of RTPTS for the reactor vessel must be calculated as follows. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline. For each material, RTPTS is the lower of the results given by Equations 1 and 2. Equation 1: RTPTS=I+M+[-10+470Cu+350 Cu Ni]f^0.270 Equation 2: RTPTS=I+M+283 f^0.194 (i) "I" means the initial reference temperature of the unirradiated material measured as defined in the ASME Code, Paragraph NB-2331. If a measured value is not available, the following generic mean values must be used: 0 F for welds made with Linde 80 flux, and -56 F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.	*[TBD]*[B-44]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.61(b)(2) continued	(ii) "M" means the margin to be added to cover uncertainties in the values of initial RTNDT, copper and nickel content, fluence and the calculational procedures. In Equation 1, M=48 F if a measured value of I was used, and M=59 F if the generic mean value of I was used. In Equation 2, M=0 F if a measured value of I was used, and M=34 F if the generic mean value of I was used. (iii) "Cu" and "Ni" mean the best estimate weight percent of copper and nickel in the material. The source of these values must be included in the assessment. The relationship of the material, on which any measurements were made, to the actual material in the pressure vessel must be described. (iv) "f" means the best estimate neutron fluence, in unit of 10^19 n/cm^2 (E greater than or equal to 1 MeV), at the clad- base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question.		
50.61(b)(3)	For each pressurized water nuclear power reactor for which the value of RTPTS for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, the licensee shall submit by April 23, 1986, an analysis and schedule for implementation of such flux reduction programs as are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2). The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated Commission approval of detailed plant- specific analyses, submitted to demonstrate acceptable risk at values of RTPTS above the screening limit due to plant modifications, new information or new analysis techniques.	*[TBD]*[B-65]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.61(b)(4)	For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent the value of RTPTS from exceeding the PTS screening criterion before the expiration date of the operating license, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine reactor vessel materials properties based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least 3 years before the value of RTPTS as defined in paragraph (b)(2) of this section is projected to exceed the PTS screening criterion or by one year after issuance of the Commission Guidance and Acceptance Criteria for these analyses, whichever is later.	*[TBD]*[B-46]	
50.61(b)(5)	After consideration of the licensee's analyses (including effects of proposed corrective actions, if any) submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the Commission may, on a case-by-case basis, approve operation of the facility at values of RTPTS in excess of the PTS screening criterion. The Commission will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.61(b)(6)	If the Commission concludes, pursuant to paragraph (b)(5) of this section, that operation of the facility at value of RTPTS in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the licensee must request and receive Commission approval prior to any operation beyond the criterion. The request must be based upon modifications to equipment,	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.61(b)(6) continued	systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.		
50.62	Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants	PARTIAL	A secondary reactor trip system that is separate and diverse from the reactor primary trip system shall be provided to automatically insert negative reactivity in event of ATWS events. Separate equipment from sensor output to final actuation device shall be used. These anticipated occurrences include events classified as Normal (N) or Anticipated (A) in Appendix C. The reactor fuel and non-fuel limits for an Extremely Unlikely class event is applicable to the ATWS event combination.
50.62(a)	Applicability. The requirements of this section apply to all commercial light-water-cooled nuclear power plants.	TOTAL	
50.62(b)	Definition. For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.	*[TBD]*[B-67]	
50.62(c)(1)	Requirements. (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.	*[TBD]*[B-68]	
50.62(c)(2)	Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram	*[TBD]*[B-69]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.62(c)(2)	system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).		
50.62(c)(3)	Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.	*[TBD]*[B-70]	
50.62(c)(4)	Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.	*[TBD]*[B-71]	
50.62(c)(5)	Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.	*[TBD]*[B-72]	
50.62(c)(6)	Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Director, Office of Nuclear Reactor Regulation.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.62(d)	Implementation. By 180 days after the issuance of the QA guidance for non-safety related components each licensee shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.63	Loss of all alternating current power		Emergency reactor core, spent fuel, and containment cooling do not depend on the availability of ac power *[HOLD]*[a-1]. Safety-related systems, that depend on a source of electrical power shall be designed to accomplish their safety function after a station blackout (total loss of normally connected off-site and on-site generated ac power) for either the time required to complete their safety function or up to 4 hours (min) and with a goal 72 hours.
50.63(a)(1)	Requirements. (1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in Sec. 50.2. The specified station blackout duration shall be based on the following factors: (i) The redundancy of the onsite emergency ac power sources; (ii) The reliability of the onsite emergency ac power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power.	TOTAL	
50.63(a)(2)	The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.63(1)(2) continued	capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.		
50.63(b)	Limitation of scope. Paragraph (c) of this section does not apply to those plants licensed to operate prior to July 21, 1988, if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.	N/A-L	This subsection applies to plants licensed to operate prior to July 21, 1988, and is not applicable to new facilities.
50.63(c)(1)	Implementation.--(1) Information Submittal. For each light-water-cooled nuclear power plant licensed to operate on or before July 21, 1988, the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by April 17, 1989. For each light-water cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.63(c)(1)(i)	A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;	*[TBD]*[B-73]	
50.63(c)(1)(ii)	A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and	*[TBD]*[B-74]	
50.63(c)(1)(iii)	A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.	*[TBD]*[B-75]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.63(c)(2)	Alternate ac source: The alternate ac power source(s), as defined in Sec. 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test. Alternate ac source(s) serving a multiple unit site where onsite emergency ac sources are not shared between units must have, as a minimum, the capacity and capability for coping with a station blackout in any of the units. At sites where onsite emergency ac sources are shared between units, the alternate ac source(s) must have the capacity and capability as required to ensure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in Sec. 50.2. If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required.	*[TBD]*[B-76]	
50.63(c)(3)	Regulatory Assessment: After consideration of the information submitted in accordance with paragraph (c)(1) of this section, the Director, Office of Nuclear Reactor Regulation, will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures, and the proposed schedule for implementing the procedures and modifications for compliance with paragraph (a) this section.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.63(c)(4)	Implementation Schedule: For each light-water-cooled nuclear power plant licensed to operate on or before June 21, 1988, the licensee shall, within 30 days of the notification provided in accordance with paragraph (c)(3) of this section, submit to the Director of the Office of	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.63(c)(4)	Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of paragraph (a) of this section. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(3) of this section. A final schedule for implementing modifications necessary to comply with the requirements of paragraph (a) of this section will be established by the NRC staff in consultation and coordination with the affected licensee.		
50.64	Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors	NONE	ANS uses HEU to satisfy research objectives.
INSPECTIONS, RECORDS, REPORTS, NOTIFICATIONS			
50.70	Inspections	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.71	Maintenance of records, making of reports	TOTAL	
50.71(a)	Each licensee and each holder of a construction permit shall maintain such records and make such reports, in connection with the licensed activity, as may be required by the conditions of the license or permit or by the rules, regulations, and orders of the Commission in effectuating the purposes of the Act, including section 105 of the Act.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.71(b)	With respect to any production or utilization facility of a type described in § 50.21(b) or § 50.22 of this part, or a testing facility, each licensee and each holder of a construction permit shall, upon each issuance of its annual financial report, including the certified financial statements, file a copy thereof with the Director of	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.71(b) continued	Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.		
50.71(c)	Records that are required by the regulations in this part, by license condition, or by technical specifications, must be retained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, these records must be retained until the Commission terminates the facility license.	TOTAL	
50.71(d)(1)	Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations. The record may also be stored in electronic media with the capability of producing legible, accurate, and complete records during the required retention period. Records such as letters, drawings, specifications, must include all pertinent information such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with and loss of records.	TOTAL	
50.71(d)(2)	If there is a conflict between the Commission's regulations in this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to § 50.12 of this part, has granted a specific exemption from the record retention requirements specified in the regulations in this part.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.71(e)	Each person licensed to operate a nuclear power reactor pursuant to the provisions of § 50.21 or § 50.22 of this part shall update periodically, as provided in paragraphs (e)(3) and (4) of this section, the final safety analysis	TOTAL	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.71(e) continued	report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.		
50.71(e)(1)	Revisions containing updated information shall be submitted on a replacement-page basis and shall be accompanied by a list which identifies the current pages of the FSAR following page replacement. One signed original and 12 additional copies of the required information shall be filed with the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.	TOTAL	
50.71(e)(2)	The submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of § 50.59 but not previously submitted to the Commission.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.71(e)(3)(i)	A revision of the original FSAR containing those original pages that are still applicable plus new replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.	TOTAL	
50.71(e)(3)(ii)	Not less than 15 days before § 50.71(e) becomes effective, the Director of the Office of Nuclear Reactor Regulation shall notify by letter the licensees of those nuclear power plants initially subject to the NRC's systematic evaluation program that they need not comply with the provisions of this section while the program is being conducted at their plant. The Director of the Office of Nuclear Reactor Regulation will notify by letter the licensee of each nuclear power plant being evaluated when the systematic evaluation program has been completed. Within 24 months after receipt of this notification, the licensee shall file a complete FSAR which is up to date as of a maximum of 6 months prior to the date of filing the revision.	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.71(e)(4)	Subsequent revisions shall be filed no less frequently than annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing.	TOTAL	
50.71(e)(5)	Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both).	TOTAL	
50.71(e)(6)	The updated FSAR shall be retained by the licensee until the Commission terminates their license.	TOTAL	
50.72	Immediate notification requirements for operating nuclear power reactors	TOTAL	The ANS control room shall be capable of providing DOE with immediate notification of any emergencies.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.72(a)(1)	General requirements. (1) Each nuclear power reactor licensee licensed under § 50.21(b) or § 50.22 of this part shall notify the NRC Operations Center via the Emergency Notification System of: (i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan; or (ii) Of those non-Emergency events specified in paragraph (b) of this section.	*[TBD]*[B-77]	These emergency classes are addressed in Appendix E of this part. Other requirements for immediate notification of the NRC by licensed operating nuclear power reactors are contained elsewhere in this chapter, in particular, § 20.205, 20.403, 50.36, and 73.71.
50.72(a)(2)	If the Emergency Notification System is inoperative, the licensee shall make the required notifications via commercial telephone service, other dedicated telephone system, or any other method which will ensure that a report is made as soon as practical to the NRC Operations Center.	*[TBD]*[B-78]	
50.72(a)(3)	The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes.	*[TBD]*[B-79]	
50.72(a)(4)	When making a report under paragraph (a)(3) of this section, the licensee shall identify: (i) The Emergency Class declared; or (ii) Either paragraph (b)(1), "One-Hour Report," or paragraph (b)(2), "Four-Hour Report," as the paragraph of this section requiring notification of the Non-Emergency Event.	*[TBD]*[B-80]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.72(b)(1)	Non-emergency events-(1) One-hour reports. If not reported as a declaration of an Emergency Class under paragraph (a) of this section, the licensee shall notify the NRC as soon as practical and in all cases within one hour of the occurrence of any of the following: (i)(A) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications. (B) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part. (ii) Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being: (A) In an unanalyzed condition that significantly compromises plant safety; (B) In a condition that is outside the design basis of the plant; or (C) In a condition not covered by the plant's operating and emergency procedures. (iii) Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant. (iv) Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal. (v) Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system). (vi) Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.	*[TBD]*[B-61]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.72(b)(2)	Four-hour reports. If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of any of the following: (i) Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety. (ii) Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or reactor operation need not be reported. (iii) Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition, (B) Remove residual heat, (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident. (iv)(A) Any airborne radioactive release that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour. (B) Any liquid effluent release that exceeds 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to Part 20 of this chapter) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour. (Immediate notifications made under this paragraph also satisfy the requirements of paragraphs (a)(2) and (b)(2) of § 20.403 of Part 20 of this chapter.)	*[TBD]*[B-43]	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.72(b)(2) continued	(v) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment. (vi) Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.		
50.72(c)	Followup notification. With respect to the telephone notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee, shall during the course of the event: (1) Immediately report (i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class. (2) Immediately report (i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant behavior that is not understood. (3) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.	*[TBD]*[B-43]	
50.73	License event report system	*[TBD]*[B-44]	
50.74	Notification of change in operator or senior operator status	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.75	Reporting and recordkeeping for decommissioning planning	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
US/IAEA SAFEGUARDS AGREEMENT			
50.78	Installation information and verification	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
TRANSFERS OF LICENSES-CREDITORS' RIGHTS-SURRENDER OF LICENSES			
50.80	Transfer of licenses	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.81	Creditor regulations	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.82	Applications for termination of licenses	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
AMENDMENT OF LICENSE OR CONSTRUCTION PERMIT AT REQUEST OF HOLDER			
50.90	Application for amendment of license or construction permit	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.91	Notice for public comment; State consultation	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
50.92	Issuance of amendment	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
REVOCATION, SUSPENSION, MODIFICATION, AMENDMENT OF LICENSES AND CONSTRUCTION PERMITS, EMERGENCY OPERATIONS BY THE COMMISSION			
50.100	Revocation, suspension, modification of licenses and construction permits for cause	N/A-L	Only those requirements relating to the design are applicable at this time. This requirement will be reevaluated during Title I.
50.101	Retaking possession of special nuclear material	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.102	Commission order for operation after revocation	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
50.103	Suspension and operation in war or national emergency	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
BACKFITTING			
50.109	Backfitting	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
ENFORCEMENT			
50.110	Violations	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
Appendix A	General Design Criteria for nuclear power plants	PARTIAL	Required by DOE 5480.6 and 5480.4. See Table B.9 of this PDR for COMPLIANCE COMMITMENT level for each specific General Design Criterion.

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
Appendix B	Quality assurance criteria for nuclear power plants and fuel reprocessing plants	TOTAL	
Appendix C	A guide for the financial data and related information required to establish financial qualifications for facility construction permits	NONE	
Appendix D	[RESERVED]	NONE	
Appendix E	Emergency planning and preparedness for production and utilization facilities	INTENT	The INTENT will be satisfied although ANS fuel limits are different since ANS uses uranium silicide fuel clad in aluminum rather than uranium oxide pellets clad in zircaloy. Medical facilities, medical supplies, and arrangement for qualified medical personnel shall be provide on site for emergency first aid. Arrangements and capability for transportation of injured personnel to off-site medical facilities shall also be provided.
Appendix F	Relating to the siting of fuel reprocessing plants and related waste management facilities	NONE	
Appendix G	Fracture toughness requirements	INTENT	The ANS reactor components that provide the function of the reactor vessel are fabricated of aluminum. See PDR Section 6.0.
Appendix H	Reactor vessel material surveillance program requirements	INTENT	
Appendix I	Numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents	TOTAL	Required by DOE 5480.6 and 5480.4.
Appendix J	Primary reactor containment leakage testing for water-cooled power reactors	TOTAL	
Appendix K	ECCS evaluation models	INTENT	
Appendix L	Information requested by the attorney general for antitrust review of facility license applications	NONE	

Table B.4. Compliance Commitment to Title 10, Part 50 of the Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," for the Design of the Advanced Neutron Source

SECTION	TITLE/TEXT	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
Appendix M	Standardization of design; manufacture of nuclear power reactors; construction and operation of nuclear power reactors manufactured pursuant to commission license	NONE	
Appendix N	Standardization of nuclear power plant designs: licenses to construct and operate nuclear power reactors of duplicate design at multiple sites	NONE	
Appendix O	Standardization of design: staff review of standard designs	NONE	
Appendix P	[RESERVED]	NONE	
Appendix Q	Pre-application early review of site suitability issues	INTENT	
Appendix R	Fire protection program for nuclear power facilities operating prior to January 1, 1979	PARTIAL	ANS will comply with the applicable parts of Appendix R as required by 50.48(b).

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE

A review of 10 CFR 50 Appendix A, "General Design Criteria," (Revised January 1, 1991) and its definitions is given below. Justification and Guidance may be given for a criterion to aid in determining applicability at the SDD level. The guidance given is to be considered as a requirement. Any requests for exceptions to the guidance must be submitted to the Chairman of the Plant Design Requirements Working Group. Each exception will be reviewed on a case-by-case basis.

Definitions required to interpret this table are:

TOTAL — *All technical requirements are applicable as written.*

PARTIAL — *Due to specific requirements in the regulations or design features in the ANS, not all of the technical requirements apply as written. The PDR shall define that part of the regulation that is and is not applicable at the overall plant level. The SDD's will provide the compliance commitment level and justification applicable to their specific system(s).*

INTENT — *Only the purpose (or intent) of the technical parts of the regulations are applicable. The PDR shall define the intent of the regulation and how it applies to the ANS at the overall plant level. The SDD's will provide the compliance commitment level and justification applicable to their specific system(s).*

NONE — *Due to specific requirements in the regulations or design features in the ANS, none of the technical requirements are applicable. The appendix provides justification why the regulation is not applicable.*

The criterion may contain statements that make it appear to be not-applicable to the design of the ANS, yet the compliance commitment level given is TOTAL, PARTIAL, or INTENT. This can occur because the regulation defines the criterion, and then, in some cases, provides an example. Therefore, statements like "such as steam generators" simply provides guidance and is not part of the criterion.

I. OVERALL REQUIREMENTS

DEF	Loss of Coolant Accidents	Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.	TOTAL	Systems, structures, and components shall be designed to assure that acceptable reactor fuel and nonfuel (e.g., core pressure boundary tube) design limits are not exceeded in the event of a loss-of-coolant accident. The limits depend on the LOCA size and its classification as defined in Appendix C of the PDR.

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
DEF	Single Failure	Safety-related systems shall be designed to withstand a single failure without total loss of the safety function. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety function. Multiple failures resulting from a single occupance are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.	TOTAL	Supplementary criteria are given in IEEE std 279/375, NRC Reg. Guide 1.53, and ANSI/ANS Std. 50.1/58.9. Additional redundancy in safety-related systems may be required to satisfy the safety reliability goals based on PRA. In general, a combination of independent failure of two passive boundaries are not considered to be within the design basis. An exception is a beam tube wall break in combination with a (proposed) outer passive seal break *[HOLD]*[B-2]. In addition, an independent single passive failure in reactor containment boundary, in combination with the limiting design basis source term (Event FD-2 in appendix c), is not considered to be within the plant design bases.
DEF	Anticipated Operational Occurrences	Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.	TOTAL	The anticipated operational occurrences that must be considered are given in Appendix C of the PDR.

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 01	Quality Standards and Records	Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented to provide adequate assurance that these structures, systems, and components will provide adequate assurance that these structures, systems, and components will satisfactorily perform their nuclear safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	TOTAL	
GDC 02	Design Bases for Protection Against Natural Phenomena	Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 03	Fire Protection	Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Non-combustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the nuclear-safety-related capability of these structures, systems, and components.	TOTAL	
GDC 04	Environmental and Missile Design Bases	Structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible with the environmental conditions associated with normal operation, maintenance, testing. and postulated accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	TOTAL	
GDC 05	Sharing of Structures, Systems, and Components	Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their nuclear safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.	NONE	GDC 05 is not applicable to the design of the ANS because there is only one facility.

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS				
GDC 10	Reactor Design	The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of any anticipated operational occurrences.	TOTAL	
GDC 11	Reactor Inherent Protection	The reactor core and associated coolant systems shall be designed so that, in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.	TOTAL	
GDC 12	Suppression Of Reactor Power Oscillations	The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	TOTAL	
GDC 13	Instrumentation and Control	Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for postulated accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	TOTAL	
GDC 14	Reactor Coolant Boundary	The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 15	Reactor Coolant System Design	The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including any anticipated operational occurrences.	TOTAL	
GDC 16	Reactor Containment Design	Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.	TOTAL	
GDC 17	Electric Power Systems	An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 17 continued		specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.		
GDC 18	Inspection and Testing of Electric Power Systems	Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as on-site power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the off-site power system, and the on-site power system.	TOTAL	
GDC 19	Control Room	A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under conditions and to maintain it in a safe condition under postulated accident conditions including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5-rem whole body, or its equivalent to any part of the body for the duration of the accident. Equipment at appropriate locations outside the control room shall be	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 19 continued		provided with a design capability for prompt hot shutdown of the reactor, (1) including anticipated operational occurrences, and including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.		
III. PROTECTION AND REACTIVITY CONTROL SYSTEMS				
GDC 20	Protection System Functions	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.	TOTAL	
GDC 21	Protection System Reliability and Testability	The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any components or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.	TOTAL	
GDC 22	Protection System Independence	The protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles ofoperation, shall be used to the extent practical to prevent loss of the protection function.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 23	Protection System Failure Modes	The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g, extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.	TOTAL	Other postulated adverse environments that should be considered include uncontrolled combustible gas-oxygen recombination, burning and/or local detonation of combustible gases, etc.
GDC 24	Separation of Protection and Control Systems	The protection system shall be separated from control systems to the extent that failure of any single control system component or channel or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control system shall be limited so as to assure that safety is not significantly impaired.	TOTAL	
GDC 25	Protection System Requirements for Reactivity Control Malfunctions	The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.	TOTAL	
GDC 26	Reactivity Control System Redundancy and Capability Worth Requirements	Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 27	Combined Reactivity Control Systems Capability	The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.	*[TBD]*[B-85]	Liquid poison addition for nuclear criticality control for the reactor and the spent fuel systems under design basis and severe accident conditions is *[TBD]*[B-86].
GDC 28	Reactivity Limits	The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition.	TOTAL	
GDC 29	Protection Against Anticipated Operational Occurrences	The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	TOTAL	
IV. FLUID SYSTEMS				
GDC 30	Quality of Reactor Coolant Pressure Boundary	Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 31	Fracture Prevention of Reactor Coolant Pressure Boundary	The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.	TOTAL	
GDC 32	Inspection of Reactor Coolant Pressure Boundary	Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.	TOTAL	
GDC 33	Reactor Coolant Makeup	A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.	INTENT	
GDC 34	Residual Heat Removal	A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable	INTENT	GDC 34 implies that safety-related ac power is required. The INTENT of GDC 34 will be met by providing sufficient capability to remove residual heat without the necessity to rely on safety-related ac power

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 34 continued		interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.		through the use of safety-related dc power and passive means (e.g., natural circulation cooling). Reactor residual heat removal is achieved using the same cooling loops that are used for normal cooling.
GDC 35	Emergency Core Cooling	A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	INTENT	GDC 35 implies that safety-related ac power is required. The INTENT of GDC 35 will be met by providing sufficient capability to remove heat from the reactor core without the necessity to rely on safety-related ac power through the use of safety-related dc power and passive means (e.g., natural circulation cooling). Emergency coolant inventory control in the event of a LOCA is provided by passive means (e.g., accumulators, limited volume cells, or submerged components).
GDC 36	Inspection of Emergency Core Cooling System	The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.	TOTAL	Emergency core cooling is achieved using the same cooling loops that are used for normal cooling with alternate emergency heat sinks. Coolant inventory in the event of a LOCA is provided by passive means. The ANS will meet the requirements of GDC 36 by designing the emergency core cooling system to permit periodic inspection of important components.

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 37	Testing of Emergency Core Cooling System	The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.	INTENT	GDC 37 implies that safety-related ac power is required. The INTENT of GDC 37 will be met by designing the system to allow periodic pressure and functional testing without the necessity to rely on safety-related ac power through the use of safety-related dc power and passive means, e.g., natural circulation cooling. (Emergency core cooling is achieved using the same cooling loops that are used for normal cooling with alternate emergency heat sinks with coolant inventory in the event of a LOCA provided by passive means.)
GDC 38	Containment Heat Removal	A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. Suitable redundancy in components and features, and suitable interconnections, leak detection isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	INTENT	GDC 38 implies that safety-related ac power is required. The INTENT of GDC 38 will be met by providing sufficient capability to remove heat from the reactor containment without the necessity to rely on safety-related ac power through the use of safety-related dc power and passive means (e.g., natural circulation cooling).

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 39	Inspection of Containment Heat Removal System	The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.	TOTAL	The ANS is designed to maintain containment heat removal and pressure control by passive means that does not depend on functioning of active components or safety-related ac electric power. The containment heat removal system will be designed to permit appropriate periodic inspection of important components.
GDC 40	Testing of Containment Heat Removal System	The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.	INTENT	The ANS is designed to maintain containment heat removal and pressure control by passive means that does not depend on functioning of active components or safety-related ac electric power. The containment heat removal system will be designed to permit appropriate periodic pressure and functional testing.
GDC 41	Containment Atmosphere Cleanup	Systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features and have suitable interconnections, leak detection, isolation, and containment capabilities to assure that for on-site electric power	INTENT	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 41 continued		system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available), its safety function can be accomplished, assuming a single failure.		
GDC 42	Inspection of Containment Atmosphere Cleanup Systems	The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.	TOTAL	
GDC 43	Testing of Containment Atmosphere Cleanup Systems	The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.	INTENT	GDC 43 implies that safety-related ac power is required. The containment atmosphere cleanup system will use safety-related dc power and passive means. Nevertheless, the system will be designed to permit appropriate periodic pressure and functional testing.
GDC 44	Cooling Water	A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 44 continued				
GDC 45	Inspection of Cooling Water System	The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.	TOTAL	
GDC 46	Testing of Cooling Water System	The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.	TOTAL	
V. REACTOR CONTAINMENT				
GDC 50	Containment Design Basis	The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by 10 CFR 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.	TOTAL	The containment design should consider other energy sources such as uncontrolled combustible gas-oxygen recombination, burning and/or local detonation of combustible gases, etc.

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 51	Fracture Prevention of Containment Pressure Boundary	The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.	TOTAL	
GDC 52	Capability for Containment Leakage Rate Testing	The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.	TOTAL	
GDC 53	Provisions for Containment Testing and Inspections	The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.	TOTAL	
GDC 54	Piping Systems Penetrating Containment	Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate nuclear safety. Determination of the appropriateness of these requirements such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.	INTENT	GDC 55 implies that safety-related ac power is required. The INTENT of GDC 55 will be met without safety-related ac power through the use of safety-related dc power and passive means (e.g., natural circulation cooling).

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GDC 56	Primary Containment Isolation	Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.	INTENT	GDC 56 implies that safety-related ac power is required. The INTENT of GDC 56 will be met without safety-related ac power through the use of safety-related dc power and passive means (e.g., natural circulation cooling).
GDC 57	Closed System Isolation Valves	Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.	TOTAL	

Table B.5. Compliance Commitment to 10 CFR 50, Appendix A, "General Design Criteria," for the Design of the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
VI. FUEL AND RADIOACTIVITY CONTROL				
GDC 60	Control of Releases of Radioactive Materials to the Environment	The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste products during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations on the release of such effluents to the environment.	TOTAL	
GDC 61	Fuel Storage and Handling and Radioactivity Control	The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.	TOTAL	The spent fuel storage pool will be designed with a sufficient pool volume to accommodate a loss of normal fuel pool decay heat removal without the need for ac power or heat sinks outside containment while satisfying the shielding and cooling requirements of the fuel for up to 4h minimum requirement [72-h goal].
GDC 62	Prevention of Criticality in Fuel Storage and Handling	Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	TOTAL	
GDC 63	Monitoring Fuel and Waste Storage	Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	TOTAL	
GDC 64	Monitoring Radioactivity Releases	Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operation, including anticipated operational occurrences and postulated accidents.	TOTAL	

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
The ANS facility design includes features that do not exist in, or are significantly different from, NRC-licensed light water power reactors. To account for these differences, additional design criteria had to be developed. These additional criteria, called Supplemental Design Criteria (SDC), reflect the safety philosophy for advanced reactors with inherent and passive safety systems. The SDCs are grouped by functional category. These categories are identical to those used for the NRCs General Design Criteria (see Table B.5) except that an additional category, VII. EXPERIMENTS, had to be added. The purpose of these SDCs is to ensure a dependable research tool while protecting the public and the investment.			
Guidance is given for each requirement to aid in determining applicability at the SDD level. The guidance given is, in essence, a requirement. Any requests for exceptions to the guidance must be submitted to the chairman of the plant design requirements working group. Each exception will be reviewed on a case-by-case basis.			
I. OVERALL REQUIREMENTS			
SDC 01	Physically Handicapped	Physically handicapped shall be accommodated in accordance with DOE 6430.1A and 41CFR101-19.6, Uniform Federal Accessibility Standard. DOE 6430.1A requires that the physically handicapped be accommodated, unless either the facility is specifically restricted to able-bodied personnel or the physical handicap would prevent rapid response and evacuation.	The reactor building, reactor support building, detritiation building and other defined areas of the ANS buildings/site should be limited to personnel able to negotiate stairs unaided and rapidly evacuate in event of accidents.
SDC 02	Common-Cause Failure Criterion	Reactor safety-related systems shall be designed against common cause/mode, multiple, and coincident events or conditions that result in the failure of multiple redundant systems to perform their safety functions. Common-mode failure is the simultaneous failure of redundant equipment caused by a single internal or external phenomena.	Redundant safety-related mechanical and electrical systems should be designed to comply with the separation criteria in NRC Reg. Guide 1.75, in order to provide protection against common-cause failures. NRC GDC 24 provides separation criteria for protection systems. Consideration should be given to avoidance of design, fabrication, maintenance, operational or installation deficiency common to multiple components. Consideration should be given in the design to the possibility of systematic, nonrandom, concurrent failures of redundant elements in the protection systems. Diverse systems that accomplish the safety function should be considered. Common cause/mode and multiple failures should be considered in probabilistic risk analysis and in deterministic analysis. Events which are the result of unique multiple or common cause failures in safety-related equipment should be identified based on industrial experience and regulatory requirements.

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 03	Investment Protection	The plant shall be designed to protect the plant investment.	The plant should be designed to protect the plant investment by (1) requiring only simple operator actions and minimum off-site assistance for anticipated events, (2) minimizing the probability of loss of heavy water inventory, (3) heavy water boundaries that may have a high probability of leakage phase not be submerged in light-water-filled pools or cells, (4) maximizing the use of component designs based on proven technology, (5) maximizing the use of simplified concepts in the reactor and plant system design in order to enhance reactor safety, operability, testability, availability, and maintainability, and (6) maximizing the use of common components, i.e., sensors, cable, valves, and piping among the various plant systems. The plant shall have a low probability of loss of ac power. The plant shall have at least two separate and independent nonsafety related off-site ac power connections. On-site generated electrical power should be provided to "critical" non-safety related systems for up to 2-h minimum [8-h goal] in the event of power interruptions or loss of off-site power. Selected plant structures shall be designed to maintain their structural integrity in the event of a 100 year seismic or tornado load. The plant shall depend on simple operator actions and minimum off-site assistance. During accidents, (1) a means for manual initiation of protective systems, in case they fail to initial automatically, shall be provided from a central main control room; (2) personnel access into other areas of the plant shall not be required to mitigate the accident unless that area and pathway to it provides adequate protection for personnel; and (3) operator access into the reactor containment building, or other potential high radiation area, to safely mitigate the accident shall not be required.

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 04	Passive Safety	Safety-related system designs shall maximize the use of passive means to accomplish their safety functions.	Passive means include the use of natural phenomena, stored energy, stored heat sinks, that do not depend on active components or operator actions. Passive spent fuel and reactor emergency cooling includes use of passive means for coolant flow, pressure, and inventory control. Passive means should be used to the maximum extent practicable for other containment functions including fission product control and combustible gas control. In those cases where passive designs are not practicable, safety-related systems design should minimize the use of active features. These active safety-related systems should use the following in order of priority: (1) components that are self actuating (e.g check valves) instead of those that require external actuation, (2) components that fail to a safe condition upon loss of actuation power or signal, (3) DC instead of AC electrical power, (4) systems/components that are used during normal operation instead of those that are in standby and need to be placed on line in event of an accident. Reactor pool and water filled cells should provide passive reactor heat sinks in event of loss of reactor heat sinks outside containment with sufficient capacity to remove decay heat from the reactor core for up to 72 hours [goal].
SDC 05	Design Margins	Systems, structures and components shall be designed to provide design margins.	The design should include margins to provide designed-in capability to accommodate transients without causing initiation of protection safety systems (i.e., scram), allowing the operator sufficient time to assess and deal with upset conditions with potential for damage, and minimizing the potential of exceeding limits that require power reduction or shutdown. Margins should also be provided for variables such as containment design leak rates and reactor system design pressures.

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 06	Design Bases for Protection Against Non-Natural Phenomena	Structures, systems, and components important to safety shall be designed to withstand the effects of non-natural phenomena such as credible aircraft crashes, explosions, and other non-natural threats external to the plant without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect (1) appropriate consideration of the most severe of the non-natural phenomena for the site and surrounding area, with sufficient margin, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the non-natural phenomena, and (3) the importance of the safety functions to be performed.	
SDC 07	Noise	The maximum noise level in the experimental areas inside containment shall be below *[TBD]*[B-57] db, and below *[TBD]*[B-58] db in offices.	
II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS			
SDC 20	Failed Fuel Monitoring	Instrumentation shall be provided to detect and monitor the level of failed fuel during reactor operations.	Failed fuel is defined as fuel that has lost leak-tight integrity of the cladding and is releasing radioactivity. The failed fuel detection instrumentation should have the capability of detecting fuel defects of a size less than that that would result in failure of adjacent fuel or cause blockage of the coolant channel. *[HOLD]*[B-3]
SDC 21	Flow Blockage	The reactor system shall be designed to assure an extremely low probability of flow blockage.	The irradiation material samples/transuranic rods and any other reactor internal component designs should assure an extremely low probability of reactor core flow blockage for any mode of failure. A fuel element inlet filter may be provided downstream of the transuranic rods to assure no core flow blockage in event of rod failure, but any such filter design should assure a low probability of core flow blockage in event of its failure.
			These components and associated structural supports should be

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 21 continued			designed to maintain their structural integrity in event of safe shutdown earthquake and other reactor-related design basis events.
III. PROTECTION AND REACTIVITY CONTROL SYSTEMS			
IV. FLUID SYSTEMS			
SDC 40	Reactor Assembly Coolant Boundaries	Those portions of the reactor assembly that form the *reactor primary coolant pressure boundary* shall be designed to assure no loss of either reflector coolant boundary, reactor shutdown, reactor cooling, or other safety system capability resulting in core damage in event of any single passive failure of reactor primary coolant pressure boundary, safe shutdown earthquake and other reactor-related design basis events/conditions. Those portions of the reactor assembly that form the *reflector coolant boundary* shall be designed to assure no loss of either reactor primary pressure boundary, reactor shutdown, or other safety system capability resulting in core damage in event of any single failure of the reflector coolant boundary, safe shutdown earthquake and other reactor-related design basis events/conditions.	The reflector coolant boundary may be considered to perform the inventory control function for the *reactor coolant pressure boundary* (as defined in 50.2 in Table B.4 in this Appendix) with normal leakage from the reactor primary coolant boundary. Other requirements that should be applied to the design of those portions of the reactor assembly coolant boundaries that form the reactor primary and reflector coolant boundaries include NRC GDCs 30, 31, and 32.
SDC 41	Reactor Assembly Cooling	The reactor assembly cooling systems shall enable cooling for up to 72-h after shutdown, considering a single failure and no heat transport outside containment.	

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 42	Passive Cooling	Means shall be provided so that safety-related cooling systems for the reactor that normally depend on a source of electrical power shall accomplish their safety function after a station blackout (total loss of normally connected off-site and on-site generated power) for *either* the time required to complete their safety function or up to 4-h (goal is 72-h).	No diesel generated ac electrical power should be required within the first 4 hours of a station blackout for safety-related cooling systems for the reactor.
V. REACTOR CONTAINMENT			
SDC 50	Reactor Containment Access	A frequently opened reactor containment is required to permit access of experimental and operations personnel to satisfied the ANS research goals and objectives. The reactor containment penetrations that are frequently operated or normally open should be sufficiently reliable to assure that containment safety functions and radiation release goals are achieved.	
SDC 51	Experimental System Isolation Valves	Experimental tubes, including guide and rabbit tubes, that penetrate the reactor containment boundary shall be provided with two automatic isolation valves. Isolation valves shall be located as close to the containment wall as possible, with one valve inside and one valve outside of containment. These isolation valves shall be designed to fail closed upon loss of actuating power.	The safety-related function of these valves are to provide a barrier against the loss of reactor primary water, reactor pool water, reflector vessel water, and the uncontrolled release of radioactivity to the environment outside the reactor containment. These isolation valves should automatically close as necessary to maintain their leak-tight integrity in event of a release to radioactivity within containment. (Also see GDC 50 and 54).
SDC 52	On-Line Monitoring of Containment Isolation Systems	The reactor containment isolation status shall be continuously monitored on-line.	
SDC 53	Reactor Containment Integrity	The reactor containment shall be designed to provide for ease of emergency closure during shutdown operation including station blackout conditions.	

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 54	Severe Accidents	Design features to prevent or mitigate severe accidents shall be evaluated and included in the design as appropriate to improve safety margins. Some of the severe accidents may be beyond those included in the containment design basis but are considered to evaluate containment design margins. Severe accidents within the design basis and severe accidents beyond the design basis are described in Appendix C.	Fuel-coolant interaction (FCIs): Reactor containment barrier structural margins should be determined for beyond design basis events such as credible steam explosions. Barriers include experimental tubes/isolation valves and the primary steel containment vessel. Missile shields over the reactor pool may be required to prevent damage to the containment barrier. Combustible gases from FCIs are covered in GDC 50, 50.44 and 50.34(f)(2)(ix). Core debris recriticality: Recriticality from core debris should be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Liquid poison may be required depending on potential locations for the core debris. Detonation of gases: Structures, systems, and components should be provided for combustible gas control. Reactor containment and local areas, structural margins should be evaluated for beyond-design-basis events such as detonation of combustible gases. Core debris cooling: Structures, systems, and components should be designed to assure adequate coolability of core debris resulting from beyond-design-basis events. Molten core-concrete interactions (MCCI's): Structures, systems, and components designs should be evaluated to minimize the generation of combustible gases resulting from interaction between core debris and concrete. Special concrete materials should be considered in selected areas, including the subpile room. A dedicated over-pressure vent from the reactor subpile room to the upper containment may also be evaluated. Fission product and radioactive release transport: Reactor containment systems design requirements, or technical specification limits may be reduced with improved understanding of fission product release from uranium-aluminum silicide fuel and transport/plateout within the reactor containment building. Containment over-pressurization: Reactor containment

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 54 continued			structural margin should be determined for safety margin events. Safety margin events are defined in Appendix C.
VI. FUEL AND RADIOACTIVITY CONTROL			
SDC 61 *[HOLD]*[B-4]	Refueling	The refueling and maintenance system shall be designed to ensure that any single equipment failure does not (1) exceed radioactivity limits, (2) increase the risk of injury to operating or maintenance personnel, and (3) result in contaminating the heavy water with light water.	
SDC 62	Tritium Concentration and Inventory	Systems shall be provided to limit the concentration of tritium in heavy water and in building/cell atmosphere in order to assure that radiation exposure to personnel is as low as reasonably achievable during operations and maintenance.	Systems should be provided to monitor concentration of tritium in heavy water, light water pools and cells, building areas, equipment cells or cell vents, and equipment vents. The goal for the tritium concentration limit in aqueous systems is 74 GBa/L (2 Ci/liter.) The requirement is 185 GBa/L (5 Ci/liter.) The total quantity of tritium stored on-site outside the containment boundary should be less than that that would result in a whole body dose greater than 5m5v (0.5 rem) or its equivalent at the site boundary in the event of the largest credible release assuming tritium is in the form of tritiated water vapor.
SDC 63	Waste Management	The plant shall be designed to enable on-site storage of all radioactive solid waste, including recovered tritium from one *[HOLD]*[B-5] year of operation. The radioactive waste systems shall be designed assuming 1.0% failed fuel *[HOLD]*.[B-6]	

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 64	Limited Volume Air and Water Cells	Reactor pool and water-filled cells shall provide reactor coolant inventory control and retention of water soluble fission products in event of breaks (LOCAs) in reactor primary coolant boundary and reflector heavy water boundary.	Reactor pool and water-filled cells should be designed to maintain their water leak-tight integrity in event of safe shutdown earthquake and a break in the reactor primary coolant boundary. Reactor and reflector coolant boundaries, water-filled pools and cells should be provided with instrumentation to detect and monitor water leakage. Limited volume air cells (enclosures) should provide passive means for reactor coolant inventory control and fission product confinement in event of a break in reactor primary coolant boundary (LOCA). These cells (enclosures) should be designed to maintain their leak-tight integrity in event of safe shutdown earthquake and a break in the reactor primary coolant boundary. The cells should be provided with instrumentation to detect water leakage into the cells and across the cell liners.
VII. EXPERIMENTS			
SDC 70	Equipment Cells	The radioactive equipment cells shall be designed to provide adequate radiation shielding and to prevent the uncontrolled release of radioactivity into operating areas or to the environment.	The cell boundary consists of the cell structure, cell penetrations, cell penetration isolation valves, and associated cell atmosphere treatment systems. The equipment cell should be designed to satisfy the single failure criteria. These cells may be metal-lined for recovery of tritiated heavy water or to satisfy the design basis leakage rate/seismic loads.
SDC 71	Beam Tube Isolation	Neutron guide, beam, and rabbit tubes that penetrate the shield wall shall be provided with redundant barriers.	Redundant barriers may be provided by using either double isolation valves or one valve and one continuously installed leak-tight barrier designed to the higher of the reflector vessel or containment design pressure. Isolation valves, if any, shall be located as close to the shield wall as possible and designed to fail closed.

Table B.6. Supplemental Design Criteria (SDC) for the Advanced Neutron Source

NUMBER	TITLE	CRITERION (REQUIREMENT)	GUIDANCE
SDC 71 continued			The safety-related function is to provide a barrier against the loss of reactor primary water, reactor pool water, reflector vessel water, and the uncontrolled release of radioactivity inside the reactor containment. These isolation valves should automatically, as necessary, close and maintain their leak-tight integrity in event of a radioactivity release within containment.
SDC 72	Hot Cells	The experiment hot cells shall be designed to provide radiation shielding and to prevent the uncontrolled release of radioactivity into operating areas.	The hot cell boundary consists of the cell structure, cell penetrations (i.e., manipulators), shield windows, and cell atmosphere cleanup systems. The cells may be designed as seismic category I-L structures, if their failure in event of an SSE could result in failure of other safety related functions.
SDC 73	Cold Source	Each cold source system shall be designed to ensure an extremely low probability of any of the following events: (1) loss of reactor containment, (2) damage to the reactor assembly, (3) loss of reactor shutdown capability, or (4) damage to any other reactor safety system.	Each cold source system should be designed to contain an internal deflagration of a stoichiometric mixture of deuterium and air initially at 0.1 MPa (14.5 psi) and 300 K (80°F). Each system should be designed to ensure no deuterium release into containment in event of (1) safe shutdown earthquake, (2) total loss of ac/dc electrical power, (3) any other reactor-related design basis event, or (4) total loss of helium refrigeration. Pressure relief devices should not vent to air inside containment.

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

A review of the Regulatory Guides (revised June, 1990) is given below. Justification and Guidance is given for each requirement to aid in determining applicability at the SDD level. The guidance given is, in essence, a requirement. Any requests for exceptions to the guidance must be submitted to the chairman of the plant design requirements working group. Each exception will be reviewed on a case-by-case basis.

Definitions required to interpret this table are:

TOTAL — *All technical requirements are applicable as written.*

PARTIAL — *Due to specific requirements in the regulations or design features in the ANS, not all of the technical requirements apply as written. The PDR shall define that part of the regulation that is and is not applicable at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

INTENT — *Only the purpose (or intent) of the technical parts of the regulation is applicable. The PDR shall define the intent of the regulation and how it applies to the ANS at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

NONE — *Due to specific requirements in the regulations or design features in the ANS, none of the technical requirements are applicable. The PDR shall provide justification why the regulation is not applicable.*

N/A-L — *The regulation sets forth only licensing procedural requirements, and does not contain any technical requirements.*

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.1 (11/70)	NPSH for emergency cooling and containment heat removal system pumps	TOTAL	RG 1.1 supports GDC 41 and requires that the suction piping for all pumps required for safe shutdown of the reactor, during both normal and accident conditions and including the cooling of both the core and the containment, is designed and located to ensure adequate net positive suction head (NPSH).
RG 1.2 (11/70)	Thermal shock to reactor pressure vessel	INTENT	This program includes the investigation into the embrittlement of carbon steel reactor vessels by irradiation and possible vessel brittle failure from high stresses caused by a large thermal gradient. Because the ANS has an aluminum pressure vessel and does not inject large quantities of cold coolant into the core pressure boundary tube (i.e., the vessel), large thermal gradients will not occur. However, because of the high neutron flux, vessel embrittlement may occur. This will be monitored.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.3 (Rev. 2, 6/74)	Assumptions for evaluating radiological consequences of loss of coolant accidents for BWRs	INTENT	RG 1.3 supports 50.34 and 10 CFR 100. This guide provides acceptable assumptions for evaluating the design basis loss of coolant accident for a boiling water reactor (BWR). The methodology, based on TID-14844, is used to determine compliance with the siting requirements of 10 CFR 100. Because the source term for the ANS is much smaller than that for a power reactor, the intent of this guide will be followed. The source terms for the ANS are given in Appendix C of the PDR.
RG 1.4 (Rev. 2, 6/74)	Assumptions for evaluating radiological consequences of loss of coolant for PWR's	INTENT	RG 1.4 supports 50.34 and 10 CFR 100. This guide provides acceptable assumptions for evaluating the design basis loss of coolant accident for a BWR. The methodology, based on TID-14844, is used to determine compliance with the siting requirements of 10 CFR 100. Because the source term for the ANS is much smaller than that for a power reactor, the intent of this guide will be followed. The source terms for the ANS are given in Appendix C of the PDR.
RG 1.5 (3/71)	Assumptions for evaluating the radiological consequence of a steam line break accident for BWRs	PARTIAL	RG 1.5 supports 50.34 and provides specific guidance for a steam line break accident for a BWR. However, sources (e.g., iodine and noble gases), dose conversions, atmospheric diffusion, and breathing rates will be reviewed to determine radiological consequences of a pipe break.
RG 1.6 (3/71)	Independence between redundant standby power source and distribution system	TOTAL	All devices relating to safety and requiring electric power (dc) and the distribution systems will be designed for separation of safety loads into independent groups.
RG 1.7 (Rev. 2, 11/78)	Control of combustible gas concentration in containment following a loss-of-coolant accident	INTENT	RG 1.7 supports GDC 35, GDC 41, GDC 50 and 50.44. This guide provides guidelines for evaluating the production of combustible gases following a LOCA in light-water reactor plants with zircaloy-clad oxide fuel. Because the ANS does not use zircaloy-clad oxide fuel, most of this RG is not applicable. However, the intent of those sections that describe methods and criteria for

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.7 (Rev. 2, 11/78) continued			combustible gas concentration control within the containment are applicable.
RG 1.8 (Rev. 2, 4/87)	Personnel selection and training	TOTAL	RG 1.8 supports 50.34(b) and 10 CFR 55 and is REQUIRED by DOE 5480.6. RG 1.8 endorses ANSI/ANS-3.1-1981, with exceptions, which provides an approach for complying with the qualifications and training requirements in 10 CFR 50 and in 10 CFR 55. RG 1.8 also endorses ANSI N18.1-1971, with exceptions.
RG 1.9 (Rev. 2, 12/79)	Selection of diesel generator set capacity for standby power supplies	INTENT	RG 1.9 supports GDC 17 and 10 CFR 50 Appendix B. RG 1.9 requires that systems are designed to withstand a complete loss of ac power (other than inverter supplied power) for at least 2 h without exceeding equipment limits. Diesel generator ac electrical power shall not be required within the first 2 h. Battery-inverter supplied ac power is used for this duration. RG 1.9 also endorses IEEE Std 387-1977, with exceptions, which delineates principal design criteria and qualification testing requirements. This standard is supplementary to IEEE Std 308-1974, which is endorsed with exceptions by RG 1.32. The ANS shall meet the intent of RG 1.9 by utilizing only stored energy sources such as dc power, compressed gas, or gravity.
RG 1.10	Mechanical (Cadweld) splices in reinforcing bars of Category I concrete structures	NONE	RG 1.10 was withdrawn and is superseded by RG 1.136 Rev. 2, 6/81.
RG 1.11 (Rev. 1, 2/72)	Instrument lines penetrating primary reactor containment	TOTAL	RG 1.11 supports GDC 55 and GDC 56. Any instrument lines penetrating containment will comply with RG 1.11. Guidance given in RG 1.11 should also be applied to any instrument lines penetrating the biological shield.

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.12 (Rev.1, 4/74)	Instrumentation for earthquakes	TOTAL	RG 1.12 is consistent with 50.36(c) and is used to satisfy the requirements of Appendix A of 10 CFR 100. RG 1.12 endorses, with exceptions, ANSI N18.5 for satisfying the seismic instrumentation requirements. Because the frequency range for the earthquake instrumentation required in C.4.b is not commercially available, the ANS may not be able to comply totally. However, proposed Rev. 2 (7/24/80) endorses ANSI/ANS 2.2-1978, which accepts most of the exceptions to Rev. 1.
RG 1.13 (Rev. 1, 12/75)	Fuel storage facility design basis	TOTAL	RG 1.13 is consistent with the requirements of GDC 61. The spent fuel pool storage facility will be designed to Seismic Category I requirements to provide protection from missiles and heavy loads. The pool will be designed with appropriate water level monitoring and with volume sufficient to provide fuel pool cooling during normal and postulated accidents.
RG 1.14 (Rev. 1, 8/75)	Reactor coolant pump flywheel integrity	TOTAL	RG 1.14 supports GDC 4. If any reactor coolant pumps have inertia flywheels, the ANS will COMPLY with RG 1.14.
RG 1.15	Testing of reinforcing bars for Category I concrete structures	NONE	RG 1.15 was withdrawn and is superseded by 1.136 Rev. 2, 6/81.
RG 1.16 (Rev. 4, 8/75)	Reporting of operating information	N/A-L	RG 1.16 supports 50.36 and relates to the technical specifications concerning health and safety, periodic routine operating reports, and "abnormal" occurrence reports for submittal to the NRC. Reporting requirements for the ANS are given in DOE 5484.1 and DOE 5484.2.
RG 1.17 (Rev. 1, 6/73)	Protection against industrial sabotage	TOTAL	RG 1.17 is consistent with 10 CFR 50.34(c). RG 1.17 endorses ANSI N18.17, with additions, which provides procedural measures for protection against industrial sabotage. The additions provide a basis for a physical security plan for the protection against industrial sabotage. The ANS will COMPLY with the requirements and

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.17 (Rev. 1, 6/73) continued			recommendations contained in ANSI N18.17, "Industrial Security for Nuclear Power Plants."
RG 1.18	Structural acceptance test for concrete primary reactor containments	NONE	RG 1.18 was withdrawn and is superseded by 1.136 Rev. 2, 6/81.
RG 1.19	Nondestructive examination of primary containment liner welds	NONE	RG 1.19 was withdrawn and is superseded by 1.136 Rev. 2, 6/81.
RG 1.20 (Rev. 2, 5/76)	Vibration measurement on reactor internals	INTENT	RG 1.20 supports GDC 1 and 50.34. RG 1.20 provides a comprehensive vibration assessment program for use in verifying the structural integrity of reactor internals for flow-induced vibrations. The overall program includes individual analytical, measurement, and inspection programs. Individual programs should be used cooperatively to verify structural integrity and to establish margin of safety. The INTENT of RG 1.20 is applicable to the ANS; however, the testing details given may not be appropriate.
RG 1.21 (Rev. 1, 6/74)	Measuring and reporting of effluents from nuclear power plants	INTENT	RG 1.21 supports GDC 60, GDC 64, 20.1, 20.106, and 50.36a. RG 1.21 describes programs for measuring, reporting, and evaluating releases of radioactive materials in liquid and gaseous effluents and guidelines for classifying and reporting the categories and curie content of solid wastes. It also provides guidelines for calculating potential annual radiation doses to individuals and populations. The program may need to be supplemented or modified because of the design of the ANS.
RG 1.22 (2/72)	Periodic testing of protection system actuation functions	TOTAL	RG 1.22 supports GDC 20 and GDC 21. RG 1.22 describes acceptable methods of including the actuation devices for protections systems. These systems must be designed to allow periodic testing.

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.23 (2/72)	Onsite meteorological programs	TOTAL	RG 1.23 supports 10 CFR 20, 10 CFR 100, 10CFR51, 10 CFR 50 Appendix I and the NEPA. RG 1.23 also endorses ANSI/ANS-2.5-1984, with exceptions, which provides guidance pertaining to the meteorological measurement program needed to provide meteorological data to be used in the evaluation of the environmental impact and the impacts of the routine radioactivity release and the accidental radioactivity release that may be associated with the construction and operation of plants. (Guidance for record retention is given in RG 1.33.) RG 1.23 consolidates existing guidance on meteorological measurements contained in RG 1.97, RG 1.101, and Supplement 1 to NUREG-0737.
RG 1.24 (3/72)	Assumptions used for evaluating potential radiological consequences of a pwr rad gas storage tank failure	TOTAL	RG 1.24 supports 50.34 and 10 CFR 100. If the ANS is going to use radioactive gas storage tanks to permit decay of radioactive gases as a means of reducing or preventing the release of radioactive materials to the atmosphere, RG 1.24 will be used to evaluate the consequences of a gas storage tank failure.
RG 1.25 (3/72)	Assumptions used for evaluating potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for BWR and PWR's	PARTIAL	RG 1.25 supports 50.34. The detailed provisions in RG 1.25 applies to oxide fuel in fuel rods. Because of the differences in fuel, fuel handling, and fuel storage, RG 1.25 is generally not applicable to the ANS. However, RG 1.25 will be reviewed for the applicability of χ/Q and thyroid dose conversion factors.
RG 1.26 (Rev.3, 2/76)	Quality group classification and standards	INTENT	RG 1.26 supports GDC 1 and 50.55a. This guide describes a quality classification system related to specified national standards that may be used to determine quality standards for satisfying GDC 1 for other safety-related components containing water, steam, or radioactive material in LWRs. Details of the Quality Group standards applicable to the ANS are given in PDR, Chap. 4.

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.27 (Rev.2, 1/76)	Ultimate heat sink	INTENT	RG 1.27 supports GDC 44 and GDC 2 for plants that use water primarily as the ultimate heat sink. Cooling towers (2) are required to satisfy the same requirements as the sink if they are used to accomplish the sink safety functions. Because heat can be transferred to the equipment cells [as a short term means (up to 72 h) of providing cooling], the detailed provisions of RG 1.27 on the design of the reactor secondary cooling system (RSCS) are applicable only where appropriate.
RG 1.28 (Rev. 3, 8/85)	Quality Assurance program requirements (design & construction)	TOTAL	RG 1.28 supports 10 CFR 50 Appendix B and endorses, with additions and modifications, ANSI/ASME NQA-1-1983 and ANSI/ASME NQA-1a-1983 Addenda. The methods described in RG 1.28, Revision 3, for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program are considered to be generally equivalent, from a programmatic standpoint, to the methods described in Revision 2 to RG 1.28 and RG 1.58, RG 1.64, RG 1.74, RG 1.88, RG 1.123, RG 1.144, and RG 1.146.
RG 1.29 (Rev. 3, 9/78)	Seismic design classification	PARTIAL	RG 1.29 supports GDC 2, 10 CFR 50 Appendix B, and 10 CFR 100 Appendix A. This guide describes a method for identifying and classifying those features that should be designed to withstand the effects of the safe shutdown earthquake (SSE). Regulatory positions C.1.e and C.1.f are not applicable to the ANS.
RG 1.30 (8/72)	Quality Assurance requirements (installation, inspection, and test of instrumentation and electrical equipment)	INTENT	RG 1.30 establishes that specific standards (IEEE 336-1971 and ANSI N45.2.4-1972) are adequate bases to comply with the pertinent requirements of 10 CFR 50 Appendix B. RG 1.30 provides guidance to ensure that appropriate quality assurance requirements that comply with 10 CFR 50 Appendix B are used for installation, inspection, and testing of instrumentation and electric equipment. Because of the passive design of the plant,

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.30 (8/72) continued			reliance on electric power is not as great as in current generation nuclear power plants. Therefore, the INTENT of RG 1.30 will be met because the QA requirement applied to ac electric equipment will be consistent with its design requirement.
RG 1.31 (Rev. 3, 4/78)	Control of stainless steel welding	TOTAL	RG 1.31 supports GDC 1, GDC 14, and 10 CFR 50 Appendix B. RG 1.31 describes an acceptable method for control of welding in fabricating and joining safety-related austenitic stainless steel components and systems. The ANS will either meet the INTENT or TOTAL compliance with RG 1.31 based on the verification method used. Rather than magnetic verification, chemical analysis provisions of ASME B&PV Code Section III, NE-2430 may be used for determining the amount of delta ferrite in each heat or lot of austenitic stainless steel filler material.
RG 1.32 (rev.2, 2/77)	Use of IEEE Std 308-1971, criteria for Class IE electric systems for nuclear generating station	INTENT	RG 1.32 supports GDC 17 and GDC 18, and endorses IEEE Std 308-1974 and IEEE Std 450-1975. IEEE 308-1974 has been replaced by IEEE 308-1980. Other RGs that must be considered while following the guidance given in RG 1.32 are: RG 1.6, RG 1.9, RG 1.75, RG 1.81 and RG 1.93. RG 1.32 presumes the existence of safety related ac power and establishes requirements relative to appropriate capability, independence, and reliability. The intent is to establish specific requirements to assure a highly reliable electrical power supply system commensurate with the functional application. RG 1.32 implies that either on-site or off-site ac power is required to meet the conditions of maintaining core cooling, containment integrity, and related support activities.
RG 1.33 (Rev. 2, 2/78)	Quality Assurance program requirements (operation)	TOTAL	RG 1.33 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.33 provides guidance for quality assurance program requirements for the operations phase of plant life.

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.34 (12/72)	Control of electroslag weld properties	TOTAL	RG 1.34 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.34 describes an acceptable method of implementing the control of weld properties when fabricating electroslag welds for nuclear components made of ferritic or austenitic materials. Because reactor cooling piping will be fabricated with austenitic stainless steel, the ANS will comply in TOTAL with RG 1.34 if electroslag weld properties are used.
RG 1.35 (Rev. 2, 1/76)	In-service surveillance of ungrouted tendons	TOTAL	RG 1.35 supports GDC 53. If the containment structure for the ANS has ungrouted tendons in prestressed concrete, RG 1.35 will be followed.
RG 1.35.1 (4/79)	Determining prestressing forces for inspection of prestressed concrete containments	TOTAL	RG 1.35.1 supports GDC 53. RG 1.35 describes a basis for developing an appropriate in-service inspection and surveillance program for ungrouted tendons in prestressed concrete containments. RG 1.35.1 expands and clarifies on how to determine prestressing forces to be used. RG 1.35.1 is applicable to the ANS if the containment uses prestressed concrete.
RG 1.36 (2/73)	Nonmetallic thermal insulation for austenitic stainless steel	TOTAL	RG 1.36 supports GDC 1, GDC 14 and GDC 31. RG 1.36 provides guidance for the selection and use of nonmetallic thermal insulation to minimize any contamination that could promote stress-corrosion cracking in the stainless steel portions of the reactor coolant pressure boundary and other systems important to safety.
RG 1.37 (3/73)	Quality Assurance requirements for cleaning fluid systems and associated equipment of water cooled nuclear plants	TOTAL	RG 1.37 supports 10 CFR 50 Appendix B and describes an acceptable method of complying with the regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled fluid systems. RG 1.37 endorses, with exceptions, ANSI N45.2.1-1973. The ANS will comply with RG 1.37.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.38 (Rev. 2, 5/77)	Quality Assurance requirements for packaging, shipping, receiving storage and handling of items for water cooled plants	TOTAL	RG 1.38 supports 10 CFR 50 Appendix B and describes acceptable quality assurance requirements for the packaging, shipping, receiving, storage, and handling of items for nuclear power plants. Because the ANS fuel is highly enriched uranium (HEU), the ANS will have TOTAL compliance with RG 1.38 with additional requirements promulgated by the DOE. RG 1.38 endorses, with additions and modifications, ANSI N45.2.2-1972.
RG 1.39 (Rev. 2, 9/77)	Housekeeping requirements for water cooled nuclear plants	TOTAL	RG 1.39 supports 10 CFR 50 Appendix B and describes an acceptable method of complying with the regulations with regard to housekeeping requirements for the control of work activities, conditions, and environments at the plant site. RG 1.39 endorses, with exceptions and modifications, ANSI N45.2.3-1973.
RG 1.40 (3/73)	Qualification test of continuous duty motors installed inside containment of water cooled nuclear plants	TOTAL	RG 1.40 supports 10 CFR 50 Appendix B. If the ANS uses continuous duty motors inside containment, the ANS shall comply with RG 1.40 (and IEEE 334-1971).
RG 1.41 (3/73)	Pre-operational testing of redundant electric power systems	TOTAL	RG 1.41 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.41 describes a method for verifying the proper assignments of redundant load groups to the related on-site power sources. RG 1.41 states that those on-site electric power systems designed in accordance with RG 1.6 and RG 1.32 should be tested to verify the existence of independence among redundant on-site power sources and their load groups. The ANS will comply with RG 1.41.
RG 1.42	Interim licensing policy on as low as practicable for gaseous radioiodine releases from light-water-cooled nuclear power reactors	NONE	RG 1.42 was withdrawn on 3/22/76.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.43 (5/73)	Control of stainless steel weld cladding of low alloy steel components	TOTAL	RG 1.43 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.43 describes methods for selecting and controlling of welding processes used for cladding ferritic steel components with austenitic stainless steel to restrict practices that could result in underclad cracking. This guide is limited to forgings and plate material and does not apply to other product forms such as castings and pipe.
RG 1.44 (5/73)	Control of the use of sensitized stainless steel	TOTAL	RG 1.44 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.44 describes the control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking.
RG 1.45 (5/73)	Reactor coolant pressure boundary leakage detection systems	TOTAL	RG 1.45 supports GDC 30. RG 1.45 provides methods for selecting leakage detection systems for the reactor coolant pressure boundary. Instrumentation shall be provided to detect and monitor leakage from all reactor and reflector coolant boundaries, including "normal" design leakage, and from water-filled pools and cells. The instrumentation shall have the capability to detect leakages less than technical specification limits and to satisfy the requirements of NRC RG 1.45.
RG 1.46	Protection against pipe whip inside containment	NONE	RG 1.46 was withdrawn on 3/11/85.
RG 1.47 (5/73)	Bypassed and inoperable status indication for nuclear power plant safety systems	TOTAL	RG 1.47 supports 50.55a and 10 CFR 50 Appendix B. RG 1.47 endorses, with exceptions and modifications, IEEE Std 279-1971. Automatic indication of the bypass or inoperability of each redundant portion of a system that performs a function important to safety shall be continuously indicated in the control room.
RG 1.48	Design limits and loading combinations for seismic Category I fluid system components	NONE	RG 1.48 was withdrawn on 3/11/85.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.49 (Rev. 1, 12/73)	Power levels of water cooled nuclear plants	INTENT	RG 1.49 supports 50.34 and 10 CFR 100. The intent of RG 1.49 is to allow for a slightly higher assumed power level to allow for possible instrument errors. Therefore, analyses of possible off-site radiological consequences of postulated design basis accidents shall be performed for at least 1.02 times the proposed core power level.
RG 1.50 (5/73)	Control of preheat temperature for welding of low-alloy steel	TOTAL	RG 1.50 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.50 describes requirements to control the welding for low-alloy steel components during initial fabrication in order to reduce the propensity for cracks (cold cracks) or reheat cracks forming in underbead areas and heat-affected zones. RG 1.50 may not be applicable to Class 2 and 3 components.
RG 1.51	In-service inspection of ASME Code Class 2 and 3 nuclear power plant components	NONE	RG 1.51 was withdrawn on 7/21/75.
RG 1.52 (Rev. 2, 3/78)	Design, testing and maintenance criteria for atmospheric cleanup system air filtration and adsorption units for light water nuclear plants	INTENT	RG 1.52 supports GDC 19, GDC 41, GDC 42, GDC 43, GDC 61 and RG 1.140. The INTENT of RG 1.52 will be met by designing the atmosphere cleanup systems to mitigate the consequences of postulated accidents by removing from the building or containment atmosphere radioactive material that may be released in an accident. However, the source term as specified in TID 14844 will be modified to reflect the ANS source term and to provide a more technically correct source term that factors in the experience learned since the TMI 2 accident. The source terms for the ANS are given in Appendix C of the PDR.
RG 1.53 (6/73)	Application of the single failure criterion to nuclear power plant protection systems	TOTAL	RG 1.53 supports 50.55a. RG 1.53 describes an acceptable method for satisfying the single-failure criterion to the design and analysis of protection systems. RG 1.53 endorses, with modifications, IEEE Std 379-1972.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.54 (6/73)	Quality Assurance requirements for protective coatings applied to water cooled nuclear plants.	NONE	RG 1.54 endorses and supplements an ANSI standard that has been withdrawn. Therefore, this RG is not applicable to the ANS. Guidance about protective coatings applied to various steel and concrete surfaces, however, is available in the ASTM subcommittee D01.43 publication: "Manual of Coating Work for Light-Water Nuclear Power Plant Primary Containment and Other Safety-Related Facilities (PCN; 03-401079-14)".
RG 1.55	Concrete placement in Category I structures	NONE	RG 1.55 was withdrawn and is superseded by RG 1.136 Rev. 2, 6/81.
RG 1.56 (Rev. 1, 7/78)	Maintenance of water purity in BWR's	NONE	RG 1.56 supports GDC 13, GDC 14, GDC 15, and GDC 31. This guide is peculiar to the operation of condensate demineralizers and reactor water cleanup systems in BWRs.
RG 1.57 (6/73)	Design limits and loading combinations for primary reactor containment system components	TOTAL	RG 1.57 supports GDC 2. RG 1.57 provides guidance for selecting combinations of loadings for design or for identifying Seismic Category I components (i.e., components that should be designed to remain functional under the effects of the safe shutdown earthquake). This guide delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of components of metal primary reactor containment systems.
RG 1.58 (Rev. 1, 9/80)	Qualification of nuclear plant inspection, examination and testing personnel	TOTAL	RG 1.58 supports Appendix B of 10 CFR 50. RG 1.58 endorses, as supplemented or modified, ANSI N45.2.6-1978, which delineates the qualifications of personnel who perform inspections, examinations, and tests during fabrication, receipt at the construction site, construction, preoperational and startup testing, and the operation phase of plants. The ANS shall comply with the methods described in RG 1.28, Revision 3 for complying with the provisions of Appendix B to 10 CFR 50 with regard to the

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.58 (Rev. 1, 9/80) continued			establishment and implementation of the requisite quality assurance program.
RG 1.59 (Rev. 02, 8/77)	Design basis floods for nuclear power plants	PARTIAL	RG 1.59 supports GDC 2, 10 CFR 100, and RG 1.29. RG 1.59 discusses the design basis floods (i.e., the most severe flood conditions that can reasonably be predicted to occur at a site as a result of severe hydro-meteorological conditions, seismic activity, or both) that plants should be designed to withstand without loss of capability for cold shutdown and maintenance thereof. The design requirements for flood protection are given in RG 1.102. RG 1.29 identifies structures, systems, and components that should be designed to withstand the effects of the safe shutdown earthquake. RG 1.59 also endorses ANSI N170-1976, with exceptions, which contains a methodology for estimating probable maximum floods along streams, probable maximum surges and seiches at estuaries and coastal areas on oceans and large lakes. Because the ANS is not along a coastline, only part of RG 1.59 is applicable.
RG 1.60 (Rev. 1, 12/73)	Design response spectra for seismic design of nuclear power plants	INTENT	RG 1.60 supports GDC 2 and 10 CFR 100. RG 1.60 describes a procedure for defining response spectra for the seismic design. The procedure given is used to define the response spectra representing the effects of the vibratory motion of the SSE, 1/2 the SSE, and the operating basis earthquake (OBE). The INTENT of RG 1.60 is applicable to the ANS. The ANS will define the response spectra for the SSE and 1/3 the SSE (see Chap. 4).
RG 1.61 (10/73)	Damping values for seismic design of nuclear plants	INTENT	RG 1.61 supports GDC 2 and 10 CFR 100. RG 1.61 delineates damping values to be used in the elastic modal dynamic seismic analysis of Seismic Category I structures, systems, and components. Because RG 1.61 specifies the operating basis earthquake = 0.5 the safe shutdown earthquake, the damping values given may not be applicable to the ANS. Nevertheless, the ANS will meet the INTENT of RG 1.61 by ensuring that structures,

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.61 (10/73) continued			systems, and components important to safety shall be designed to withstand the effects of earthquakes.
RG 1.62 (10/73)	Manual initiation of protective actions	TOTAL	RG 1.62 supports 50.55a(h). RG 1.62 endorses Section 4.17, "Manual Initiation," of IEEE Std 279-1971, which requires that protection systems include means for manual initiation of each protective action at the system level and that the single failure criterion (as set forth in Sect. 4.2) be met. The ANS shall provide the means for manually initiating each protective action at the system level and meet the single failure criterion.
RG 1.63 (Rev. 3, 2/87)	Electric penetration assemblies in containment structures for water cooled nuclear plants	TOTAL	RG 1.63 supports GDC 50 and 10 CFR 50 Appendix B. RG 1.63 endorses, with exceptions, IEEE Std 317-1976, which prescribes design, qualification, construction, installation, and testing requirements for electric penetration assemblies in containment structures.
RG 1.64 (Rev. 2, 6/76)	Quality Assurance program requirements for the design of nuclear power plants	NONE	The ANS shall comply with the methods described in RG 1.28, Revision 3 for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program.
RG 1.65 (10/73)	Materials and inspections for reactor vessel closure studs	INTENT	RG 1.65 supports GDC 1, GDC 30, GDC 31, 50.55a, and 10 CFR 50 Appendix B. RG 1.65 supplements Section III and Section XI of the Boiler and Pressure Vessel Code for requirements associated with reactor vessel closure stud bolting. The material specifications for reactor studs may not be applicable to the ANS. Nevertheless, the ANS shall meet the INTENT of RG 1.65 by designing the reactor coolant pressure boundary with sufficient margin to assure that when stressed the boundary will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized.
RG 1.66	Nondestructive examination of tubular products	NONE	RG 1.66 was withdrawn on 10/6/77.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.67	Installation of overpressure protective devices	NONE	RG 1.67 was withdrawn on 4/27/83.
RG 1.68 (Rev. 2, 8/78)	Pre-operational and initial startup test program for water cooled power reactors	INTENT	RG 1.68 supports GDC 1, 50.34 and 10 CFR 50 Appendix B. While Appendix A of 10 CFR 50 requires that all structures, systems, and components important to safety be tested, it is not required that all of them be tested to the same stringent requirements (i.e., they are tested to quality standards commensurate with the importance of the safety functions to be performed). RG 1.68 describes the general scope and depth of initial test programs. Some parts of Appendix A to RG 1.68 are not applicable to the ANS due to the different design. The ANS will comply with the INTENT of RG 1.68 by ensuring that an initial test program will demonstrate that the plant can be operated in accordance with design requirements important to safety. Therefore, features not in LWRs must be accounted for in the test program.
RG 1.68.1 (Rev. 1, 1/77)	Pre-operational and initial startup testing of feedwater and condensate systems for BWRs	INTENT	RG 1.68.1 supports GDC 1, 10 CFR 50 Appendix B, and RG 1.68. RG 1.68.1 provides preoperational and initial startup program guidance and test requirements for feedwater and condensate systems and components to verify operability in a BWR. The INTENT of RG 1.68.1 is applicable to the ANS.
RG 1.68.2 (rev. 1, 7/78)	Initial startup test program to demonstrate remote shutdown capability for water cooled nuclear power plants	TOTAL	RG 1.68.2 supports GDC 1 and Appendix B of 10 CFR 50. RG 1.68.2 describes an initial startup test program for demonstrating hot shutdown capability and the potential for cold shutdown from outside the control room. GDC 19 requires the ability to shut down from outside the control room. RG 1.68 has been interpreted by some not to include demonstration of reactor trip capability from outside the control room, but only maintenance of the hot shutdown condition. Thus, RG 1.68.2 provides the scope of testing, documentation of procedures, and the reporting of test results pertaining to demonstration of remote shutdown capability.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.68.3 (4/82)	Pre-operational testing of instrument and control air systems	TOTAL	RG 1.68.3 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.68.3, which replaces RG 1.80, describes a method for complying with the regulations with respect to preoperational testing to verify that instrument and control air systems and the loads they supply will operate properly at normal system pressures and to ensure the operability of functions important to safety in the event that system pressure is lost, reduced below normal operating level, or increased above the design pressure of the air system components to the upstream safety valve accumulation pressure. RG 1.68.3 endorses ANSI/ISA S7.3-1975 with respect to oil, water, and particulate matter contained in the product air.
RG 1.69 (12/73)	Concrete radiation shields for nuclear plants	TOTAL	RG 1.69 supports GDC 1, 10 CFR 50 Appendix B, and 10 CFR 20. RG 1.69 endorses, with exceptions, ANSI N101.6-1972, which discusses aggregates, design of concrete mixtures and forms, placement of concrete, design and installation of penetrations, embedments, metal liners, and penetration plugs and also outlines testing and quality assurance provisions needed to verify that the desired quality of design and construction has been achieved. The standard does not include detailed treatments of structural design or determination of shield thickness, but does reference other standards covering structural design. If the shielding structure provides structural function necessitating heavy reinforcing, the limitation of 3 in. maximum slump does not apply (ANSI 101.6, Sect. 5.4).
RG 1.70 (Rev. 3, 11/78)	Standard format and content of SAR's	PARTIAL	RG 1.70 is REQUIRED by DOE 5480.6. The SAR will be modified to reflect the structures, systems, and components that are in the ANS. The standard format will be maintained as much as possible.
RG 1.71 (12/73)	Welder qualifications for areas of limited accessibility	TOTAL	RG 1.71 supports GDC 1, 50.55a, and 10 CFR 50 Appendix B. RG 1.71 describes a method to control welding for nuclear components. Because RG 1.71 is specific to welding of low-alloy and high-alloy steels. Other

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
			material that must be considered in addition to those listed in this RG are hafnium, aluminum, Zircaloy, Inconnel, and 17-4 PH stainless steel.
RG 1.72 (Rev. 2, 11/78)	Spray pond plastic piping	TOTAL	RG 1.72 supports GDC 1, 50.55a, and 10 CFR 50 Appendix B. RG 1.72 describes a method for implementing requirements with regard to the design, fabrication, and testing of fiberglass-reinforced thermo-setting resin (RTR) piping for spray pond applications. RG 1.72 endorses, with exceptions, Code Case N-155-1 and ASTM D-1599-74. (Note: RG 1.48 provides general guidance for loading combinations relative to design limits for Class 3 piping.) If the ANS uses spray pond plastic piping, it will comply with RG 1.72.
RG 1.73 (1/74)	Qualification tests of electric valve operators installed inside the containment of nuclear power plants	TOTAL	RG 1.73 supports 10 CFR 50 Appendix B. RG 1.73 endorses, with exceptions, IEEE Std 382-1972, which delineates specific procedures for the qualification testing of Class I electric valve operators to demonstrate design adequacy for service with the containment. The procedure provides for testing under conditions simulating (1) those that would be imposed during and after a design basis loss-of-coolant accident and (2) those occurring during normal operating conditions. Effects to be considered include radiation, temperature, pressure, humidity, and chemical sprays.
RG 1.74 (2/74)	Quality Assurance terms and definitions	NONE	The ANS shall comply with the methods described in RG 1.28, Revision 3, for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program.
RG 1.75 (Rev. 2, 9/78)	Physical independence of electric systems	TOTAL	RG 1.75 supports 50.55a (and thus IEEE Std 279-1971), GDC 3, GDC 17 and GDC 21, and endorses IEEE Std 384-1974. RG 1.75 establishes the detailed requirements for electrical system independence, including the on-site ac

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
			power supply system. The intent is to ensure that there is adequate independence of the electrical power supplies required for safe shutdown, including consideration of accident conditions, to assure that no single failure results in the loss of a required protective function. RG 1.75 assumes that either on-site or off-site ac power is required to attain a safe shutdown condition. (Note: The determination of which circuits and equipment are redundant and the degree of redundancy required is outside the scope of RG 1.75 and IEEE Std 384-1974.) Safety related ac and dc power systems will comply with RG 1.75.
RG 1.76 (4/74)	Design Basis Tornado for Nuclear Power Plants	TOTAL	R293G 1.76 supports GDC 2 and 10 CFR 100, Section 100.10(c). Even though the probabilities of tornadoes in the vicinity of the ANS are low and their expected intensity is smaller in comparison to those recorded in "tornado alley," the wind speeds are still based on a 1.76×10^{-7} probability per year tornado. Missile Spectrum I of SRP 3.5.1.4 will be used to analyze the effects of tornado-generated missiles.
RG 1.77 (5/74)	Assumptions used for evaluating a control rod ejection accident for PWR's	INTENT	RG 1.77 supports GDC 28 and 50.34. RG 1.77 provides methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium-oxide-fueled PWRs. Because of different fuel, different source terms, different driving mechanism to force a rod ejection, etc., only the INTENT of RG 1.77 is applicable to the ANS. Therefore, the ANS shall determine if a rod ejection accident is feasible, and if so, evaluate the consequences of the accident. The source term for the ANS is given in Appendix C of the PDR.
RG 1.78 (6/74)	Assumptions for evaluating the habitability of a nuclear plant control room during a postulated hazardous chemical release	TOTAL	RG 1.78 supports GDC 4 and GDC 19. The control room shall be protected from hazardous chemicals that may be discharged as a result of equipment failures, operator errors, or events and conditions outside the control of plant personnel.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.79 (Rev. 1, 9/75)	Pre-operational testing of ECCS for PWR's	INTENT	RG 1.79 supports GDC 1, 10 CFR 50 Appendix B, and RG 1.68. Because the ECCS system is normally in operation, uses dc safety-related power, and relies on passive safety features, only the INTENT of RG 1.79 is applicable. This guide does provide test conditions that can impact the design.
RG 1.80	Pre-operational testing of instrument air systems	NONE	RG 1.80 was withdrawn on 5/4/82 and reissued as RG 1.68.3, a renumbered revision to this guide with an expanded scope that addresses control air systems.
RG 1.81 (Rev. 1, 1/75)	Shared emergency and shutdown electric systems for multi-unit nuclear power plants	NONE	RG 1.81 supports GDC 5. Because the site for the ANS is not a multi-unit site and because the ANS does not share standby electric power systems with another facility (i.e., a nonreactor facility), RG 1.81 is not applicable.
RG 1.82 (Rev.1, 11/85)	Sumps for emergency cooling and containment spray systems	NONE	RG 1.82 supports GDC 1, GDC 35, GDC 36, GDC 37, GDC 38, GDC 39, and GDC 40. RG 1.82 provides guidance for providing long-term cooling by recirculating water from containment sumps. Because the ANS does not recirculate water from containment sumps, this RG is not applicable.
RG 1.83 (Rev. 1, 7/75)	Inservice inspection of PWR steam generator tubes	INTENT	RG 1.83 supports GDC 14, GDC 15, GDC 31, and GDC 32. RG 1.83 provides a method for reducing the probability and consequences of steam generator tube failures through periodic inservice inspection for early detection of defects and deterioration. Even though the ANS does not have steam generators, it does have heat exchangers that provide the same function — a large heat transfer area. The INTENT of RG 1.83 will be followed by ensuring that heat exchangers and their tubes can be inspected and, if necessary, plugging the tubes. Inadequate control of coolant chemistry must be enlarged to consider effects of D_2O.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.84 (Rev. 24, 6/86)	Design and fabrication code case acceptability ASME Section III, Division I	TOTAL	RG 1.84 supports GDC 1, GDC 30, 50.55a, and 10 CFR 50 Appendix B. RG 1.84 lists those Section III, ASME Code cases oriented to design and fabrication of components. Additional requirements will be required to cover design features unique to the ANS.
RG 1.85 (Rev. 24, 6/86)	Materials code case acceptability-ASME Division 1	TOTAL	RG 1.85 supports GDC 1, GDC 30, 50.55a, and 10 CFR 50 Appendix B. RG 1.85 lists those Section III, ASME Code cases oriented to materials and testing for components. Additional requirements will be required to cover design features unique to the ANS.
RG 1.86 (6/74)	Termination of operating licenses for nuclear reactors	N/A-L	RG 1.86 supports 50.51 and 50.82. Because RG 1.86 deals with the licensing agency (i.e., the Nuclear Regulatory Commission), it is not applicable to the ANS.
RG 1.87 (Rev. 1, 6/75)	Guidance for construction of Class 1 components in elevated temperature reactors (supplement to ASME Code Cases 1592, 1593,1594, 1595, 1596)	NONE	RG 1.87 supports GDC 1 and 50.55a. This guide provides guidelines for construction of ASME Class 1 components operating at elevated temperatures. This guide applies to high temperature gas cooled reactors, liquid metal fast breeder reactors, and gas cooled fast breeder reactors. Because the ANS is a low pressure, low temperature facility, none of the Class 1 components will see service at elevated temperatures. Therefore, the RG does not apply to the ANS.
RG 1.88 (Rev. 2, 10/76)	Collection, storage and maintenance of nuclear power plant Quality Assurance records	NONE	RG 1.88 supports 10 CFR 50 Appendix B. The methods described in RG 1.28, Revision 3, for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program are considered to be generally equivalent, from a programmatic standpoint, to the methods described in Revision 2 to RG 1.28 and RG 1.58, RG 1.64, RG 1.74, RG 1.88, RG 1.123, RG 1.144, and RG 1.146.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.89 (Rev. 1, 6/84)	Qualification of class IE equipment for nuclear plants	INTENT	RG 1.89 supports GDC 1, GDC 2, GDC 4, GDC 23, 50.49, and 10 CFR 50 Appendix B. RG 1.89 describes a method acceptable for complying with 50.49 with regard to qualification of electric equipment important to safety to ensure that the equipment can perform its safety function during and after a design basis accident. RG 1.89 endorses, with exceptions, IEEE Std 323-1974 that describes basic procedures for qualifying Class IE equipment and interfaces that are to be used, including components or equipment of any interface whose failure could adversely affect any Class IE equipment. [Note, IEEE Std 323 was replaced with IEEE Std 344. The differences between the standards and NRC exceptions as they apply to IEEE Std 344 will be evaluated in the PSAR.] The INTENT of RG 1.89 shall be met to ensure that equipment important to safety can perform its safety function during and after a design basis accident. The source term and the methods for calculating mass and energy release rates shall be modified to represent conditions applicable to the ANS. The source terms for the ANS are given in Appendix C of the PDR.
RG 1.90 (Rev. 1, 8/77)	In-service inspection of prestressed concrete containments structures with grouted tendons	NONE	RG 1.90 supports GDC 53. This guide describes an acceptable basis for developing an appropriate surveillance and in-service inspection program for prestressed concrete containment structures with grouted tendons. Because the ANS does not have a prestressed concrete containment structure, this RG is not applicable to the ANS.
RG 1.91 (Rev. 1, 2/78)	Evaluation of explosions postulated to occur transportation routes near nuclear plants	TOTAL	RG 1.91 supports GDC 2 and GDC 4. RG 1.91 describes methods for determining whether the risk of damage because of an explosion on a nearby transportation route is sufficiently high to warrant a detailed investigation. Acceptable methods for evaluating structural adequacy when an investigation is warranted are also described. This guide is limited to solid explosives and hydrocarbons liquified under pressure and is not applicable to cryogenically liquified hydrocarbons. It considers the effects of airblasts on highway, rail, and water routes, but excludes pipelines and fixed facilities.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.92 (Rev. 1, 2/76)	Combining modal responses and spatial components in seismic response analysis	INTENT	RG 1.92 supports GDC 2 and 10 CFR 100. RG 1.92 describes methods acceptable for (1) combining the values of the response of individual modes in a response spectrum modal dynamic analysis to find the representative maximum value of a particular response of interest for the design of a given element of a structure, system or component, and (2) combining the maximum values (in the case of time-history dynamic analysis) or the representative maximum values (in the case of spectrum dynamic analysis) of the response of a given element of a structure, system, or component, when such values are calculated independently for each of the three orthogonal spatial components (two horizontal and one vertical) of an earthquake. The combined value will be the representative maximum value of the combined response of that element of the structure, system, or component to simultaneous action of the three spatial components.
RG 1.93 (12/74)	Availability of electric power source	TOTAL	RG 1.93 supports 50.36(c)(2) and GDC 17. RG 1.93 establishes limiting conditions of operation (LCO) for the development of plant technical specifications under conditions in which one or more of the on-site or off-site ac power supplies or dc power supplies are not available. The intent is to ensure that adequate electrical power is available for safe shutdown, including consideration of accident conditions. RG 1.93 assumes that either on-site or off-site ac power is required to attain a safe shutdown condition. Safety related ac and dc power systems will comply with RG 1.93.
RG 1.94 (Rev. 2, 9/79)	Quality Assurance requirements for installation, inspection, and testing of structural concrete and structural steel during construction phase of nuclear plants	TOTAL	RG 1.94 supports 10 CFR 50 Appendix B. RG 1.94 describes a method for complying with the regulations with regard to quality assurance requirements for installation, inspection, and testing of structural concrete and structural steel during the construction phase. RG 1.94 endorses, with exceptions, ANSI/ASME N45.2.5-1978.
RG 1.95 (Rev. 1, 1/77)	Protection of nuclear power plant control room against an accidental chlorine release	TOTAL	RG 1.95 supports GDC 4 and RG 1.78. RG 1.95 describes design features and procedures for the protection of control room operators against an accidental chlorine

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
			release. RG 1.78 identifies chlorine as a hazardous chemical which, if present in the control room atmosphere in sufficient quantity, could result in the control room becoming uninhabitable. It is the purpose of RG 1.95 to describe specific design features and procedures to mitigate hazards to control room operators from an accidental chlorine release. Although this guide was developed to provide protection from an on-site chlorine release, the protection provisions are sufficient for an off-site chlorine release.
RG 1.96 (Rev. 1, 6/76)	Design of main steam isolation valve leakage control systems for BWR nuclear plants	NONE	RG 1.96 supports GDC 54 and 10 CFR 100. Because this guide is peculiar to BWRs that have installed quick closing isolation valves in the steam supply lines directly from the reactor vessel to the turbine generator, it is not applicable to the ANS, which has no steam lines.
RG 1.97 (Rev. 3, 5/83)	Instrumentation for light water cooled nuclear plants to assess plant conditions during and following an accident	PARTIAL	RG 1.97 supports GDC 13, GDC 19, and GDC 64. RG 1.97 describes a method to provide instrumentation to monitor plant variables and systems during and following an accident. RG 1.97 endorses, with exceptions, ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors." RG 1.97 is not fully applicable to the ANS because water level instrumentation is not required, and the source term will be modified to reflect the ANS source term. The source terms for the ANS are given in Appendix C of the PDR. Other parameters required to be monitored are ***(TBD)***.[B-69]
RG 1.98 (3/76)	Assumptions used for evaluating the potential radiological consequences of a radioactive off-gas system in a BWR	INTENT	RG 1.98 supports GDC 61 and 50.34. The radioactive off-gas systems discussed in this guide and the assumed release break downstream of the steam jet air ejectors are specific to BWRs. Therefore, only the intent of this RG is applicable to the ANS.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.99 (Rev. 1, 4/77)	Effects of residual elements on predicted radiation damage to reactor vessel materials	INTENT	RG supports GDC 31, 10 CFR 50 Appendix G, and 10 CFR 50 Appendix H. This guide describes procedures for predicting the effects of the residual elements, such as copper and phosphorus, on neutron radiation damage to the low alloy steel pressure vessel. Because the ANS does not have a low alloy steel pressure vessel, this RG is not applicable. However, radiation damage to the aluminum pressure tube must be evaluated to determine its replacement interval.
RG 1.100 (Rev. 1, 8/77)	Seismic qualification of electric equipment for nuclear power plants	TOTAL	RG 1.100 supports 10 CFR 50 Appendix B. RG 1.100 describes a method for complying with the regulations with respect to verifying the adequacy of the seismic design of electric equipment. RG 1.100 endorses, with exceptions, IEEE Std 344, which includes principles, procedures, and methods of seismic qualification which, when satisfied, confirm the adequacy of the equipment design for the performance of Class IE functions during and after the time the equipment is subjected to forces resulting from an SSE preceded by a number of OBEs.
RG 1.101 (Rev. 2, 10/81)	Emergency planning for nuclear power plants	TOTAL	RG 1.101 supports 50.4, 50.47, and the FEMA. RG 1.101 endorses Revision 1 to NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," which provides specific acceptance criteria for complying with the standards set forth in 50.47 of 10 CFR 50. These criteria provide a basis for NRC, state and local governments to develop acceptable radiological emergency plans and improve emergency preparedness.
RG 1.102 (Rev. 1, 9/76)	Flood protection for nuclear power plants	TOTAL	RG 1.102 supports GDC 2, 10 CFR 100, RG 1.29, and RG 1.59. RG 1.102 describes types of flood protection for safety related structures, systems, and components identified in RG 1.29. In addition, RG 1.102 describes methods of protecting nuclear power plants from the effects of probable maximum precipitation (PMP) falling directly on the site.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.103	Post-tensioned prestressing systems for concrete reactor vessels and containments	NONE	RG 1.103 was withdrawn and is superseded by RG 1.136 Rev 2, 6/81.
RG 1.104	Overhead crane handling systems for nuclear power plants	NONE	RG 1.104 was withdrawn on 8/22/79.
RG 1.105 (Rev. 2, 2/86)	Instrument setpoints	TOTAL	RG 1.105 supports GDC 13, GDC 20, and 50.36(c). RG 1.105 describes a method for ensuring that instrument set points are initially within and remain within the technical specification limits. RG 1.105 endorses, with exceptions, ISA-S67.04-1982, which contains the minimum requirements to be used for establishing and maintaining set points of individual instrument channels in safety-related systems.
RG 1.106 (Rev. 1, 3/77)	Thermal overload protection for electric motors on motor operated valves	TOTAL	RG 1.106 supports GDC 1, GDC 4, GDC 13, and 10 CFR 50 Appendix B. RG 1.106 describes a method for complying with the regulations with regard to the application of thermal overload protection devices that are integral with the motor starter for electric motors on motor-operated valves. This method would ensure that the thermal overload protection devices will not needlessly prevent the motor from performing its safety-related function. RG 1.106 endorses sections of IEEE Std 279-1971.
RG 1.107 (Rev. 1, 2/77)	Quality of cement grouting for prestressing tendons in containment structures	NONE	RG 1.107 supports GDC 1 and describes quality standards for the use of portland cement grout as the corrosion inhibitor for prestressing tendons in prestressed concrete containment structures. Because the ANS does not have a prestressed containment, this RG is not applicable.
RG 1.108 (Rev. 1, 8/77)	Periodic testing of diesel generator units used as on-site electric power system at nuclear plants	TOTAL	RG 1.108 supports GDC 17, GDC 18, and 10 CFR 50 Appendix B. RG 1.108 provides a method with regard to periodic testing of diesel electric power units to ensure that the diesel electric power systems will meet their availability requirements. The purpose of on-site electric power systems are (1) to provide power promptly to engineered safety features if a loss of offsite power and an accident occur during the same time period and (2) to

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.108 (Rev. 1, 8/77) continued			provide power to equipment needed to maintain the plant in a safe condition if an extended loss of off-site power occurs. If diesel generator units are selected for ANS, they will comply with RG 1.108.
RG 1.109 (Rev. 1, 10/77)	Calculation of annual doses to man from routine releases of reactor effluents for purpose of compliance with 10 CFR 50 Appendix I	TOTAL	RG 1.109 supports 10 CFR 50 Appendix I. RG 1.109 describes basic features of models used to calculate preoperational estimates of effluent releases, dispersion of the effluent in the atmosphere and different water bodies, and estimation of the associated radiation doses to man and suggests parameters for the estimation of radiation doses to man from effluent releases. The methods used are general approaches in lieu of specific parameters for individual sites. Portions of RG 1.109 supersede RG 1.42, which was withdrawn.
RG 1.110 (3/76)	Cost-benefit analysis for radwaste systems for light water cooled nuclear plants	TOTAL	RG 1.110 supports 10 CFR 50 Appendix I. RG 1.110 describes a method for performing a cost-benefit analysis for liquid and gaseous radwaste system components.
RG 1.111 (Rev. 1, 7/77)	Methods for estimating atmospheric transport and dispersion of gaseous effluents from LWR's	TOTAL	RG 1.111 supports 50.34a, 10 CFR 50 Appendix I, 20.106, and 20.1(c). RG 1.111 is also REFERENCED in RG 1.23. RG 1.111 describes basic features of calculational models and assumptions for the estimation of atmospheric transport and dispersion of gaseous effluents in routine releases. Sections of RG 1.111 supersede RG 1.42, which was withdrawn.
RG 1.112 (5/77)	Calculation of releases of radioactive materials in gaseous and liquid effluents from lwr's	INTENT	RG 1.112 supports 50.34a, 10 CFR 50 Appendix I, 20.106, and 20.1(c). RG 1.112 endorses NUREG-0016 and NUREG-0017 which provide methods for calculating annual average expected releases of radioactive material in liquid and gaseous effluents. Because these NUREGs are specific to BWRs (NUREG-0016) and PWRs (NUREG-0017), the models and calculations will be modified to reflect the radioactive source term calculations for the ANS. The source terms for the ANS are given in Appendix C of the PDR. Therefore, the INTENT of RG 1.112 will be met.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.113 (Rev. 1, 4/77)	Estimating aquatic dispersion of effluents from accidental and routine reactor releases for purposes of implementing Appendix I	NONE	RG 1.113 supports 50.34a, 10 CFR 50 Appendix I, 20.106, and 20.1(c). This guide is concerned with reactors that receive and/or discharge cooling water to rivers or cooling ponds. The ANS uses secondary and auxiliary tower water systems that are devoted entirely to reactor heat removal.
RG 1.114 (Rev. 1, 11/76)	Guidance on being operator at the controls of a nuclear power plant	TOTAL	RG 1.114 supports 50.54 and 10 CFR 55. RG 1.112 requires the presence of an operator at the controls and a senior operator in the control room. The control room must be designed so that all controls, instrumentation displays, and alarms required for the safe operation, shutdown, and cooldown are readily available to the operator. The senior operator must be able to spend most of the time in that portion of the control room where there is direct and prompt access to information on current unit conditions.
RG 1.115 (Rev. 1, 7/77)	Protection against low trajectory turbine missiles	NONE	RG 1.115 supports GDC 4 and 10 CFR 100. This guide describes methods for protecting safety-related structures, systems, and components against low-trajectory missiles resulting from turbine failure by appropriate orientation and placement of the turbine-generator set. Because the ANS does not have a turbine-generator set, the RG is not applicable.
RG 1.116 (5/77)	Quality Assurance requirements for installation, inspection, and testing of mechanical equipment and systems	TOTAL	RG 1.116 supports 10 CFR 50 Appendix B and endorses, with exceptions, ANSI N25.2.8-1975. RG 1.116 describes a method to meet quality assurance requirements for the installation, inspection, and testing of mechanical equipment and systems and endorses ANSI N45.2.8-1975. The relationship of the quality assurance requirements included in Sections III and XI of the ASME Boiler and Pressure Vessel Code with those included in ANSI N45.2.8-1975 must be considered.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.117 (Rev. 1, 4/78)	Tornado design classification	TOTAL	RG 1.117 supports GDC 2 and 10 CFR 100. RG 1.117 describes a method for identifying those structures, systems, and components that should be protected from the effects of the design basis tornado (as determined from RG 1.76).
RG 1.118 (Rev. 2, 6/78)	Periodic testing of electric power and protection systems	TOTAL	RG 1.118 supports GDC 18, GDC 21, 50.55a, 10 CFR 50 Appendix B, and RG 1.105; and endorses, with exceptions, IEEE Std 338-1977. RG 1.118 provides guidance of the periodic testing of the protection system and electric power systems for systems important to safety (as defined in RG 1.105). It also provides supplementary guidance to that included in RG 1.32 regarding the periodic testing of electric power systems. This revision of RG 1.118 also accounts for changes made between IEEE Std. 338-1975 and IEEE Std. 338-1977. IEEE Std. 338-1977 provides acceptable methods for the periodic testing of electric power and protection systems (with several exceptions, as listed in RG 1.118).
RG 1.119	Surveillance program for new fuel assembly designs	NONE	RG 1.119 was withdrawn on 6/30/77.
RG 1.120 (Rev. 1, 11/77)	Fire protection guidelines for nuclear plants	TOTAL	RG 1.120 supports GDC 3 and supplements RG 1.75. RG 1.120 addresses fire protection programs for safety-related systems and equipment and for other plant areas containing fire hazards that could adversely affect safety-related systems. It does not give guidance for protecting the life safety of the site personnel or for protection against economic or property loss. This guide endorses numerous NFPA standards.
RG 1.121 (8/76)	Bases for plugging degraded PWR steam generator tubes	INTENT	RG 1.121 supports GDC 14, GDC 15, GDC 31, and supplements RG 1.83. This guide describes a method for establishing the limiting safe conditions of tube degradation for steam generator tubing. RG 1.83 defines defective tubes (i.e., tubes with wall thickness less than the minimum acceptable thickness) as being unacceptable for continued service and recommends that these and leaking tubes be plugged. Partially degraded tubes with a wall

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.121 (8/76) continued			thickness greater than the minimum acceptable tube wall thickness are acceptable for continued service. The ANS will comply with the INTENT of RG 1.121 by monitoring heat exchanger tubes for degradation (i.e., defects, cracks, wall thinning, and wastage).
RG 1.122 (Rev. 1, 2/78)	Development of floor design response spectra for seismic design of floor-supported equipment or components	TOTAL	RG 1.122 supports GDC 2 and 10 CFR 100. RG 1.122 describes methods acceptable for developing two horizontal and one vertical floor design response spectra at various floors or other equipment support locations of interest from the time-history motions resulting from the dynamic analysis of the supporting structure. These floor design response spectra are needed for the dynamic analysis of the systems or equipment supported at various locations of the supporting structure.
RG 1.123 (Rev. 1, 7/77)	Quality Assurance requirements for control of procurement of items and services for nuclear plants	NONE	The ANS shall comply with the methods described in RG 1.28, Revision 3 for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program.
RG 1.124 (Rev. 1, 1/78)	Service limits and loading combinations for Class 1 Linear-type component supports	TOTAL	RG 1.124 supports GDC 2. RG 1.124 delineates acceptable levels of service limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of Class 1 linear type component supports.
RG 1.125 (Rev. 1, 10/78)	Physical models for design and operation of hydraulic structures and systems for nuclear plants	TOTAL	RG 1.125 supports 50.34. RG 1.125 describes the desired coordination with the NRC and the detail and documentation of data and studies that should be included in the PSAR to support the use of physical hydraulic model testing for predicting performance of safety-related hydraulic structures and systems. This guide is applicable only to physical models used to predict the action or interaction of surface waters with features located outside of containment. This guide is not applicable to internal plant systems or structures.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.126 (Rev. 1, 3/78)	An acceptable model and related statistical methods for the analysis of fuel densification	NONE	RG 1.126 supports 10 CFR 50 Appendix K. This guide provides an analytical model and related assumptions and procedures for predicting the effects of fuel densification of pellet-type reactor fuel. This guide is applicable only to UO_2, UO_2-PuO_2, and UO_2-Gd_2O_3 fuel pellets. Because the ANS fuel is in plate rather than pellet form, and uses U_3Si_2 in an aluminum matrix, clad with type 6061 aluminum, this RG is not applicable.
RG 1.127 (Rev. 1, 3/78)	Inspection of water control structures associated with nuclear power plants	NONE	RG 1.127 supports GDC 45, 50.34, and 50.36. This guide describes a basis for developing an appropriate in-service inspection and surveillance program for dams, reservoirs, canals, and other water-control structures associated specifically with emergency cooling water systems or flood protection of nuclear power plants. Because the ANS does not utilize dams or reservoirs for any emergency cooling water sources or flood protection design, the RG is not applicable.
RG 1.128 (Rev. 1, 10/78)	Installation, design and testing of large lead storage batteries for nuclear plants	TOTAL	RG 1.128 supports GDC 1, GDC 17, and 10 CFR 50 Appendix B. RG 1.128 describes a method for performing the installation design and installation of large lead storage batteries. RG 1.128 endorses, with exceptions, IEEE Std. 484-1975 that includes design practices and procedures for storage, location, mounting, ventilation, instrumentation, preassembly, and assembly of large lead sealed cell storage batteries. Safety practices are also included.
RG 1.129 (Rev. 1, 2/78)	Maintenance, testing and replacement of large lead storage batteries for nuclear plants	TOTAL	RG 1.129 supports GDC 18, 10 CFR 50 Appendix B, and 50.34(b). RG 1.129 endorses, with exceptions, IEEE Std. 450-1975 that provides a basis for maintenance, testing, and replacement of large lead storage batteries. The standard includes provisions for maintenance, test schedules, testing procedures, and replacement criteria. The standard does not include surveillance and testing of the dc system, even though the battery is part of that system. RG 1.129 also requires a battery service test in addition to the battery discharge test.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.130 (Rev. 1, 10/78)	Service limits and loading combinations for class I plate and shell type component supports	TOTAL	RG 1.130 supports GDC 2. RG 1.130 delineates acceptable levels of service limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of Class 1 plate-and-shell-type component supports.
RG 1.131 (8/77)	Qualification tests of electric cables, field splices and connections for light water cooled nuclear power plants	PARTIAL	RG 1.131 supports 10 CFR 50 Appendix B. RG 1.131 endorses, with exceptions, IEEE Std. 383-1974, which is concerned with cables both inside and outside the containment and establishes procedures for simulating operating conditions so that type testing will be adequate for the intended service conditions. It covers normal operating and design basis event conditions, except that the fire test is a reference configuration and flame rather than the design basis configuration and flame. RG 1.131 requires that the source term is to be taken from RG 1.89 rather than from IEEE Std. 383-1974. The source term and the methods for calculating mass and energy release rates shall be modified to represent conditions applicable to the ANS. The source terms for the ANS are given in Appendix C of the PDR.
RG 1.132 (Rev. 1, 3/79)	Site investigations for foundations of nuclear plants	TOTAL	RG 1.132 supports 10 CFR 100. RG 1.132 describes programs of site investigations that would normally meet the needs for evaluating the safety of the site from the standpoint of the performance of foundations and earthworks under most anticipated loading conditions, including earthquakes. It also describes site investigations required to evaluate geotechnical parameters needed for engineering analysis and design. RG 1.132 does not discuss detailed geologic fault investigations, nor does it deal with hydrologic investigations, except for groundwater measurements.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.133 (rev. 1, 5/81)	Loose-part detection program for the primary system of light water cooled reactors	TOTAL	RG 1.133 supports GDC 1, GDC 13, 50.36, and 20.1(c). RG 1.133 describes a method for detecting a potentially safety-related loose part during normal operation. This guide also outlines a program that can help meet the Part 20 criterion that exposures to radiation during routine operation will be "as low as reasonably achievable" (ALARA). The sensitivity of the system will be at least as sensitive as that required in Regulatory Position 1(b) *[HOLD]*[B-7] *[Note: The ANS may require more sensitive instrumentation than required by RG 1.133.]* Instrumentation shall be provided to detect the presence of loose parts in the parts in the reactor assembly and associated cooling systems, as well as in the reflector cooling system.
RG 1.134 (Rev. 2, 4/87)	Medical evaluation of nuclear plant personnel requiring operator licenses	TOTAL	RG 1.134 supports 10 CFR 55. RG 1.134 endorses ANSI/ANS-3.4-1983, which prescribes minimum requirements necessary to determine that the medical condition and general health of operators will not cause operational errors.
RG 1.135 (9/77)	Normal water level and discharge at nuclear plants	TOTAL	RG 1.135 supports GDC 2 and 10 CFR 100. RG 1.135 sets forth methods for determining normal ground and surface water levels and surface water discharges for use in analyses of the effect of design basis natural and accidental events on structures and systems. (Design basis events are those natural events such as probable maximum flood, safe shutdown earthquake, and other natural events of low probability of occurrence, and accidental events, such as explosions, plane crashes, and fires.) RG 1.135 also suggests data sources and acceptable methods for determining such water levels and discharges.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.136 (Rev. 2, 6/81)	Material for concrete Containments	PARTIAL	RG 1.136 supports GDC 1 and 10 CFR 50 Appendix B. This guide provides supplemental requirements to the ASME Code, Section III, Division 2 "Code for Concrete Reactor Vessels and Containments," also known as ACI 359-80. RG 1.136 provides acceptable bases with regard to the materials, construction, and testing of concrete containments. Depending on the type containment, not all of this guide is applicable to the ANS (e.g., the reference to RG 1.107 and grouted tendon systems).
RG 1.137 (Rev. 1, 10/79)	Fuel oil systems for standby diesel generators	TOTAL	RG 1.137 supports GDC 17 and 10 CFR 50 Appendix B. RG 1.137 describes a method to assure the adequacy of fuel-oil quality. RG 1.137 endorses ANSI N195-1976 for complying with GDC 17. Because ANSI N195-1976 does not address quality assurance, RG 1.137 recommends using RG 1.28 (which endorses ANSI N45.2-1977) for the design, construction, and maintenance of the fuel-oil system.
RG 1.138 (4/78)	Laboratory investigations of soils for engineering analysis and design of nuclear plants	TOTAL	RG 1.138 supports 10 CFR 100 for conducting site investigations to permit an evaluation of the site and provide information needed for seismic response analyses and engineering design. Safety-related site characteristics are identified in RG 1.70 and RG 4.7. RG 1.132 discusses programs of field studies, exploratory borings, and sampling needed to provide geotechnical data for site evaluation and engineering analysis and design. RG 1.138 describes laboratory investigations and testing practices for determining soil and rock properties and characteristics needed for engineering analysis and design for foundations and earthworks.
RG 1.139 (5/78)	Guidance for residual heat removal	TOTAL	RG 1.139 supports GDC 19 and GDC 34. RG 1.139 describes a method for complying with regulations with regard to the removal of decay heat and sensible heat after a reactor shutdown. The design should allow the reactor to be taken from normal operating conditions to cold shutdown using only safety-grade systems that satisfy GDC 1, GDC 2, GDC 3, GDC 4 and GDC 5. System

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.139 (5/78) continued			testing should meet the requirements of IEEE Std. 338 and the recommendations of RG 1.22 and RG 1.68.
RG 1.140 (Rev. 1, 10/79)	Design, testing and maintenance criteria for normal ventilation exhaust system air filtration and adsorption units for light water cooled nuclear power plants	TOTAL	RG 1.140 supports GDC 60, GDC 61, 50.34(a), 50.36(a), and 10 CFR 50 Appendix I. RG 1.140 only applies to atmosphere cleanup systems designed to collect airborne radioactive materials during normal plant operation, including anticipated operational occurrences, and addresses the atmosphere cleanup systems, including the various components and ductwork in the normal operating environment. RG 1.140 does not apply to the post-accident engineered safety-feature atmosphere cleanup systems that are designed to mitigate the consequences of postulated accidents. For this, see RG 1.52.
RG 1.141 (4/78)	Containment isolation provisions for fluid system	TOTAL	RG 1.141 supports GDC 54, GDC 55, GDC 56, and GDC 57. RG 1.141 endorses ANSI N271-1976, which provides minimum design, testing, and maintenance requirements for the isolation of fluid systems that penetrate the primary containment. The INTENT of RG 1.141 is also applicable to the biological shield and nonfluid systems penetrating either containment or the biological shield. Exception to RG 1.141 is the source term to be used is that from RG 1.89. For the ANS, the source term and the methods for calculating mass and energy release rates shall be modified to represent conditions applicable to the ANS. The source terms for the ANS are given in Appendix C of the PDR.
RG 1.142 (Rev. 1, 10/81)	Safety related concrete structures for nuclear power plants (other than reactor vessels and containments)	TOTAL	RG 1.142 supports GDC 1, GDC 2, and GDC 4. RG 1.142 applies to structures required to withstand pressures and to maintain a certain degree of leaktightness during operating and accident conditions (e.g., cells). For those concrete structures used to provide radiation shielding, RG 1.142 endorses ANSI/ANS 6.4-1977. This RG also endorses numerous other standards.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.143 (Rev. 1, 10/79)	Design guidance for radioactive waste management systems, structures and components installed in light water cooled nuclear power plants	TOTAL	RG 1.143 supports GDC 1, GDC 2, GDC 60, and 50.43. RG 1.143 furnishes design guidance relating to seismic and quality group classification and quality assurance provisions for radioactive waste management systems, structures, and components. Further, it describes provisions for controlling releases of liquids containing radioactive materials, e.g., spills or tank overflows, from all plant systems outside reactor containment. RG 1.143 endorses, with exceptions, ANSI N199-1976/ANS-55.2.
RG 1.144 (Rev. 1, 9/80)	Auditing of Quality Assurance programs for nuclear plants	NONE	The ANS shall comply with the methods described in RG 1.28, Revision 3, for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program.
RG 1.145 (Rev. 1, 11/82)	Atmospheric dispersion models for potential accident consequence assessment at nuclear plants	TOTAL	RG 1.145 supports 50.35, 10 CFR 100, RG 1.3 or RG 1.4. RG 1.145 is also REFERENCED in RG 1.23. RG 1.145 provides a method for determining site-specific relative concentrations (χ/Q). RG 1.145 provides methods for (1) calculating χ/Q values, (2) determining χ/Q values on a directional basis, (3) determining χ/Q values on an overall site basis, and (4) choosing χ/Q values to be used in the evaluation of the types of events described in RG 1.3 and RG 1.4.
RG 1.146 (8/80)	Qualification of Quality Assurance program audit personnel for nuclear power plants	NONE	The ANS shall comply with the methods described in RG 1.28, Revision 3, for complying with the provisions of Appendix B to 10 CFR 50 with regard to the establishment and implementation of the requisite quality assurance program.
RG 1.147 (Rev. 5, 8/86)	In-service inspection code case acceptability ASME section XI division I	TOTAL	RG 1.147 supports GDC 1 and 50.55a. RG 1.147 lists those Section XI ASME Code cases that are acceptable for implementation in the in-service inspection of components and their supports.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.148 (3/81)	Functional specification for active valve in systems important to safety in nuclear plants	PARTIAL	RG 1.148 supports GDC 1 and 10 CFR 50 Appendix B. RG 1.148 is applicable to active valve assemblies in systems important to safety. RG 1.148 endorses ANSI N278.1-1975, which establishes a consistent approach to specifying the minimum information needed to identify the requirements for function and operability of valve assemblies in systems important to safety. The quality group classifications will be modified to reflect the classifications given in the main body of the PDR. [These quality group classifications relate to those given in RG 1.26 (Groups A, B and C) and those given in the ASME Code Class (Groups 1, 2, and 3)].
RG 1.149 (Rev. 1, 4/87)	Nuclear power plant simulator for use in operator training	TOTAL	RG 1.149 supports 55.45(a) and 55.45(b). RG 1.149 endorses ANSI/ANS-3.5-1985, which specifies minimum performance and configuration criteria for a simulator, for comparing a simulator to its reference plant, and for upgrading simulators to reflect changes to the reference plant response or control room configuration.
RG 1.150 (Rev. 1, 2/83)	Ultrasonic testing of reactor vessel welds during pre-service and in-service examination	INTENT	RG 1.150 supports GDC 1, 50.55a, and Appendix B of 10 CFR 50. RG 1.150 describes procedures for implementing the above requirements with regard to the preservice and in-service examinations of reactor vessel welds by ultrasonic testing (UT). The scope of this guide is limited to reactor vessel welds and does not apply to other structures and components such as piping. The INTENT of RG 1.150 is applicable to the core pressure boundary tube.
RG 1.151 (7/83)	Instrument sensing lines	TOTAL	RG 1.151 supports GDC 1, GDC 13, GDC 24, 50.34, and 10 CFR 100. RG 1.151 describes a method for complying with regulations with regard to the design and installation of safety-related instrument sensing lines. RG 1.151 endorses ISA-S67.02, which provides for the design, physical protection, and installation requirements for instrument sensing line piping and tubing. It also establishes the applicable ASME code requirements and boundaries for the design and installation of instrument

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.151 (7/83) continued			sensing lines that interconnect safety-related piping and vessels with safety-related instrumentation and nonsafety-related instrumentation. The requirements of RG 1.151 will be met by substituting the appropriate quality group classifications given in the main body of the PDR with the ASME Classes of 1, 2, or 3.
RG 1.152 (11/85)	Programmable digital computer system software in safety-related systems of nuclear power plants	TOTAL	RG 1.152 supports GDC 21 and 10 CFR 50 Appendix B. RG 1.152 describes a method for promoting high functional reliability for safety-related systems using programmable digital computer systems. This method is applicable to designing software, verifying software, implementing software, and validating computer systems. RG 1.152 endorses ANSI/IEEE-ANS-7.4.3.2-1982. This standard does not address any follow-on activities such as testing and validation of computer systems beyond the design, implementation, and integration phases. RG 1.118 provides guidance on periodic testing of hardware and software.
RG 1.153 (12/85)	Power, instrumentation, and control portions of safety systems	TOTAL	RG 1.153 supports GDCs 2, 4, 10, 12, 13, 15, 17, 18, 20, 21, 22, 23, 24, 25, 29, 34, 37, 40, 43, 46, and 54; 50.49, and 50.55a. RG 1.153 describes a method for complying with the regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems. RG 1.153 endorses IEEE Std. 603-1980, with minor exceptions.
RG 1.154 (1/87)	Format and content of plant-specific pressurized thermal shock safety analysis reports for pressurized water reactors	NONE	RG 1.154 supports 50.61. Because the ANS has an aluminum pressure tube instead of a steel pressure vessel and because the ANS is a low pressure, low temperature facility, this RG is not applicable.
RG 1.155 (8/88)	Station blackout	TOTAL	RG 1.155 supports GDC 1, GDC 17, GDC 18 and 50.63; and endorses NUMARC-8700. RG 1.155 describes a method for plants to cope with a station blackout for a specified duration. RG 3.2 provides guidance for determining the length of time that a plant is able to cope with a station blackout.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 1.156 (11/87)	Environmental Qualification of Connection Assemblies for Nuclear Power Plants	TOTAL	
RG 1.157 (05/89)	Best-Estimate Calculation of Emergency Core Cooling System Performance	INTENT	RG 1.157 supports 50.46 and Appendix K of 10 CFR 50. RG 1.157 describes models, correlations, data, model evaluation procedures, and methods for performing realistic or best estimate calculations of ECCS performance during a loss-of-coolant accident and for estimating the uncertainty in that calculation. RG 1.157 also permits the use of other models, data, model evaluation procedures, and methods if they are supported by appropriate experimental data and technical justification. Therefore, the ANS will meet the requirements of this guide even though it cannot meet the specifics due to the differences in the ECCS.
RG 1.158 (02/89)	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	TOTAL	RG 1.158 supports GDC 1, GDC 2, GDC, GDC 23, and Appendix B of 10 CFR 50. RG 1.158 supports IEEE Std. 535-1986 with some exceptions, which describes qualification methods for Class 1E lead storage batteries and racks to be used outside of primary containment. Safety-related batteries undergo a program of qualification as part of an overall quality assurance program that also includes requirements for design, production, quality control, installation, maintenance, and periodic testing. This RG addresses only the qualification portion of the overall quality assurance program.
RG 1.159 (08/90)	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
RG 2.1	Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors	TOTAL	
RG 2.2 (11/73)	Development of Technical Specifications for Experiments in Research Reactors	TOTAL	

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 2.4 (5/77)	Review of Experiments for Research Reactor	TOTAL	
RG 2.5 (10/77)	Quality Assurance Program Requirements for Research Reactors	TOTAL	
RG 2.6 (Rev. 1, 3/83)	Emergency Planning for Research Reactors	TOTAL	
RG 3.2	Efficiency Testing of Air-Cleaning Systems Containing Devices for Removal of Particles	TOTAL	
RG 3.9	Concrete Radiation Shields	TOTAL	
RG 3.42 (Rev. 1, 9/79)	Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70	TOTAL	
RG 3.43 (Rev. 1, 4/79)	Nuclear criticality safety in the storage of fissile materials	TOTAL	
RG 4.1 (Rev. 1, 4/75)	Programs for monitoring radioactivity in the environs of nuclear power plants	TOTAL	
RG 4.2 (Rev. 2, 7/76)	Preparation of environmental reports for nuclear power Plants	TOTAL	
RG 4.7 (Rev. 1, 11/75)	General site suitability for nuclear power stations	TOTAL	
RG 4.8 (12/75)	Environmental technical specification for power plants	TOTAL	
RG 4.11 (Rev. 1, 8/77)	Terrestrial environmental studies for nuclear power plants	TOTAL	
RG 4.15 (Rev. 1, 2/79)	Quality assurance for radiological monitoring programs (normal operations) - effluent streams and the environment	TOTAL	
RG 5.1 (12/72)	Serial numbering of fuel assemblies for light-water-cooled nuclear power plants	NONE	The ANS uses two fuel assemblies (upper and lower) that are different sizes. Light water reactors use numerous fuel assemblies that are usually identical. Therefore, RG5.1 is not specifically applicable to the design, construction, or operation of the ANS. However, the fuel cores will be numbered.
RG 5.7 (Rev. 1, 5/80)	Entry/exit control for protected areas, vital areas, and material access	TOTAL	

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NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 5.10 (7/73)	Selection and use of pressure-sensitive seals on containers for on-site storage of special nuclear material	TOTAL	
RG 5.12 (11/73)	General use of locks in the protection and control of facilities and special nuclear materials	TOTAL	
RG 5.13 (11/73)	Conduct of nuclear material physical inventories	TOTAL	
RG 5.15 (1/74)	Security seals for the protection and control of special nuclear materials	TOTAL	
RG 5.20 (1/74)	Training, equipping, and qualifying of guards and watchmen	TOTAL	
RG 5.29 (Rev. 1, 6/75)	Nuclear material control systems for nuclear power plants	TOTAL	
RG 5.43 (1/75)	Plant security force duties	TOTAL	
RG 5.44 (Rev. 2, 5/80)	Perimeter intrusion alarm systems	TOTAL	
RG 5.54 (3/78)	Standard format and content of safeguard contingency plans for nuclear power plants	TOTAL	
RG 5.62 (2/81)	Reporting of physical security events	N/A-L	The licensing requirements given in this section of 10 CFR 50 do not provide any design requirements, only requirements for dealing with the licensing agency (i.e., the Nuclear Regulatory Commission).
RG 5.65 (9/86)	Vital access controls, protection of physical security equipment, and key and lock controls	TOTAL	
RG 7.1 (6/74)	Administrative guide for packing and transporting radioactive material	TOTAL	
RG 7.4 (6/75)	Leakage tests on packages for shipment of radioactive materials	TOTAL	
RG 7.6 (Rev. 1, 3/78)	Design criteria for the structural analysis of shipping cask containment vessels	TOTAL	
RG 7.7 (8/77)	Administrative guide for verifying compliance with packaging requirements for shipments of radioactive materials	TOTAL	

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 7.8 (5/77)	Load combinations for the structural analysis of shipping casks	TOTAL	
RG 7.9 (Rev. 1, 1/80)	Standard format and content of Part 71 applications for approval of packaging of Type B, large quantity, and fissile radioactive material	TOTAL	
RG 7.10 (Rev. 1, 6/86)	Establishing quality assurance programs for packaging used in the transport of radioactive materials	TOTAL	
RG 8.1	Radiation Symbol	TOTAL	
RG 8.2 (2/73)	Guide for administrative practice in radiation monitoring	TOTAL	
RG 8.3 (2/73)	Film badge performance criteria	TOTAL	
RG 8.4 (2/73)	Direct-reading and indirect-reading pocket dosimeters	TOTAL	
RG 8.5 (Rev. 1, 3/81)	Criticality and other interior evacuation alarms	TOTAL	
RG 8.6 (5/73)	Standard test procedures for Geiger-Muller counters	TOTAL	
RG 8.7 (5/73)	Occupational radiation exposure records systems	TOTAL	
RG 8.8 (Rev. 3, 6/78)	Information relative to ensuring that occupational radiation exposures at nuclear power station will be as low as reasonably achievable	TOTAL	
RG 8.9 (9/73)	Acceptable concepts, models, equations, and assumptions for a bioassay program	TOTAL	
RG 8.10 (Rev. 1-R, 5/77)	Operating philosophy for maintaining occupational radiation exposures as low as reasonably achievable	TOTAL	
RG 8.12 (Rev. 1, 1/81)	Criticality accident alarm systems	TOTAL	
RG 8.13 (Rev. 1, 11/75)	Instruction concerning prenatal radiation exposure	TOTAL	
RG 8.14 (Rev. 1, 8/77)	Personnel neutron dosimeters	TOTAL	

Table B.7. Compliance Commitment to NRC Regulatory Guides for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
RG 8.15 (10/76)	Acceptable programs for respiratory protection	TOTAL	
RG 8.19 (Rev. 1, 6/79)	Occupational radiation dose assessment in light-water-cooled reactor power plants design stage man-rem estimates	TOTAL	
RG 8.27 (3/81)	Radiation protection training for personnel at light-water-cooled nuclear power plants	TOTAL	
RG 8.29	Instruction Concerning Risks from Occupational Radiation Exposure	TOTAL	

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

A review of Standard Review Plan (NUREG-0800) and the Branch Technical Positions, last published in 1984, is given below. Justification and Guidance is given for each requirement to aid in determining applicability at the SDD level. The guidance given is, in essence, a requirement. Any requests for exceptions to the guidance must be submitted to the chairman of the plant design requirements working group. Each exception will be reviewed on a case-by-case basis.

Definitions required to interpret this table are:

TOTAL — *All technical requirements are applicable as written.*

PARTIAL — *Due to specific requirements in the regulations or design features in the ANS, not all of the technical requirements apply as written. The PDR shall define that part of the regulation that is and is not applicable at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

INTENT — *Only the purpose (or intent) of the technical parts of the regulation is applicable. The PDR shall define the intent of the regulation and how it applies to the ANS at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

NONE — *Due to specific requirements in the regulations or design features in the ANS, none of the technical requirements are applicable. The PDR shall provide justification why the regulation is not applicable.*

N/A-L — *The regulation sets forth only licensing procedural requirements, and does not contain any technical requirements.*

The SRP sections or the BTP's may contain statements that make it appear to be not-applicable to the design of the ANS, yet the compliance commitment level given is TOTAL, PARTIAL, or INTENT. This can occur because the regulation may, in some cases, provide an example. Therefore, statements like "such as steam generators" simply provides guidance and is not part of the requirement it self.

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
Chapter 1	Introduction and general description	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
Chapter 2	Site characteristics	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 2.3.3, Appendix A (Rev. 2, 7/81)	Recommended format for hourly meteorological data to be placed on magnetic tape	INTENT	This appendix provides a recommended format for magnetic tapes that are to be used to record meteorological data to support safety analysis reports prior to operation and for making atmospheric airborne dispersion estimates for accidental and expected routine airborne releases subsequent to reactor startup. This BTP will be reviewed to ensure that all data required for site evaluation are being collected and documented under DOE and OR orders.
SRP 2.4.1, Appendix A (Rev. 2, 7/81)	Hydrologic engineering site visits	N/A-L	Because this appendix deals with NRC staff internal procedures for reviewing an applicant's site and regional hydrologic characteristics, it is not applicable to the ANS.
SRP 2.4.12, BTP HGEB-1 (Rev. 2, 7/81)	Safety-related permanent dewatering	NONE	This BTP provides acceptable geotechnical and hydrologic design bases and criteria for permanent dewatering systems used for safety-related buildings and structures. Because the ANS does not have a dewatering system, this BTP is not applicable.
Chapter 3	Design of structures, components	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 3.2.2, Appendix A (Rev. 1, 7/81)	Classification of main steam components other than reactor coolant boundary for BWR plants	NONE	This BTP provides clarification on the quality group classification criteria for main steam components for BWR plants to assure that appropriate standards and codes are applied to this high-energy fluid piping system. Because this BTP specifically addresses BWR main steam components, it is not applicable to the ANS.
SRP 3.2.2, Appendix B (Rev. 1, 7/81)	Classification of BWR/6 main steam and feedwater components	NONE	This appendix to the SRP provides classification of BWR/6 main steam and feedwater components that are not classified as safety-related items, are not designed to seismic Category I standards, and are not housed in seismic Category I structures. Because this appendix is peculiar to BWR plants, it is not applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 3.5.3, Appendix A (7/81)	Permissible ductility ratio for overall damage prediction	TOTAL	This appendix provides criteria foe permissible ductility ratios for reinforced concrete and steel structural elements subjected to impacted and impulsive loads. It is therefore applicable to the ANS for use in evaluating the overall response of reinforced concrete and steel structural elements subjected to impactive or impulsive loads.
SRP 3.6.1, BTP ASB 3-1 (Rev. 1, 7/81)	Protection against postulated piping failures in fluid systems outside containment	TOTAL	This BTP provides an approach for the design, including the arrangement, of fluid systems located outside of containment to ensure that the plant can be safely shut down in the event of piping failures outside containment. This BTP is applicable to the ANS.
SRP 3.6.2, BTP MEB 3-1 (Rev. 1, 7/81)	Postulated rupture locations in fluid system piping inside and outside containment	TOTAL	This BTP provides guidance for selecting postulated rupture locations for breaks in fluid system piping, analytically demonstrating the adequacy of fluid system piping design, and demonstrating that essential systems and components required to shut down the reactor safely and mitigate the consequences of postulated pipe failures are not adversely affected. This BTP is applicable to the ANS.
SRP 3.8.4, Appendix A (Rev. 0, 7/81)	Interim criteria for safety-related masonry wall evaluation	TOTAL	This appendix provides minimum design consideration and criteria for safety-related masonry walls. This appendix is applicable to the ANS.
SRP 3.8.4, Appendix B (Rev. 0, 7/81)	Structural design audits	INTENT	This appendix provides internal guidance to NRC staff for performing structural design audits.
SRP 3.8.4, Appendix C (Rev. 0, 7/81)	Category I structures design reports, table of contents	INTENT	The primary objective of the design report is to provide the NRC reviewer with design and construction information more specific than that contained in the SAR, which can assist the staff to plan and conduct a structural audit as per SRP 3.8.4, Appendix B.
SRP 3.8.4, Appendix D (Rev. 0, 7/81)	Technical position on spent fuel pool racks	TOTAL	This appendix provides minimum requirements and criteria for spent fuel pool racks and associated structures. It is applicable to the spent fuel storage facilities at ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 3.9.3, Appendix A (Rev. 1, 4/84)	Stress limits for ASME Class 1, 2, and 3 components and component supports of safety-related systems and Class CS core support structures under specified service loading combinations	TOTAL	This appendix improves the consistency and understanding of the basic approach in the selection of load combinations applicable to safety-related systems, and establishes acceptable relationships between plant postulated events, plant and system operating conditions, component and component support design, and service stress limits, functional capability, and operability. This appendix is applicable to the ANS.
Chapter 4	Reactor	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 4.2, Appendix A (7/81)	Evaluation of fuel assembly structural response to externally applied forces	TOTAL	This appendix provides guidance for evaluating the adequacy of a fuel assembly structural response to earthquakes and postulated pipe breaks. Because this appendix provides overall general guidelines covering the analysis of loads, determination of assembly strength, and acceptance criteria, it is applicable to the ANS.
SRP 4.3, BTP CPB 4.3-1 (Rev. 2, 7/81)	Westinghouse constant axial control (CAOC)	INTENT	This BTP addresses analysis and test requirements for determining allowed power shapes and power densities relative to the design bases, peaking factors from in-core maps during load-follow transients, and valid power distribution envelopes for all normal and anticipated reactor conditions.
Chapter 5	Reactor coolant system and connected systems	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 5.2.2, BTP RSB 5-2 (Rev. 0, 8/81)	Overpressurization protection of pressurized water reactors while operating at low temperatures	INTENT	This BTP is concerned with a special system required for licensed PWR plants to protect them from overpressurization while operating at low temperatures (primarily during startup and shutdown conditions). Therefore, the BTP is not applicable to the ANS.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 5.3.2, BTP MTEB 5-2 (Rev. 1, 7/81)	Fracture toughness requirements	INTENT	This BTP summarizes and clarifies the ASME Code and 10 CFR 50 requirements regarding fracture toughness, pressure-temperature limits, and material surveillance for ferritic materials in pressure-retaining components of the reactor coolant pressure boundary (RCPB). This BTP is applicable to those portions of the RCPB for the ANS that are ferritic materials (i.e., this BTP does not apply to the removable portion of the reactor coolant pressure tube). The INTENT of this BTP should be expanded to include the aluminum reactor assembly components that form the RCPB, including the reflector vessel.
SRP 5.4.2.1, BTP MTEB 5-3 (Rev. 2, 7/81)	Monitoring of secondary side water chemistry in PWR steam generators	INTENT	This BTP provides recommendations for design and water chemistry programs for steam generators to ensure that the integrity of tubes and tube sheets will be maintained under operating, maintenance, and testing conditions. This BTP is applicable to the design and water chemistry programs for the facility's heat exchangers.
SRP 5.4.7, BTP RSB 5-1 (Rev. 2, 7/81)	Design requirements of the residual heat removal system	TOTAL	Because the design of residual heat removal (RHR) systems vary, this BTP provides guidance on functional, isolation, pressure relief, pump protection, and test requirements for the RHR system. This position is applicable to the ANS.
SRP 5.4.12 (Rev. 0, 7/81)	Reactor coolant system high point vents	TOTAL	See also Table B.4, section 50.34(f)(2)(vi)
Chapter 6	Engineered safety systems	TOTAL	The COMPLIANCE COMMITMENT level for each section of this Chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 6.1.1, BTP MTEB 6-1 (Rev. 2, 7/81)	pH for emergency coolant water for PWRs	INTENT [HOLD][a]	This BTP provides criteria for post-accident emergency coolant water pH control to reduce the probability of stress-corrosion cracking in austenitic stainless steel components (i.e., nonsensitized, sensitized, nonstressed, or stressed). The INTENT of guidance in the BTP may be applicable to the ANS if borated water is used for poison injection.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 6.2.1.1.C, Appendix A (Rev. 2, 1/83)	Steam bypass for Mark I, II, and III containments	NONE	This appendix provides design criteria for quenching steam that bypasses the suppression pool during a loss-of-coolant accident in pressure-suppression-type containments.
SRP 6.2.1.5, BTP CSB 6-1 (Rev. 2, 7/81)	Minimum containment pressure model for PWR ECCS performance evaluation	TOTAL	This BTP provides guidance for performing minimum containment pressure analysis. This BTP is applicable to the ANS.
SRP 6.2.3, BTP CSB 6-3 (Rev. 2, 7/81)	Determination of bypass leakage paths in dual containment plants	TOTAL	This BTP provides guidance in the determination of that portion of the primary containment leakage that can circumvent the secondary containment boundary and escape directly to the environment.
SRP 6.2.4, BTP CSB 6-4 (Rev. 2, 7/81)	Containment purging during normal plant operations	TOTAL	This BTP provides design criteria for systems used to purge the containment during reactor operation.
SRP 6.2.5, Appendix A (Rev. 2, 7/81)	Description of combustible gas analyzer program	PARTIAL	This appendix describes a digital computer program, COGAP (Combustible Gas Analyzer Program) that is used for determining hydrogen-oxygen concentrations within reactor containment following loss-of-coolant accidents. Those sections that concern zirconium fuel cladding or zinc corrosion are not applicable. The evaluation of combustible gases also needs to be expanded for the ANS because of heavy-water being its primary coolant.
SRP 6.3, BTP RSB 6-1 (Rev. 1, 7/81)	Piping from the RWST (or BWST) and containment sump(s) to the safety injection pumps	INTENT	This BTP provides criteria for the design of piping systems from the refueling water storage tank (RWST) or the borated water storage tank (BWST) to the safety injection pumps of PWRs. Even though the ECCS at the ANS does not take suction from a borated source of water, the intent of this position regarding the evaluation of the design of the piping systems for the suction lines to the safety injection pumps is applicable.
SRP 6.4, Appendix A (Rev. ?, 7/81)	Acceptance criteria for valve or damper repair alternative	TOTAL	This appendix describes acceptable single-failure criteria for valves and repair of valve dampers used in control room ventilation systems. This appendix is applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
Chapter 7	Instrumentation and controls	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 7.1, Appendix A (Rev. 1, 2/84)	Acceptance criteria and guidelines for instrumentation and control systems important to safety	TOTAL	This appendix identifies (1) the General Design Criteria and IEEE 279 criteria and (2) the RGs which are applicable to instrument and control systems important to safety. The GDCs include 1, 2, 4, 13, 19, 20-25, and 29. RGs include 1.22, 1.47, 1.53, 1.75, 1.97, 1.105, 1.118, and 1.151. This appendix is applicable to the ANS.
SRP 7.1, Appendix B (Rev. 0, 7/81)	Guidance for evaluation of conformance to IEEE Std 279	TOTAL	This appendix discusses the requirements of IEEE 279, Sect. 3 (Design Basis) and Sect. 4 (Requirements) as they pertain to the reactor trip system and engineered safety features actuation system. This appendix is applicable to the ANS.
SRP Appendix 7-A, BTP ICSB 3 (Rev. 2, 7/81)	Isolation of low pressure systems from the high pressure reactor coolant system	NONE	This BTP provides design guidance for avoiding damage by overpressurization to low-pressure systems that are connected to the high-pressure reactor coolant system. Because the ANS does not have a high-pressure RCS, this BTP is not applicable.
SRP Appendix 7-A, BTP ICSB 4 (Rev. 2, 7/81)	Requirements of motor-operated valves in the ECCS accumulator lines	INTENT	This BTP provides specific guidance in meeting the intent of IEEE 279 for motor-operated isolation valves in ECCS accumulator injection lines. The intent of this BTP is applicable to the ANS for those motor-operated valves in lines considered to be "operating bypasses" (i.e., the protective function of that line is defeated when the motor-operated valve is closed).
SRP Appendix 7-A, BTP ICSB 12 (Rev. 2, 7/81)	Protection system trip point changes for operation with reactor coolant pumps out of service	TOTAL	This BTP provides additional guidance on the application of Sect. 4.15 of IEEE 279, which states that if a mode of reactor operation requires a more restrictive set point, the means of ensuring use of the more restrictive set point shall be positive (i.e., either automatic adjustment during operation or shutting the plant down before manually adjusting the set points).

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP Appendix 7-A, BTP ICSB 13 (Rev. 2, 7/81)	Design criteria for auxiliary feedwater systems	INTENT	This BTP states that the auxiliary feedwater system in a PWR should be capable of satisfying its functional requirements assuming a break in the auxiliary feedwater piping inside containment together with a single electric failure as defined in IEEE 279. This BTP may be applicable to the ANS because it concerns the removal of afterheat from the core after a loss of off-site power.
SRP Appendix 7-A, BTP ICSB 14 (Rev. 2, 7/81)	Spurious withdrawals of single control rods in pressurized water reactors	INTENT	The intent of this BTP is to provide specific guidance toward an acceptable interpretation and application of GDCs 20 and 25 with regard to inadvertent single control rod withdrawals. This BTP is applicable to the design of the reactivity control system.
SRP Appendix 7-A, BTP ICSB 20 (Rev. 2, 7/81)	Design of instrumentation and controls provided to accomplish changeover from injection to recirculation mode	TOTAL	This BTP is concerned with the design of instrumentation and controls provided to accomplish changeover from injection to recirculation mode following a loss-of-coolant accident.
SRP Appendix 7-A, BTP CSB 21 (Rev. 2, 7/81)	Guidance for application of Regulatory Guide 1.47	TOTAL	This BTP provides supplemental guidance for the implementation of the RG 1.47, which describes a method of implementing IEEE 279. Because IEEE 279 and RG 1.47 provide criteria for indicating bypasses of protection systems, this BTP is applicable to the ANS.
SRP Appendix 7-A, BTP ICSB 22 (Rev. 2, 7/81)	Guidance for application of Regulatory Guide 1.22	TOTAL	This BTP provides guidance in the application of RG 1.22, regarding protection system functions that are not tested while the reactor is operating at power. This BTP requires the identification of all actuated equipment not tested during operation, and an assessment of how each item conforms to the provisions of paragraph D.4 of RG 1.22. This BTP is applicable to the ANS.

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NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP Appendix 7-A, BTP ICSB 26 (Rev. 2, 7/81)	Requirements for reactor protection system anticipatory trips	TOTAL	NRC created this BTP because several plants incorporated a number of anticipatory or "back-up" trips for which no credit was taken in the accident analyses. These trips were typically not designed to the requirements of IEEE 279, and therefore introduced nonsafety grade equipment into the reactor protection system. This BTP is applicable to the ANS for ensuring that all reactor trips incorporated in the reactor protection system will meet the requirements of IEEE 279.
SRP Appendix 7-B (Rev. 1, 4/81)	General agenda, station site visits	INTENT	Only the INTENT of this appendix is applicable to the ANS because it deals with NRC staff internal procedures for review of design and evaluation of the actual implementation of the design as installed at the site.
Chapter 8	Electric Power	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 8.2, Appendix A (Rev. 0, 7/83)	Guidelines for generator circuit breakers/load break switches	NONE	This appendix provides guidance for circuit breaker installation that will allow a plant to power its off-site power supply circuits directly from its unit generator. Because the ANS does not produce electric power (and consequently does not have a unit generator), this appendix is not applicable.
SRP Appendix 8-A, BTP ICSB 4 (Rev. 2, 7/81)	Requirements on motor-operated valves in the ECCS accumulator lines	INTENT	This BTP provides specific guidance in meeting the intent of IEEE 279 for motor-operated isolation valves in ECCS accumulator injection lines. This position is applicable to the accumulator injection lines. In addition, the intent of this BTP is applicable to all motor-operated valves that are considered to be "operating bypasses."
SRP Appendix 8-A, BTP ICSB 8 (Rev. 2, 7/81)	Use of diesel-generator sets for peaking	NONE	This BTP is concerned with the use of emergency power diesel-generator sets for providing peaking power service to utility transmission systems. Because the mission of the ANS is not to generate electric power, this BTP is not applicable.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP Appendix 8-A, BTP ICSB 11 (Rev. 2, 7/81)	Stability of offsite power systems	TOTAL	This BTP is concerned with the loss of electrical transmission grid stability and the availability of off-site power with the loss of the largest operating unit on the grid. This BTP is applicable to the ANS because the facility must be able to withstand voltage fluctuations on the grid and the possibility of these fluctuations causing a reactor scram.
SRP Appendix 8-A, BTP ICSB 18 (Rev. 2, 7/81)	Application of the single-failure criterion to manually controlled electrically operated valves	TOTAL	This BTP provides guidelines for disconnecting power to electrical components of a fluid system as one acceptable means of designing against a single failure that might cause an undesirable component action. This BTP is applicable to those safety-related electrically operated valves at the ANS.
SRP Appendix 8-A, BTP ICSB 21 (Rev. 2, 7/81)	Guidance for application of Regulatory Guide 1.47	TOTAL	This BTP provides supplemental guidance for the implementation of the RG 1.47, which describes a method of implementing IEEE 279. Because IEEE 279 and RG 1.47 provide criteria for indicating bypasses of protection systems, this BTP is applicable to the ANS.
SRP Appendix 8-A. BTP PSB 1 (Rev. 0, 7/81)	Adequacy of station electric distribution system voltages	TOTAL	This BTP provides guidance for protecting Class 1E power systems from low voltage conditions. Because sustained low grid voltage conditions can result in adverse effects on Class 1E loads, this BTP is applicable to the ANS.
SRP Appendix 8-A, BTP PSB 2 (Rev. 0, 7/81)	Criteria foe alarms and indications associated with diesel-generator unit bypassed and inoperable status	TOTAL	This BTP provides guidance for meeting the provisions of RG 1.47 as it pertains to diesel-generator units. Because standby or emergency power generating units are required to power safety-related systems and components, this BTP is applicable to the ANS.
SRP Appendix 8-B (Rev. 0, 7/81)	General agenda, station site visits	INTENT	This appendix deals with NRC staff internal procedures for review of design and evaluation of the actual implementation of the design as installed at the site.
Chapter 9	Auxiliary Systems	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 9.2.5, BTP ASB 9-2 (Rev. 2, 7/81)	Residual decay energy for light-water reactors for long-term cooling	INTENT	This position provides equations, assumptions, and graphs which are to be used for determining fission product decay heat and heavy element decay heat in terms of fractions of full reactor operating power. Because this BTP provides assumptions and formulations for calculating the residual decay energy for long-term cooling of light-water-cooled reactors, only the intent of this position is applicable to the ANS.
SRP 9.5.1, BTP CMEB 9.5-1 (Rev. 2, 7/81)	Guidelines for fire protection for nuclear power plants	TOTAL	This BTP provides guidance for the development of a fire protection program for safety-related systems and other plant areas containing fire hazards that could adversely affect safety-related systems. The requirements of this BTP are applicable to the ANS and should be cross-checked against DOE 5480.1A.
Chapter 10	Steam and Power Conversion	NONE	
SRP 10.4.7, BTP ASB 10-2 (Rev. 3, 4/84)	Design guidelines for avoiding water hammers in steam generators	INTENT	This position provides guidance for eliminating water hammer in top-feed steam generator designs, preheat steam generator designs, and in once-through steam generator designs. Because of the similarities between heat exchangers and steam generators, this BTP is applicable to the ANS.
SRP 10.4.9, BTP ASB 10-1 (Rev. 2, 7/81)	Design guidelines for auxiliary feedwater system pump drive and power supply diversity for pressurized water reactor plants (PWRs)	*[TBD]*[B-90]	This BTP provides design guidelines acceptable for auxiliary feedwater system pump drives and power supplies for PWRs. This system in PWRs is used for heat removal following a reactor trip and loss of off-site power. It provides makeup water to the steam generators, which produce steam that is released to the atmosphere. Compliance will be provided at a later date.
Chapter 11	Radioactive Waste Management	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 11.3, BTP ETSB 11-5 (Rev. 0 7/81)	Postulated releases due to a waste gas system leak or failure	TOTAL	This position provides guidelines for limiting postulated radioactive releases due to a radioactive waste gas system leak or failure. The goal is to ensure that the radiological consequences of a single failure in a waste gas system will not result in the total body exposure exceeding 5m5v (0.5 rem) to an individual at the nearest exclusion area boundary. This BTP is applicable to the ANS and will be broadened to evaluate tritium releases.
SRP 11.4, BTP ETSB 11-3 (Rev. 2, 7/81)	Design guidance for solid radioactive waste management systems in light-water-cooled nuclear power reactor plants	TOTAL	This BTP provides design requirements for solid radioactive waste systems. Solid waste systems process both dry and wet wastes. Dry wastes include activated charcoal, HEPA filters, rags, paper, and clothing. Wet wastes include spent bed resins, filter sludge, evaporator and reverse osmosis concentrates, and spent cartridge filter elements. This BTP is applicable to the ANS.
SRP 11.4, Appendix 11.4-A (Rev. 0, 7/81)	Design guidance for temporary onsite storage of low level radioactive waste	TOTAL	This BTP provides guidance for the design of temporary on-site storage of low-level radioactive wastes at nuclear facilities. This position addresses the storage requirements for wet waste, solidified wet waste, and dry low-level waste and reemphasizes the safety considerations in the storing, handling, and disposal of low-level radioactive waste. This BTP is applicable to the ANS.
SRP 11.5, Appendix A (Rev. 1, 7/81)	Design guidance for radiological effluent monitors providing signals for initiating termination of flow or other modification of effluent stream properties	TOTAL	This appendix provides design guidance for monitors used to initiate signals for actuation of systems to control the release of radioactive materials in effluents to the environment. This appendix does not apply to effluent monitors used to mitigate the consequences of a design basis accident. This appendix is applicable to the ANS.
Chapter 12	Radiation Protection	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
Chapter 13	Conduct of Operations	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
Chapter 14	Initial Test Program	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
Chapter 15	Accident Analysis	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 15.1.5, Appendix A (Rev. 2, 7/81)	Radiological consequences of main steam line failures outside containment of a PWR	INTENT	This appendix provides guidance for the accident analysis of a PWR main-steam-line break accident (MSLB) outside containment. It covers specifically the evaluation of the radiological consequences of the MSLB accident.
SRP 15.4.8, Appendix A (Rev. 1, 7/81)	Radiological consequences of a control rod ejection accident (PWR)	PARTIAL	This appendix provides guidance for the evaluation of a control rod ejection accident analysis at a PWR. Its twofold objective is to evaluate specific plant conformance with 10CFR100 regulations and to verify the plant's capacity to mitigate the radiological consequences of a rod ejection accident. This appendix is applicable to the ANS.
SRP 15.4.9, Appendix A (Rev. 2, 7/81)	Radiological consequences of control rod drop accident (BWR)	NONE	This appendix provides guidance for evaluating the radiological consequence analysis of a postulated control rod drop accident at a BWR. Because the ANS is more closely related to a PWR rather than a BWR, it follows SRP 15.4.8, Appendix A and not this appendix.
SRP 15.6.5, Appendix A (Rev. 1, 7/81)	Radiological consequences of a design basis loss-of-coolant accident including containment leakage contribution	TOTAL	This appendix provides guidance for evaluating the fission product releases and radiological consequences resulting from a postulated design basis loss-of-coolant accident (LOCA) and the containment leakage doses that contribute to the total LOCA doses. This appendix is applicable for the evaluation of radiological consequence analyses from a postulated LOCA together with containment leakage.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 15.6.5, Appendix B (Rev. 1, 7/81)	Radiological consequences of a design basis loss-of-coolant accident: leakage from engineered safety feature components outside containment	PARTIAL	This appendix provides guidance for the evaluation of radiological consequences from fission product leakage from engineered safety features (ESF) equipment located outside the primary containment following a LOCA. The general guidelines of this appendix are applicable to the ANS.
SRP 15.6.5, Appendix D (Rev. 1, 7/81)	Radiological consequences of a design basis loss-of-coolant accident: leakage from main steam isolation valve leakage control system (BWR)	INTENT	This appendix provides guidance for evaluating the radiological consequences of a LOCA with leakage past the main steam isolation valves (MSIV) in a BWR. The intent of this appendix is applicable to the ANS because the ANS must evaluate leakage past all isolation valves on piping (or beam tubes) that penetrate containment.
Chapter 16	Technical Specifications	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
Chapter 17	Quality Assurance	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
Chapter 18	Human Factors	TOTAL	The COMPLIANCE COMMITMENT level for each section of this chapter will be provided at a later date. Until then, the entire chapter is applicable to the ANS.
SRP 18.1, Appendix A (Rev. 0, 9/84)	Evaluation criteria for detailed control room design reviews (DCRDR)	INTENT	This purpose of the detailed control room design review (DCRDR) is to improve the ability of control room operators to prevent accidents or to cope with accidents if they occur by improving the information provided to them. Even though this appendix was written for existing facilities, the intent of this appendix with respect to identifying deficiencies in human engineering is applicable to the ANS.

Table B.8. Compliance Commitment to the Standard Review Plan and Branch Technical Positions for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	COMMENTS
SRP 18.2, Appendix A (Rev. 0, 11/84)	Human factors review guidelines for the safety parameter display system (SPDS)	TOTAL	This appendix requires licensed power reactors to install a safety parameter display system (SPDS). This system is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This appendix is applicable to the ANS.

Table B.9. Compliance Commitment to Other NRC Regulations and Publications of Interest for the Design of the Advanced Neutron Source

SECTION	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE

A preliminary review of other NRC regulations and publications of interest is given below. Because this review is preliminary, designers should review all of 10 CFR 20 and 10 CFR 100 for applicability. Justification and Guidance will be given at a later date for each requirement to aid in determining applicability at the SDD level.

Definitions required to interpret this table are:

TOTAL — *All technical requirements are applicable as written.*

PARTIAL — *Due to specific requirements in the regulations or design features in the ANS, not all of the technical requirements apply as written. The PDR shall define that part of the regulation that is and is not applicable at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

INTENT — *Only the purpose (or intent) of the technical parts of the regulation is applicable. The PDR shall define the intent of the regulation and how it applies to the ANS at the overall plant level. The SDDs will provide the compliance commitment level and justification applicable to their specific system(s).*

NONE — *Due to specific requirements in the regulations or design features in the ANS, none of the technical requirements are applicable. The PDR shall provide justification why the regulation is not applicable.*

N/A-L — *The regulation sets forth only licensing procedural requirements, and does not contain any technical requirements.*

The regulations listed below may contain statements that make it appear to be not-applicable to the design of the ANS, yet the compliance commitment level given is TOTAL, PARTIAL, or INTENT. This can occur because the regulation may, in some cases, provide an example. Therefore, statements like "such as steam generators" simply provides guidance and is not part of the requirement it self.

10 CFR 20

20.102	Radiation dose standards for individuals in restricted areas	TOTAL	
20.103	Exposure of individuals to concentration of radioactive materials in air in restricted areas	TOTAL	
20.105	Permissible levels of radiation in unrestricted areas	TOTAL	
20.106	Radioactivity in effluents to unrestricted areas	TOTAL	
20.303	Disposal by release into sanitary sewage system	TOTAL	

Table B.9. Compliance Commitment to Other NRC Regulations and Publications of Interest for the Design of the Advanced Neutron Source

SECTION	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
20.305	Treatment or disposal by incineration	TOTAL	
20.306	Disposal of specific wastes	TOTAL	
10 CFR 61			
61.56	Waste characteristics	TOTAL	Resolved Unresolved Safety Issue (USI) C-17
10 CFR 70			
70.24	Criticality accident requirements	TOTAL	
10 CFR 73			
73.55	Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage	TOTAL	
10 CFR 100			
100.10	Factors to be considered when evaluating sites	TOTAL	
100.11	Determination of exclusion area, low population zone, and population center distance	TOTAL	The source term will reflect the source term expected from the ANS and not from a PWR or a BWR. The source term for the ANS is given in Appendix C of the PDR.
NRC SECY PAPERS			
SECY 80-31	Accident considerations under NEPA	TOTAL	Resolved USI A-33

Table B.9. Compliance Commitment to Other NRC Regulations and Publications of Interest for the Design of the Advanced Neutron Source

SECTION	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
SECY 88-272	Technical resolution of unresolved safety issues A-3, A-4, and A-5 regarding steam generator tube integrity	INTENT	Resolved USI A-3 and USI A-4
SECY 89-122	Resolution of unresolved safety issue (USI) A-48, hydrogen control measures and effects of hydrogen burns on safety equipment. Combustible gases include hydrogen, deuterium, CO, etc. Sources of combustible gases include fuel-clad interactions, core-concrete interactions, cold source inventory, etc. Therefore, any "hydrogen" control must be expanded to include "combustible gas" control.	TOTAL	Resolved USI A-48
GENERIC LETTERS[a]			
GL 88-14	Instrument air supply system problems affecting safety-related equipment	TOTAL	Resolved Generic Safety Issue (GSI) 43
GL 88-15	Electric power system - inadequate control over design processes	TOTAL	
GL 88-20	Individual plant exams for severe accident vulnerabilities	TOTAL	
GL 89-06	Task Action Plan item I.D.2 safety parameter display system	TOTAL	Resolved GSI 125.I.3
GL 89-13	Service water systems problems affecting safety-related equipment	TOTAL	Resolved GSI 51
GL 89-19	Request for action related to resolution of unresolved safety issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants"	TOTAL	Resolved USI A-47
GL 89-22	Resolution of generic safety issue no. 103, "Design for Probable Maximum Precipitation"	TOTAL	Resolved GSI 103

[a] These letters are addressed to NRC licensees and, in most cases, the subjects do not apply to research reactors. The commitment by ANS to follow these letters is limited to technical issues that apply to the design.

Table B.9. Compliance Commitment to Other NRC Regulations and Publications of Interest for the Design of the Advanced Neutron Source

SECTION	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
NRC INFORMATION BULLETINS			
IEB 79-06A	Review of operation errors and system misalignments identified during the Three Mile Island accident	TOTAL	Resolved TMI Action Plan Item II.K.1(17)
NRC PUBLICATIONS			
NUREG 0737	Clarification of TMI Action Plan requirements	TOTAL	Resolved TMI Action Plan Item II.K.3(1); and GSIs 5, 9, 10, 11, 12, 15, 19, 22, 27, 31, 44, and 46
NUREG 0737, Supplement 1	Clarification of TMI Action Plan requirements	TOTAL	Resolved TMI Action Plant Item III.A.1.2 and GSI 67.3.3

Table B.10. Compliance Commitment to MMES Policy Procedures for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
A preliminary review of MMES Policies and Procedures is given below. Guidance is given to aid in determining applicability at the SDD level.			
ENVIRONMENTAL SAFETY AND HEALTH			
ESH-7	Control of Occupational Exposure to Chemical and Physical Stresses	TOTAL	An effective program shall be maintained for control of employee exposure to chemical, radiation, and physical stresses consistent with the requirements of Martin Marietta Corporation, DOE and OSHA. Work areas shall be maintained free from or protected against hazards that cause illness, impaired health or significant discomfort. Employees shall be informed of hazards that may be encountered in their work areas and will be trained in techniques to maintain exposures as low as reasonably achievable (ALARA). The mandatory standards specified in DOE Orders 5480.4, 5480.10, 1324.2, 5480.1B, 5480.11, 5482.1B, 5483.1A, and 5484.1 must be met. Similar OR Orders are 5480.4, 5480.10, and 5484.1.
ESH-8	Safety Analysis and Review System	TOTAL	The safety analysis and review system must ensure that potential hazards are systematically identified; potential impacts and risks are analyzed and evaluated; reasonable measures have been taken to eliminate, control, or mitigate hazards; comparison has been made with applicable design criteria, codes, standards, and regulations and deviations have been explained and justified; the above processes along with management authorization of the operation are formally documented. The SAR will conform with appropriate NRC Regulatory Guides. Applicable standards and criteria are DOE Orders 5480.5, 5480.6, 5481.1B, 5000.3, and 6430.1A; LA-10294-MS; and MMES Policies and Procedures ESH-20, ESH-23, and GP-13.

Table B.10. Compliance Commitment to MMES Policy Procedures for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
ESH-14	Environmental Protection and Waste Management	TOTAL	Effluents will be managed, controlled, monitored, and reported in accordance with the provisions of DOE Orders and Directives; the applicable regulations of local, state, and federal regulatory agencies; and company environmental policies. Any actions that may potentially have a significant effect upon the quality of the environment shall be reviewed for environmental impacts, and this review shall be documented. Significant environmental compliance issues shall be identified. Numerous DOE Orders are required by ESH-14.
ESH-15	Substantial Risk/Significant Adverse Reaction Recordkeeping and Reporting Under the Toxic Substances Control Act	TOTAL	ESH-15 provides guidance for ensuring the proper recordkeeping and/or reporting of information that may relate to a substantial risk/significant adverse reaction to health or the environment from toxic chemicals. This procedure shall meet the specific requirements of Sections 8(c) and 8(e) of the Toxic Substances Control Act.
ESH-16	Control of Hazardous Materials	TOTAL	ESH-16 establishes guidelines to ensure the identification and control of materials that are classified as hazardous due to their toxicity, flammability, reactivity, or other special hazards. It does not apply to radioactive hazards, biological hazards, or hazardous wastes that are addressed in *[TBD]*[B-91] ESH-16 provides a mechanism for control and accountability of hazardous materials that are classified as Category II selected chemicals. Primary control of Category I chemicals is to be accomplished through "DEA Certificate Applications and Renewals."

Table B.10. Compliance Commitment to MMES Policy Procedures for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
ESH-17	Receipt and Shipment of Hazardous Materials and Hazardous Wastes (Including Radioactive Materials)	TOTAL	ESH-17 provides MMES policy to control the receipt, shipment, and transport of all materials and equipment to and from company installations to ensure accountability, safe transport, and compliance with appropriate international, federal, state, and local transportation regulations, as well as applicable DOE Manual provisions and directives. Associated Property Management, Financial, and Security Policies apply. Not included in this policy is intra-plant transport.
ESH-20	Safety of Nuclear Facilities	TOTAL	ESH-20 requires that the ANS will not operate until a proper safety evaluation has been made; the risks associated with operation are acceptable; approved operating and maintenance procedures are in place; and operators, supervisors, and support personnel have been properly trained.
ESH-21	Fire Protection Program	TOTAL	ESH-21 requires that the ANS maintain the necessary staff, service, and advisory groups to implement the installation fire protection, prevention, and control program; ensure that annual fire protection objectives, plans, and budgets are established; develops, documents, and implements standard operating procedures for fire protection that prevents and minimizes failures that might occur; reviews design for life safety compliance and multiple fire protection considerations; and conducts appropriate job-related training programs to ensure that designated employees can safely perform fire protection related tasks.
ESH-22	Nuclear Criticality Safety Program	TOTAL	The purpose of ESH-22 is to prevent the occurrence of an accidental nuclear criticality, to mitigate the consequences of a nuclear criticality accident such that the impact on facilities and the environment a5re as minor as reasonably achievable, and to protect the health and safety of on-site and off-site people.

Table B.10. Compliance Commitment to MMES Policy Procedures for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
ESH-23	Approval of Modifications or Additions to Nuclear Facilities	TOTAL	ESH-23 requires that procedures are implemented to provide for the appropriate review by technical experts and/or environmental safety and health specialists of all proposed modifications or changes to nuclear facilities. Reviews must be documented. The safety analysis and review system must be implemented whenever proposed changes that affect facility safety or result in a change to existing safety documentation.
GENERAL POLICIES			
GP-5	Quality Assurance Program	TOTAL	Quality assurance programs within Energy Systems shall be prepared, organized, and implemented in accordance with the requirements of ANSI/ASME NQA-1 (latest version), "Quality Assurance Program Requirements for Nuclear Facilities." The quality assurance of software is to be specifically addressed in the quality assurance program for any used for design verification.
GP-13	Occurrences Notifications, Investigation, and Reporting	TOTAL	It is MMES policy to assure that vigilance is exercised over all company activities; that every reasonable precaution is taken to protect the public, employees, environment, property, and company interest; that management evaluates and judges the significance of occurrences; that all occurrences are promptly investigated and accurately reported; and that management takes responsive action to implement the requirements of this procedure.

Table B.10. Compliance Commitment to MMES Policy Procedures for the Design of the Advanced Neutron Source

NUMBER	TITLE	COMPLIANCE COMMITMENT	JUSTIFICATION AND GUIDANCE
GP-14	Emergency Preparedness Program	TOTAL	MMES shall maintain an emergency preparedness program to protect the health and safety of employees, on-site personnel, members of the public, the environment, and property in the event of emergencies involving company activities. The emergency preparedness program will meet or exceed regulatory requirements. Emergency plans and procedures must be in general conformance with standards and criteria set forth in the following DOE Orders: 5000.3A, 5500.1A, 5500.2A, 5500.3A, 5500.4A, 5500.7A, 5500.8 and 5500.9. Provisions of the following Policy Procedures are applicable: GP-12, GP-13, PER-53, ESH-12, ESH-15, ESH-16, ESH-20, and ESH-22. Provisions of the following documents are applicable: Y/JA-346 and ANSI NQA-1. Provisions of the following documents are applicable as references: NFPA 1500; 29 CFR 1910; and INPO 85-001, Rev. 1.
GP-39	Security Requirements for the Shipment of Classified and Strategically Important Material	TOTAL	Classified and strategically important material shall be shipped in accordance with security requirements for its protection while in transit, and shall be in accordance with an approved security plan or as specifically detailed by the installation security department concerned.

Table B.11. Standards Required or Referenced by the MMES Policy Procedures

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE

A listing of standards and codes either required or recommended by MMES Policy Procedures are given below. The Justification and Guidance column indicates which Policy Procedure requires or recommends that particular standard or code. REQUIRED means that the Policy Procedure specifies that this standard "shall be" used in the design. RECOMMENDED means that the Policy Procedure states that this standard "shall be considered" in the design.

Some of the codes or standards listed below may contain statements that make it appear to be not-applicable to the design of the ANS, yet the compliance commitment level given is TOTAL, PARTIAL, or INTENT. This can occur because the code or standard may, in some cases, provide an example. Therefore, statements like "such as steam generators" simply provides guidance and is not part of the requirement it self.

American National Standards Institute (ANSI)		
ANSI Z4.1	Sanitation	REQUIRED by MMES ESH-7.
Code of Federal Regulations (CFR)		
10 CFR	Hazardous Materials Regulations as appropriate	REQUIRED by MMES ESH-17.
29 CFR 1910.20	Access to Empolyee Exposure and Medical Records	REQUIRED by MMES ESH-7.
29 CFR 1910, Subpart Z	Toxic and Hazardous Substances	REQUIRED by MMES ESH-7.
40 CFR	Hazardous Materials sections as appropriate	REQUIRED by MMES ESH-17.
40 CFR 717	EPA Reporting and Recordkeeping Requirements for Allegations that Chemicals Cause Significant Adverse Reactions to Health and Environment	REQUIRED by MMES ESH-15
49 CFR	Hazardous Materials sections as appropriate	REQUIRED by MMES ESH-17.

Table B.11. Standards Required or Referenced by the MMES Policy Procedures

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
DOE/TIC-11603	Non-Reactor Nuclear Facilities Standards and Guidelines	REQUIRED by MMES ESH-8 and MMES ESH-20.
Manufacturing Chemists Association (MCA)		
MCA	Manufacturing Chemists Association: Chemical Safety Data Sheets	REFERENCED by MMES ESH-16.
Martin Marietta Energy Systems (MMES)		
MMES ESS.ENV.1	Environmental, Safety, Health, and Waste Management Quality Assurance Program	REQUIRED by MMES ESH-7, MMES ESH-20, and MMES ESH-22.
MMES	Energy Systems Severity Index for Nuclear Criticality Safety Incidents	REQUIRED by MMES ESH-22.
MMES	Martin Marietta Energy Systems, Inc. Nuclear Criticality Safety Policy Statement (August 1988)	REQUIRED by MMES ESH-22.
National Fire Protection Agency (NFPA)		
NFPA	Fire Protection Guide on Hazardous Materials	REFERENCED by MMES ESH-16.

Table B.11. Standards Required or Referenced by the MMES Policy Procedures

NUMBER	TITLE	JUSTIFICATION AND GUIDANCE
Oak Ridge National Laboratory (ORNL)		
ORNL	ORNL Industrial Hygiene Manual	REFERENCED by MMES ESH-16.
ORNL-TM-5721	Material Safety Data Sheets, Vol. I	REFERENCED by MMES ESH-16.
ORNL-TM-5722	Material Safety Data Sheets, Vol. II	REFERENCED by MMES ESH-16.

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Appendix C:

DESIGN-BASIS EVENTS AND BEYOND DESIGN-BASIS EVENTS

Appendix C
DESIGN BASIS EVENTS AND CONDITIONS AND BEYOND DESIGN-BASIS EVENTS

PART I: DESIGN BASIS EVENTS AND CONDITIONS

1.1 INTRODUCTION

This appendix provides further definition of the design-basis events (DBEs) listed in Table 4.8-2 of the body of this specification. A wide range of some 22 design-basis event categories, with an average of about five different events per category, has been specified. These DBEs are intended to represent a limiting set of challenges to plant features and capabilities necessary for plant safety.

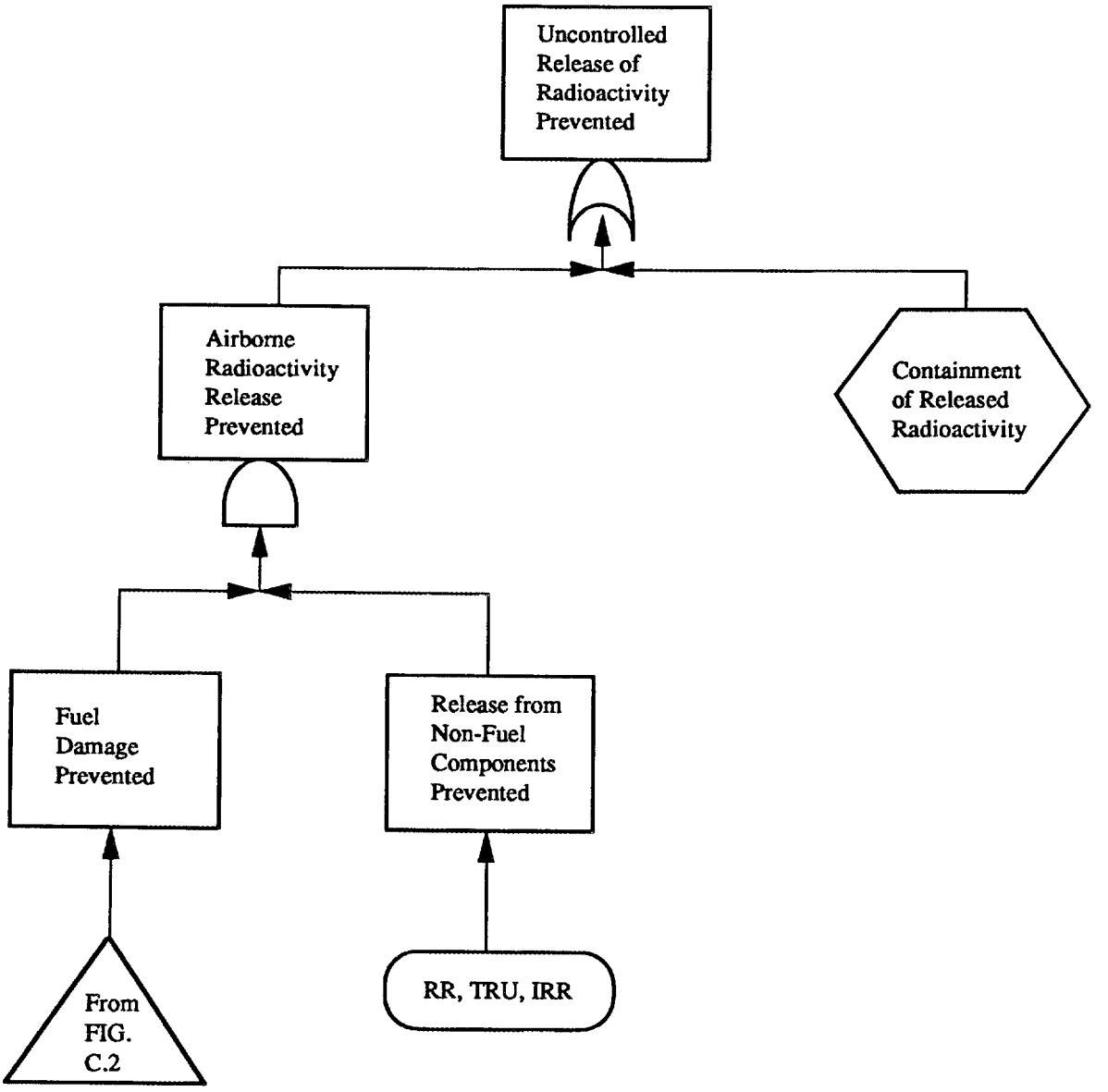
Figures C.1–C.3 present a simple fault tree that illustrates the relationship of the DBEs to the functions necessary for maintaining the barriers against uncontrolled release of radioactivity. The DBEs included in these figures are defined in Table 4.8-2. The emphasis is on fuel integrity, the primary and preferred barrier. To maintain fuel integrity, three functions must be maintained or controlled: reactor power (heat generation), fuel cooling, and coolant chemistry. The AND-gate above those functions in Fig. C.1 designates that all three functions must be successful for integrity to be maintained. Of the 22 DBE categories, 14 can directly affect fuel integrity.

A somewhat unique concern for ANS, the cooling of nonfuel components and structures inside the reactor, is also addressed in Fig. C.1 and accounts for four of the DBE categories. Component cooling is especially critical for ANS because of the intense radiation fields in the reactor and reflector vessel. These DBEs are important because failure of component cooling, much of which is supplied by the reflector vessel flow, could conceivably initiate an event that would challenge fuel integrity or lead to mechanical failures of systems needed to protect the reactor. For example, inadequate cooling of the core pressure boundary tube (CPBT) could possibly lead to a loss-of-coolant accident; or, inadequate cooling of a control rod could lead to control rod system failures.

The rationale for grouping the DBEs into frequency categories is discussed in Section 4.8.2 of this PDR. The basic purpose of the groupings is to be sure that appropriate acceptance criteria are applied to each condition or event. The frequency categories are repeated here for convenience:

1. NORMAL: planned, expected operational events.
2. TEST: planned tests involving the reactor facility.
3. ANTICIPATED EVENTS: off-normal events of frequency as low as 10^{-2}/year.
4. UNLIKELY EVENTS: accidents and events of frequency less than 10^{-2}/year but greater than 10^{-4}/year.
5. EXTREMELY UNLIKELY EVENTS: accidents and events of frequency less than 10^{-4}/year but greater than 10^{-6}/year.

Safety analyses must demonstrate that the acceptance criteria of Table 4.8-1 are satisfied for each event.



AND – gate

OR – gate

Fig. C.1. ANS design-basis events—prevention of uncontrolled release of radioactivity.

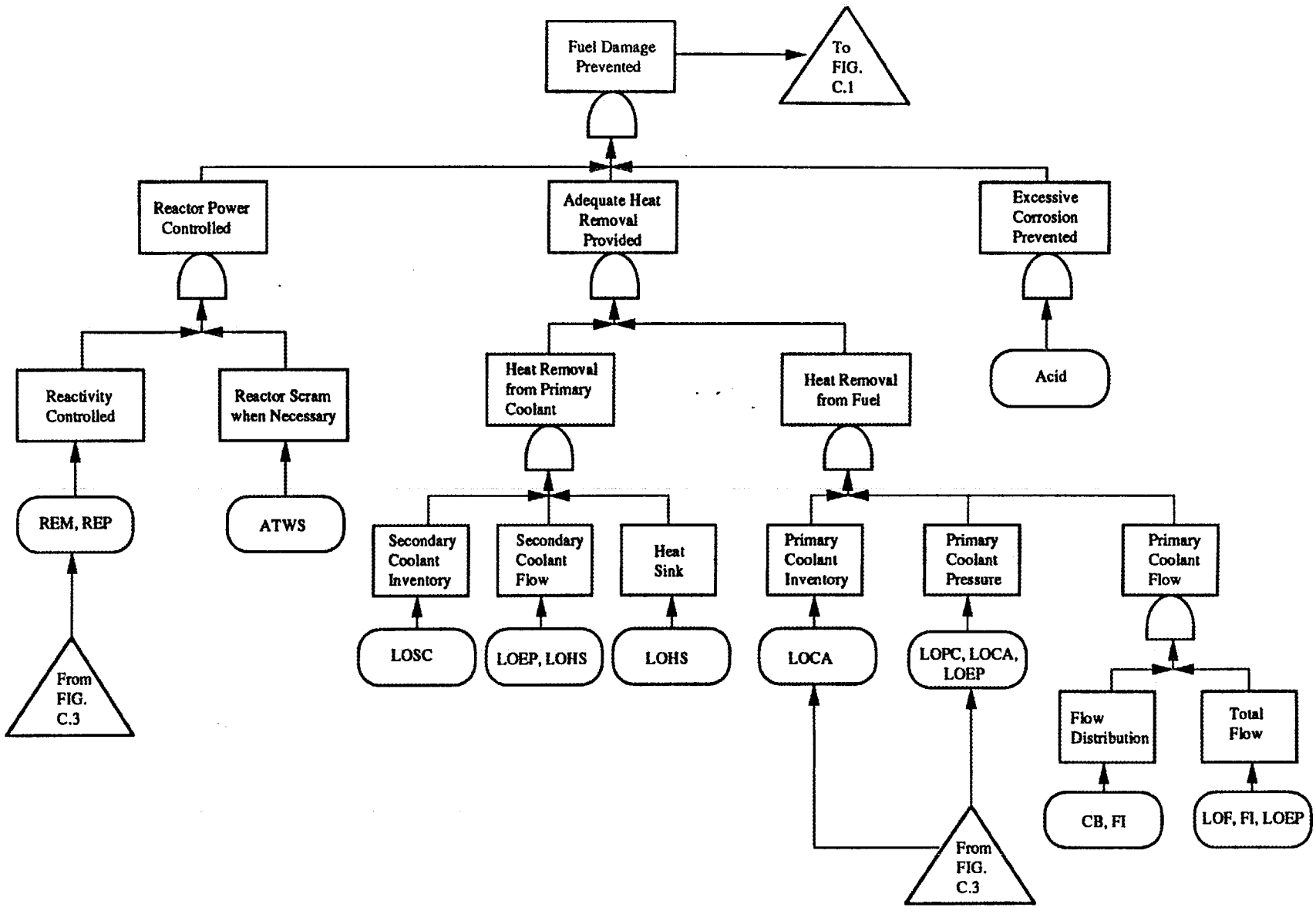


Fig. C.2. ANS design-basis events—prevention of fuel damage.

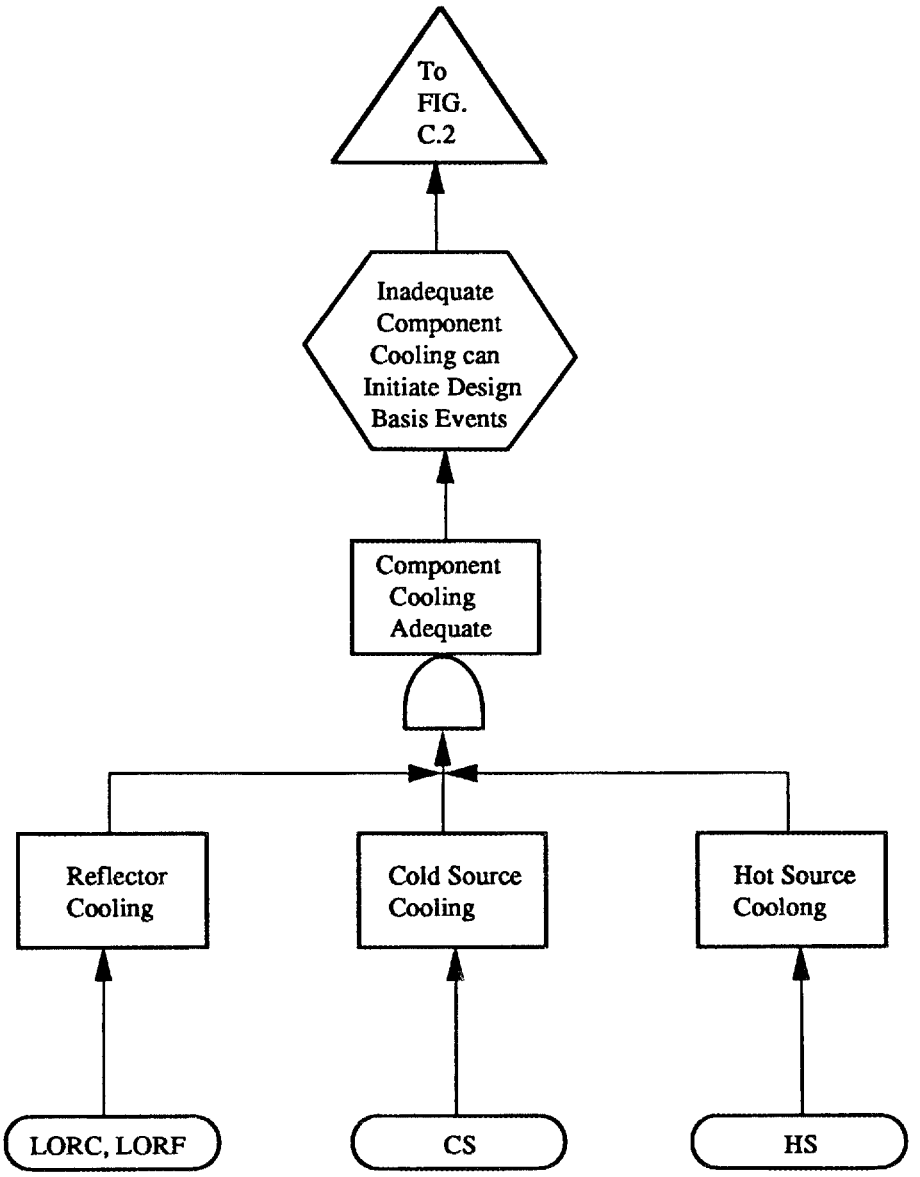


Fig. C.3. ANS design-basis events—component cooling.

1.2 COINCIDENT OCCURRENCES AND SINGLE FAILURES

An adequate and responsible analysis of DBEs shall consider not only the additional failures caused by the initiating event, but also coincident occurrences (COs) and single failures (SFs) that are independent of the initiating event. Dependent failures caused by the initiating event are considered specifically in the analysis of each DBE. Independent failures shall be considered utilizing the generic methodology provided by ANSI/ANS-51.1-1983. The basic approach is that independent failures and occurrences shall be considered in the analysis; but, if they are not very probable, the resultant event may be evaluated under the less restrictive acceptance criteria of a lower frequency event. The specific rules are given in Table C.1. Table C.2 lists specific examples of coincident occurrences and single failures that must be accounted for in analyzing DBEs.

Table C.1. Methodology for determining the plant condition of an event[a]

Initiating occurrence (IO) category	Category assuming coincident occurrence $CO > 10^{-2}/IO^b$	Category assuming $CO \leq 10^{-2}/IO^c$	Category assuming both $CO > 10^{-2}/IO^b$ and single failure	Category assuming both $CO \leq 10^{-2}$ IO^c and SF
Anticipated (A), freq $> 10^{-1}$	A	A	A	U
Anticipated (A)	A	U	U	EU[(d)]
Unlikely (U)	U	EU	EU	EU
Extremely unlikely (EU)	EU	EU	EU[(e)]	EU[(e)]

Legend
IO = initiating occurrence (see Table C.2 for examples of IOs and COs).
CO = coincident occurrence (or combination of coincident occurrences).
SF = single failure.

Notes:
[a]This table is adapted from ANSI/ANS-51.1-1983.
[b]$CO > 10^{-2}/IO$ means that the CO has probability of occurrence, concurrently with the IO, that is greater than 10^{-2}.
[c]$CO \leq 10^{-2}/IO$ means that the CO has probability of occurrence, concurrently with the IO, that is less than or equal to 10^{-2}.
[d]Any credible IO, when combined with an SF, CO, or both, must stay within allowable off-site dose (Table 4.8-1) criteria (except as noted in Sect. 3.2.3 of ANSI/ANS-51.1-1983 for beyond design-basis events).
[e]Some events, if considered on an individual basis as allowed by ANSI/ANS-5, may result in frequencies less than 10^{-6}/year.

Table C.2. Single failures and coincident occurrences for analysis of DBEs and conditions

1. Coincident occurrences, probability of occurring with initiating event > 10^{-2}

- Any one channel of plant or reactor control in manual (holds constant output demand but does not respond to changing conditions).
- All plant and reactor control variables (especially reactor power, inlet temperature and pressure) biased from nominal by expected error (including drift and calibration errors).
- Xenon reactivity feedback lower or higher than nominal by the amount expected as a result of normal startup or by brief operation at reduced power levels (affects assumed initial control rod position).

2. Coincident occurrences, probability of occurring with initiating event < 10^{-2}

- Any one plant control variable (especially reactor power, inlet temperature, and pressure) at the alarm annunciator set point (set point for annunciation to the operator of abnormal condition in need of correction).
- Any system or subsystem inoperative under conditions allowed by the plant's technical specifications; for example:
  - one pony motor subsystem (motor, batteries, or charger)
  - standby pressurizer pump
  - electrical power supply
    - one battery
    - diesel generator
    - others *[TBD]*[C-1]
- Seal leakage (reactor main circulating pump shaft seals, and/or CPBT labyrinth seals) at the maximum rate allowed by technical specifications.
- Heat exchanger tube leakage at the maximum amount allowed by technical specifications.
- Loss of off-site power.

3. Single failures

- All channels of plant or reactor control in manual (holds constant output demand but does not respond to changing conditions). .
- Any one plant or reactor control panel control button, switch, or adjustment manipulated in an inappropriate manner by inadvertent operator action.
- Any one plant or reactor control channel at the reactor scram set point (unless specific single failure analysis shows that this could not occur via a single failure within the control electronics or actuating mechanisms).
- Any one reactor coolant accumulator inoperative (gas space filled with water).

Note: The SFs and COs of this table are events that are independent of the initiating event. Any SFs or COs known to result from the basic initiating event must be included within the definition of the event itself.

1.3 EVENT DESCRIPTIONS

Table C.3 lists the ANS DBEs along with SFs and/or COs that are especially relevant to each event.

Table C.3. ANS design-basis events

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
N-1	A	Fuel loading	RF	
N-2	A	Approach to criticality: reactor depressurized; special low-power RPS set points enabled by mode switch.	SD	
N-3	A	Startup for extended low-power operation at < 10% FP.	SD	
N-4	A	Startup to low power with escalation to full power.	SD	
N-5	A	Controlled shutdown to low power.	FP	
N-6	A	Fast runback to 10% power with hold at 10% power; followed by escalation back to full power.	FP	
N-7	A	Reactor manual or automatic scram.	FP	
N-8	A	Fuel unloading.	RF	
REN-1	A	Partial drop of single outer control rod: reactivity < $0.80 (max. amount that can be compensated under automatic control).	FP	OP[e] operator oversight and intervention (i.e., the operator does not terminate the initiating event or initiate a manual reactor scram).
REN-2	A	Spurious action of one shutdown system.	FP	
REN-3	*[HOLD]*[C-1]	Liquid poison injection: desirability for liquid poison injection system has not been established for the ANS reference design.	N/A	N/A
REN-4	U	Light-water injection into the reflector tank.	FP	
REP-1	A	Start-up accident: reactor initially subcritical prior to withdrawal of inner control rods. Withdrawal interrupted by scram signal.	SD	REGROD[e] normal, nonsafety automatic reactivity control (i.e., no compensatory inner control rod motion) + OP[e].

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
REP-2	A	Spurious withdrawal of inner control rods.	FP	REGROD[e]
REP-3	EU	Outer control rods withdrawn after the inner rods have been fully withdrawn. Scram stops outer rods moving out of core; they reinsert under pull of gravity.	SD	OP[e] + safety grade interlocks that prevent inner rods from withdrawing until outer rods are full out.
REP-4	N/A	Rapid expulsion of control rods is not possible with present reference design.	N/A	N/A
REP-5	A	Single beam tube flooding: step reactivity insertion of $0.17.	FP	REGROD[e]
REP-6	U	Maximum credible cold source reactivity disturbance: reactivity of $0.59 and rate of insertion *[TBD]*[C-2]	FP	REGROD[e]
REP-7	EU	Multiple beam tube flooding or catastrophic cold source destruction: bounded by step reactivity insertion of $0.80 *[HOLD]*[C-2].	FP	
REP-8	U	Light-water injection via the pressurizer pumps; reactivity insertion rate = $Rho_o/[C_o*(V_{pcs}/B_{mu})]$, where Rho_o is the light water reactivity at the peak concentration (C_o), V_{pcs} is the primary coolant system pressurized volume, and B_{mu} is the pressurizer pump (make-up) flow.	FP	REGROD[e] + is included in OP[e]
REP-9	EU	D_2O/H_2O interface reaches core following start of H_2O-contaminated spare primary cooling loop.	0.5 FP	
LOPC-1	A	One letdown valve spuriously strokes fully open.	FP	NAPC: Normal automatic, nonsafety pressure control (i.e., the other letdown valves do not stroke closed to compensate for the failure), plus failure of the downstream letdown block valve to close (LDBV).

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
LOPC-2	A	All letdown valves go fully open.	FP	OP[c] + LDBV[c] [Note: If a high flow CPBT seal is employed, it will be necessary to assume a failure of one primary coolant pump low pressure trip (PCPLPT[c]) for this event.]
LOPC-3	A	All operating pressurizer pumps shut down.	FP	OP[c] + (NAPC[c] or LDBV[c]), and possibly PCPLPT[c] (see note with LOPC-2).
LOPC-4	U	Overpressure relief valve fails to open after having been actuated (perhaps by LOPC-5 or -6).	FP	OP[c] + NAPC[c]
LOPC-5	A	One letdown valve spuriously strokes closed.	FP	OP[c] + NAPC[c]
LOPC-6	A	All letdown valves spuriously stroke closed (possibly by failure of pressure control); no reduction in pressurizer pump speed.	FP	OP[c] + one primary coolant pressure relief device fails to open if challenged (RELIEF[c]).
LOPC-7	A	Inadvertent start of one or more pressurizer pumps.	FP	OP[c] + NAPC[c]
LOPC-8	A	Pressurizer pump runaway to *[TBD]*%[C-3] overspeed.	FP	(NAPC[c] or spurious closure of all letdown valves or failure of pressurizer pump excess discharge pressure trip) + RELIEF[c]
FI-1	A	One primary coolant pump started spuriously.	FP	OP[c]
FI-2	U	One of the internal structures that limits core bypass flow undergoes structural failure: geometry and severity *[TBD]*[C-4]. Duration: one entire fuel cycle.	FP	N/A, unless there are consequent failures due to additional bypass flow.
FI-3	U	Inadvertent start of one primary coolant pump during refueling.	RF	
LOF-1	A	Single primary coolant main ac motor trips, pump coasts down to pony motor speed. Reactor power ramped down to 50%. Loop with bad motor is isolated, spare primary coolant loop brought on, and power escalated back to FP.	FP	N/A: This is an operability event.

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
LOF-2	A	All primary coolant pumps coast down to pony motor speed.	FP	Failure of normal automatic runback to ~ 10% flow + loss of a single primary coolant loop by failure of pony motor or spurious closure of loop isolation valve.
LOF-3	U	Single primary pump shaft breaks and causes very rapid decrease in primary coolant flow.	FP	Failure of automatic reactor power adjustment to the lower flow.
LOF-4	EU	All primary coolant pump motors coast down to zero rpm (failure of all pony motors).	FP	
LOF-5	A	Single primary coolant loop isolation valve closes.	FP	Failure of automatic power adjustment to the lower flow.
LOF-6	U	Flow strainer in one loop totally blocked over period of *[TBD]*[C-5]. The blockage occurs instantly.	FP	Failure of automatic power adjustment to the lower flow.
LOF-7	U	Two isolation valves stroke closed spuriously.	FP	Failure of automatic power adjustment to the lower flow.
LORF-1	A	Reflector coolant pumps coast down to pony flow.	FP	Failure of a single reflector coolant pump pony motor.
LORF-2	A	One reflector coolant loop isolated.	FP	Failure of a single reflector coolant pump pony motor in an unaffected loop.
CB-1	A	Foreign object in coolant at core inlet. Size *[TBD]*[C-6].	FP	
CB-2	U	Experiment or transuranic target structural failure in such a manner as to perturb flow at core inlet: geometry *[TBD]*[C-7].	FP	
CB-3	U	Core inlet strainer structural failure: mode *[TBD]*[C-8] and geometry *[TBD]*[C-9].	FP	
CB-4	EU	Major core inlet flow blockage: complete inlet blockage of one fuel element.	FP	
CB-5	U	Blockage of CPBT cooling passage: extent and location *[TBD]*[C-10].	FP	
CB-6	U	Blockage of inner control rod cooling passage: extent and location *[TBD]*[C-11].	FP	

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
LOHS-1	A	One secondary coolant isolation valve closes or one secondary coolant pump trips, causing loss of all forced flow in one loop.	FP	OP[c] + (loss of an additional secondary coolant loop or failure of automatic reactor power reduction).
LOHS-2	U	Forced flow through all secondary coolant loops is lost for up to 72 hours.	FP	RELIEF[c]
ACID-1	A	High pD due to loss of DNO_3 addition.	FP	OP[c] (no operator intervention for *[TBD]*[C-12] minutes after first alarm annunciation).
ACID-2	A	Low pD due to excessive DNO_3 addition.	FP	No operator intervention for *[TBD]*[C-13] minutes after the first alarm annunciation.
LOC-1		Loss of coolant accident (LOCA) involving loss of coolant from reactor to reflector coolant; frequency category for any break location depends on the break size (categories defined below).	FP	Depends on size (e.g., LBLOC-1 requires assumption of failure of *[TBD]*[C-14] outer rods to insert upon scram, due to postulated outer CPBT deformation).
LOC-2		LOCA involving loss of coolant from reactor to reactor pool.	FP	
LOC-3		LOCA involving loss of coolant from reactor to water cell.	FP	
LOC-4		LOCA involving loss of coolant from reactor to limited-volume air cell.	FP	
LOC-5		LOCA involving loss of coolant from reactor to elevated air cell (i.e., a break associated with the accumulator or its piping).	FP	
LOC-6		LOCA involving loss of coolant from reactor to secondary coolant via a main heat exchanger tube break.	FP	
LOC-7		LOCA involving loss of coolant from reactor to pool water via an emergency heat exchanger tube break.	FP	
LOC-8		LOCA involving loss of coolant from reactor to subpile room limited-volume enclosure.	FP	Leak in subpile room limited-volume enclosure.
LOC-9		LOCA involving loss of gas from the accumulator gas space.	FP	

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
SBLOC-i	A	Small break: initial leak rate less than normal letdown flow; depressurization insufficient to cause immediate reactor scram. Locations 1 through 9, as specified above.	FP	(Loss of off-site power) + (failure of one pressurizer pump or one pony motor).
MBLOC-i	U	Medium break: initial leak rate greater than normal letdown flow; prompt low pressure scram likely, but pressure remains high enough for continued operation of one primary coolant pump on its ac main motor.	FP	Failure of one primary coolant ac main motor to trip, or failure of one pressurizer pump, or failure of one pony motor, or loss of off-site power.
LBLOC-i	U-EU	Large break: initial leak rate so large that depressurization is essentially immediate (note: frequency category depends on location of break). For double-ended EU breaks in stainless steel, it is assumed that a period of 1000 ms is required for the initiating through-wall leak to become a double-ended break.	FP	None for EU large breaks. See LOC-1, above, for outer CPBT break. Also, any U breaks must assume one of the three accumulators are water-filled (missing its normal gas charge) prior to the break or a loss of off-site power.
LORC-1		Loss of reflector coolant from the reflector to a beam tube or to a cold source thimble; frequency depends on break size (categories defined below).	FP	Depends on size (see SBLORC-i, below).
LORC-2		Loss of reflector coolant from the reflector to the reactor pool.	FP	
LORC-3		Loss of reflector coolant to a water or air cell.	FP	
LORC-4		Loss of reflector coolant because of a heat exchanger break.	FP	
SB-LORC-i	A	Loss of reflector coolant at a rate insufficient to cause immediate degradation of safety-related reflector cooling functions; locations specified above.	FP	Loss of off-site power and failure of one make-up pump or one reflector coolant pump motor.
LB-LORC-i	U	Loss of reflector coolant at a rate sufficient to cause immediate degradation or loss of safety-related reflector cooling functions.	FP	Failure of one make-up pump or one reflector coolant pump motor.

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
LOSC-1		Loss of secondary coolant inside the reactor support building; frequency category depends on break size (categories defined below).	FP	Depends on break size and effects of secondary coolant flooding.
LOSC-2		Loss of secondary coolant inside the pipe chase.	FP	
LOSC-3		Loss of secondary coolant at the cooling tower basin, on the pump suction side.	FP	
LOSC-4		Loss of secondary coolant at the cooling tower or basin.	FP	
LOSC-5		Loss of secondary coolant inside the reactor building.	FP	
SB-LOSC-i	A	Small break: loss of secondary coolant at a rate insufficient to cause immediate degradation of heat removal from the primary coolant.	FP	Failures caused by flooding, if any, plus failure of one monitoring instrument channel and one secondary coolant pump or valve or loss of off-site power.
LB-LOSC-i	U	Large break: loss of secondary coolant with immediate degradation of heat removal from the primary coolant.	FP	Failures caused by flooding, if any, plus loss of off-site power.
NCG-1	A	Total of 4.7 m^3 (166 ft^3) @ STP of dissolved gases released by primary coolant as a result of depressurization: 1 m^3 (35 ft^3) of H_2 plus 3.7 m^3 (131 ft^3) of O_2.	FP	Failure of one letdown valve.
NCG-2	U	Failure of a gas cooled irradiation experiment in such a manner that 5.7 m^3 (201 ft^3) @ STP of inert gas escapes into primary coolant system.	FP	Loss of primary coolant pressure.
NCG-3	U	Accumulator excess gas supply: 5 m^3 (177 ft^3) @ STP *[HOLD]*[C-3].	SD/LP/FP	OSI[e]
ATWS-1	EU	Loss of off-site power with failure of inner rods to insert or scram.	FP	N/A
ATWS-2	EU	REP-5 with failure of inner control rods to insert or scram.	FP	N/A

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
FHA-1	EU	Fuel handling accident: single 24-h old spent-fuel element dropped during handling in D_2O. Criticality control is maintained, but the element is assumed to land on its side, compromising coolability. Presumed 100% fuel damage of the dropped element.	RF	*[TBD]*[C-15]
FHA-2	EU	Fuel handling accident: single fresh-fuel element dropped during handling in D_2O, becoming separated from neutron poison elements. Criticality control not maintained. Presumed prompt criticality terminated by energetic disassembly of the superheated element with consequent damage to primary and secondary containment boundaries.	RF	
FHA-3	*[TBD]*[C-16]	Fuel handling accident: single spent-fuel element dropped inside primary containment after transfer out of D_2O pool. Effect on decay heat removal and extent of fuel damage *[TBD]*[C-17].	RF	*[TBD]*[C-16]
FHA-4	*[TBD]*[C-18]	Fuel handling accident: single spent-fuel element dropped during handling outside primary containment. Effect on decay heat removal and extent of fuel damage *[TBD]*[C-19].	RF	*[TBD]*[C-18]
LOEP-1	A	Loss of off-site power.	FP	Failure of one primary coolant pony motor + either of the following coincident occurrences: reactor power runback does not occur or primary coolant pressure is lost.
LOEP-1.U1	U	Loss of off-site power with reactor operating on two loops.	0.5 FP	Pre-existing pump failure or tube leak has caused isolation of one coolant loop with reduction to 0.5 FP. After the LOEP, another pony fails, leaving only one loop running.

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
LOEP-2	U	Station blackout: complete loss of off-site and on-site ac power for *[TBD]*[C-20] hours.	FP	Failure of one primary coolant pony motor.
LOEP-3	U	Loss of all non-1E power.	FP	
FD-1	U	An event occurs at the end of a fuel cycle and causes 1% (by volume of the fuel) to exceed the melting (liquidus) point of aluminum 6061, causing the following fractions of fission products within the melted fuel to be released to the primary coolant: 100% of noble gases, 50% of halogens, and 1% of all others.	FP	Coincident occurrences in coolant cleanup or containment or containment isolation must be considered (per Table C.1).
FD-2	EU	Definition same as for FD-1, except that the extent of fuel damage is 100%.	FP	Same as for FD-1.
CS-1	A	Loss of cold source cooling: the helium-based cooling system fails, or the D_2 pump stops pumping.	FP	
CS-2	U	Fracture of any one cold source pressure boundary. Possible consequent failures of other boundaries must be considered.	FP	
CS-3	EU	Cold source pressure boundary rupture or other major failure results in depressurization of the D_2 followed by in-leakage of air until a stoichiometric concentration is reached, and the D_2/air mixture undergoes rapid oxidation.	FP	
CS-4	EU	Cold source pressure FP boundary rupture or other major failure results in depressurization of the D_2 and accumulation of the D_2 in a local area of the primary containment building. Rapid oxidation of the D_2/air mixture follows.		
HS-1	A	Malfunction or failure of the system that maintains the desired hot source temperature.	FP	

Table C.3 (continued)

Event and frequency category[a]		Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
HS-2	U	Fracture of any one hot source pressure boundary. Possible consequent failures of other boundaries must be considered.	FP	
TRU-1	A	Pinhole leak in any one transuranic target.	FP	
TRU-2	U	Pinhole leak in any one transuranic target with water logging.	FP	
TRU-3	U	Major perforation in target rod cladding leak diameter > 1 mm (.04 in).	FP	
TRU-4	U	Structural failure of target or mounting hardware.	FP	
TRU-5	U	Loading error: manufacturing error (excessive fissile transuranic): max error *[TBD]*[C-21].	FP	
IRR-1	A	Interruption of cooling gas flow to an irradiation capsule.	FP	
IRR-2	A	Loss of the experiment's primary containment boundary.	FP	
IRR-3	U	Loss of experiment's primary and secondary boundary integrity.	FP	
IRR-4	U	Major structural failure of an experiment.	FP	
RR-1	U	Major structural failure in radioactive liquid waste system.	N/A	
RR-2	U	*[TBD]*[C-22] major structural failure in gaseous waste system.		
RR-3	EU	A beam tube ruptures at the end nearest the reactor. D_2O rapidly flows into the beam tube, impacting, but not breaking, the beam window (a sealed, hemispherical aluminum cap).	N/A	

Table C.3 (continued)

Event and frequency category[a]	Description	Initial condition[b]	Typical SFs and/or COs for safety analysis[c]
OTHER INTERNAL EVENTS			
Fires	*[TBD]*[C-23]		
Equipment	*[TBD]*[C-24]		
Flooding	*[TBD]*[C-25]		
Pools			
Water cells			
Secondary coolant			
Heavy object drop	*[TBD]*[C-26]		
EXTERNAL EVENTS			
Tornado			
Seismic	*[TBD]*[C-27]		
OBE	Anticipated		
SSE	Unlikely/Extremely unlikely		
Floods	Unlikely		
HFIR or transuranic facility accidents			

[a]N > 1/y
A > .01/y
U > 10^{-4}/y
EU > 10^{-6}/y

[b]FP=full power, LP=low power, SD=shutdown, RF=refueling

[c]OP=failure of normal operator oversight and intervention
REGROD=failure of normal, nonsafety automatic reactivity control
NAPC=failure of normal automatic pressure control
LDBV=failure of letdown block valve to close
PCPLPT=failure of primary coolant pump to trip on low pressure
RELIEF=failure of a pressure relief valve to open

2. PART II: BEYOND-DESIGN-BASIS (BDB) EVENTS

2.1 INTRODUCTION

This section defines two scenarios that exhibit severe accident phenomena and containment loads that may be possible for a wide range of postulated severe accidents. Derivation of the two scenarios, designated BDB-1 and BDB-2, is explained below. The basic purpose of defining BDB scenarios is to facilitate the assessment of severe accident mitigation capabilities of the design by providing specific information about severe accidents. These assessments utilizing the BDB accidents might also point out additional features that would further enhance the severe accident mitigation capability of the design. Such additional features would be considered on a case-by-case basis for inclusion in the design.

The accident sequences defined by this section are described as BDB because the expected frequency of either of the sequences is less than 10^{-6}/year, the lower limit for DBEs per ANSI/ANS-51.1-1983. This frequency is about one order of magnitude lower than the 10^{-5}/year ANS goal for limiting the total frequency of core damage. The reason for this lower frequency for the BDB (i.e., 10^{-6}/year vs 10^{-5}/year) is that the outcomes of all severe accident phenomena have been resolved on the very conservative side throughout the development of the sequences. Phenomena resolved in such a bounding include: (1) energetic steam explosion, (2) immediate escape to containment atmosphere of noble gas and halogen fission products, (3) melt-through of debris from primary coolant system, and (4) 100% efficient transport of molten debris to the subpile room without significant escape of primary coolant to the subpile room. The resolution of all these phenomena concomitantly in the conservative direction is sufficient to ensure that the net frequencies of the sequences developed below are in the BDB range.

2.2 DERIVATION OF THE BDB ACCIDENTS

A comprehensive approach is taken to ensure that the selected BDB accidents cover a wide range of possible sequences. A debris cooling event tree (Fig. C.4) was constructed to help in classifying severe accident sequences. It follows a severe accident from the initial severe fuel damage to a stable, or quasi-stable, end state. The tree has six branches to account for significant events: steam explosion, pool integrity, debris coolability in the primary coolant system or in the reflector vessel, and debris and water transport to the subpile room. The overall severity of each branch depends on how many of the events are resolved in the direction of increasing challenge to containment integrity or enhanced release of radioactivity. Table C.4 lists the barriers against release of radioactivity that are breached by each branch.

The method for establishing the BDB events is to group the event tree branches by severity level or similarity of phenomena involved. A DBE or BDB event is then defined in such a manner that it bounds the consequences of the two or more sequences that it represents. A bounding scenario is indicated in Table C.4 for each branch of the severe accident event tree.

Branches 1, 3, and 4 of the Fig. C.4 event tree are relatively benign. It is a virtual certainty that conditions (the radiological release and containment loads) represented by these sequences are bounded by an existing design basis accident for fuel damage, designated FD-2, the source term for which is specified by Table C.3 of this appendix.

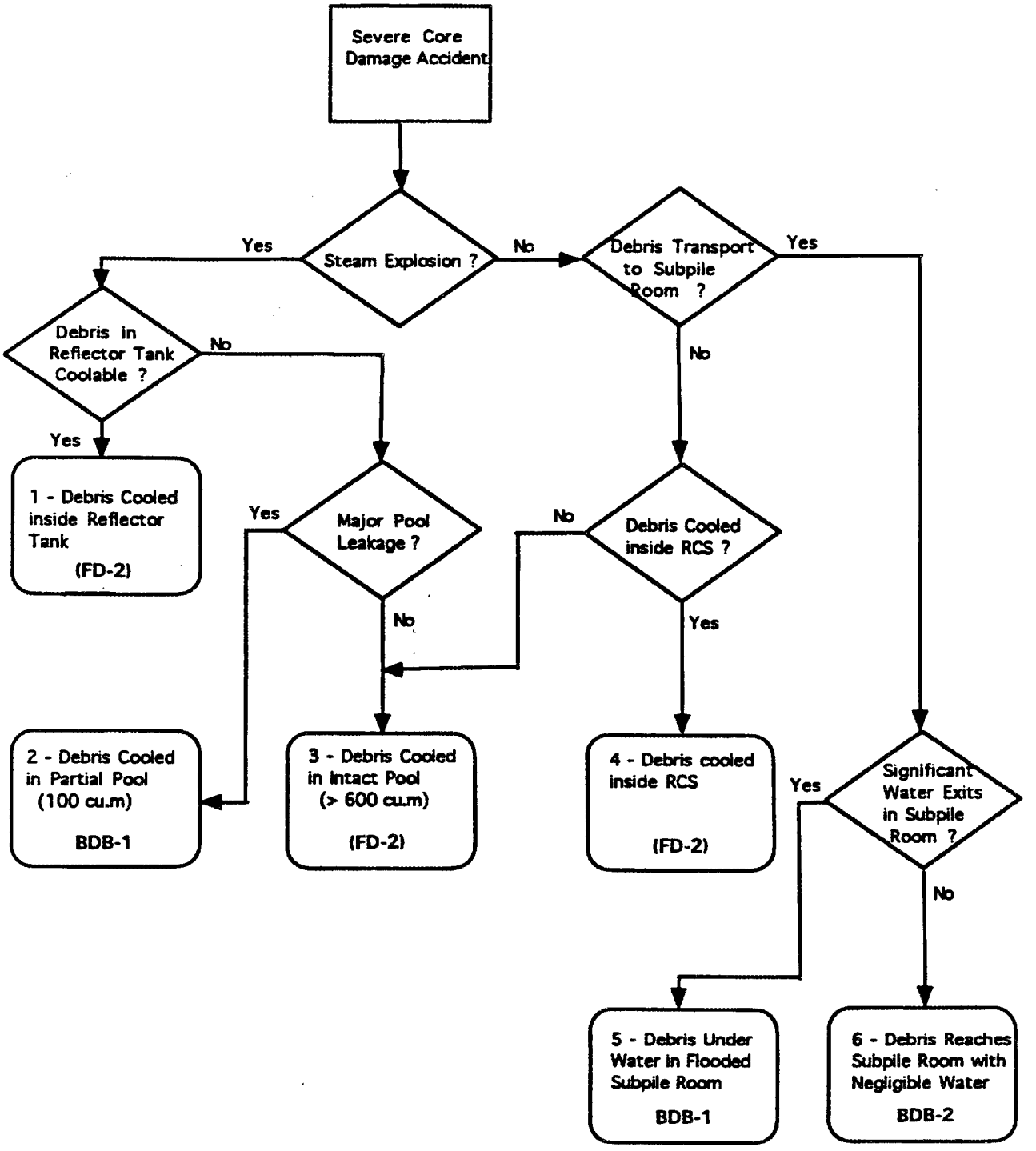


Fig. C.4. Severe accident debris cooling event tree.

Table C.4. Bounding evaluation scenarios—design basis and beyond

Event tree branch[a]	Barriers compromised	Bounding scenario
1	Fuel	FD-2[b]
2	Fuel + coolant system + part of pool	BDB-1
3	Fuel + coolant system	FD-2[b]
4	Fuel	FD-2[b]
5	fuel + coolant system + part of pool	BDB-1
6	Fuel + coolant system + all of pool	BDB-2

[a]Refer to Fig. C.4 for severe accident debris cooling event tree.
[b]FD-2 is a DBE; its source term is defined in Table C.3 of this appendix. BDB-1 and -2 are BDB accidents that are defined in this section.

Branches 2, 5, and 6 of the Fig. C.4 event tree are relatively severe. Branches 2 and 5 result in debris cooled in a limited pool of water and, thereby, are sufficiently similar to be grouped into one composite sequence that is designated BDB-1. Branch 6, which involves debris reaching the subpile room floor without a significant amount of water (i.e., in an essentially dry state), is designated as BDB-2.

The justification for classifying three of the event tree sequences as design basis and classifying the other three as BDB is the higher degree of improbability associated with the more severe events. The reason for studying these sequences is to satisfy the defense-in-depth principle.

2.3 DEFINITION OF BDB ACCIDENT SCENARIOS

The specific definition of the two BDB accidents, BDB-1 and -2, is given in Tables C.5 and C.6. The guiding principle behind the formulation of these scenarios has been to err on the conservative side. For each base scenario, there are three variations that reflect various containment failure modes, designated with a subscript.

2.4 USE OF THE BDB

The basic purpose of BDBs is to provide sequences that can be used to demonstrate defense-in-depth against the release of radioactivity. Table C.7 lists specific containment design features that may contribute to or emphasize defense-in-depth.

Table C.5. Definition of BDB-1

Sequence designators	BDB-1, BDB-1_{cf1}, BDB-1_{cf2}, BDB-1_{cf3}
Scenario definition	A severe fuel damage accident has occurred at a sufficiently high power level to result in a violent molten FCI that has ruptured the CPBT and a sufficient number of beam tubes to result in the rapid draining of the reactor pool to the level of the beam tubes. The amount of water left in the pool is 100 m^3. For the base sequence, containment isolation occurs. For sequence BDB-1_{cf1} there is a primary containment failure (i.e., a greater than design basis leak between primary and secondary containment). For sequence BDB-1_{cf2}, there is, in addition to the primary containment failure, a failure of secondary containment blowers and/or leak rate. Sequence BDB-1_{cf3} represents a primary failure that results in greater than design-basis leakage direct to the environment.
Fission product distribution	The fission products are distributed widely across the reactor pool area so that the debris temperature does not become high enough to cause concrete degradation. The explosion may have driven some of the debris upward in such a manner that there are fission products in the reactor pool above the water level. These fission products will volatilize much sooner and more readily than those under water.
Source term to containment atmosphere	Based on the above scenario, it is assumed that all of the noble gas fission products and 50% of the volatile halogen group escape to containment atmosphere (high-bay area) immediately after the explosion. The balance of the fission products stays behind in the water. As the water heats up, some of the more volatile of these fission products escape from the water into the containment atmosphere.

Table C.6. Definition of BDB-2

Sequence designators	BDB-2, BDB-2_{cf1}, BDB-2_{cf2}, BDB-2_{cf3}
Scenario definition	A severe fuel damage accident has occurred at a sufficiently low power level that there is no violent molten FCI. The molten debris coalesces in the bottom of the CPBT, where it reaches temperatures high enough to degrade stainless steel. Under the influence of gravity, the molten debris sinks downward toward the subpile room. After a brief time, the debris melts through the bottom of the CPBT boundary and falls into the subpile room. Prior to melt-through, a solid crust formed on the debris mass, and this crust prevents any significant amount of water from the primary coolant system from reaching the subpile room. For the base sequence, containment isolation occurs. For sequence BDB-2_{cf1}, there is a primary containment failure: a greater than design basis leak between primary and secondary containment. For sequence BDB-2_{cf2}, there is, in addition to the primary containment failure, a failure of secondary containment blowers and/or leak rate. Sequence BDB-2_{cf3} represents a primary containment failure that results in leakage directly into the environment (bypassing secondary containment).
Fission product distribution	It is assumed that all the fission products remain in the fuel debris until the molten debris mass falls onto the subpile room floor. This assumption maximizes the opportunity for fission products to escape from the primary coolant system. Without much water cover and with the debris in a small pile, the debris reaches temperatures high enough to degrade the concrete floor while driving volatile fission products to escape into the subpile room atmosphere. The fission products that stay behind continue to degrade concrete, leading to the release of CO_2, CO, and H_2 as well as water vapor. Flammable atmospheres are reached in the subpile room.
Source term to containment atmosphere	Based on the above scenario, it is conservatively assumed that all of the noble gas fission products and a sufficient percentage of the volatile halogen, cesium, and tellurium groups escape into containment atmosphere (high-bay area). Half of the fission product heat immediately after the shutdown is released into the containment atmosphere, and the other half, into the debris. The balance of the fission products stays behind in the debris, keeping it hot enough to degrade concrete.

Table C.7. Mitigative features to be evaluated for BDB severe accidents

Sequence	Additional barrier failures	Features
BDB-1	N/A: basic sequence (primary containment leakage < 0.5%/day, secondary containment leakage < 10%/day with > 99% efficient off-gas filtration).	69 kPa above atmospheric (10 psig) primary containment pressure loading capability; missile shield to ensure protection of primary containment; capability to replenish reactor pool from outside containment.
$BDB\text{-}1_{cf1}$	> 0.5%/day leakage from primary to secondary containment.	Secondary containment blower capacity; allowable filter heat loading.
$BDB\text{-}1_{cf2}$	Secondary containment: various failure modes.	Same as BDB-1; plus, secondary containment filter bypass prevention.
$BDB\text{-}1_{cf3}$	Leakage direct to environment.	Back-draft prevention in ventilation supply ducts, ventilation system intake, and/or exhaust filtration.
BDB-2	N/A: basic sequence (primary containment leakage < 0.5%/day, secondary containment leakage < 10%/day with > 99% efficient off-gas filtration.)	Same as BDB-1 (except for missile shield); plus, flooding system and refractory concrete floor lining for the subpile room.
$BDB\text{-}2_{cf1}$	see $BDB\text{-}1_{cf1}$	see $BDB\text{-}1_{cf1}$
$BDB\text{-}2_{cf2}$	see $BDB\text{-}1_{cf2}$	see $BDB\text{-}1_{cf2}$
$BDB\text{-}2_{cf3}$	see $BDB\text{-}1_{cf3}$	see $BDB\text{-}1_{cf3}$

Appendix D:

HOLD/TBD RESPONSIBILITY-STATUS LIST

Appendix D

HOLD/TBD RESPONSIBILITY-STATUS LIST

The following procedure is to be followed for establishing and controlling *[HOLDS]*s and *[TBD]*s for the ANS PDR and SDDs.

When either a *[HOLD]* or a *[TBD]* is identified for an entity (parameter, table, appendix, etc.), it shall be tagged with a superscript with two digits (the first referring to the chapter number within the document and the second referring to the sequence within the designated chapter) and entered into the corresponding table within the appropriate appendix of the document. When a [HOLD] or a [TBD] is resolved, the information is to be added or corrected, the number is to be removed within the document, and the [HOLD] and [TBD] tables are to be updated with the actual date of resolution recorded in the table.

TBD responsibility-status list

TBD number	PDR section	Description	Responsible organization	Responsible document	Responsible person or title	Date or project phase for resolution, S-Scheduled A-Actual
TBD 2-1	2.2 (Table 2.1-1)	Conceptual design value for damage rate of neutrons in the central irradiation facility.	ANS Project	PDR	Tom McManamy	Title I (S)
TBD 2-2	2.2 (Table 2.1-1)	Conceptual design value for damage rate of neutrons in the reflector vessel facility.	ANS Project	PDR	Tom McManamy	Title I (S)
TBD 2-3	2.2 (Table 2.1-1)	Conceptual design value for the peak heat flux in the isotope production facility.	ANS Project	PDR	Tom McManamy	Title I (S)
TBD 2-4	2.2 (Table 2.1-1)	Conceptual design value for the temperature in the polyethylene rabbit tube.	ANS Project	PDR	Tom McManamy	Title I (S)
TBD 4-1	4.8.1.1	Fuel integrity limits for design basis events associated with the effects of corrosion on the outcome of unlikely events.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-2	4.8.1.1	Additional acceptance criteria for design basis events associated with structural deflection of the fuel plates.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-3	4.8.2 (Table 4.8-2)	Determining the frequency category for reactivity events caused by liquid poison injection accidents.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-4	4.8.2 (Table 4.8-2)	Determining the frequency category for reactivity events caused by rapid expulsion of a single control rod.	ANS Project	PDR	Mike Harrington	Title I (S)

TBD responsibility-status list (continued)

TBD 4-5	4.8.2 (Table 4.8-2)	Determining the frequency category for a spent fuel element dropped inside containment.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-6	4.8.2 (Table 4.8-2)	Determining the frequency category for spent fuel element dropped outside containment.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-7	4.8.2 (Table 4.8-2)	Determining the frequency category for loss of spent fuel cooling.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-8	4.8.2 (Table 4.8-2)	Determining the frequency category for a fuel element being stuck during transport.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-9	4.8.2 (Table 4.8-2)	Completing the list and determining the frequency categories for "other internal events."	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 4-10	4.8.2 (Table 4.8-2)	Determining the frequency category for tornados.	ANS Project	PDR	Mike Harrington	Title I (S)
TBD 5-1	5.1	Determining the ANS global availability including major maintenance.	ANS Project	PDR	ANS PRC	Advanced Conceptual Design
TBD 5-2	5.2	Determining the percent of the scheduled operating time that the ANS shall be at full power operation.	ANS Project	PDR	ANS PRC	Advanced Conceptual Design
TBD 6-1	6.2.1	Incorporate this section into the PDR.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD 6-2	6.2.2	Incorporate this section into the PDR.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD B-1 through B-7	Appendix B, Table B.1	Provide guidance for the application of several DOE orders.	ANS Project	PDR	Mike Muhlheim	Advanced Conceptual Design
TBD B-8	Appendix B, Table B.1	Determine emergency facility requirements and communications requirements to meet DOE Order 5500.2.	ANS Project	PDR	Mike Muhlheim	Advanced Conceptual Design
TBD B-9 through B-11	Appendix B, Table B.1	Provide guidance for the application of several DOE orders.	ANS Project	PDR	Mike Muhlheim	Advanced Conceptual Design
TBD B-12 through B-84	Appendix B, Table B.4	Determine the compliance commitment for various sections of 10 CFR 50.	ANS Project	PDR	Mike Muhlheim	Advanced Conceptual Design
TBD B-85	Appendix B, Table B.5	Determine the compliance commitment for GDC 27.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design

TBD responsibility-status list (continued)

TBD B-86	Appendix B, Table B.5	Determine if liquid poison addition will be needed for nuclear criticality control in the reactor and spent fuel storage system during accident conditions.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD B-87	Appendix B, Table B.6	Determine the maximum noise level in the experimental areas inside containment.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD B-88	Appendix B, Table B.6	Determine the maximum noise level in office.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD B-89	Appendix B, Table B.7	Decide what parameters should be monitored to satisfy RG 1.97.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD B-90	Appendix B, Table B.8	Decide on the compliance commitment for SRP 10.4.9.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD B-91	Appendix B, Table B.10	Decide how to address hazardous materials not covered under Martin Marietta Policy Procedure ESH-16.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
TBD C-1	Appendix C, Table C.2	Complete the list of coincident occurrences under item 2 in the table.	ANS Project	PDR	Mike Harrington	Title I
TBD C-2	Appendix C, Table C.3	Define the rate of reactivity insertion for the design-basis accident REP-6.	ANS Project	PDR	Mike Harrington	Title I
TBD C-3	Appendix C, Table C.3	Define the pump runaway speed for design-basis accident LOPC-8 REP-6.	ANS Project	PDR	Mike Harrington	Title I
TBD C-4	Appendix C, Table C.3	Define the geometry and severity of the structural failure for design-basis accident FI-2.	ANS Project	PDR	Mike Harrington	Title I
TBD C-5	Appendix C, Table C.3	Define the time period for blockage of the flow strainer for design-basis accident LOF-6.	ANS Project	PDR	Mike Harrington	Title I
TBD C-6	Appendix C, Table C.3	Define the size of the foreign object in the primary coolant for design-basis accident CB-1.	ANS Project	PDR	Mike Harrington	Title I
TBD C-7	Appendix C, Table C.3	Define the flow blockage for design-basis accident CB-2.	ANS Project	PDR	Mike Harrington	Title I
TBD C-8	Appendix C, Table C.3	Define the mode of structural failure for design-basis accident CB-3.	ANS Project	PDR	Mike Harrington	Title I
TBD C-9	Appendix C, Table C.3	Define the geometry associated with design-basis accident CB-3.	ANS Project	PDR	Mike Harrington	Title I

TBD responsibility-status list (continued)

TBD C-10	Appendix C, Table C.3	Define the extent and location for the cooling blockage for design-basis accident CB-5.	ANS Project	PDR	Mike Harrington	Title I
TBD C-11	Appendix C, Table C.3	Define the extent and location for the coolant blockage for design-basis accident CB-6.	ANS Project	PDR	Mike Harrington	Title I
TBD C-12	Appendix C, Table C.3	Define the time for no operator intervention for design-basis accident ACID-1.	ANS Project	PDR	Mike Harrington	Title I
TBD C-13	Appendix C, Table C.3	Define the time for no operator intervention for design-basis accident ACID-2.	ANS Project	PDR	Mike Harrington	Title I
TBD C-14	Appendix C, Table C.3	Define the number of outer rods that fail in design-basis accident LOC-1.	ANS Project	PDR	Mike Harrington	Title I
TBD C-15	Appendix C, Table C.3	Define the failure mode for design-basis accident FHA-1.	ANS Project	PDR	Mike Harrington	Title I
TBD C-16	Appendix C, Table C.3	Define the frequency category and failure mode for design-basis accident FHA-3.	ANS Project	PDR	Mike Harrington	Title I
TBD C-17	Appendix C, Table C.3	Define the extent of fuel damage for design-basis accident FHA-3.	ANS Project	PDR	Mike Harrington	Title I
TBD C-18	Appendix C, Table C.3	Define the frequency category and failure mode for design-basis accident FHA-4.	ANS Project	PDR	Mike Harrington	Title I
TBD C-19	Appendix C, Table C.3	Define the extent of fuel damage for design-basis accident FHA-4.	ANS Project	PDR	Mike Harrington	Title I
TBD C-20	Appendix C, Table C.3	Define the duration of station blackout design-basis accident LOEP-2.	ANS Project	PDR	Mike Harrington	Title I
TBD C-21	Appendix C, Table C.3	Define the maximum fissile loading error for design-basis accident TRU-5.	ANS Project	PDR	Mike Harrington	Title I
TBD C-22	Appendix C, Table C.3	Define the major structural failure for design-basis accident RR-2.	ANS Project	PDR	Mike Harrington	Title I
TBD C-23	Appendix C, Table C.3	Define the design-basis accidents due to fires.	ANS Project	PDR	Mike Harrington	Title I
TBD C-24	Appendix C, Table C.3	Define the design-basis accidents due to other equipment failures.	ANS Project	PDR	Mike Harrington	Title I
TBD C-25	Appendix C, Table C.3	Define the design-basis accidents due to flooding.	ANS Project	PDR	Mike Harrington	Title I
TBD C-26	Appendix C, Table C.3	Define the design-basis accidents due to heavy objects.	ANS Project	PDR	Mike Harrington	Title I
TBD C-27	Appendix C, Table C.3	Define the design-basis accidents due to seismic events.	ANS Project	PDR	Mike Harrington	Title I

Hold responsibility-status list

HOLD number	PDR section	Description	Responsible organization	Responsible document	Responsible person or title	Date or project phase for resolution, S-Scheduled A-Actual
HOLD 4-1	4.1.3	Size of piping for the design basis pipe break	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 4-2	4.1.6	The percent of iodine fission product escaping from the fuel to the containment atmosphere for postulated accidents dealing with the site suitability source term.	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 4-3	4.7.6.1, 4.7.9 (Table 4.7-3, note *c*)	Specification of the OBE for the ANS	ANS Project	Reference 3 of Chap. 6 of PDR, Rev. 4.	Earl Stone	Advanced Conceptual Design (S)
HOLD 4-4	4.7.9 (Table 4.7-3, note *d*)	Specification of the DBT for the ANS	ANS Project	Reference 3 of Chap. 6 of PDR, Rev. 4.	Earl Stone	Advanced Conceptual Design (S)
HOLD 4-5	4.8.1.1 and (Table 4.8-1)	Maximum acceptable fuel temperature and time at that temperature for anticipated events.	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 4-6	4.8.1.1 and (Table 4.8-1)	Maximum acceptable fuel temperature and time at that temperature for unlikely events.	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 4-7	4.8.2 (Table 4.8-2)	Specification of the design basis accident dealing with reactivity events from liquid poison injection.	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 4-8	4.8.2 (Table 4.8-2)	Specification of the design basis accident dealing with loss of reflector coolant in the reflector to water/air cell.	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 5-1	5.4.2	Specification of the ground-level acceleration for the ILE.	ANS Project	PDR	Mike Harrington	Title I (S)
HOLD 6-1	6.1.2 (Table 6.1-1)	Specification of detailed design requirements for the secondary shutdown system.	ANS Project	SDD-33	John Anderson	Advanced Conceptual Design
HOLD 6-2	6.1.2 (Table 6.1-1)	Determining if a liquid poison injection system will be needed for the ANS.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD 11-1	11.4	Determining the time that spent fuel will be stored in the containment building.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design

Hold responsibility-status list (continued)

HOLD 11-2	11.4	Determining how long spent fuel will be retained before it is shipped off-site.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-1	Appendix B, Table B.4	Verify that emergency cooling for the reactor core, spent fuel, and containment do not depend on the availability of ac power.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-2	Appendix B, Table B.5	Determine if a beam tube wall break in combination with a proposed outer passive seal break is a single failure.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-3	Appendix B, Table B.6	Determine the sensitivity required for the failed fuel detection instrumentation.	ANS Project	PDR	Ray Booth	Title I
HOLD B-4	Appendix B, Table B.6	Decide if SDC 61 is needed.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-5	Appendix B, Table B.6	Decide how long to require on-site storage of solid waste and recovered tritium.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-6	Appendix B, Table B.6	Decide what percentage of failed fuel to assume in order to design the capacity of radioactive waste systems.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-7	Appendix B, Table B.7	Decide on the sensitivity required for the loose-parts detection system.	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD B-8	Appendix B, Table B.8	Verify the compliance commitment for SRP 6.1.1	ANS Project	PDR	Ray Booth	Advanced Conceptual Design
HOLD C-1	Appendix C, Table C.3	Verify the event and frequency category for the design-basis accident REN-3 dealing with liquid poison injection.	ANS Project	PDR	Mike Harrington	Advanced Conceptual Design
HOLD C-2	Appendix C, Table C.3	Verify the reactivity insertion value for the design-basis accident REP-7.	ANS Project	PDR	Mike Harrington	Title I
HOLD C-3	Appendix C, Table C.3	Verify the excess gas supply to the accumulator for design-basis accident NCG-3.	ANS Project	PDR	Mike Harrington	Title I

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