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OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

Plant Design Requirements

Advanced Neutron Source

MANAGED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

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ADVANCED NEUTRON SOURCE

PLANT DESIGN REQUIREMENTS

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ADVANCED NEUTRON SOURCE PLANT DESIGN REQUIREMENTS

APPROVALS

REVISION 2

C. D. West
ANS Project Director
Chairman, ANS Project Review Committee

10-2-91
Date

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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 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 tank systems, reactor control systems, and the experimental systems.

Becquerel (Bq): The SI unit for radioactivity equal to one nuclear transformation per second (s^{-1}).

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 NRC.

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.

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* Sources for this list are:

Advanced Light Water Reactor Utility Requirements Document, Vol. 1, ALWR Policy and Summary of Top-Tier Requirements, Electric Power Research Institute, NP-6780, March 1990.

Glossary of Terms in Nuclear Science and Technology, prepared by ANS-9, the American Nuclear Society Standards Subcommittee on Nuclear Terminology and Units, Harry Alter, Chairman, ISBN 0-89948-553-9, 1988.

Clinch River Breeder Reactor Plant, Overall Plant Design Description (OPDD-10), prepared by the Clinch River Breeder Reactor Plant Project Office, August 1983.

American National Standard, Nuclear Safety Design Criteria for Light Water Reactors, ANSI, ANS50.1 - Draft 3, March 1991.

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

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

Coolant: The fluid circulated through the reactor core, and nearby systems in the reflector tank, which transfers 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 such 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 Pressure Boundary Tube (CPBT): That component in the ANS which 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-specific licensing and safety criteria; a measure by which one can determine if a goal is achieved.

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 backup 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 Goals: General objectives that the design is trying to achieve. When something is a goal, it will be stated as such in the design documentation. Achieving a design goal is desirable but not mandatory.

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 regulation.

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

Dose: Quantity of radiation absorbed, in units of rem or sieverts, by the body or by any portion of the 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 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 radioactive byproduct of nuclear fission.

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, burnup, fluence, 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.

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

Integrating System Design Descriptions (ISDD): The top-level ANS design document for generic plant functions, like containment. Only the Plant Design Requirements (PDR), which define plant level requirements, have higher authority. The ISDDs augment the PDR by assigning functions to the 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 setpoint for automatic protection-device action that is closer to the normal operating range for the variable than 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).

Non-safety-related (non-safety-class): All systems which are <u>not</u> safety-related (see below).

Operating-Basis Earthquake (OBE): (TBD).

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

Plant Design Requirements (PDR): 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: Those components such as heat exchangers, piping, pumps, and valves which are part of the reactor coolant system or 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 which is normally shut is acceptable if failure does not prevent normal reactor inventory control), and

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 assure 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 the ANS is the sievert (Sv), which is equal to 100 rems.

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, App. A).

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 sitting, 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 safety analysis report (SAR).

Safety Limits: Limits on important process variables that are necessary to reasonably protect 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 ANS 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 Plant Design Requirements (PDR), which define plant level requirements, have 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.

Technical Specifications: Limits, controls, and surveillance requirements on process variables and equipment in an operating nuclear plant that 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 plant lifetime. No fuel damage should result from unlikely events even if licensing conservations are used in assessing consequences.

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
ANS	Advanced Neutron Source
ANSI/ANS	American National Standards Institute/American Nuclear Society
CDR	Conceptual Design Report
CPBT	Core Pressure Boundary Tube
CSAR	Conceptual Safety Analysis Report
DOE	Department of Energy
EAB	Exclusion Area Boundary
EIS	Environmental Impact Statement
EPA	Environmental Protection Agency
ER	Environmental Report
FSAR	Final Safety Analysis Report
HFIR	High Flux Isotope Reactor
ILL	Institut Laue-Langevin
INZ	Immediate Notification Zone
ISDD	Integrating System Design Description
LPZ	Low-Population Zone
NE	Office of Nuclear Energy
NEPA	National Environmental Protection Act
NOA	Notice of Availability
NOI	Notice of Intent
NRC	Nuclear Regulatory Commission
NSCANS	National Steering Committee for the Advanced Neutron Source
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
PDR	Plant Design Requirements
PMP	Project Management Plan
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
REDC	Radiochemical Engineering Development Center
ROD	Record of Decision
QA	Quality Assurance
SAR	Safety Analysis Report
SDD	System Design Description
SEMP	Systems Engineering Management Plan

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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 the 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, and references. The first two numbers are those of the two-digit section in which the cite first occurs. 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, and references are renumbered as the document is revised, while avoiding excessively long and cumbersome reference numbers.

ABSTRACT

The Advanced Neutron Source (ANS) is a new, world class facility for research using hot, thermal, cold, and ultra-cold neutrons. At the heart of the facility is a 350-MW_{th}, heavy water cooled and moderated reactor. The reactor is housed in a central reactor building, with supporting equipment located in an adjoining reactor support building. An array of cold neutron guides fans out into a large guide hall, housing about 30 neutron research stations. Office, laboratory, and shop facilities are included to provide a complete users facility. The ANS is scheduled to begin operation at the Oak Ridge National Laboratory at the end of the decade.

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

1. SCOPE

1. SCOPE

1.1 INTRODUCTION

This Plant Design Requirements (PDR) document defines the plant-level requirements for the design, construction, and operation of the Advanced Neutron Source (ANS). This document also defines and provides input to the individual System Design Description (SDD) documents. Together, the PDR document and the SDD documents define and control the baseline configuration of the 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 the 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.[1.1.1] The project organization and management structure is defined in the Project Management Plan (PMP),[1.1.2] the project quality assurance program is defined in the Quality Assurance Plan (QA Plan),[1.1.3] and other assessments and supporting documents are described in the Systems Engineering Management Plan (SEMP).[1.1.4]

1.2 PROJECT PURPOSE

The ANS will meet the recognized national need for an intense, steady-state, broad spectrum source of neutrons for research.[1.2.1-11] The 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 top-level technical objectives of the project are listed in Table 1.2.1.

1.3 PROJECT SCOPE

The ANS Project includes all aspects of the design and construction of the Advanced Neutron Source defined in the construction project data sheet 92-ORNL-KC(AF)-1, dated April 1990. The project 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 (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 the 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 that has a minimum unperturbed thermal flux in the reflector exceeding the best currently available in the world [unperturbed thermal flux of 1.5×10^{19} $m^{-2} \cdot s^{-1}$ at the Institute Laue Langevin (ILL) High Flux Reactor and the High Flux Isotope Reactor (HFIR)] by at least a factor of 5. The reactor will also provide materials irradiation and transuranic isotope production capabilities that match or exceed the capabilities of the 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 to the facilities offered locally or by other DOE operations.

Table 1.2.1. ANS Project technical objectives.

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

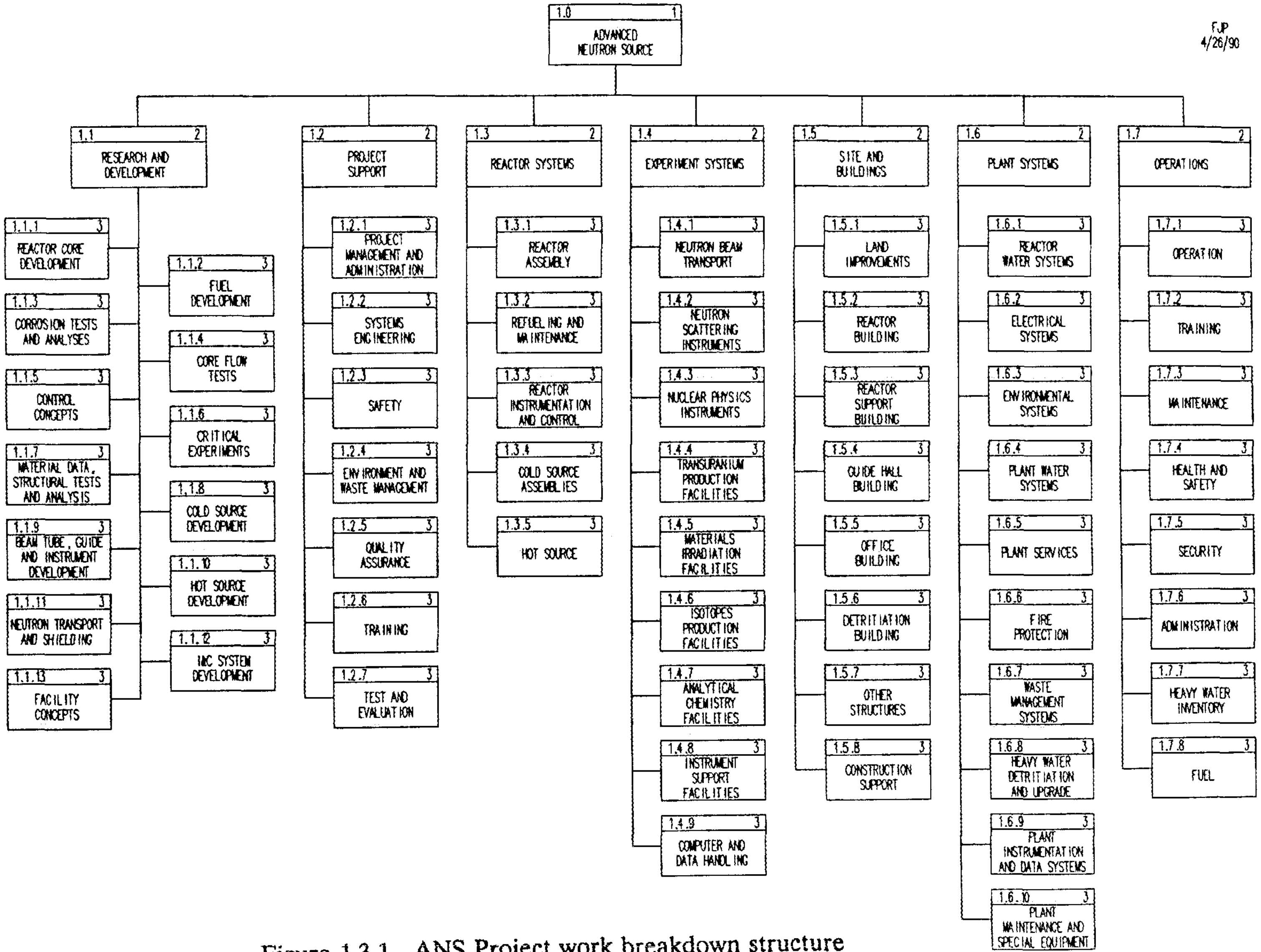

Figure 1.3.1. ANS Project work breakdown structure

2. DESIGN GOALS

2. DESIGN GOALS

2.1 RESEARCH FACILITY GOALS

The ANS will provide equipment and facilities for

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;
4. Activation analysis and related materials analysis capabilities.

Specific major criteria for the implementation of the project technical objectives (Table 1.2.1) are given in Table 2.1.1. The design criteria follow the recommendations of the ANS user community as defined by NSCANS and others.[2.1.1-43] The ability to adapt varied experimental facilities to changing future priorities through modular design and operational adjustments is a fundamental objective 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

If the project is to fulfill its purpose, the source of neutrons (i.e., the reactor) must meet certain minimum performance specifications for neutron flux, neutron spectrum, and experimental space. Consideration will also be given to performance beyond minimum specifications, to maximize the research value of the ANS by providing for optional research capability during initial plant design. This means that if additional performance or facilities, beyond the minimum specification, 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, to the extent feasible, be retained. Table 2.1.1 lists the major flux and spectrum specifications.

Table 2.1.1. ANS design goals

Parameter[a,b]	Criterion
Neutron scattering	
Hot neutrons	
Thermal flux at hot source	≥1.0
Number of hot sources	1
Number of hot beams	2
Thermal neutrons	
Peak thermal flux in reflector	≥7.5
Thermal:fast ratio	≥80:1
Number of thermal tangential tubes	6
Number of thermal radial tubes	1
Cold neutrons	
Thermal flux at cold sources	≥2.0
Number of cold sources	2
Number of horizontal cold guides	14
Number of slant cold beams	1
Nuclear and fundamental physics	
Number of thermal through tubes	1
Number of slant thermal beams	1
Number of slant very cold beams	2
Materials irradiation	
Small specimens	
Fast flux	≥1.4
Fast:thermal ratio	≥1:2
Total number of positions	10
Number of instrumented positions	5
Damage rate (dpa/y in stainless steel)	≥30
Nuclear heating rate (w/g in stainless steel)	≤54
Axial flux gradient over 200 mm	≤30%
Available diameter (mm)	≥17
Available length (mm)	≥500
Larger specimens[c]	
Fast flux	≥0.5
Fast:thermal ratio	≥1:3
Number of instrumented positions	≥8
Damage rate (dpa/y in stainless steel)	≥8
Nuclear heatingf rate (w/g in stainless steel)	≤15
Axial flux gradient over 200 mm	≤30%
Available diameter (mm)	≥48
Available length (mm)	≥500

Table 2.1.1. (continued)

Parameter[a,b]	Criterion
Isotope production	
Transuranium production	
Epithermal flux	≥0.6
Epithermal:thermal ratio	≥1:4
Allowable peak heat flux (MW/m^2)	≥4.0
Total annual production:	
^{252}Cf (g)	1.5
^{254}Es (μg)	40
Epithermal hydraulic rabbit tube	Epithermal flux peak position
Epithermal:thermal ratio	≥1:4
Allowable peak heat flux (MW/m^2)	≥1.75
Other isotopes	
Thermal flux	≥1.7
Number of reflector positions	≥4
Materials analysis	
Activation analysis pneumatic tubes	
40 cm^3 rabbits in reflector	4
1 cm^3 rabbits in reflector	1
Thermal flux at reflector rabbit positions	≥0.2
Heating rate:	
Temperature in a 40 cm^3 high density polyethylene rabbit (°C)	≤120
Rabbit tubes in light water pool	2
Thermal flux at light water rabbit positions	≥0.04
Prompt-gamma activation analysis cold neutron stations	
Low-background (multiple beam) guide system	1
Neutron depth profiling	
Number of slant cold beams	1

Notes:

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

[b] Neutron spectra are defined as follows:
Fast > 0.1 MeV   100 eV ≥ Epithermal > 0.625 eV   Thermal ≤ 0.625 eV

[c] The large materials irradiation specimens are intended to replace irradiation facilities in the HFIR removable beryllium region. It is unlikely that ANS will be able to meet 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 (see Sect. 6.6.2).

The size and configuration of the core must be such that the necessary space is available to accommodate and cool the transuranium production and materials irradiation facilities (see Sects. 2.2, 2.1, respectively). Access must be provided for electrical leads and gas lines for the instrumented irradiation capsules.

The size and configuration of the reflector tank must be such that the space and cooling requirements of the isotope production facilities, materials analysis facilities (rabbit tubes), cold sources, cold neutron guides, and neutron beam tubes are accommodated.

The hot source can, as dictated by safety and neutronic requirements, be placed outside or inside the reflector tank.

To avoid the coolant flashing to steam in the event of depressurization, the bulk coolant outlet temperature will be below the normal boiling point.

To minimize safety questions and technical risks, the reactor design must 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.

2.4 USER AND SYSTEM SUPPORT GOALS

The ANS facility will be capable of handling at least 1000 short-term (1 to 2 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 ILL), including the availability of adequate in-house scientific and technical staff support. To facilitate achieving this goal, the ANS plant will be designed and constructed with secure physical barriers between beam research and related support areas, on the one hand, and reactor operations areas on the other. 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 normal laboratory security requirements.

The ANS facility, in context with facilities available at Oak Ridge National Laboratory (ORNL) and the other DOE sites at Oak Ridge, will provide the necessary offices, shops, change facilities, 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 the Oak Ridge Reservation (ORR), including

Table 2.3.1. Design choices for the ANS Reactor

Type	Compact undermoderated core in a reflector region
Power level	≤ 350 MW (thermal power into fuel coolant)
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

Sources:

Proceedings of the workshop on An Advanced Steady-State Neutron Facility, "Report on the Working Group on Critique of Source Concepts," J. A. Lake and C. D. West, *Nuclear Instrumentation and Methods*, Vol. A249, No. 1, pp. 125-131, August/September 1986.

Advanced Neutron Source Project Annual Report, April 1987 - March 1988, Appendix B, "Core Comparison Workshop Summary," ORNL/TM-10860, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., February 1989

Advanced Neutron Source Final Preconceptual Reference Core Design, ORNL/TM-11234, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., August 1989

Report of the Advanced Neutron Source Safety Workshop, October 25-26, 1988, CONF-8810193, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1988

J. A. Young and J. U. Koppel, "Slow Neutron Scattering by Molecular Hydrogen and Deuterium," *Phys. Rev.*, 135, p. A603, August 1964 roadways, utilities, and monitoring systems. Security and fire protection 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, and 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 which involve interaction with 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 the ANS.

3. ECONOMIC REQUIREMENTS

4. SAFETY AND LICENSING REQUIREMENTS

4. SAFETY AND LICENSING REQUIREMENTS

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, the ANS shall be "licensable," and the standards, codes, and guides applied to the ANS design must include those by which the NRC would judge the ANS.

4.1.2 Risk Limitation Goals

The ANS risk limitation goals are based on the NRC Policy Statement dated August 21, 1986[4.1.1] and upon the similar DOE *Nuclear Safety Objectives Policy Statement* Draft, dated February 9, 1989.[4.1.2] 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, and latent fatality refers to an initially sublethal dose of radiation that may cause cancer that, in turn, causes 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 accidents risk, and 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.[4.1.3] 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 only be evaluated after a detailed multistep calculational process. 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 due to internal events shall not exceed 1×10^{-5} per year. The DOE *Safety Objectives Policy*[4.1.2] specifies this goal for new production reactors. A major NRC risk study[4.1.4] 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 shall 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 >200 rem (2 Sv)] to workers outside the reactor containment or to the general public. This definition is consistent with that applied in *NUREG-1150*.[4.1.4]
3. Inherent safety characteristics. ANS safety-related systems (see Sect 4.4 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 upon 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 the containment or retention of fission products.

Table 4.1.1. Radiological accident risk goals for the 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	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 Oak Ridge Reservation (ORR). The current preferred site for the ANS is in eastern Melton Valley, about 2 km from the southern boundary of the 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.

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*[4.1.5] [*NUREG*-0800, as amended by *CFR* 52 (167), (1987)].
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]*. Risk of pipe break exceeding the design basis is minimized by the provisions of paragraphs (1) and (3) of this subsection.

3. The total mean probability of catastrophic rupture of the core pressure boundary tube or any primary-coolant pipe larger than the design-basis break (see Sect. 4.1.3, item 2) 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.[4.1.6] The goal stated in this paragraph for the ANS is significantly lower 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 such 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 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 the 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 deleterious effect on the environment, as determined by the Environmental Impact Statement (EIS).

4.1.6 Ensured Site Suitability

The ANS reactor and containment shall be designed such that the emergency planning zones for the ANS are compatible with the existing ORNL planning zones,[4.1.7] 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"[4.1.8] must be satisfied. This means that the ANS design-basis site suitability source-term (SSST) accident (a severe 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.

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

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.[4.1.9] 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. The escaped, vaporized radionuclides are then available for 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 atmospheric dispersion conditions (e.g., Pasquill Type F conditions with 1 m/s windspeed for the first 8 hours of the accident) sufficiently conservative that conditions more favorable to dispersion—therefore 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 *Regulatory Guide* 1.4 prescription of a 25% iodine release exceeds what would be expected for an ANS severe accident due to the large amount of water submersion employed in the ANS design. Nevertheless, to ensure a solid site 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]* 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 the 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 in any way compromised by the proposed facility. Sufficient information must be provided in the SAR to enable a technically competent outside reviewer to verify independently that DOE and applicable NRC regulations, standards, and guides are followed. Per DOE Order 5480.6, the ANS SARs will follow the format specified by *NRC Regulatory Guide 1.70*, as supplemented by *NUREG*-0800.[4.1.5]

The SAR is written in three phases: the Conceptual SAR (CSAR), the Preliminary SAR (PSAR) and the Final SAR (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 the 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 (NEPA) Process

The NEPA process consists of a number of 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 scoping 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 environmental report (ER) will be 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*. Data in the ER is also used by the independent organization preparing the EIS. Since the data needed to prepare the draft EIS is a subset of the requirements of *Regulatory Guide 4.2*, the ER for the 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, the ANS will obtain all permits required under the Federal Facilities Agreement. This 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 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, including the ANS, is properly analyzed, evaluated and 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 the 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 DOE regulatory agencies.

In Sect. 8.b, DOE 5480.6 directs that the General Design Criteria 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 the ANS.

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

Sect. 8.e provides considerable direction for operating personnel training and qualification. While not generally applicable to the design of the 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, *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 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. 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.

However, there are two key reasons to use DOE 6430.1A as well as DOE 5480.6 in the design of the 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 the ANS shall in general follow the design criteria given in DOE 6430.1A. 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. 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. A list of key DOE Orders relevant to the design of the ANS is provided in Table 4.3.1.

4.4 NRC REGULATIONS AND GUIDES

4.4.1 10 *CFR* 50

As directed in DOE Order 5480.6, the General Design Criteria given in Appendix A of 10 *CFR* 50 shall be applied to the design of the ANS. Application of those General Design Criteria that address the design at a plant level are given in Appendix B of this PDR. Application of those General Design Criteria that address the functions of specific systems are addressed in the appropriate System Design Description (SDD) document.

As directed in DOE Order 5480.6, a set of Technical Specifications will be prepared for the 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-15.1-1990.

The ANS Quality Assurance (QA) program shall meet the intent of 10 *CFR* 50, Appendix B, as outlined in Ch. 7 of this PDR.

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

In addition to the specific paragraphs and appendices outlined above, all of 10 *CFR* 50 shall be reviewed for applicability to the ANS. Paragraphs and appendices of 10 *CFR* 50 that address issues directed at power reactors, but which are also issues important to the ANS, shall be followed in the design of the 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 the ANS. Those requirements that apply to the ANS at the overall plant level will be defined in 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 the 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

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

As directed in DOE Order 5480.6, the classifications for Safe Shutdown Earthquake (SSE) and OBE shall be applied to the ANS. Consideration of differences in the missions of power and research reactors may be taken into account in defining the rationale for selection of an OBE. Definitions of the SSE, and an appropriate equivalent to the OBE, are given in Sect. 6.7.

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 the 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 the 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 the general design criteria outlined in Section 4.5 and Appendix B of this PDR, and those applicable to an individual system of the ANS 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 the ANS, or for which a modified approach toward addressing the issue is proposed, shall be indicated, and 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 safety analysis report or the associated documentation supporting the DOE equivalent of a license to construct and operate the ANS.

4.4.4 NRC *Regulatory Guides*

Regulatory Guides are issued to describe methods acceptable to the NRC staff of 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 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 *Regulatory Guides* considered fully applicable shall be indicated. All *Regulatory Guides* considered not applicable to the ANS, or for which a modified approach is proposed, shall be indicated and 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, Appendix B, and SDDs (possibly as an appendix). A fully detailed report shall be provided as part of the SAR or associated documentation supporting the licensability of the ANS. The draft *Reference Document List for the ANS* (available from the ANS Project office) is a preliminary source of such documentation.

Table 4.4.1. NRC requirements applicable to the 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 which 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 from selected criteria may be justified in the CSAR.

[d]Compliance with 10 *CFR* 100 is intended, but exception from the 10 *CFR* 100, Appendix A, definition of the Operating Basis Earthquake 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-licenses facilities, the ANS will recommend to DOE designs and operating procedures that meet the NRC requirements.

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)*[4.1.5] is prepared to provide guidance to NRC staff reviewers in performing safety reviews of applications to construct or to operate light-water (LWR) nuclear power plants. The principal purpose of the *Standard Review Plan* is to assure 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 *Standard Review Plan* 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 the nuclear power industry. *Standard Review Plans* are not substitutes for regulatory guides or NRC regulations, and compliance with them is not required. However, the *Standard Review Plans* do provide an indication of the contents of the SARs that the NRC reviews as part of its licensing process. *Branch Technical Positions* are published along with the *Standard Review Plans* and further amplify the expectations of specific NRC review branches as to the contents of the SARs submitted with the license applications.

Although the *Standard Review Plans* address light water power reactors 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 the ANS. They also provide insight into the design requirements for the overall ANS plant and for specific ANS systems and components. Thus, the *Standard Review Plans* and *Branch Technical Positions* provide a checklist of requirements and information to be generated for the ANS. In cases where the plans and positions are deemed not to be applicable to the ANS, or where modified approaches are proposed, these shall be documented. The draft *Reference Document List for the ANS* (available from the ANS Project office) is a preliminary source of such documentation.

4.5 GENERAL DESIGN CRITERIA UNIQUE OR ADAPTED TO ANS

4.6 OTHER APPLICABLE REGULATIONS AND STANDARDS

4.7 SAFETY SYSTEM CLASSIFICATIONS

4.7.1 NRC Quality Group Classifications

The 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), with a minor modification (indicated in italics below) to reflect the systems that compose the ANS reactor plant.

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

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

1. systems or portions of systems important to safety that are 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 that are 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 that are connected to the reactor-coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation 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*.

Quality Group C includes water- 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 but part of:

1. cooling water systems or portions of systems important to safety that are 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 (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 that are designed for 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 are capable of being isolated from that boundary during all modes of normal reactor operation 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 (1.) through (3.) of this quality group that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential off-site doses that exceed 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 in Sect. III of the *ASME Boiler and Pressure Vessel Code.*

Quality Group D includes water-containing components not part of the reactor-coolant pressure boundary or included in quality Groups B or C but 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 guidance such as the NRC Quality Groups in an attempt to establish a consistent set of classifications and requirements for light-water reactor 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 in 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*, to the needs of the modest-pressure, low-temperature, high-power-density heavy-water reactor of the ANS.

4.7.3 Plant Conditions

A wide range of normal, off-normal, and accident events must be considered in the design of the ANS. In order 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.[4.7.1] The ANS has retained the use of the descriptive names for each of the categories — Normal Operations, Anticipated Events, Unlikely Events, and Extremely Unlikely Events — in place of the ANSI/ANS PC-1 through PC-5 designators. As explained below, 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 the ANS are given in Section 4.8.

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.

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 to 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 which 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.

<u>Unlikely Events</u>: This category corresponds to ANSI/ANS category PC-4. Events in the Unlikely category will:

1. be selected using PRA methods complemented by traditional engineering judgement, and will include internal events with estimated mean frequency less than 10^{-2}/year, but more than 10^{-4} per year,
2. include a comprehensive selection of external events (see Sect. 4.8.2 for a minimum list of such events), and
3. be subject to single failure criteria and other licensing conservatisms (no credit for non-safety-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 probably replacement of the reactor core would be necessary prior to restart following an Unlikely Event.

<u>Extremely Unlikely Events</u>: 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 using probabilistic risk assessments (PRAs) complemented by engineering judgement. This is consistent with guidance provided in the NRC Safety Goal [4.1.1] and Severe Accident Policies [4.7.2] which encourage the use of PRA methods to supplement engineering judgement and deterministic (nonmechanistic) analyses.[4.7.3]

Specifically, Extremely Unlikely Events would include internal events with estimated mean frequently 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 should, to the extent practicable, be minimized by design — as guided by the ANS fuel damage and large release goals (see Sect. 4.1.2).

Beyond-Design-Basis Accidents: In order to ensure that the ANS has an adequate degree of defense-in-depth against severe accidents, a specific group of severe accidents has been defined: the Beyond-Design-Basis Accidents (BDBAs). The 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; due to 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 to (1) guide the development of mitigative design features to prevent uncontrolled release of radioactivity, and (2) 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 for 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 and Components Affecting Reactor Safety

Equipment that is necessary to accomplish a nuclear safety function shall be assigned to one of three 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 the ANS and a pressurized light-water power reactor. Because of the small core, the high-power density, and the high-coolant flowrates of the 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 Safety Class-1 (SC-1).

The 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.

Detailed lists of the safety classifications of all components that are important to reactor safety shall be prepared as an appendix to the SDD that covers the design of the individual component.

Table 4.7.1. Classifications of components affecting reactor safety

Class		Definition
SC-1		Those systems or portions of systems:
		that compromise part of the reactor-coolant pressure boundary, except as noted under SC-2, below;
		that are used to perform reactivity scram functions under any plant conditions;
	*	that maintain core geometry or provide core support and whose failure could initiate a core disruptive accident; and
	*	whose single failure could cause a loss of safety function of another SC-1 component.
SC-2		Those systems or parts of systems not in SC-1:
		that are part of reactor-coolant pressure boundary but excluded by 10 *CFR* 50.55a(c);
		that 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;
	*	that are required to maintain an adequate reactor-coolant inventory following a reactor-coolant boundary leak;
	*	that are a part or an extension of the reactor containment boundary;
		that are required for emergency core cooling, postaccident containment heat removal, or postaccident fission product removal;

Table 4.7.1. (continued)

Class	Definition
	that are required for reactor shutdown or residual heat removal;
SC-2 (Cont.)	that 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 that are not normally operating or cannot be tested adequately during normal power operation; and
	whose single failure could cause a loss-of-safety function of another SC-2 component.
SC-3	Those systems or parts of systems not in SC-1 or SC-2:
	with secondary coolant systems that are required for emergency core cooling, postaccident containment heat removal, postaccident containment atmosphere clean-up or to remove residual heat from the reactor core or from spent fuel storage;
	whose failure could result in the loss-of-safety function of another SC-3 component;
	that 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 remote closure; or
	whose failure could result in release to the environment of radioactivity and would result in calculated potential exposures at the site boundary in excess of 0.5 rem whole body (or its equivalent).

* This category is additional to those not included in the NRC Quality Groups defined in Regulatory Guide 1.26, because of the smaller core, higher-power density, and higher flowrates in the ANS reactor.

4.7.5 Safety Classifications for Systems and Components Not Affecting Reactor Safety

Systems and components not affecting reactor safety shall be classified as nonnuclear safety class 1 (NNS-1) or nonnuclear safety class 2 (NNS-2), as shown in Table 4.7.2. This notation corresponds to the nonnuclear safety class of ANSI/ANS-51.1.1983. Two classes are provided so that a graded approach to the requirements of DOE 6430.1A, which applies to nonreactor systems and components, may be addressed. Items in NRC Quality Group D are included in the NNS-1 classification.

The classification of systems and components not affecting reactor safety shall be identified in an appendix to the SDD that covers the design of the individual system or component.

4.7.6 Seismic Classifications

Seismic Category I includes all ANS structures, systems, and components whose function is important to reactor safety. This classification includes all components classified as SC-1, SC-2, or SC-3, and all structures whose integrity is necessary to allow the safety class systems to perform their safety function. The classification of seismic category I structures, systems, and components is to conform with the requirements of *NRC Regulatory Guide 1.29, Seismic Design Classification*. Seismic category I structures, systems, and components are to be designed to remain functional during an SSE (defined in Sect. 6.7) without loss of capability to perform their safety function (and shall also be capable of withstanding the effects of the OBE defined in Sect. 6.7) without loss of capability to remain functional.

Seismic Category II includes those ANS structures, systems, and components that are to be designed to withstand the OBE. Seismic category II includes features that are required to permit continued reactor operation but that are not included in seismic category I, and includes any systems, structures, and components not in seismic category I but whose failure could damage a category I component. It also includes those items identified as requiring protection against the OBE so as to protect the plant investment.

Seismic Category III covers those structures, systems, and components that are not included in either seismic category I or II but that are essential for maintaining support of normal operations. Seismic category III structures shall be designed to the *Uniform Building Code*, in compliance with DOE 6430.1A.

4.7.7 Tornado Classifications

Tornado Category I includes all ANS structures, systems, and components whose function is important to reactor safety. This classification includes all components classified as SC-1, SC-2, or SC-3, and all structures whose integrity is necessary to allow the safety class systems to perform their safety function. The classification of tornado category I structures, systems, and components is to conform with the requirements of *NRC Regulatory Guide 1.117, Tornado Design*

Classification. Tornado category I structures, systems and components must remain functional during a Design Basis Tornado (DBT) (defined in Sect. 6.7) without loss of capability to perform their safety function and shall also be capable of withstanding the effects of the Operating Basis Tornado (OBT) (including tornado-generated missiles) and Winds and Pressure Changes (defined in Sect. 6.7) without loss of capability for normal operations.

Tornado Category II includes those ANS structures, systems, and components that are to be designed to withstand the OBT (including tornado-generated missiles and Winds and Pressure Changes). Tornado category II includes features that are required to permit continued reactor operation but that are not included in tornado category I, and includes any systems, structures, and components not in tornado category I but whose failure could damage a category I component. It also includes those items identified as requiring protection against the OBT, and Winds and Pressure Changes so as to protect the plant investment.

Tornado Category III covers those structures, systems, and components that are not included in either tornado category I or II but that are essential for maintaining support of normal operations. Tornado category III structures shall be designed to the *Uniform Building Code*, in compliance with DOE 6430.1A.

Table 4.7.2. Classifications of components not affecting reactor safety

Class[a]		Definition
NNS-1		Those components not in SC-1, SC-2, or SC-3:
		that contain or may contain radioactive materials;
	*	that are not safety related but that provide a significant contribution to plant investment protection;
	*	that require standards higher than commercial standards in accordance with DOE Order 6430.1.
NNS-2	*	Nonnuclear safety class 2 applies to all components not included in the above and are procured to commercial standards in accordance with DOE 6430.1A.

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

[a]In any mode of failure, equipment that does not otherwise perform a function affecting reactor safety shall not cause loss of the safety function of SC-1, SC-2, or SC-3 equipment.

4.7.8 Industry Classifications

ASME Section III, Class 1 shall be applied to all pressure vessels, piping, and valves classified as SC-1 components. **ASME Section III, Class CS** shall be applied to all core support components classified as SC-1 components.

ASME Section III, Class 2 shall be applied to all pressure vessels (except primary containment), piping, and valves classified as SC-2 components, including primary containment penetrations. **ASME Section III, Class MC** shall be applied to the metal primary containment vessel.

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

ASME Section VIII shall be applied to all pressure vessels (as defined by the *ASME Code*) classified as NNS-1 or NNS-2 components.

ANSI B31.1, power piping, or **ANSI B31.3**, chemical plant piping, as indicated in the SDD for that system, shall be applied to all piping systems classified as NNS-1.

IEEE Class 1E, as defined in IEEE Standard 378-1980, 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.

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 tornado classifications, and key industry standards is given in Table 4.7.3.

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.

Table 4.7.3. Correlation of reactor and nonreactor safety classifications and industry codes

Safety classification	NRC quality group	Seismic class	Tornado class	Pressure vessel code	Electrical class
SC-1	A	I	I	ASME Section III, Class 1 or CS	Class 1E
SC-2	B	I	I	ASME Section III, Class 2 or MC	Class 1E
SC-3	C	I	I	ASME Section III, Class 3	Class 1E
NNS-1	D	II	II	ASME Section VIII[a]	Non-class 1E
NNS-2	D	III	III	ASME Section VIII	Non-class 1E

[a]Piping systems classified as NNS-1 shall be designed and constructed in accordance with ANSI B31.1 or ANSI B31.3, as specified in the SDD.

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 safety features, e.g. isolation valves, water cells and air filtration units, comprise the third barrier against uncontrolled release of radioactivity to 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 subsections below explain the corresponding bases.

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 reference 4.1.5).

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 and Anticipated events. Temperature excursions to above 400°C may be allowable for Unlikely Events; the durations and magnitudes of allowable excursions are [TBD] but in any case will not exceed the 582°C solidus temperature of the A1-6061 cladding.

Corrosion of the aluminum cladding by conversion to Al_2O_3 is known to occur in aluminum-fueled reactors. A primary concern for the ANS is spallation of the oxide layer. The acceptance criterion that prevents oxide spallation[4.8.1] 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].

Limiting the structural deflection of the fuel plates is essential because sufficient plate-to-plate separation is critical to maintain 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 must 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].

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 does not lead to steam blanketing at any point, and subsequent fuel damage, the heat flux must be limited to below the critical heat flux (CHF) at the most unfavorable point in the core. The CHF limit, in accordance with the acceptance criteria in Section 4.4 of the *Standard Review Plan*,[4.1.5] is expressed as follows for Anticipated Events:

Throughout the core, heat flux at all points must remain below that for critical heat flux (per the Gambill-Weatherhead[4.8.2] or other acceptable CHF correlation), and heat flux over all hot streaks must remain below that for parallel channel-flow instabilities (per Costa [4.8.3] 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 must be met: minimum 95% nonexceedance probability at 95% confidence level or minimum 99.9% nonexceedance at 50% confidence level.

The CHF limitation is also important for Unlikely Events. NRC General Design Criterion 35 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. For the Unlikely Events, no more than 1% of the area of any fuel plate may be allowed to exceed the CHF heat-flux limit.

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 should be given to include some events at the top of any frequency category into the next highest 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 must always apply this very important acceptance criterion.

Combination of loads from certain design-basis events must 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 Sect. 6.7.3, Structural Design.

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 core pressure boundary tube 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 must 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, must be maintained as low as is reasonable achievable (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."[4.8.4] Under the ALARA principle, release limits are below the 25 mrem per year limit imposed by 40 *CFR* 61, Subpart H.[4.8.5] ANS/ANSI 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 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 the 10 *CFR* 100, Reactor Site Criteria.

Also see Sect. 6.7.4, Radiation Protection, 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 the 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 document.

The grouping of events into frequency categories is based on regulatory requirements as well as on available data from research, test, or power reactors. Data sources expressed in the ANS event category grouping include PRA studies conducted for the ANS, the High Flux Isotope Reactor (HFIR) Level I PRA, and applicable power reactor experience. The NRC General Design Criteria (GDC, 10 *CFR* 50, Appendix A) provide 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 should 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 must be designed to accommodate these test conditions.

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. <400°C; temp. difference across Al_2O_3 < 119°C	B (Upset)	(a) Appendix I if frequency > 0.1/year (b) If frequency < 10^{-1}/yr, 10 *CFR* 20 Unrestricted area limit
Unlikely (10^{-4}< frequency < 10^{-2}/year)	No CHF over > 1% of any fuel plate. Transient temperature increase as a function of time-at-temperature [TBD]	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[d]

[a]See Sect. 4.8.1.1 for detailed explanation of the fuel integrity limitations.

[b]In addition to not exceeding the *ASME Code* stress limitations, 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 Oak Ridge Boundary.

[d]See also Table 4.1.2 for exposure limitation goals and requirements referenced to locations other than the ORR boundary.

Table 4.8.2. ANS design-basis conditions and events

Event	Frequency category[1]	Identification number
NORMAL REACTOR OPERATIONS	Normal	
Fuel loading		N-1
Approach to criticality		N-2
Startup to low power		N-3
Startup to full power		N-4
Controlled shutdown to low power		N-5
Fast runback		N-6
Scram		N-7
Fuel unloading		N-8
Reduced power and operation with only two primary loops		N-9
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 system)		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]	TBD	REN-3
Light-water injection into reflector tank	Unlikely	REN-4
POSITIVE REACTIVITY (REP)		
Shim/safety withdrawal at normal speed from startup or low-power conditions	Anticipated	REP-1
Shim/safety withdrawal at normal speed from full power	Anticipated	REP-2
All-rod withdrawal at normal speed	Unlikely	REP-3
Rapid expulsion of a single rod (may be precluded by design)	TBD	REP-4
Single beam-tube flooding	Anticipated	REP-5
Cold source catastrophic failure with D_2 replaced by D_2O	Unlikely	REP-6
Multiple beam-tube flooding	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

Table 4.8.2. (continued)

Event	Frequency category[1]	Identification number
LOSS OF COOLANT PRESSURE CONTROL (LOPC)		
PRESSURE DECREASE		
One letdown valve goes fully open[2]	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[2]	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 (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
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
LOSS OF FLOW PATH		
Single isolation valve closed	Anticipated	LOF-5
Flow strainer in one loop blocked	Unlikely	LOF-6
Multiple isolation valve closure	Extremely unlikely	LOF-7
(Multiple isolation valve closure prevented by interlock)		
LOSS OF REFLECTOR COOLANT FLOW (LORF)		
All pumps shutdown	Anticipated	LORF-1
All flow control or isolation valves closed	Anticipated	LORF-2
CORE FLOW BLOCKAGE (CB)		
Experiment or transuranic target structural failure	Anticipated	CB-1
Foreign object in coolant	Anticipated	CB-2
Core inlet strainer structural failure	Unlikely	CB-3
Major core inlet flow blockage	Extremely unlikely	CB-4

Table 4.8.2. (continued)

Event		Frequency category[1]	Identification number
LOSS OF COOLANT INVENTORY			
See Loss of Coolant Accidents, below.			
LOSS OF HEAT SINK (LOHS)			
Loss of one normal heat sink[2]		Anticipated	LOHS-1
Loss of all normal heat sinks outside containment		Anticipated	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
Medium	(Rapid depressurization to below scram setpoint, but pressure adequate for ac motor operation of one primary-coolant pump)	Unlikely	MBLOC-i
Large	(Immediate depressurization to ambient pressure)	Extremely unlikely	LBLOC-i
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

Table 4.8.2. (continued)

Event	Frequency category[1]	Identification number
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]		LORC-3
Reflector auxiliary heat exchanger tube break		LORC-4
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 (ATWS)		
Anticipated Event with failure of primary scram system	Unlikely	ATWS-1
Unlikely Event with failure of primary scram system	TBD	ATWS-2
FUEL HANDLING ACCIDENTS		
New Fuel Storage	TBD	
Fuel Transfer Operations	TBD	
Fuel Hot-Cell Operations	TBD	
Spent-Fuel Storage	TBD	
Loss of criticality control	TBD	
Loss of spent-fuel cooling	TBD	
Fuel element stuck	TBD	
Fuel element drop	TBD	

Table 4.8.2. (continued)

Event	Frequency category[1]	Identification number
LOSS OF ELECTRICAL POWER (LOEP)		
Loss of all offsite power	Anticipated	LOEP-1
Station blackout	Unlikely	LOEP-2
Loss of all non-1E power	Anticipated	LOEP-3
EXPERIMENT ACCIDENTS		
Cold source (CS)		
Loss of cooling	Anticipated	CS-1
Pressure boundary fracture	Unlikely	CS-2
Internal D_2-air explosion	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)		
Inadequate cooling	Anticipated	IRR-1
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:		
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, tritium and D_2O release	Unlikely	RR-3

Table 4.8.2. (continued)

Event	Frequency category[1]	Identification number
OTHER INTERNAL EVENTS		
Fires	TBD	
Equipment generated missiles	TBD	
Flooding	TBD	
Pools		
Water cells		
Secondary coolant		
Heavy object drop	TBD	
EXTERNAL EVENTS		
Tornado	TBD	
Seismic		
OBE	Anticipated	
SSE	Unlikely/Extremely unlikely	
Floods	Unlikely	
HFIR or Transuranic Facility Accidents		

[1]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.

[2]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.

5. OPERATION AND MAINTENANCE REQUIREMENTS

5. OPERATION AND MAINTENANCE REQUIREMENTS

5.1 PLANT AVAILABILITY GOAL

Other than during scheduled major maintenance periods, the plant availability will be at least 80%. Global availability (including major maintenance) will be at least [TBD] %.

5.2 PLANT PREDICTABILITY GOAL

Reactor cycles and maintenance schedules will be predictable and scheduled with a lead time of at least 9 months once normal reactor operation is achieved. The reactor should be in operation, at full power, at least [TBD] % of the scheduled operating times.

5.3 PLANT LIFETIME REQUIREMENTS

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

6. PLANT SYSTEM DESIGN REQUIREMENTS

6.0 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 document. In other cases, the requirements are specified in ISDDs, also defined here, which cover generic functional areas like containment. The requirements given here and in the ISDDs complement the safety and licensing requirements found in Ch. 4. Requirements in the ISDDs carry the same authority as those in this PDR. Any inconsistencies in requirements should be resolved at the PDR level.

Section 6.1 presents the ISDDs and SDDs in tabular form for easy reference. Sections 6.2 through 6.6, arranged by general category of ISDDs and SDDs, present more detailed information concerning functions, safety classifications, and interfaces.

6.1 DEFINITION OF SYSTEMS

6.1.1 Integrating Systems (ISDD)

Table 6.1.1 identifies the ISDD by number and title, defines their responsibilities, and identifies the SDDs that implement their requirements. As shown in Table 6.1.1, ISDDs are numbered ISDD 21 through ISDD 25, consistent with the WBS shown in Figure 1.3.1.

6.1.2 Individual Systems (SDD)

Table 6.1.2 identifies the SDD by number and title, defines their functions and responsibilities, and identifies their primary interface systems. As done with the ISDDs, the SDDs are assigned numbers which follow the WBS, Figure 1.3.1. Specifically, the general categories of SDDs include Reactor Systems (SDD 31 to SDD 35), Experiment Systems (SDD 41 to SDD 49), 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 number	Title and Implementing Responsibilities	Systems (SDDs)
21	Reactor Containment Systems • fission-product control • heat removal and pressure control • combustible-gas control • personnel protection • equipment protection	31,32,33,41,52,61,63,69
22	Reactor Shutdown and Cooling Systems • shutdown • coolant-flow control • coolant-pressure control • coolant-inventory control • coolant-heat sink • coolant-purity control	31,33,41,52,53,61,69
NOTE:	Reactor = reactor assembly including both reactor primary and reflector cooling.	
23	Environmental Compliance and Monitoring (TBD)	TBD
24	Security • 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	Plant Integrated I&C Systems • I & C architecture • I & C integration	33, 69, and I/C portions of other systems

Table 6.1.2. Functions, Responsibilities, and Primary Interface Systems for the SDDs

SDD Number	Title, Functions, and Responsibilities	Primary Interface Systems (SDDs)
31	Reactor Assembly • Reactor core — neutron generation — heat generation — fission-product barrier — heat transport to primary coolant • Structural support — fuel elements — transuranic rods (1) — material irradiation samples (1) — control/shutdown mechanisms (1) — cold source (1) — hot source (1) — reflector tank — reactor assembly coolant piping • Heat transfer to reflector coolant • Fission-product barriers — primary to reflector — primary to H_2O pool — reflector to H_2O pool • Coolant inventory control — submerged operation — subpile room limited-volume enclosure	32,33,34,35,41,44,45, 52,61,62,64,67,68,69

Note (1). Other systems provide these components.

SDD Number	Title, Functions, and Responsibilities	Primary Interface Systems (SDDs)
32	Refueling and Maintenance • New fuel management — nuclear criticality control — receipt inspection — storage — reactor loading • Spent fuel management — reactor unloading — nuclear criticality control — cooling — shielding — storage — disposal • Special maintenance equipment	31,33,34,35,52,53,61,62, 63,64,65,66,67,69 site external facilities

Table 6.1.2. (continued)

33	Reactor Instrumentation and Controls • Reactor and reflector monitoring — shutdown rod positions — neutron flux — coolant-valve positions — coolant flow — coolant temperature — coolant pressure — coolant inventory — failed-fuel detection — experimental-tube leakage — cold source — other experiments related to reactor safety • Reactor and reflector control — reactivity — coolant temperature — coolant pressure — neutron flux — thermal power • Reactor and reflector protection — reactor reactivity shutdown - primary shutdown - secondary shutdown - fast secondary shutdown [HOLD] - liquid poison injection [HOLD] — reactor primary-coolant isolation — reactor secondary-coolant isolation — reactor main pump motor shutdown — experimental tube isolation	31,32,34,35,41,69,52,53, 61,62,63,64,65,66,67,68,69
34	Cold-Source Assemblies • Generate cold (40 K) neutrons • Cold source cooling • Deuterium containment	31,33,41,52,62,63,64, 66,67,69
35	Hot Source Assembly • Generate hot (2000 K) neutrons • Hot-source temperature control • Hot-source containment	31,33,41,52,62,65, 67,69

Table 6.1.2. (continued)

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	31,33,34,35,42,43,47, 52,54,61,62,67,69,610
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 — detectors — multidetectors • Instrument data collection and control • Personnel safety	41,48,49,52,54,62,64, 65,67,610
43	Nuclear and Fundamental Physics Instruments • Beam or guide stations — beam preparation — shielding — personnel safety	41,48,49,52,54,62,64, 65,67,10,69,610

Table 6.1.2. (continued)

	• Specialized stations — liquid hydrogen facility — small fissile target handling — activated target handling • Neutron beam monitoring	
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 tank materials irradiation — hot-cell loading and unloading — cooling — transfer to/from irradiation stations • Handling — reactor pool — storage pool	31,32,33,52,53,61, 62,67
46	Isotopes Production Facilities • Irradiation in reflector tank — 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

Table 6.1.2. (continued)

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 — cryogenic support — storage • Instrument preparation — off-line assembly and testing — shop support — storage, maintenance, and calibration • Crystal preparation — growth — characterization — mounting — storage	42,43,47,52,54,62,63, 64,65,67,610
49	Computer and Data-Handling 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
51	Land Improvements • Site characterization and monitoring • Land erosion control • Vehicle access and parking • Personnel walkways	24,62,64,66,67,69

Table 6.1.2. (continued)

	• Storm water management • Site security control	
52	Reactor Building • Equipment space, arrangement, and structural support • Equipment protection • 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 — management • 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	51,53,54,55,61,62,63,64, 65,66,69,610, all systems with penetrations
53	Reactor and 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 & egress • Personnel movement in building • Personnel change rooms • Personnel restrooms	32,51,52,55,61,62,64,65, 63,66,69,610

Table 6.1.2. (continued)

54	Guide Hall and Research Support Buildings • 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 and Interface Control 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 • Equipment protection • Tritium control • Heavy-water control • Personnel protection • Personnel access and egress	51,61,62,63,64,65,66, 67,68,69,610
57	Other Structures • Equipment space and structural support • Equipment protection • Cooling-tower basins • Pump house • Waste storage areas • Diesel-generator building • Vent stack	51,62,63,64,65,66,67,69

Table 6.1.2. (continued)

	• Security portals • Electrical switch yards	
58	Construction Support • Construction material storage • Construction personnel offices • Construction shops	51
61	Reactor Water System • Heavy-water storage and management • Reactor primary cooling (2) (F,PR,I,PU,HS) • Reflector-tank cooling (F,I,PU,HS) • Fission-product control — submerged operation — leak tight cells — isolation valves	31,33,52,53,62,63,64,65, 66,67,68,69

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, T = temperature, PR = pressure, I = inventory, PU = purity, HS = heat sinks).

62	Electrical Power and Communications System • 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

Table 6.1.2. (continued)

63	Environmental Control System • 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) • 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)	51,52,53,54,55,56, 57,61,62,64,67,69

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 • Reactor pool water service • Refueling water cooling • Auxiliary safety-related cooling water • Secondary cooling system • Process water supply • Nonsafety related cooling water • Potable water supply • Sanitary sewer water collection and disposal	51,52,53,57,61,62, 63,65,66,67,69 site external services

Table 6.1.2. (continued)

	• Equipment and floor drain • Safety-related chilled water • Nonsafety related chilled water • Building heating water • Cooling-tower makeup water • Cooling-tower blowdown treatment and disposal • Demineralized water management	
65	Plant Services • 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	Fire Protection • Fire protection program management • Fire hazards analyses • Fire detection • Fire suppression • Fire barriers • Fire exits • Fire water storage and distribution • Fire brigade	52,53,54,55,56,57,61, 62,63,64,65,69 site external services
67	Waste Management • 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 Detritiation and Upgrade • Receipt, sampling, and analysis • Protium removal and disposal • Tritium removal and disposal • Heavy-water recovery • Radioactivity confinement	52,53,56,61,62,63,64, 65,66,67,69,610, site external services

Table 6.1.2. (continued)

69	Plant Instrumentation, Control, and Data • Plant monitoring — meteorological — leak detection — radiation monitoring — security - intrusion detection and alarms - video monitoring - computers/communications — health physics — post-accident monitoring — 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, valve 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 • 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	31,32,33,34,35,52,54, 55,56,57,61,62,63,64, 65,66,67,68,610
610	Plant Maintenance, General Purpose, and Handling 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.

7. QUALITY ASSURANCE PROGRAM REQUIREMENTS

7. QUALITY ASSURANCE PROGRAM REQUIREMENTS

7.1 QUALITY ASSURANCE PROGRAM

The QA Program for the ANS Project shall meet the requirements of DOE Order 5700.6 "Quality Assurance"[7.1.1] and the intent of 10 *CFR* 50, "Appendix B",[7.1.2] as elaborated in Sect. 17 of the *USNRC Standard Review Plan* (NUREG-0800).[4.1.5] The ANS QA Program is documented in the *ANS Quality Assurance Plan* (ORNL/TM-11446)[1.1.3] and addresses the basic and supplemental requirements of ASME NQA-1.

The responsibility for the quality of ANS structures, systems, components, or activities shall reside with the line management of the project and the various supporting organizations and contractors. Subtier ANS QA Plans shall be required of contractor support organizations with extensive multidisciplinary scopes of work. Such plans shall be reviewed for acceptability by the ANS Project organization before contractors begin project design work.

A major component of the ANS QA Program shall be a group of quality controls collectively referred to as the Configuration Management System. The Configuration Management System shall ensure that the performance requirements and design features of the facility are identified, documented in a controlled reference source, and distributed to project organizations involved in the design of the facility.

The performance requirements and design features for the ANS are initially defined in this PDR document and shall be further detailed in the SDDs. These requirements shall establish the objectives of the facility design effort and serve as the benchmarks by which the success and acceptability of the design will be measured. They shall also serve as the standards of acceptance for the operability performance testing. As the facility design evolves, established requirements may not be achievable. To ensure proper interfacing of systems and components, the requirements documents shall be revised to reflect the achievable parameter.

Configuration Management shall involve multiple aspects of the project quality assurance program including: control of personnel training, control of design, control of computer codes, control of drawings, control of safety analyses, control of procurements, control of procedures, control of documents, control of materials, control of tests, control of nonconformances and control of quality records.

Configuration Management shall be practiced throughout the life of the facility. The 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 facility documentation (drawings, reports, analyses, etc.).

7.2 DEFINITION OF QUALITY LEVELS

The ANS QA program provisions shall apply to ANS structures, systems, components, and activities in a graded manner commensurate with the assigned Quality Level and Safety Classification designations defined in Sect. 4.5.

Nonsafety classified structures, systems, components, or activities shall have QA Program provisions selectively applied to provide an appropriate level of confidence that the items function in accordance with their design or that activities are performed as intended.

8. EXTERNAL INTERFACE REQUIREMENTS

9. SURVEILLANCE AND ACCEPTANCE TEST REQUIREMENTS

10. DECOMMISSIONING REQUIREMENTS

11. FUEL REPROCESSING AND FABRICATION REQUIREMENTS

12. REFERENCES

12. REFERENCES

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