ORNL/TM-10949

ornl

OAK RIDGE NATIONAL LABORATORY

MARTIN MARIETTA

Advanced Neutron Source (ANS) Regulatory Requirements and Safety-Related Considerations: August 1988 Draft

R. M. Harrington
Oak Ridge National Laboratory
and
C. A. Flanagan
Westinghouse

OAK RIDGE NATIONAL LABORATORY
CENTRAL RESEARCH LIBRARY
CIRCULATION SECTION
4500N ROOM 175
LIBRARY LOAN COPY
DO NOT TRANSFER TO ANOTHER PERSON
If you wish someone else to see this report, send in name with report and the library will arrange a loan.

Advanced Neutron Source

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

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
NTIS price codes—Printed Copy: A04 Microfiche A01

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

ORNL/TM-10949

Advanced Neutron Source (ANS) Regulatory Requirements and Safety-Related Considerations: August 1988 Draft

R. M. Harrington
Advanced Neutron Source Project
Oak Ridge National Laboratory

and

C. A. Flanagan
Westinghouse

August 1988

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
operated by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0281881 4

CONTENTS

Page

LIST OF TABLES v

INTRODUCTION AND PURPOSE 1

PART A. REGULATORY CONSIDERATIONS 3

A.1 APPLICABLE REGULATORY REQUIREMENTS 3

A.1.1 SUMMARY OF REQUIREMENTS 3

A.1.2 NUCLEAR REGULATORY COMMISSION QUALITATIVE AND QUANTITATIVE SAFETY GOALS 10

A.1.3 SUMMARY OF GENERAL DESIGN CRITERIA (GDC) ... 11

A.2 LICENSING BASIS EVENTS 11

PART B. DESIGN CONSIDERATIONS 17

B.1 SAFETY-RELATED EQUIPMENT 17

B.2 REACTOR CORE 18

B.2.1 FUNCTIONS 18

B.2.2 APPLICABLE GENERAL DESIGN CRITERIA 19

B.2.3 PRINCIPAL DESIGN CRITERIA 19

B.3 REACTOR SAFETY PROTECTION SYSTEM 20

B.3.1 FUNCTIONS 20

B.3.2 APPLICABLE GENERAL DESIGN CRITERIA 21

B.3.3 PRINCIPAL DESIGN CRITERIA 21

B.4 CORE PRESSURE BOUNDARY TUBE AND HEAVY-WATER REFLECTOR TANK 21

B.4.1 FUNCTIONS 22

B.4.2 APPLICABLE GENERAL DESIGN CRITERIA 23

B.4.3 PRINCIPAL DESIGN CRITERIA 23

B.5 PRIMARY COOLANT SYSTEM 24

B.5.1 FUNCTIONS 24

B.5.2 APPLICABLE GENERAL DESIGN CRITERIA 25

B.5.3 PRINCIPAL DESIGN CRITERIA 25

B.6 CONTAINMENT SYSTEM 26

B.6.1 FUNCTIONS 27

B.6.2 APPLICABLE GENERAL DESIGN CRITERIA 28

B.6.3 PRINCIPAL DESIGN CRITERIA 28

B.6.4 CONTAINMENT PERFORMANCE 29

Page

APPENDIX I. NRC REGULATORY GUIDES THAT MAY BE APPLICABLE TO THE ANS PROJECT 31

APPENDIX II. RADIATION STANDARDS FOR PROTECTION OF THE PUBLIC IN THE VICINITY OF DOE FACILITIES 49

APPENDIX III. TEXT OF GENERAL DESIGN CRITERIA 53

LIST OF TABLES

Table		Page
A.1	DOE mandatory and reference standards for reactor safety	4
A.2	NRC and EPA regulatory sources	7
A.3	Examples of top-level regulatory criteria for ANS	8
A.4	General design criteria for nuclear power plants	12
A.5	Examples of potential licensing basis events for ANS	15
B.1	Examples of potential ANS safety-related equipment	18

INTRODUCTION AND PURPOSE

The Advanced Neutron Source (ANS) represents a new user experimental facility for neutron research. The ANS will provide a steady-state source of neutrons from a reactor of unprecedented flux.

The ANS Project objectives are

- to design and construct the world's best research reactor for neutron scattering,
- to provide isotope production facilities that are as good as, or better than, those of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL), and
- to provide materials irradiation facilities that are as good as, or better than, those of HFIR.

The ANS concept calls for steady-state operations at reactor thermal power levels on the order of 300 MW(t) in a compact, high-power-density (5- to 8-MW/L) core. Each core has an operating lifetime of approximately 2 weeks.

The ANS design and operation must ensure safety to the general public and to facility personnel and equipment. A design requirement for the ANS is that applicable regulatory requirements be satisfied. The ANS will be a U.S. Department of Energy (DOE) facility and, as such, must be designed and operated in compliance with DOE orders and regulations. A primary DOE order applicable to ANS is DOE Order 5480.6, dated September 23, 1987, "Safety of Department of Energy-Owned Nuclear Reactors." That order incorporates by reference applicable U.S. Nuclear Regulatory Commission (NRC) regulations.

The purpose of this document is to outline the safety-related considerations for ANS and, by so doing, to indicate the regulatory requirements and design considerations that must be addressed by the project in the design and during subsequent operations. As such, this document represents a road map of how the project intends to analyze and document nuclear safety concerns and design considerations that will have to be addressed by the ANS Project. As the project matures and the design becomes better defined and projected performance better understood, the safety-related considerations will also become better defined and interpreted. Appropriate safety-related analyses and documentation will be produced (e.g., preliminary and final safety analysis reports). This document summarizes the sources of applicable regulatory requirements, indicates recent policy guidance from the NRC, provides the NRC general design criteria that must be met, suggests which of the major ANS systems components are likely to be categorized as "safety related," and provides descriptive material that is of safety interest and importance.

Part A of this document is focused on regulatory requirements that must be addressed and satisfied by the ANS Project. Part B is focused on design and performance considerations within which trade-offs are possible and desirable.

This document does not describe how the project should proceed so as to ensure that these requirements and considerations are addressed. The applicable documents covering such implementation will include, *inter alia*, the Project Management Plan and the Quality Assurance Plan.

In Part A, Sect. A.1 summarizes the present understanding and interpretation of applicable regulatory requirements for ANS. Section A.2 discusses licensing basis events. In Part B, Sect. B.1 suggests systems that may be categorized as "safety related." Sections B.2 to B.6 discuss these systems. Appendix I lists NRC regulatory guides and branch technical positions from the NRC Standard Review Plan (NUREG-0800) which may be applicable to the ANS facility. Appendixes II and III provide the text of (a) Radiation Standards for Protection of the Public in the Vicinity of DOE Facilities and (b) the General Design Criteria for Nuclear Power Plants from the *Code of Federal Regulations*.

PART A. REGULATORY CONSIDERATIONS

A.1 APPLICABLE REGULATORY REQUIREMENTS

A.1.1 SUMMARY OF REQUIREMENTS

This section addresses the applicable regulatory requirements for ANS. The responsibility for awareness of and compliance with all applicable federal, state, and local regulations resides with the project and DOE. The ANS is a DOE facility, and, as such, it must be designed and operated in compliance with DOE orders and regulations. Table A.1 specifies design standards that the DOE has made mandatory for application to new reactors, as well as reference standards of good practice. ANS designers should view the reference standards as required except under extenuating circumstances (e.g., conflicting NRC requirement).

A primary DOE order applicable to ANS is Order 5480.6, dated September 23, 1986, "Safety of Department of Energy-Owned Nuclear Reactors." The order incorporates, by reference, NRC regulations 10 *CFR* 50, 10 *CFR* 100, *Regulatory Guide* 1.8, and *Regulatory Guide* 1.13A, Rev. 1. Other regulatory guides may be applicable. See Appendix I for a discussion and listing of NRC regulatory guides. This listing in Appendix I also includes branch technical positions from the NRC Standard Review Plan (NUREG-0800). Even if ANS, as a DOE facility, is not required to go through the process for obtaining an operating license, it is the project's policy that the design satisfy these regulatory requirements. Other federal, state, and local regulations judged applicable by the project and DOE will then also require compliance.

Table A.2 lists applicable regulatory sources for ANS. The final design and subsequent operation of ANS will have to satisfy all applicable regulations. In the process of performing the design, guidance to the project will be provided by establishing specific, top-level regulatory requirements for ANS. During the early stages of conceptual design, the project will finalize and adopt firmly established, top-level criteria along the lines of the examples cited in Table A.3.

The DOE has established "Radiation Standards for Protection of the Public in the Vicinity of DOE Facilities." These limits cover all pathways, including air pathways, during normal planned operations only (as per 40 *CFR* 61, Subpart H). These standards became effective in 1985 and are currently being incorporated into a DOE order for radiation protection. These standards are included as Appendix II.

Table A.1. DOE mandatory and reference standards for reactor safety

(Ref. DOE 5480.4, 5-15-84)

Mandatory Standards

A. ANSI/IEEE 279–1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (ANSI).

B. "Boiler and Pressure Vessel Code" (ASME).

C. ANS 15.1–1983, "Development of Technical Specifications for Research Reactors" (American Nuclear Society).

D. IEEE 308–1980, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" (IEEE).

E. IEEE 603–1980, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (IEEE).

Reference Standards

A. "Applicable Nuclear Energy Reactor Development Technology (RDT) Standards" [see index of NE (RDT) standards] (DOE).

B. Title 10 *CFR* 50, "Licensing of Production and Utilization Facilities," and appendices, including regulatory guides issued to describe a method of implementing these regulations (NRC).

C. Title 10 *CFR* 55, "Operator's Licenses" (NRC).

D. Title 10 *CFR* 100, "Reactor Site Criteria" (NRC).

E. Safety Series No. 35, "Safe Operation of Critical Assemblies and Research Reactors," 1971 edition (IAEA).

F. Safety Series No. 31, "Safe Operation of Nuclear Power Plants" (IAEA).

G. Institute of Electrical and Electronics Engineers Standards, as follows:

1. IEEE 317–1976, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations" (IEEE).

2. IEEE 323–1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (IEEE).

3. IEEE 336–1980, "Installation, Inspection, and Testing Requirement for Class 1E Instrumentation and Electric Equipment at Nuclear Power Generating Stations" (IEEE).

4. IEEE 338–1977, "Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems" (ANSI/IEEE).

5. IEEE 344–1974, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" (ANSI/IEEE).

Table A.1 (continued)

Reference Standards (continued)

6. IEEE 352–1975, "Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems" (ANSI/IEEE).

7. IEEE 379–1977, "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems" (ANSI/IEEE).

8. IEEE 382–1980, "Standard for Qualification of Safety Related Valve Actuators" (IEEE).

9. IEEE 384–1977, "Standard Criteria for Independence of Class 1E Equipment and Circuits" (IEEE).

10. IEEE 387–1977, "Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations" (ANSI/IEEE).

11. IEEE 450–1980, "Recommended Practice for Large Lead Storage Batteries for Generating Stations and Substations" (revision of ANSI/IEEE Std 450-1975) (IEEE).

12. IEEE 467–1980, "Standard Quality Assurance Program Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations" (IEEE).

13. IEEE 484–1975, "Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations" (ANSI/IEEE).

14. IEEE 498–1980, "Standard Requirements for the Calibration and Control of Measuring and Test Equipment Used in the Construction and Maintenance of Nuclear Power Generating Stations" (IEEE).

15. IEEE 567, "Trial-Use Standard Criteria for the Design of the Control Room Complex for a Nuclear Power Generating Station" (ANSI/IEEE).

16. Other IEEE Standards, if required.

H. American Nuclear Society (ANS) Standards, as follows:

1. ANS-2.2-19/8, "Earthquake Instrumentation Criteria for Nuclear Power Plants" (Revision 1 of N18.5–1974)

2. ANS-2.17–1980, "Evaluation of Radionuclide Transport in Ground Water for Nuclear Power Sites."

3. ANS-3.5–1979, "Nuclear Power Plant Simulators for Use in Operator Training."

Table A.1 (continued)

Reference Standards (continued)

4. ANS-3.7.2-1979, "Emergency Control Centers for Nuclear Power Plants."
5. ANS-4.1-1978, "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations."
6. ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors."
7. ANS-6.4-1977, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants."
8. ANS-11.13/N101.6-1972, "Concrete Radiation Shields."
9. ANS-14.1/N394-1975, "Operation of Fast Pulse Reactors."
10. ANS-15.2/N398-1974, "Quality Control for Plate-Type Uranium-Aluminum Fuel Elements."
11. ANS-15.3/N399-1974, "Records and Reports for Research Reactors."
12. ANS-15.8/N402-1976, "Quality Assurance Program Requirements for Research Reactors."
13. ANS-15.11-1977, "Radiological Control at Research Reactor Facilities."
14. ANS-15.12-1977, "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents."
15. ANS-15.15-1978, "Criteria for the Reactor Safety System of Research Reactors."
16. ANS-15.16-1978, "Emergency Planning for Research Reactors."
17. ANS-15.18-1979, "Administrative Controls for Research Reactors."
18. ANS-59.1-1979, "Safety-Related Cooling Water Systems in Nuclear Power Plants."
19. ANS-59.3-1977, "Safety-Related Control Air Systems."
20. ANS-59.4-1979, "Generic Requirements for Light Water Nuclear Power Plant Fire Protection."
21. ANS-59.51/N195-1976, "Fuel Oil Systems for Standby Diesel-Generators."

I. BNL-50831-I, "Design Guide; Critical Facilities" (BNL).

J. BNL-50831-II, "Design Guide; Light and Heavy Water Cooled Reactor" (BNL).

K. BNL-50831-III, "Design Guide; Pool Type Reactors" (BNL).

Table A.2. NRC and Environmental Protection Agency (EPA) regulatory sources

Source	Area Covered
51 *FR* 30028	Safety goals for nuclear power plant operation (similar to earlier version in NUREG-0880)
10 *CFR* 20	Permissible dose levels and activity concentrations in restricted and unrestricted areas
10 *CFR* 50	All sections not inherently applicable only to power reactors [per DOE 5480.6 (9/23/86), Sect. 1.6]
10 *CFR* 50.36	Technical specifications [per DOE 5480.6 (9/23/86), Sect. 8.d]
10 *CFR* 50, Appendix I	General design criteria
10 *CFR* 50, Appendix I	Numerical dose guidelines for meeting the criterion "as low as reasonably achievable" (ALARA) for routine power reactor effluents
10 *CFR* 100	Site suitability evaluation, numerical dose guidelines for determining the exclusion area boundary, low population zone, and population center distances.
NRC Regulatory Guide 1.70	Standard format and content of safety analysis reports for nuclear power plants [per DOE 5480.6 (9/23/86), Sect. 8.c]
EPA-520/ 1-75-001	Protection action guide doses for protective actions for nuclear incidents
40 *CFR* 61, Subpart H	National emission standards for radionuclide emissions from DOE facilities

Table A.3. Examples of top-level regulatory criteria for ANS

Source	Criterion
51 *FR* 30028 (initially published in NUREG-0880)	1. The risk to an average individual in the vicinity of a nuclear power plant from prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
	2. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.
	Note: Criteria 1 and 2 require interpretation by ANS based on the prompt fatality risk in the United States and the latent fatality risk near ANS.
10 *CFR* 20	1. Permissible levels of radiation in unrestricted areas:
	whole-body dose < 0.5 rem/calendar year, whole-body dose < 0.002 rem/any one hour, and whole-body dose < 0.1 rem/any seven consecutive days.
	2. Radioactivity in effluents to unrestricted areas: limits as specified in Appendix II, Table II, 10 *CFR* 20.
	3. Permissible airborne levels in restricted areas: intake < 520 MPC-h in a calendar quarter.
	Note: These are all routine requirements.
10 *CFR* 50, Appendix I	Control room and vital access and occupancy: whole-body dose (or equivalent) < 5 rem for duration of accident for required vital access (general design criteria number 19).
10 *CFR* 50, Appendix I	1. Estimated annual dose from liquid effluents < 0.003 rem whole body or 0.010 rem to any organ.
	2. Estimated annual dose from gaseous effluents < 0.005 rem whole body or 0.015 rem to the skin or any organ.
	3. Estimated annual dose from radioactive material in particulate form in effluents to the atmosphere < 0.015 rem to any organ.

Table A.3 (continued)

10 *CFR* 100	The exclusion area boundary, 2-h, and low population zone, 30-d accident doses < 25 rem whole body and 300 rem thyroid. For ANS design purposes, the exclusion area and the low population zone must be designated such that their borders extend only to the boundary of the DOE property surrounding the reactor.
EPA-520/ 1-75-001	1. Intervention during hypothetical accidents indicated for population if whole-body dose projected to exceed 1 to 5 rem or thyroid dose exceeds 5 to 25 rem from airborne radioactive materials. 2. Guides for exposure from foodstuffs, water, and material deposited on property and equipment: to be determined.
40 *CFR* 61, Subpart H	Emissions of radionuclides to air from DOE facilities shall not exceed a dose equivalent of 25 mrem/year to the whole body or 75 mrem/year to a critical organ of any member of the public.

A.1.2 NUCLEAR REGULATORY COMMISSION QUALITATIVE AND QUANTITATIVE SAFETY GOALS

A safety goal policy statement was issued by the NRC (10 *CFR* 50) on August 21, 1986. The policy statement is focused on the risks to the public from nuclear power plant operation. The objectives are to establish goals that broadly define an acceptable level of radiological risk. Two qualitative safety goals were established; these are supported by two quantitative objectives. The two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks. The goals and objectives are the following:

Qualitative Goals

- Individuals should bear no significant additional risk to life and health resulting from the consequences of nuclear power plant operation.

- Society should bear no significant addition to other societal risks resulting from nuclear power plant operations.

Quantitative Objectives

- The risk to an average individual within 1 mile of the site exclusion area boundary of prompt fatalities from reactor accidents should be limited to 0.1% of the sum of prompt fatality risks resulting from all other accidents to which the general population is exposed.

- The risk of cancer fatalities to the population within 10 miles which might result from nuclear power plant operation should be limited to 0.1% of the sum of cancer fatality risks resulting from all other causes.

The NRC also proposed the following general performance guideline for evaluation by the staff:

- The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be $<10^{-6}$/year of reactor operation.

The ANS project intends to conduct a probabilistic risk assessment (PRA) to confirm that these goals and objectives are satisfied. In addition, the proposed general guideline (referred to the NRC staff for evaluation) is one on which the ANS project has taken the following position:

- A "large release of radioactive materials to the environment from a reactor accident" is a release that would result in radiation exposure in excess of the 10 *CFR* 100 limits.

- Given the definition above, the overall mean frequency of such an event is to be $<10^{-6}$/year.

All project work impacted by the NRC policy statement should incorporate this guidance.

A.1.3 SUMMARY OF GENERAL DESIGN CRITERIA (GDC)

The DOE Order 5480.6, Sect. 8.0 (Program Requirements), dated September 23, 1986, states that the GDC specified in 10 *CFR* 50, Appendix A, shall be applied to all DOE-owned reactors in the case of ALL NEW CONSTRUCTION OF REACTOR FACILITIES. A summary listing of these criteria is given in Table A.4. Appendix III contains the complete text of the General Design Criteria.

In Part B, Sects. B.1 to B.6, of this document, a brief discussion of the safety-related equipment of ANS is provided. Included in each of these sections is an identification of which of the GDC apply to each equipment area. It should be noted that some of the GDC apply to more than one equipment area.

A.2 LICENSING BASIS EVENTS

Licensing basis events are abnormal or accident events used in the analysis of the facility to demonstrate compliance with the top-level regulatory criteria. The selection of these events will be based on risk assessment techniques to identify those classes of events that have the potential for radionuclide release in excess of the limits specified by the regulatory criteria.

Regulatory Guide 1.70 provides guidance for examining a broad spectrum of accidents in the safety analysis report to assess the effects of anticipated process disturbances and postulated component failures. These analyses are to be performed to assess the capability of the plant to accommodate such events and their consequences and to identify limitations of expected performance. The situations to be analyzed would include anticipated operational occurrences, off-design transients that induce fuel failures above those expected from normal operational occurrences, and postulated accidents of low probability. The analyses should include an assessment of the consequences of an assumed fission product release resulting in potential hazards beyond those from any credible accident [i.e., for the purposes of site suitability analysis per 10 *CFR* 100.11(a)].

Guidance is provided to ensure that a sufficiently broad spectrum of initiating events has been considered. The events are to be categorized by type and expected frequency of occurrence to facilitate the analysis of limiting cases. Specific acceptance criteria are to be applied consistently to each postulated initiating event.

To assist in categorizing events, *Regulatory Guide* 1.70 provides the following list of classes to which postulated initiating events should be assigned:

Table A.4. General design criteria for nuclear power plants (10 *CFR* 50, App. A)

Major Classification	Subject	Criterion Number
I.	Overall Requirements	
	Quality standards and records	1
	Design bases for protection against natural phenomena	2
	Fire protection	3
	Environmental and missile design bases	4
	Sharing of structures, systems, and components	5
II.	Protection by Multiple Fission Product Barriers	
	Reactor design	10
	Reactor inherent protection	11
	Suppression of reactor power oscillations	12
	Instrumentation and control	13
	Reactor coolant pressure boundary	14
	Reactor coolant system design	15
	Containment design	16
	Electric power systems	17
	Inspection and testing of electric power systems	18
	Control room	19
III.	Protection and Reactivity Control Systems	
	Protection system functions	20
	Protection system reliability and testability	21
	Protection system independence	22
	Protection system fault modes	23
	Separation of protection and control systems	24
	Protection system requirements for reactivity control malfunctions	25
	Reactivity control system redundancy and capability	26
	Combined reactivity control systems capability	27
	Reactivity limits	28
	Protection against anticipated operational occurrences	29
IV.	Fluid Systems	
	Quality of reactor coolant pressure boundary	30
	Fracture prevention of reactor coolant pressure boundary	31

Table A.4 (continued)

Major Classification	Subject	Criterion Number
IV.	Fluid Systems (continued)	
	Inspection of reactor coolant pressure boundary	32
	Reactor coolant makeup	33
	Residual heat removal	34
	Emergency core cooling	35
	Inspection of emergency core cooling system	36
	Testing of emergency core cooling system	37
	Containment heat removal	38
	Inspection of containment heat removal system	39
	Testing of containment heat removal system	40
	Containment atmosphere cleanup	41
	Inspection of containment atmosphere cleanup system	42
	Testing of containment atmosphere cleanup system	43
	Cooling water	44
	Inspection of cooling water system	45
	Testing of cooling water system	46
V.	Reactor Containment	
	Containment design basis	50
	Fracture prevention of containment pressure boundary	51
	Capability for containment leakage rate testing	52
	Provisions for containment testing and inspection	53
	Systems penetrating containment	54
	Reactor coolant pressure boundary penetrating containment	55
	Primary containment isolation	56
	Closed system isolation valves	57
VI.	Fuel and Radioactivity Control	
	Control of releases of radioactive materials to the environment	60
	Fuel storage and handling and radioactivity control	61
	Prevention of criticality in fuel storage and handling	62
	Monitoring fuel and waste storage	63
	Monitoring radioactivity releases	64

- increase in heat removal by the secondary coolant system,
- decrease in heat removal by the secondary coolant system,
- decrease in primary coolant system flow rate and core flow blockage,
- reactivity and power distribution anomalies,
- increase in primary coolant inventory,
- decrease in primary coolant inventory,
- radioactive release from a subsystem or component, and
- anticipated transient without scram (ATWS).

For the ANS reactor, the following additional event categories are needed:

- cold source accidents and
- reflector tank accidents.

Each initiating event should be evaluated as to its expected frequency of occurrence, and only limiting faults need be analyzed completely; others can be treated qualitatively.

Because some events are more probable than others, the result of the risk analyses will be used to identify events as a function of probable frequency. These events can then be categorized into one of two categories. The purpose of the categories is to ensure that more stringent acceptance criteria are applied to events that might reasonably be expected to happen in the lifetime of the plant.

1. Anticipated operational occurrences (AOOs). These are events that are expected to occur one or more times during the facility lifetime; events with a probability of occurrence exceeding 0.0125/year,* including uncertainties, would fit this category. AOOs must be analyzed, including uncertainties, to demonstrate that the consequences are limited to minimal unplanned release of radioactivity and that acceptable fuel design limits are not exceeded; analysis results must demonstrate compliance with 10 *CFR* 50, Appendix I (numerical dose guidelines for meeting the ALARA criteria for power reactor effluents).

---

*Based on probability of about 0.5 of one or more occurrences some time during an assumed 40-year facility life.

2. Design basis events. These are events that are not expected to occur in the lifetime of ANS; the probability of occurrence ranges from 0.0125/year down to a tentative lower value* of 10^{-6}/year, including uncertainties. These events must be analyzed, including uncertainties, to demonstrate compliance with the 10 *CFR* 100 dose limits for off-site radiation exposure (see Table A.3).

The ANS project must determine what the licensing basis events are and which category contains which events. Examples of events that might fit into each of the two categories are given in Table A.5.

Table A.5. Examples of potential licensing basis events for ANS[a]

1.	Anticipated Operational Occurrences
	Power transient with forced-flow cooling
	Primary coolant pump trip
	Inadvertent control element withdrawal
	Small primary coolant pipe break
	Secondary coolant pump trip
	Loss of off-site power
2.	Design Basis Events
	Primary coolant pump seizure
	Anticipated transient without reactor scram
	Spurious control element withdrawal
	Earthquakes and other natural disasters
	Small primary coolant pipe break with failure of emergency injection system

[a]This list is for example only and is not all-inclusive. The final list will be developed from the results of the PRA within the general framework of *Regulatory Guide* 1.70, Sect. 15.

*If the PRA finds events of potential off-site consequences of probability below the lower cutoff, they will be considered in the development of the facility emergency planning.

PART B. DESIGN CONSIDERATIONS

B.1 SAFETY-RELATED EQUIPMENT

The safety-related equipment for ANS will comprise those systems, structures, and components that perform the functions required to limit the probability of core damage or to limit the releases of radionuclides to those allowed by 10 *CFR* 100. The equipment that is safety related must be determined for ANS by first identifying the functions that must be performed and then by identifying the equipment to perform those functions. The definition of safety-related equipment should correspond to that in *Regulatory Guide* 1.26. *Regulatory Guide* 1.29 provides guidance in determining which systems must be designed to withstand the effects of the safe-shutdown earthquake (SSE, to be designated). Generally, safety-related equipment includes systems, components, and structures needed to shut down the reactor, maintain a safe-shutdown condition, and mitigate the radiological consequences from a release of radioactive materials. Also, any system, component, or structure that could lead to an off-site equivalent dose of 0.5 rem as the result of failure during operation is considered "important to safety."

The overall design of ANS will be in accordance with 10 *CFR* 50, Appendix B. The ANS Quality Assurance (QA) program (to be developed) will be structured to assure compliance with 10 *CFR* 50, Appendix II. In structuring the QA program, consideration will be given to categorization of systems with respect to licensing basis events as follows:

Safety Related — Those items needed to prevent or mitigate design basis events. (Equipment or systems in this category will have to meet the most stringent requirements.)

Important to Safety — Those items needed only to prevent or mitigate anticipated operational occurrences — somewhat less stringent than the above.

Nonsafety Related — Those items not included in either of the previous classifications. These would have less stringent requirements from a safety aspect, and their requirements may be set by ANS availability.

Examples of the types of equipment that are likely to be included as safety-related equipment for ANS are given in Table B.1.

Table B.1. Examples of potential ANS safety-related equipment

Reactor core
Reactor protection system
Heavy water reflector tank
Primary coolant system
Containment system

B.2 REACTOR CORE

The reactor core consists of the fuel element, associated control systems, the irradiation experiment assemblies, and the associated support structures.

The nuclear heat is generated in the reactor fuel element. Removal of the heat energy is provided by the heat transport system, with the main circulating pumps providing the driving force to supply primary coolant, heavy water, to the fuel element. The reactor fuel element is contained within a pressure container (the core pressure boundary tube) to enable operation of the primary coolant system at a sufficiently high pressure.

Control of the fission energy heat generation rate during reactor power operation and assurance of adequate shutdown margin is provided by moveable control elements driven by control element drive mechanisms.

Various irradiation experiments will be inserted into specified locations designed into the fuel assembly. These target assemblies will be used for various experimental purposes during power operation.

As discussed in Sect. B.4, the reactor core will be surrounded by a heavy-water reflector to provide a source of thermal neutrons for neutron scattering experiments. Neutron guide tubes will be positioned adjacent to the reactor core so that a high population of neutrons will be available to the neutron guide tubes. One or more cold sources will be located in close proximity to the reactor core; these will contain liquid deuterium and will operate at temperatures of about 20 to 30 K.

B.2.1 FUNCTIONS

The primary functions of the reactor core are to

- provide a source of neutrons for experimental purposes, including neutron scattering experiments and target irradiations;

- transfer the heat generated by the fission process to the primary coolant flowing within the fuel element; and

- provide control of radionuclides generated as the result of the fission process and neutron activations. These control measures include retention of radionuclides in the fuel, removal of core heat, control of chemical attack, and control of the heat generation rate.

B.2.2 APPLICABLE GENERAL DESIGN CRITERIA (See Table A.3)

The general design criteria from Table A.3 which apply to the reactor core are the following:

Category I: 1-5
Category II: 10-16, 19
Category III: 25-28
Category IV: 30-32, 34
Category V: None
Category VI: None

B.2.3 PRINCIPAL DESIGN CRITERIA (See Table A.3)

In addition to the general design criteria from Table A.3 which apply, there may be other principal design criteria that should be established for the ANS reactor core. Examples of such criteria are as follows:

- Design, fabricate, and operate the fuel element such that any fission product releases to the primary coolant system will not exceed acceptable values for both normal and off-normal conditions.

- Select reactor design parameters (dimensions, power densities, coolant pathways to the environment) to limit fuel element temperature to acceptable values, including conditions after reactor shutdown and during core decay heat removal.

- Design the reactor to ensure that inadvertent reactivity, temperature, pressure, or other undesirable excursions result in negative feedback so as to restore the reactor to a desired set of conditions. An example is to design the reactor core such that it has a negative temperature coefficient of reactivity at all times in core life.

- Provide control equipment to ensure control of reactor thermal power during normal and off-normal conditions such that fuel clad melting or fuel melting is not expected to occur during core life.

- Maintain fuel geometry and coolant flow path integrity to ensure that the reactor can be safely shut down and adequately cooled during all operational occurrences.

B.3 REACTOR SAFETY PROTECTION SYSTEM

The reactor safety protection system of ANS will consist of a system of hardware and software necessary to protect the general public, facility personnel, and facility systems, subsystems, and components. This system will provide active control or will initiate systems to control the heat generation rate during startup, power level operation, shutdown, and during periods requiring decay heat removal after shutdown.

The reactor safety protection system provides those features essential to monitor and implement commands related to operating reactor safety. This includes the ability (1) to sense reactor and associated plant process variables, (2) to detect abnormal reactor and associated plant conditions, and (3) to initiate those protective actions required to minimize the consequences of the abnormal conditions.

The reactor safety protection system is maintained in an operating condition during all reactor and plant modes. The status of the reactor and plant is monitored at all times; protective actions are initiated as required. Operation of the reactor and plant during testing and maintenance periods with portions of the safety protection system out-of-service must be governed by operations technical specifications.

The functional components of this system include sensors and associated actuated equipment; signal processing, computer logic, surveillance and testing equipment; interface equipment with the main control room such as processing channel readouts, status indication, accident monitoring, and ability for operator manual intervention; and remote operator interface equipment for shutdown initiation and monitoring.

The safety protection system should provide for automatic shutdown of the reactor upon detection of abnormal conditions.

B.3.1 FUNCTIONS

The primary functions of the reactor safety protection equipment are related to radionuclide control and power generation. These functions are as follows:

- To protect the general public, facility personnel, and facility equipment against radionuclide releases by sensing process variables to detect abnormal conditions and to take corrective action to control the heat generation rate within acceptable limits.

- To protect facility equipment to limit risk of damage from abnormal power generation by sensing process variables to detect abnormal conditions and to take corrective action to maintain parameters within acceptable limits.

- To provide instrumentation to give sufficient information to determine status of protection system at all times so as to allow initiation of safety-related actions when necessary.

B.3.2 APPLICABLE GENERAL DESIGN CRITERIA (See Table A.3)

The general design criteria from Table A.3 that apply to the reactor-safety protection system are the following:

Category I: 1-5
Category II: 12, 13, 15, 19
Category III: 20-29
Category IV: 37, 46
Category V: None
Category VI: 63, 64

B.3.3 PRINCIPAL DESIGN CRITERIA

In addition to the general design criteria, there may be other principal design criteria for the reactor safety protection equipment that should be established for ANS. Examples of such criteria are the following:

- Design, fabricate, and provide the equipment to ensure that the thermal power is controlled during normal and abnormal conditions.

- Sense process variables to initiate a safe shutdown of the reactor when necessary and subsequently to maintain a safe shutdown condition.

- Design, fabricate, and install the reactor safety protection system to standards that ensure the ability to withstand specified environmental conditions without undue risk to the general public and the facility (per *Regulatory Guide* 1.97).

- Ensure the ability of the reactor safety protection system to perform its safety functions before, during, and for an adequate time after being subjected to all normal, abnormal, and safety-related design conditions.

- Design the protection system with consideration for redundancy, diversity, and single and multiple failures and design the system to be "fail safe."

B.4 CORE PRESSURE BOUNDARY TUBE AND HEAVY-WATER REFLECTOR TANK

The ANS fuel element and irradiation experiments that comprise the reactor core will be located within a closely surrounding core pressure boundary tube. Surrounding the core pressure boundary tube will be a

larger tank containing the heavy-water reflector, the cold sources, and neutron guide tubes. The portion of the core pressure boundary tube adjacent to the reactor core will be subjected to very high levels of neutron and gamma ray irradiation. Significant gamma- and neutron-originated heat deposition will have to be removed from the core pressure boundary tube. Because of the high neutron flux, the core pressure boundary tube may have to be replaced approximately every 6 months to a year.

The core pressure boundary tube must be designed to withstand internal pressure up to its design pressure [to be determined (TBD)]. A pressure relief system must also be provided. In accordance with ASME rules, the pressure relief system must relieve within 2% of the design pressure and must prevent primary coolant pressure within the core pressure boundary tube from exceeding 110% of design pressure for all normal and upset transients.

The heavy-water reflector tank will operate at low pressure. One or more cold sources will be located in close proximity to the core pressure boundary tube (and the reactor core) within the reflector tank. One, at least, may contain liquid deuterium and operate at a temperature of about 20 to 30 K. Significant heat removal capability must be provided for the cold sources. Neutron guide tubes will be positioned to direct neutrons from the cold sources to various experiments located in the guide hall.

B.4.1 FUNCTIONS

The primary functions of the core pressure boundary tube and heavy-water reflector tank are related to local heat generation and radiological control functions and are as follows:

- The core pressure boundary tube shall contain the primary coolant inventory flowing through the tube during power production, shutdown, refueling, and startup/shutdown transient conditions.

- The core pressure boundary tube system shall limit the release of radionuclides during all modes of normal operation, including AOOs. The system must also maintain sufficient integrity during design basis events to prevent major configuration changes that would preclude providing adequate coolant to the fuel element or safe shutdown of the reactor.

- The heavy-water reflector tank and associated fluid system must contain the heavy water, including relatively significant amounts of tritium generated during reactor operation.

- The cold source must provide a source of cold neutrons for experimental purposes.

- The neutron guide tubes must direct the neutrons from the source to the experiments located in the guide hall.

B.4.2 APPLICABLE GENERAL DESIGN CRITERIA (See Table A.3)

The general design criteria from Table A.3 which apply to the core pressure boundary tube and heavy-water reflector tank system are the following:

Category I: 1-5
Category II: 14, 15
Category III: None
Category IV: 30-32, 34, 44-46
Category V: None
Category VI: 60, 61

B.4.3 PRINCIPAL DESIGN CRITERIA

The principal design criteria for the core pressure boundary tube and heavy-water reflector tank are related to power generation and radionuclide control. Examples are the following:

- The core pressure boundary tube system must be designed in accordance with the ASME boiler and pressure vessel code for nuclear vessels to ensure integrity during normal and abnormal operating conditions.

- The core pressure boundary tube must be designed to respond elastically to short-term pressure changes up to the specified maximum pressure (to be determined).

- The design of the core pressure boundary tube must accommodate in-service inspection.

- The design of the core pressure boundary tube must include consideration of transient events, which may represent conditions more severe than steady-state operation.

- The cold source must be designed to handle the heat deposition within the cold source; the operating temperature of the cold source should be 20 to 30 K.

- The ANS design must accommodate the off-normal condition of a failure in the cold source cooling system.

- The maximum reactivity effect associated with the sudden filling or the sudden voiding of any beam tube, guide, cold source, or vacuum chamber must be limited such that the reactor protection system has time to respond and prevent fuel damage.

- The cold source design must be such that any accidental energetic event, or explosion, that originates in the cold source is not capable of breaching the pressure boundary of the primary coolant system (including core pressure boundary tube) or of interfering with the ability to scram.

- The cold source shall be designed to preclude leakage of explosive moderator material and to detect such leakage if it occurs.

B.5 PRIMARY COOLANT SYSTEM

The primary coolant system consists of the heat exchangers, primary coolant circulation pumps, pressurizer pumps, and associated valves, filters, fill systems, and makeup systems. This system is designed to remove virtually all of the energy from the reactor core. The primary coolant, heavy water, is pumped through the reactor fuel element at a high mass flow rate. The coolant absorbs heat from the fuel element, passes to the primary heat exchangers, where the heat is given up to a secondary coolant (water), which is then circulated to a cooling tower, where the heat is dissipated to the atmosphere.

The reactor core of ANS operates at a very high power density. The reactor fuel element must be maintained under forced-flow cooling conditions at all times subsequent to initiation of the neutron fission process, including some time after core shutdown.

The primary coolant system operates at elevated pressure conditions [about 4 MPa (600 psi)], but only at moderate temperatures. The core outlet temperature is <100°C under normal operating conditions.

B.5.1 FUNCTIONS

The functions of the primary coolant system are related to heat generation and radionuclide control. These functions are as follows:

- To provide the means to transfer the heat generated within the reactor core to the secondary coolant.

- To provide for removal of heat deposited in the pressure vessel, associated structural elements, the control elements, and all internal experimental assemblies.

- To limit release of radionuclides during normal operations and during AOOs.

B.5.2 APPLICABLE GENERAL DESIGN CRITERIA (See Table A.3)

The general design criteria from Table A.3 that apply to the primary coolant system are the following:

Category I: 1-5
Category II: 14, 15, 17
Category III: 29
Category IV: 30-37, 44-46
Category V: 55-57
Category VI: 64

B.5.3 PRINCIPAL DESIGN CRITERIA

Examples of principal design criteria for the primary coolant system are as follows:

- Design the primary coolant system to satisfy specified operational parameters for steady-state, full-load, and part-load power production operations. The design must accommodate uncertainties such as flow resistances, flow distribution, and heat losses.

- During reactor core shutdown conditions, the primary coolant system must be designed to remove the core decay heat for either pressurized or depressurized conditions.

- During reactor core refueling operations, the primary coolant system must be designed to accommodate the refueling procedure and, as necessary, provide the ability to remove the core decay heat.

- The primary coolant system must provide the ability to go from shutdown, depressurized conditions to startup, pressurized, part-power conditions to pressurized, full-power conditions according to specified steps (TBD).

- The system must be designed to transfer heat energy to the heat exchangers for all postulated conditions of heat removal, including normal, pressurized power conditions, as well as depressurized conditions.

- The system must be designed to withstand cyclic loads resulting from postulated transients (TBD).

- The system must be designed to mitigate the consequences of internally and externally generated missiles.

- The fluid-containing portions of the system must be designed to prevent rupture during an SSE. Safety-related components shall be designed and qualified for environmental conditions.

- The system must be designed to respond to postulated transients without the loss of forced-flow cooling; these conditions constitute the anticipated operating occurrences for the primary system and are to be determined. This will include events such as loss of cooling due to some failure within the plant or loss of off-site power.

- The primary coolant system must be designed to maintain coolant inventory and pressure control in the event of primary coolant leakage; the level of leakage of the system must be determined.

- The reactor coolant system pressure boundary must be designed to the *ASME Boiler and Pressure Vessel Code* for nuclear components.

- The reactor coolant system design shall accommodate in-service inspection.

- The reactor coolant system pressure boundary should be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage or rapidly propagating failure.

- Purification systems will be provided to maintain the water quality and purity of the primary coolant and reflectors.

B.6 CONTAINMENT SYSTEM

The ANS must be designed to provide barriers to the release of core-generated fission products and to activation products within the core and primary coolant system.

The dominant source of radioactivity is from radionuclides generated within the fuel. Another significant source of radioactivity is the isotope production rods positioned near the core. The design of the fuel elements and the isotope production rods and associated reactivity control and cooling systems should provide high assurance that there is no release of fission products from the fuel. The fuel material in the fuel element combined with the fuel plate cladding represent the first barrier to release of fission products.

Should fission or isotope production products be released to the primary coolant, they will be transported throughout the primary circuit by the coolant. The coolant system pressure boundary is an additional barrier to the release of fission products. A purification system must be provided to remove gaseous and metallic fission products from the coolant. Condensible fission products will be removed by deposition, or plateout, on the various wetted surfaces within the primary coolant system.

Activation product radionuclides will be produced within the reactor core and circulating primary coolant system. These will include impurities in the circulating coolant, release of tritium generated

within the core and primary coolant system as well as in the reflector tank, and particulate matter (gases, liquids, and solids) that find their way into the circulating primary coolant.

The containment system consists of the containment building, or buildings, which may comprise primary and secondary structural barriers and the associated air handling and treatment systems.

The containment buildings housing the reactor and the primary coolant system must serve to limit or control the release of radionuclides which are generated within these systems and which find their way into the circulating coolant system or leak into the building atmosphere. The containment buildings represent the final barrier to release of radionuclides.

B.6.1 FUNCTIONS

The containment systems are related to both the neutron generation and radionuclide control functions. These functions are as follows:

- With respect to the neutron generation function, the containment system must provide the space to accommodate the reactor core and the systems and components associated with the fission process.

- The containment system must provide an appropriate environment, including specified equipment radiation limits, for the systems housed within the buildings.

- The containment system maximum design pressure will accommodate the most severe design basis event.

- The containment system must protect its own capabilities and those of the contained systems and components from various hazards, both internally and externally generated.

- Regarding radionuclide control, the containment systems must maintain the required reactor, core pressure boundary tube, and primary coolant system geometry. The design must ensure that these systems maintain structural integrity for specified design basis conditions.

- The consequences of accident conditions imposed on the containment system must be such as to ensure continued operability of the reactor safety protection system.

- The containment system must limit, or control, both on-site and off-site radioactive exposure on a routine basis; the source of such exposure would be the release of radionuclides stemming from leakage or the production of activation products.

- The containment building also serves to limit either personnel access or area radiation levels as required to control occupational radiation exposure.

B.6.2 APPLICABLE GENERAL DESIGN CRITERIA (See Table A.3)

The general design criteria from Table A.3 which apply to the containment systems are the following:

Category I: 1-5
Category II: 16
Category III: None
Category IV: 38-43
Category V: 50-57
Category VI: 60, 64

B.6.3 PRINCIPAL DESIGN CRITERIA

The principal design criteria for the containment system are related to both the neutron generation and radionuclide control functions. Examples are as follows:

- The containment system must be designed to ensure continued functional performance capability of the systems and components for the following design basis conditions:

  - wind and tornado loadings,
  - flooding,
  - missiles,
  - dynamic effects associated with postulated pipe rupture,
  - seismic design conditions, and
  - structural design considerations for combinations of the preceding design basis conditions.

- The containment system must be designed to satisfy 10 *CFR* 100 limits, with the borders of the exclusion area and low population zone defined to extend only to the boundary of the DOE property surrounding the reactor.

- Routine off-site radionuclide releases are to be limited to ALARA to meet the dose guidelines of 10 *CFR* 50, Appendix I.

- Radiation levels in the generally accessible areas of the containment building during power generation operations must be limited to (TBD) mrem/h during all modes of normal plant operation; these radiation levels, for occupational radiation control, should permit access to these areas for at least (TBD) hours a week. (Note: The values TBD should be consistent with ALARA requirements.)

- During reactor core shutdown, the radiation levels throughout the containment buildings are to be limited as required to maintain an overall facility population exposure to no more than (TBD) percent of the 10 *CFR* 20 limits. These radiation levels should permit access and enough time to accomplish anticipated maintenance, inspection, repair, and refueling activities. (Note: The values TBD should be consistent with ALARA requirements.)

- Automatic systems shall be provided to detect and extinguish fires in any part of the containment buildings that contain, or might contain, combustible materials.

- Provisions should be included for preventing the accumulation of potentially explosive mixtures of gases that may be generated or released.

- The maximum design pressure for the containment design will be greater than the most challenging anticipated accident pressures calculated for design basis events (see Sect. A.2).

B.6.4 CONTAINMENT PERFORMANCE

The design of the containment building for containment performance is determined by four basic considerations: (1) leak rate, (2) internal fission product removal, (3) fission product removal, and (4) effluent filtration efficiency. Each of these considerations must be addressed in the design of the systems and components comprising the containment building.

There are several approaches to the design of the containment building; these include single containment, ventilated or unventilated or with internal ventilation, and double containment with ventilated outer containment or with ventilated outer and inner containment. The efficiency of these various approaches must be determined by evaluating the impact of various containment design parameters on the level of off-site doses.

The dose limits regulating off-site releases for design basis events are specified in 10 *CFR* 100. At the specified exclusion area boundary, the dose limits are 25-rem whole body and a 300-rem thyroid dose in a 2-h period following postulated fission product release. At the low population zone boundary, the limit is 25-rem whole body and 300-rem thyroid for the duration of the postulated accident.

The evaluations performed to date for ANS indicate that a tight containment alone is not sufficient to assure acceptable containment performance; filtration and/or in-containment removal of fission products are needed. The results indicate that filtration appears a most effective removal mechanism. The combination of tight containment and filtration should provide the containment performance desired and required for ANS.

Numerical goals for containment functional performance have been tentatively set as the following:

- A maximum containment atmosphere leak rate of 4% of the containment free volume per day at containment design pressure and either of the following:

- 99% efficient nonnoble gas nuclide (semivolatiles) removal from containment effluent or

- 1300%/h in-containment removal rate.

APPENDIX I

NRC REGULATORY GUIDES THAT MAY BE APPLICABLE TO THE ANS PROJECT

Division 2 — Research and Test Reactors

Guide Number	Title
2.1	Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors
2.2	Development of Technical Specifications for Experiments in Research Reactors
2.3	Quality Verification for Plate-Type Uranium-Aluminum Fuel Elements for Use in Research Reactors
2.4	Review of Experiments for Research Reactors
2.5	Quality Assurance Program Requirements for Research Reactors
2.6	Emergency Planning for Research Reactors (for Comment) (Draft: HF 201-1 Proposed Revision 1, published 3/82)

Division 1 — Power Reactors

The following pages listing Regulatory Guides for power reactors were excerpted directly from "Screening of NRC Regulatory Guides, Standard Review Plan Appendices, and Branch Technical Positions for Applicability to DOE Category A Reactors," by W. J. Brynda and C. E. Tanguay (Brookhaven National Laboratory Informal Report BNL-38524, September 1986). These pages list the regulatory guides under the section of the safety analysis report (SAR) that they most directly deal with. The ANS SAR will have the same section structure because DOE reactors follow the NRC Standard Format.

"Excerpted from BNL-38524"

16.0 CROSS-INDEX

This section contains a cross-index between the information in this summary report and the USNRC Division 1 Regulatory Guides (R.G.) or Standard Review Plant (SRP) Branch Technical Position (BTP) and Appendix source documents. The index is arranged according to the sections in the report and provides the reader with a list of the source documents pertinent to each of the subjects summarized. Revision levels, dates, and supplement designations have been omitted.

"Excerpted from BNL-38524"

Should the reader desire additional information on the summarized reactor safety subjects, the assessments of the R.G. and SRP BTP/ Appendix documents considered potentially applicable to Category A reactors appears in consecutive order in Appendices A and B, respectively. No attempt was made to embody all of the indepth information contained in each Regulatory Guide or Standard Review Plan source document into its related assessment summary. It was assumed that individuals concerned with applying the safety guidelines and criteria of the Regulatory Guides or SRP documents would want to scrutinize the published documents.

SUMMARY REPORT SECTION	R.G./SRP DOCUMENT	DOCUMENT TITLE
2. SITING		
2.1 Geography and Demography		
2.1.1	R.G. 1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants.
2.2 Meteorology		
2.2.1	R.G. 1.23	Onsite Meteorological Program.
2.2.2	R.G. 1.76	Design Basis Tornado for Nuclear Power Plants.
2.3 Hydrology		
2.3.1	R.G. 1.59	Design Basis Floods for Nuclear Power Plants.
2.4 Geology and Seismology		
2.4.1	R.G. 1.132	Site Investigations for Foundations of Nuclear Power Plants.
2.4.2	R.G. 1.138	Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants.
3. DESIGN, FABRICATION, AND MATERIALS		
3.1 Classification of Systems and Structures		
3.1.1	R.G. 1.26	Quality Group Classification and Standards for Water, Steam, and Radioactive Waste-Containing Components of Nuclear Power Plants.

3.1.2	R.G. 1.29	Seismic Design Classification.
3.1.3	SRP 3.6.1	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.
3.1.4	R.G. 1.117	Tornado Design Classification.

3.2 Wind and Tornado Design Including Missile Protection

3.2.1	R.G. 1.76	Design Basis Tornado for Nuclear Power Plants.
3.2.2	R.G. 1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plants.
3.2.3	R.G. 1.115	Protection Against Low-Trajectory Turbine Missiles.
3.2.4	SRP 3.5.3	Permissible Ductility Ratio for Overall Damage Prediction.

3.3 Flood Protection and Water Level Design

3.3.1	R.G. 1.102	Flood Protection for Nuclear Power Plants.
3.3.2	R.G. 1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants.
3.3.3	R.G. 1.135	Normal Water Level and Discharge at Nuclear Power Plants.

3.4 Seismic Design

3.4.1	R.G. 1.12	Instrumentation for Earthquakes.
3.4.2	R.G. 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants.
3.4.3	R.G. 1.61	Damping Values for Seismic Design of Nuclear Power Plants.
3.4.4	R.G. 1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis.

3.4.5	R.G. 1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.

3.5 Design of Seismic Structures

3.5.1	R.G. 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.
3.5.2	R.G. 1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments).
3.5.3	SRP 3.8.4	Interim Criteria for Safety-Related Masonry Wall Evaluation.

3.6 Mechanical System and Component Design

3.6.1	SRP 3.6.2 BTP MEB 3-1	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.
3.6.2	SRP 3.9.3 Appendix A	Stress Limits for ASME Class 1, 2, and 3 Components and Component Supports of Safety-Related Systems and Class CS Core Support Structures Under Specified Service Loading Combinations.
3.6.3	R.G. 1.84	Design and Fabrication of Code Case Acceptability-ASME Section III, Division 1.
3.6.4	R.G. 1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors.
3.6.5	R.G. 1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports.
3.6.6	R.G. 1.130	Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports

3.6.7	R.G. 1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants.

3.7 Seismic and Environmental Qualification of Mechanical and Electrical Equipment

3.7.1	R.G. 1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants.
3.7.2	R.G. 1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants.
3.7.3	R.G. 1.89	Environmental Qualification of Certain Equipment Important to Safety for Nuclear Power Plants.
3.7.4	R.G. 1.100	Seismic Qualification of Electrical Equipment for Nuclear Power Plants.
3.7.5	R.G. 1.131	Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants.

3.8 Welding and Materials

3.8.1	R.G. 1.31	Control of Ferrite Content in Stainless Steel Weld Metal.
3.8.2	R.G. 1.34	Control of Electroslag Weld Properties.
3.8.3	R.G. 1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel.
3.8.4	R.G. 1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components.
3.8.5	R.G. 1.44	Control of the Use of Sensitized Stainless Steel.

3.8.6	R.G. 1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel.
3.8.7	R.G. 1.65	Materials and Inspections for Reactor Vessel Closure Studs.
3.8.8	R.G. 1.71	Welder Qualification for Areas of Limited Accessibility.
3.8.9	R.G. 1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin.
3.8.10	R.G. 1.85	Materials Code Case Acceptability—ASME Section III, Division 1.
3.8.11	R.G. 1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials.

4. DECOMMISSIONING

4.1 Termination

4.1.1	R.G. 1.86	Termination of Operating Licenses for Nuclear Reactors.

5. REACTOR AND REACTOR COOLANT SYSTEMS

5.1 Reactor

5.1.1	R.G. 1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing.
5.1.2	SRP 4.2 Appendix A	Evaluation of Fuel Assembly Structural Response to Externally Applied Forces.
5.1.3	R.G. 1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification.

5.2 Reactor Coolant and Residual Heat Removal Systems

5.2.1	R.G. 1.2	Thermal Shock to Reactor Pressure Vessels.
5.2.2	SRP 5.3.2 BTP MTEB 5-2	Fracture Toughness Requirements.

5.2.3	R.G. 1.14	Reactor Coolant Pump Flywheel Integrity.
5.2.4	R.G. 1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems.
5.2.5	R.G. 1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors.
5.2.6	R.G. 1.139	Guidance for Residual Heat Removal (RHR).
5.2.7	SRP 5.4.7 BTP RSB 5-1	Design Requirements of the Residual Heat Removal System.
5.2.8	SRP 9.2.5 BTP ASB 9-2	Residual Decay Energy for Light-Water Reactors for Long-Term Cooling.

6. ENGINEERING SAFETY FEATURES

6.1 Emergency Core Cooling Systems

6.1.1	R.G. 1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps.
6.1.2	R.G. 1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.
6.1.3	SRP 6.1.1 BTP MTEB 6-1	pH for Emergency Coolant Water for PWRs.
6.1.4	SRP 6.3 BTP RSB 6-1	Piping from the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps.

6.2 Reactor Building

6.2.1	R.G. 1.7	Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident.
6.2.2	SRP 6.2.5 Appendix A	Description of Combustible Gas Analyzer Program.
6.2.3	R.G. 1.11	Instrument Lines Penetrating Primary Reactor Containment.

6.2.4	R.G. 1.63	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants.
6.2.5	R.G. 1.141	Containment Isolation Provisions for Fluid Systems.
6.2.6	SRP 6.2.1.5 BTP CSB 6-1	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation.

6.3 ESF Ventilation System

6.3.1	R.G. 1.52	Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atomsphere Cleanup System Air Filtration and Adsorption Units for Light-Water-Cooled Nuclear Power Plants.

7. INSTRUMENTATION AND CONTROL

7.1 Protection Systems and Safety-Related Instrumentation

7.1.1	R.G. 1.22	Periodic Testing of Protection System Actuation Functions.
7.1.2	SRP Appendix 7-A BTP ICSB 22	Guidance for Application of Regulatory Guide 1.22.
7.1.3	R.G. 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.
7.1.4	SRP Appendix 7-A BTP ICSB 21	Guidance for Application of Regulatory Guide 1.47.
7.1.5	SRP Appendix 7-A BTP ICSB 4	Requirements of Motor-Operated Valves in the ECCS Accumulator Lines.
7.1.6	SRP Appendix 7-A BTP ICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System.
7.1.7	R.G. 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems.

"Excerpted from BNL-38524"

7.1.8	SRP Appendix 7-A BTP ICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors.
7.1.9	R.G. 1.62	Manual Initiation of Protective Actions.
7.1.10	SRP Appendix 7-A BTP ICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service.
7.1.11	SRP Appendix 7-A BTP ICSB 26	Requirements for Reactor Protection System Anticipatory Trips.
7.1.12	R.G. 1.75	Physical Independence of Electric Systems.
7.1.13	R.G. 1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants and Environs – Conditions During and Following an Accident.
7.1.14	SRP Appendix 7-A BTP ICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode.
7.1.15	R.G. 1.105	Instrument Set Points.
7.1.16	R.G. 1.118	Periodic Testing of Electric Power and Protection Systems.
7.1.17	R.G. 1.151	Instrument Sensing Lines.
7.1.18	R.G. 1.152	Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants.
7.1.19	R.G. 1.153	Criteria for Power, Instrumentation, and Control Portions of Safety Systems.
7.1.20	SRP 7.1, Appendix A	Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety.

7.1.21	SRP 7.1, Appendix B	Guidance for Evaluation of Conformance to IEEE STD 279.

8. ELECTRIC POWER SYSTEMS

8.1 Class 1E Electric Power Systems

8.1.1	R.G. 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems.
8.1.2	R.G. 1.9	Selection, Design, and Qualification of Diesel-Generator Units Used As Standby (Onsite) Electric Power Systems at Nuclear Power Plants.
8.1.3	SRP Appendix 8-A BTP PSB 2	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status.
8.1.4	R.G. 1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants.
8.1.5	SRP Appendix 8-A BTP PSB 1	Adequacy of Station Electric Distribution System Voltages.
8.1.6	R.G. 1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments.
8.1.7	R.G. 1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants.
8.1.8	R.G. 1.93	Availability of Electric Power Sources.
8.1.9	R.G. 1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves.
8.1.10	SRP Appendix 8-A BTP ICSB 18	Application of the Single-Failure Criterion to Manually-Controlled Electrically-Operated Valves.

8.1.11	R.G. 1.108	Periodic Testing of Diesel Generator Units Used As Onsite Electric Power Systems at Nuclear Power Plants.
8.1.12	R.G. 1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants.
8.1.13	R.G. 1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants.

9. AUXILIARY SYSTEMS

9.1 Steam Generators, Auxiliary Feedwater Systems, and the Ultimate Heat Sink

9.1.1	R.G. 1.27	Ultimate Heat Sink for Nuclear Power Plants.
9.1.2	SRP Appendix 7-A BTP ICSB 13	Design Criteria for Auxiliary Feed-water System.
9.1.3	SRP 10.4.9 BTP ASB 10-1	Design Guidelines for Auxiliary Feed-water System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants.
9.1.4	SRP 10.4.7 BTP ASB 10-2	Design Guidelines for Avoiding Water Hammer in Steam Generators.
9.1.5	SRP 5.4.2.1 BTP MTEB 5-3	Monitoring of Secondary Side Chemistry in PWR Steam Generators.

9.2 Diesel Fuel Systems and Reactor Fuel Handling and Storage Systems

9.2.1	R.G. 1.13	Spent Fuel Storage Facility Design Basis.
9.2.2	R.G. 1.137	Fuel Oil Systems for Standby Diesel Generators.
9.2.3	SRP 3.8.4 Appendix D	Technical Position on Spent Fuel Pool Racks.

9.3 Hazardous Gas Protection and Normal Ventilation Filtration Systems

9.3.1	R.G. 1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release.
9.3.2	R.G. 1.140	Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.

9.4 Fire Protection Systems

9.4.1	R.G. 1.120	Fire Protection Guidelines for Nuclear Power Plants.
9.4.2	SRP 9.5.1 BTP CMEB 9.5-1	Guidelines for Fire Protection for Nuclear Power Plants.

10. RADIOACTIVE WASTE MANAGEMENT

10.1 Radioactive Waste (Radwaste) Systems

10.1.1	R.G. 1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors.
10.1.2	R.G. 1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.
10.1.3	SRP 11.4 BTP ETSB 11-3	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Plants.
10.1.4	SRP 11.3 BTP ETSB 11-5	Postulated Releases Due to a Waste Gas System Leak or Failure.
10.1.5	SRP 11.4 Appendix 11.4-A	Design Guidance for Temporary Onsite Storage of Low Level Radioactive Waste.

10.1.6	SRP 11.5 Appendix 11.5-A	Design Guidance for Radiological Effluent Monitors Providing Signals for Initiating Termination of Flow or Other Modification of Effluent Stream Properties.

11. RADIATION PROTECTION

11.1 Radiation Protection Methods

11.1.1	R.G. 1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants.
11.1.2	R.G. 1.69	Concrete Radiation Shields for Nuclear Power Plants.
11.1.3	R.G. 1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.
11.1.4	R.G. 1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors.
11.1.5	R.G. 1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I.

12. CONDUCT OF OPERATIONS

12.1 Qualification and Training

12.1.1	R.G. 1.8	Qualification and Training of Personnel for Nuclear Power Plants.
12.1.2	R.G. 1.134	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses.

12.2 Operational Conduct

12.2.1	R.G. 1.114	Guidance on Being Operator at the Controls of a Nuclear Power Plant.

12.3 Emergency Preparedness

12.3.1	R.G. 1.101	Emergency Planning and Preparedness for Nuclear Power Reactors.

13. INITIAL TEST PROGRAM

13.1 General Guidance for Initial Test Programs

13.1.1	R.G. 1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants.

13.2 Specific Initial and Preoperational Tests

13.2.1	R.G. 1.168.2	Initial Startup Program to Demonstrate Shutdown Capability for Water-Cooled Nuclear Power Plants.
13.2.2	R.G. 1.168.3	Preoperational Testing of Instrument and Control Air Systems.
13.2.3	R.G. 1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors.
13.2.4	R.G. 1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Power Load Assignments.

14. ACCIDENT ANALYSIS

14.1 Reactor Power Level

14.1.1	R.G. 1.49	Power Levels for Nuclear Power Plants.

14.2 Postulated Accidents, Diffusion Models, and Assumptions

14.2.1	R.G. 1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

"Excerpted from BNL-38524"

14.2.2	R.G. 1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors.
	SRP 15.6.5 Appendix A	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution.
	SRP 15.6.5 Appendix B	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment.
14.2.3	R.G. 1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors.
	SRP 15.4.8 Appendix A	Spectrum of Rod Ejection Accidents (PWR).
14.2.4	SRP 15.1.5	Radiological Consequences of Main Steam Line Failures Outside Containment of PWR.
14.2.5	R.G. 1.24	Assumptions Used for Evaluating the Potential Consequences of a Pressurized Water Reactor Gas Storage Tank Failure.
14.2.6	R.G. 1.25	Assumptions for Evaluating the Potential Consequences of a Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.
14.2.7	R.G. 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During Postulated Hazardous Chemical Releases.

"Excerpted from BNL-38524"

15. QUALITY ASSURANCE

15.3 General

15.3.1	R.G. 1.74	Quality Assurance Terms and Definitions.
15.3.2	R.G. 1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records.
15.3.3	R.G. 1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants.
15.3.4	R.G. 1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants.

15.4 Design and Construction

15.4.1	R.G. 1.28	Quality Assurance Program Requirements (Design and Construction).
15.4.2	R.G. 1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants.
15.4.3	R.G. 1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.
15.4.4	R.G. 1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants.
15.4.5	R.G. 1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants.

15.4.6 Installation, Inspection, and Testing

15.4.6.1	R.G. 1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Test Personnel.
15.4.6.2	R.G. 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment.

		15.4.6.3	R.G. 1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.
		15.4.6.4	R.G. 1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems.
	15.4.7		R.G. 1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.
15.5	Operations			
	15.5.1		R.G. 1.33	Quality Assurance Program Requirements (Operations).
	15.5.2	Inservice Inspection		
		15.5.2.1	R.G. 1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations.
		15.5.2.2	R.G. 1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes.
		15.5.2.3	R.G. 1.147	Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1.

APPENDIX II

RADIATION STANDARDS FOR PROTECTION OF THE PUBLIC IN THE VICINITY OF DOE FACILITIES

A. DOSE LIMITS

1. All Pathways

The effective dose equivalent for any member of the public from all routine DOE operations* (natural background and medical exposures excluded) shall not exceed the values given below.

Length of exposure	Effective dose equivalent‡ [mrem/year (mSv/year)]
Occasional annual exposure	500 (5)
Prolonged period of exposure	100 (1)

No individual organ shall receive a committed effective dose equivalent of 5 rem/year (50 mSv/year) or greater.

2. Air Pathway Only (Limits of 40 CFR 61, Subpart H)

Type of exposure	Effective dose equivalent [mrem/year (mSv/year)]
Whole-body dose	25 (0.25)
Any organ	75 (0.75)

B. ACTION LEVELS

1. To preclude an individual in the general population from receiving more than 100-mrem/year effective dose equivalent, a DOE administrative action level is established at 25 mrem/year (excluding medical and natural background exposures) for its routine operations. This dose value is not a limit but an administrative threshold that will require a specific evaluation of the magnitude of identifiable exposures to an exposed individual by the responsible DOE field office. A copy of the investigation report will be transmitted to

---

*Routine DOE operations mean normal, planned operations and do not include actual or potential accidental or unplanned releases.

‡Effective dose equivalent will be expressed in rem (or millirem), with the corresponding value in sievert (or millisievert) in parenthesis.

the relevant Program Office(s) and the Deputy Assistant Secretary for Environment, Safety and Health, PE-20.

2. To preclude exceeding the air pathway limits in item A.2 above, field offices shall notify the relevant program office and EH-10 of calculated or anticipated doses to individual members of the population in excess of one-half the specified dose equivalent limit (12.5 mrem).

C. AS LOW AS REASONABLY ACHIEVABLE (ALARA)

Field offices and contractors shall implement programs to assure that exposures resulting from DOE operations to members of the public are maintained ALARA. The ALARA programs shall be documented. Each field office shall periodically audit contractor ALARA programs and contractor progress in attaining ALARA conditions. Assessments of ALARA must include best estimates of actual effective dose equivalent to individual members of the populations as well as collective dose equivalent to a distance of 80 km from the site.

D. DEMONSTRATION OF COMPLIANCE

Demonstration of compliance with these criteria shall normally be done by the following procedure:

a. Calculation of external exposure and internal intakes by use of effluent data and environmental pathway models approved by the Office of Operational Safety, EH, and/or environmental measurements.

b. Calculation of total effective dose equivalent using the Draft Final Committed Dose Equivalent Tables.

For DOE facilities with airborne releases subject to 40 CFR 61, Subpart H, the AIRDOS-EPA model must be used except as otherwise approved by EPA. Compliance will be determined by calculation of the dose to members of the public at the point of maximum annual concentration in an unrestricted area where any member of the public resides or abides.

E. ACCIDENTS

The exposure limits given above are for routine DOE operations and are not intended for use as criteria to evaluate the acceptability of postulated accidents. Planning for the prevention or mitigation of accidents and their effects shall be accomplished in accordance with the requirements of DOE 5480.1A, Chapter V, "Safety of Nuclear Facilities," and Chapter VI, "Safety of Department of Energy-Owned Reactors."

F. APPLICABLE DEFINITIONS

The following definitions are derived from ICRP Publication No. 42 (1984).

Dose equivalent — The product of the absorbed dose (in rads) in the tissue of interest, a quality factor (specified by the ICRP), and any other modifying factors specified by the ICRP.

Effective dose equivalent — A quantity defined by the sum

$$\Sigma_T W_T H_T ,$$

where W_T is the weighting factor specified by the ICRP to represent the production of the stochastic risk resulting from irradiation of tissue T to the total risk when the whole body is irradiated uniformly and H_T is the mean dose equivalent in tissue T. H_T may be from external or internal sources. Values of W_T have been specified by the ICRP and were used in preparing the Draft Final Committed Dose Equivalent Tables.

Committed dose equivalent — The time integral of the dose-equivalent rate in a particular tissue following an intake of radioactive material into the body. In keeping with ICRP recommendations, this period is set at 50 years for DOE.

Committed effective dose equivalent — The sum of the committed dose equivalents to individual tissues resulting from an intake, each multiplied by the appropriate weighting factor W_T.

Collective effective dose equivalent — The collective effective dose equivalent is equal to the integrated sum of individual effective dose equivalents times the number of individuals exposed. For purposes of this directive only, the collective effective dose equivalent shall be truncated at 80 km distance from site boundaries and at 50 years following each year's release.

G. EFFECTIVE DATE

These criteria shall be used for all DOE dose calculations effective July 1, 1985, including annual summary reports for CY 1985 and subsequent years.

APPENDIX III

TEXT OF GENERAL DESIGN CRITERIA

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

INTRODUCTION

DEFINITIONS

Nuclear Power Unit.
Loss of Coolant Accidents.
Single Failure.
Anticipated Operational Occurrences.

CRITERIA

	Number
I. *Overall Requirements:*	
Quality Standards and Records	1
Design Bases for Protection Against Natural Phenomena	2
Fire Protection	3
Environmental and Missile Design Bases	4
Sharing of Structures, Systems, and Components	5
II. *Protection by Multiple Fission Product Barriers:*	
Reactor Design	10
Reactor Inherent Protection	11
Suppression of Reactor Power Oscillations	12
Instrumentation and Control	13
Reactor Coolant Pressure Boundary	14
Reactor Coolant System Design	15
Containment Design	16
Electric Power Systems	17
Inspection and Testing of Electric Power Systems	18
Control Room	19
III. *Protection and Reactivity Control Systems:*	
Protection System Functions	20
Protection System Reliability and Testability	21
Protection System Independence	22
Protection System Failure Modes	23
Separation of Protection and Control Systems	24
Protection System Requirements for Reactivity Control Malfunctions	25
Reactivity Control System Redundancy and Capability	26
Combined Reactivity Control Systems Capability	27
Reactivity Limits	28
Protection Against Anticipated Operational Occurrences	29
IV. *Fluid Systems:*	
Quality of Reactor Coolant Pressure Boundary	30
Fracture Prevention of Reactor Coolant Pressure Boundary	31
Inspection of Reactor Coolant Pressure Boundary	32
Reactor Coolant Makeup	33
Residual Heat Removal	34
Emergency Core Cooling	35
Inspection of Emergency Core Cooling System	36
Testing of Emergency Core Cooling System	37
Containment Heat Removal	38
Inspection of Containment Heat Removal System	39
Testing of Containment Heat Removal System	40
Containment Atmosphere Cleanup	41
Inspection of Containment Atmosphere Cleanup Systems	42
Testing of Containment Atmosphere Cleanup Systems	43
Cooling Water	44
Inspection of Cooling Water System	45
Testing of Cooling Water System	46
V. *Reactor Containment:*	
Containment Design Basis	50
Fracture Prevention of Containment Pressure Boundary	51
Capability for Containment Leakage Rate Testing	52
Provisions for Containment Testing and Inspection	53
Systems Penetrating Containment	54
Reactor Coolant Pressure Boundary Penetrating Containment	55
Primary Containment Isolation	56
Closed System Isolation Valves	57
VI. *Fuel and Radioactivity Control:*	
Control of Releases of Radioactive Materials to the Environment	60
Fuel Storage and Handling and Radioactivity Control	61
Prevention of Criticality in Fuel Storage and Handling	62
Monitoring Fuel and Waste Storage	63
Monitoring Radioactivity Releases	64

INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of

nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which 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 electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in pressurized water reactors may be excluded from the design basis when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18—Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with

the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, test-

ing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of

remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1242, Pub. L. 94-79, 89 Stat. 413 (42 U.S.C. 5841))

[36 FR 3256, Feb. 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, Feb. 12, 1976; 43 FR 50163, Oct. 27, 1978; 51 FR 12505, Apr. 11, 1986]

Internal Distribution

1. C. W. Alexander
2. R. G. Alsmiller
3. B. R. Appleton
4. Y. Y. Azmy
5. J. R. Buchanan
6. N. C. J. Chen
7. G. L. Copeland
8. F. C. Difilippo
9 H. L. Dodds
10. F. F. Dyer
11. W. W. Engle
12. W. R. Gambill
13. R. K. Genung
14. J. A. Getsi
15. S. R. Greene
16. R. M. Harrington
17. D. L. Henderson
18. J. A. Johnson
19. R. A. Lillie
20. A. L. Lotts
21. B. S. Maxon
22-121. M. R. McBee
123. B. H. Montgomery
124. F. R. Mynatt
125. R. K. Nanstad
126. L. C. Oakes
127. R. E. Pawel
128. F. J. Peretz
129. R. T. Primm, III
130. C. C. Queen
131. J. S. Rayside
132. H. Reutler
132. A. Ruggles
133. T. L. Ryan
134. D. L. Selby
135. H. B. Shapira
136. W. D. Shults
137. P. B. Thompson
138. K. R. Thoms
139. D. B. Trauger
140. R. E. Uhrig
141. C. D. West
142. M. K. Wilkinson
143. G. T. Yahr
144. G. L. Yoder
145. A. Zucker
146. ORNL Patent Office
147. Central Research Library
148. Document Reference Section
149-150. Laboratory Records
151. Laboratory Records (RC)

External Distribution

152. Jack Dorning, University of Virginia, Department of Nuclear Engineering and Engineering Physics, Charlottesville, VA 22901.
153. B. Chalmers Frazer, Materials Sciences Division, U.S. Department of Energy, Germantown, ER.132, Washington, DC 20545.
154. Ralph Fullwood, Brookhaven National Laboratory, Upton, NY 11973.
155. J. A. Lake, Manager, Nuclear Engineering and Reactor Design, Idaho National Engineering Laboratory, P. O. Box 1625, Idaho Falls, ID 83415.
156. John Marks, Babcock and Wilcox Co., P. O. Box 785, Lynchburg, VA 24505.
157. John M. Ryskamp. Idaho National Engineering Laboratory, P. O. Box 1625, Idaho Falls, ID 83415.
158. J. L. Snelgrove, Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL 60439.
159. Iran Thomas, Director, Materials Sciences Division, U.S. Department of Energy, Germantown, ER.13, Washington, DC 20545.
160. D. K. Wilfert, Energy Programs, U.S. Department of Energy, Oak Ridge Operations, P. O. Box 2001, Oak Ridge, TN 37831.
161. R. J. Willard, U.S. Department of Energy, Oak Ridge Operations, P. O. Box 2001, Oak Ridge, TN 37831.
162-191. Technical Information Center, DOE, Oak Ridge, TN 37831