NRC REGULATIONS AND FUSION POWER

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INTRODUCTION

To take advantage of the differences between fusion and fission power in order to anticipate and influence possible Nuclear Regulatory Commission (NRC) regulation, one must understand how the NRC regulates fission power plants. This report provides an overview of the hierarchy of NRC regulations and describes those regulations. In addition, a discussion of reactor siting and emergency planning provides the basis for the NRC's position on these topics.

NRC REGULATIONS

LAWS

At the top of the hierarchy are the laws (acts) which provide the NRC with its mission and the authority to carry out that mission. The major laws that defined the Atomic Energy Commission (AEC) role and now define the NRC role in licensing nuclear power plants are:

- The Atomic Energy Act of 1954[1]
- The Energy Reorganization Act of 1974[2]
- The National Environmental Policy Act (NEPA) of 1969[3]

CODE OF FEDERAL REGULATIONS

Title 10 Parts 0 through 199 of the Code of Federal Regulations (10 CFR 0 through 10 CFR 199) addresses all the specific areas covered in laws enacted by Congress and provides the practices, policies and procedures on how the NRC will carry out its mission. The Code of Federal Regulations is federal law insofar as the laws enacted by Congress have given the NRC authority. In general the CFR describes facility configuration in broad terms rather than the specifics that are needed to develop a design. The needed specificity is usually provided in the lower tier documents discussed below.

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1 Atomic Energy Act of 1954 (Public Law 703, Act of August 30, 1954).

2 Energy Reorganization Act of 1974 (Public Law 93-438, Act of October 11, 1974).

3 National Environmental Policy Act (NEPA) of 1969 (Public Law 91-190, Act of January 1, 1970).

If the NRC has jurisdiction over fusion power plants, the Parts of 10 CFR that could affect the design, construction, operation and decommissioning of fusion power plant are listed in Table 1.

One of the more important parts of 10 CFR is 10 CFR 50, Appendix A ("General Design Criteria for Nuclear Power Plants") which provides both general and specific nuclear power plant design requirements for a water-cooled reactor. The General Design Criteria (GDCs) discuss the entire plant configuration, not just the cooling system. The cooling system, in the case of fusion, would probably not be water. A review was conducted of the GDCs with respect to a fusion power reactor design. Table 2 delineates those GDCs whose generic aspects should be considered in fusion power reactor design.

Appendices A, B and C provide additional information about the GDCs. Appendix A is a copy of 10 CFR 50, Appendix A. Appendix B is a copy of Section 3.1 of the Clinch River Breeder Reactor (CRBR) Project Preliminary Safety Analysis Report (PSAR). Since the CRBR was a Liquid Metal Fast Breeder Reactor (LMFBR) not all the existing GDCs were applicable. Through discussions with the NRC, alternate GDCs were developed and addressed in the LMFBR design as discussed in PSAR Section 3.1. Appendix C is a copy of Section 3.1 of the General Atomic Standard Safety Analysis Report (GASSAR). The GASSAR is for a High-Temperature Gas-cooled Reactor (HTGR) standard plant that is water-cooled (i.e., steam cycle for electric power generation). Since the GASSAR is for a standard plant from a reactor vendor, Section 3.1 only discusses those GDCs that are related to the nuclear reactor part of the design. The utility (applicant) would have to discuss the remaining GDCs in its PSAR. In the GDCs that are discussed, some are not applicable to the HTGR design.

REGULATORY GUIDES

Regulatory Guides ("Reg Guides") are documents that have been developed to describe such information as: methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations; techniques used by the NRC staff in evaluating specific problems or postulated accidents; and data needed by the NRC staff in its review. It is important to note that Reg Guides are not requirements only guidance. However, since it is the NRC's preferred way of accomplishing a 10 CFR requirement, many applicants comply with Reg Guides whenever reasonably possible, rather than taking an alternate approach. This not to say alternate approaches have not been accepted by the NRC, its just that if the guidance of a Reg Guide is complied with, the review and approval cycle is much shorter. There are numerous cases where utilities have taken alternate approaches than laid out in a Reg Guide and have still received NRC approval.

Unfortunately, in issuing the Reg Guides, the AEC and later the NRC changed their implementation methodology over time, which requires some explanation.

Safety Guides

When the AEC first issued Reg Guides in 1970, they were called "Safety Guides", dealt only with power reactor issues, and were numbered sequentially. Later when it became apparent that guidance would have to be supplied to other types of nuclear facilities as well as non-safety issues, the name was changed to "Regulatory Guides" and the numbering was changed to a decimal system where the number to the left of the decimal refers to the Regulatory Guide Division (discussed below) and the number to the right of the decimal are sequential within each division. All existing Safety Guides became part of Division 1 ("Power Reactors") and had "1." added in front of their existing numbers. Currently, there are less than ten still active Safety Guides, and while they appear in listings as Division 1 Reg Guides and are commonly called Reg Guides, the documents are still the originally issued Safety Guides.

For Comment

In the mid-70s, when new revisions to Reg Guides were proposed, the NRC would issue these proposed revisions as if they were new revisions but with "FOR COMMENT" diagonally in gray on the pages and the NRC Reg Guide listing would delineate these as the next revision to the Reg Guide with the words "for Comment" in parenthesis after the title. However, the NRC became lax in approving the "FOR COMMENT" revisions and lax in reissuing these revisions without the "FOR COMMENT" underlay. In addition, the NRC staff reviewers started treating the "FOR COMMENT" revisions as approved revisions. This new practice by the NRC staff raised many concerns in the nuclear industry as to what constituted official approved NRC guidance. The NRC addressed these concerns by revising its method for issuing draft Reg Guides in the following manner: 1) having a separate draft Reg Guide listing, 2) giving these drafts Task numbers instead of Reg Guide numbers (i.e., Task DG-1021 instead of "FOR COMMENT" Regulatory Guide 1.9, Revision 3), and 3) issuing them with cover sheets that clearly state that these are draft Reg Guides and that they do not represent official NRC positions. However, the NRC neither rescinded nor approved the existing "FOR COMMENT" Reg Guides. Presently there are less than thirty still active "FOR COMMENT" Reg Guides and the NRC and the nuclear industry treat them as approved Reg Guides.

The approximately 350 approved Reg Guides (this includes the "FOR COMMENT" Reg Guides) that are currently active are grouped in the following ten broad divisions:

Division #	**Division Title**
1	Power Reactors
2	Research and Test Reactors
3	Fuels and Materials Facilities
4	Environmental and Siting
5	Materials and Plant Protection
6	Products
7	Transportation
8	Occupational Health
9	Antitrust and Financial Review
10	General

A further discussion of Reg Guides is provided in Appendices D and E. Appendix D is a copy of NUREG/BR-0070 ("Guide to Types of NRC Formal Documents and Their Uses," USNRC, May 1984) and Appendix E is a copy of "Descriptions of Standing Order Items" (Office of the Secretary, USNRC, September 1993).

STANDARD REVIEW PLANS AND BRANCH TECHNICAL POSITIONS

The Standard Review Plans (SRPs) which are compiled in NUREG-0800 are the NRC's formal methodology for how the NRC staff is to conduct reviews of sections of a PSAR or an FSAR. The SRPs were developed to assure consistency in the NRC's review and identify which NRC Branch has primary review responsibility and if any NRC Branches have secondary review responsibility (the NRC is divided into "Offices" and some offices are further subdivided into "Branches"). While at first glance it may seem that SRPs don't affect design of a nuclear power plant, in reality the SRPs are just like Reg Guides, that is, since it is the NRC's preferred way of accomplishing their review, many applicants conform to the SRP guidance whenever practical, rather than taking an alternate approach.

In each SRP, the NRC discusses what NRC requirements/guidance (i.e., GDCs, Reg Guides, NUREGs, etc.) and what other guidance (i.e., codes, standards, reports, etc.) should be used by the NRC reviewer in approving a particular PSAR or FSAR section. Because the SRPs are the formal NRC approval mechanism and the SRPs provide a cross-reference to the accepted NRC guidance, the SRPs have become the "starting point document" for determining what NRC requirements/guidance affect any particular structure, system and/or component in a nuclear power plant.

Both the NRC and the nuclear industry realize that guidance constantly undergoes change (i.e., revised, superseded, retired, etc.) and that, after the fact ,the SRPs have to be revised (via page updates to NUREG-0800). Therefore, it is common practice to use the latest version of the guidance in the SRP cross-reference unless the NRC has formally stated otherwise.

In addition, attached to some SRPs are Branch Technical Positions (BTPs). The NRC Branches develop the BTPs to further define what is needed by the NRC Branch in order to complete the review. BTPs are just like Reg Guides and SRPs, that is, since it is the NRC's preferred way of accomplishing the review, many applicants conform to the BTP guidance whenever practical, rather than taking an alternate approach.

It is important to note that while the SRPs (and BTPs) are usually the starting point for determining what are the NRC requirements that will affect the design of a nuclear power plant, the SRPs are only mentioned in passing as part of the Regulatory Guide discussion in Appendix D and are briefly mentioned in the NUREG-0800 discussion in Appendix E.

INFORMATION NOTICES, BULLETINS, GENERIC LETTERS, ADMINISTRATIVE LETTERS

These documents are discussed in Appendices D and E, except for Administrative Letters (ALs) which are only discussed in Appendix E (ALs were first issued in July 1993). Additional information about these documents is provided below:

Information Notices

While Information Notices ("Info Notices" or "Notices") do not require a written response to the NRC, the addressees are required evaluate each Notice for applicability and to take appropriate action if necessary. In addition, the addressees must document the evaluation and any actions.

Inspection and Enforcement

Bulletins and Notices used to be issued by the NRC Office of Inspection and Enforcement ("IE" or "I&E"). The Office of Inspection and Enforcement was dissolved during an NRC reorganization. Its duties and Bulletins and Notices have been assumed by other NRC Offices. However, Bulletins and Notices are still unofficially called "IE Bulletins", "I&E Bulletins", "IE Notices" and "I&E Notices".

Bulletins and Generic Letters

While their descriptions in Appendices D and E make it sound like they have different functions, Generic Letters (GLs) and Bulletins are similar in their affect on and implementation by addressees. Also, some Bulletins and GLs provide additional design guidance to that already provided in SRPs, BTPs and Reg Guides. The NRC usually incorporates this guidance into the SRPs and/or the Reg Guides but not always (e.g., GL 86-10 on Fire Protection).

NUREGs

Appendices D and E provide descriptions about NUREGs in general and about specific NUREGs. By itself, a NUREG does not constitute guidance. However, if a Reg Guide, an SRP, a BTP, a Bulletin or a GL states that a particular NUREG should be used as guidance, that NUREG becomes guidance.

REACTOR SITING AND EMERGENCY PLANNING

For new nuclear power plants, the criteria for siting and emergency planning with respect to radiological releases, are contained in: 10 CFR 50.47[4]; 10 CFR 50.54(s)(1)[5]; 10 CFR 100.3[6]; 10 CFR 100.11[7] and NUREG-0396[8].

10 CFR 100.3 — This section contains the following three definitions:

> "*Exclusion area* means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion and removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to a facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations,provided that no significant hazards to the public health and safety will result."
>
> Examples of exclusion areas that contain non-licensee owned regions are San Onofre 2 & 3 (San Clemente, CA) and Waterford 3 (Taft, LA). San Onofre has agreements with California and the US Marine Corps to evacuate and close those portions of Interstate 5 and Camp Pendleton that are inside the exclusion area boundary. Waterford 3 has a similar agreement with the US Coast Guard with respect to the Mississippi River.

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4 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," § 50.47, "Emergency Plans."

5 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," § 50.54, "Conditions of Licenses," the "Conditions of Licenses" (commonly called "License Conditions") are requirements imposed on the plant operating license.

6 10 CFR 100, "Reactor Site Criteria," § 100.3, "Definitions."

7 10 CFR 100, "Reactor Site Criteria," § 100.3, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

8 NUREG-0396; EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978.

"*Low population zone* means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken on their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents in the area."

"*Population center distance* means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents."

10 CFR 100.11 — This section contains the method for determining the boundaries/distances for the above three definitions:

"An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure."

"A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure."

"A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration."

10 CFR 50.47(c)(2) and 10 CFR 50.54(s)(1) — Both these sections state the following about Emergency Planning Zones (EPZs):

> "Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway."

NUREG-0396, Appendix I, Section B.1, pages I-4 through I-6 — This section provides the rationale for the size of the plume exposure pathway[9] EPZ and the ingestion exposure pathway[10] EPZ, as follows:

> "Under NRC Regulations, the site/reactor design combination must be such that the consequences of design basis accidents are below the plume exposure guidelines of 10 CFR Part 100. The design basis loss-of-coolant accident (DBA-LOCA) has been typically the most severe design basis accident in that it results in the largest calculated of any accident in this class. The DBA-LOCA is not a realistic accident scenario in that the release magnitudes are much severe than would be realistically expected and may exceed that of some core-melt type accidents. A best estimate assessment of the release following a LOCA would be significantly smaller than the DBA-LOCA used for siting purposes. An analysis of this accident has been performed for most of the power plants licensed or under review by NRC to determine the dose/distance relationships as computed by traditionally conservative assumptions used under 10 CFR Part 100 requirements. Results of this study are presented later in this appendix. The study concluded that the higher PAG[11] plume exposures of 25 rem (thyroid) and 5 rem (whole body) would not be exceeded beyond 10 miles for any site analyzed. Even under the most restrictive PAG plume exposure values of 5 rem to the thyroid and 1 rem whole body, over 70 percent of the plants would not require any consideration of emergency responses beyond 10 miles. It should be noted that even for the DBA-LOCA, the lower range of the plume PAGs would likely not be exceeded outside the low population zone (LPZ) for average meteorological conditions."

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9 Plume exposure pathway — The principal exposure sources from this pathway are (a) whole body exposure to gamma radiation from the plume and from deposited material and (b) inhalation exposure from the passing radioactive plume. The time of potential exposure could range from hours to days. (NUREG-0396, Glossary, page 28)

10 Ingestion exposure pathway — The principal exposure from this pathway would be from ingestion of contaminated water or foods such as milk of fresh vegetables. The time of potential exposure could range from hours to months. (NUREG-0396, Glossary, page 27)

11 Protective Action Guide — Projected absorbed dose to individuals in the general population which warrants protective action following a contaminating event. (NUREG-0396, Glossary, page 28)

"For the ingestion pathways, under the same DBA-LOCA conditions, the downwind range within which a PAG of 1.5 rem thyroid could be exceeded would be limited to within 50 miles even under conservative 10 CFR 100 assumptions. The 50 mile distance is also justified as a maximum planning distance because of likely significant wind shifts within this distance that would further restrict the radius of spread of radioactive material."

The above raises the following question:

Is there a relationship or a conflict between the exclusion area, the low population zone and the population center distance of 10 CFR 100 and the plume exposure pathway EPZ and the ingestion pathway EPZ of 10 CFR 50?

The answer is:

No. The regions discussed in 10 CFR 100 address a different set of issues than the regions discussed in 10 CFR 50.

The 10 CFR 100 siting criteria[12] for a nuclear power plant require that the consequences from the most severe design basis accidents analyzed should not result in exposures in excess of those provided in 10 CFR 100.11. The 10 CFR 50 EPZ requirements establish what type of emergency planning activities should be implemented in the vicinity of a nuclear power plant.

12 Besides 10 CFR 100 and 10 CFR 100, Appendix A, additional siting guidance can be found in USNRC Regulatory Guide 4.7 ("General Site Suitability Criteria for Nuclear Power Stations," Revision 1, November 1975).

NUREG-0396[13] provides the following clarification:

> "The dose guideline values in 10 CFR Part 100 do not constitute acceptable limits for emergency doses to the public under accident conditions. The numerical values of 25 rem whole body and 300 rem thyroid[14] can be considered values above which prevention of serious health effects would be the paramount concern. Good health physics practice would indicate that radiological exposures of these magnitudes should not be allowed to take place if reasonable and practical measures can prevent such exposures."
>
> "The assumptions used for siting purposes in calculating the doses that could result from design basis accidents are conservative. The actual doses that would result from releases postulated to occur from a design basis accident therefore would be expected to be much lower than the dose guidelines of 10 CFR Part 100 under most meteorological conditions. The inhalation and direct exposure doses from the releases postulated for design basis accidents are not likely to exceed the PAG levels beyond the LPZ under average meteorological conditions. It has been, however, the NRC's position that a spectrum of postulated conditions be considered in emergency planning including adverse meteorological conditions."

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13 NUREG-0396, Appendix III, Issue C, pages III-9 and III-10.

14 These values are the dose values at the outer boundary of the low population zone, see 10 CFR 100.11.

The table below summarizes the siting and emergency planning dose limits.

Region	Thyroid Dose	Whole Body Dose	Exposure Duration
Exclusion Area	300 rem	25 rem	First 2 hours of plume passage.
Low Population Zone	300 rem	25 rem	Entire duration of plume passage.
Plume EPZ (Nominal PAG)	25 rem	5 rem	Entire duration of plume passage.
Plume EPZ (Restrictive PAG)	5 rem	1 rem	Entire duration of plume passage.
Ingestion EPZ	1.5 rem	NA	Dependent on ingestion pathway.

The above doses are based on a plume release that contains radioactive fission products in the form of gases, aerosols, particulates, vapors, etc. that settles to the ground. If for a fusion power plant there is no iodine, the thyroid doses disappear (and it could be argued that the Ingestion EPZ can be eliminated). In addition, if the release is essentially tritiated gas (i.e., HT, DT and T_2) the plume may never return to the ground. However, if the release is essentially tritiated water vapor (i.e., HTO, DTO and T_2O) the plume may return to the ground.

Finally, regardless of the plume composition, if it can be shown that the Plume EPZ boundary whole body dose of 5 rem (or better yet, the 1 rem dose) can be met at the plant site boundary (i.e., the plant fence) for the worst-postulated fusion DBA release, the need for an off-site emergency plan could be rendered moot.

Finally, in 10 CFR 20, Appendix B, Tables 1, 2 and 3[15], the NRC treats tritiated gas as tritiated water vapor with respect to radiation protection while DOE still provides separate limits for tritiated gas and tritiated water vapor.[16, 17, 18, 19]

SUMMARY

Although 10 CFR 0 - 199 contains the regulations that a nuclear power plant must comply with, in practice the method and verification of compliance are done with lower tier documents. Regulatory Guides, Standard Review Plans and Branch Technical Positions, supplemented as necessary by Generic Letters, Bulletins and NUREGs, are the actual vehicles used by the designer to develop the facility and these same documents are used by the NRC to verify that the design complies with NRC criteria. Therefore, to anticipate and influence possible NRC regulation of fusion power, one must cognizant of not only 10 CFR 0-199 but also the lower tier documents which is how the NRC determines if compliance is met.

For fusion power plant designers working on structures, systems and/or components that are similar or identical to those used in a nuclear power plant, a recommended starting point for NRC regulatory criteria is the Standard Review Plans (NUREG-0800).

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15 10 CFR 20, "Standards for Protection Against Radiation," Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure: Effluent Concentrations; Concentrations for Release to Sewerage," Table 1,"Occupational Values," Table 2, "Effluent Concentrations," Table 3, "Releases to Sewers."

16 10 CFR 835, "Occupational Radiation Protection," Appendix A, "Derived Air Concentrations (DAC) for controlling Radiation Exposure to workers at DOE Facilities."

17 Proposed Rule (58FR16268, March 25, 1993) 10 CFR 834, "Radiation Protection of the Public and the Environment," Appendix A, "Derived Concentration Guides for Air and Water," Table A-1a, "Derived Concentration Guides (DCGs) for Members of the Public from Ingested Water and Inhalation Resulting in an EDE of 100 mrem/yr," Table A-1b, "Derived Concentration Guides (DCGs) for Members of the Public from Ingested Water and Inhalation Resulting in an EDE of 1 mSv/yr," Table A-2, "Alternate Absorption Factors and Lung Retention Classes for Specific Compounds."

18 DOE Order 5400.5, "Radiation Protection of the Public and the Environment," Chapter III, "Derived Concentration Guides for Air and Water," Figure III-1, "Derived Concentration Guides (DCGs) for Members of the Public from Ingested Water and Inhalation Resulting in 100 mrem/yr," Figure III-2, "Alternative Absorption Factors and Lung Retention Classes for Specific Compounds."

19 DOE Order 5480.11, "Radiation Protection for Occupational Workers," Attachment 1, "Derived Air Concentrations for Controlling Radiation Exposure to Workers at DOE Facilities," Table 1, "Derived Air Concentrations (DAC) for Controlling Radiation Exposure to Workers at DOE Facilities," Table 2, "Alternative Absorption Factors and Lung Retention Classes for Specific Compounds."

For fusion power plant designers working on structures, systems and/or components that are unique to a fusion power plant, the SRPs may be able to provide some guidance on what the regulatory criteria might be.

If the estimated radiological exposures from the worst-postulated fusion DBA release are minimal, there may be a significant effect on the present accepted practice for plant siting and emergency plans. If only low population exposures are postulated, it may be possible to site a fusion power plant in populated areas and eliminate off-site emergency plans.

TABLE 1

PARTS OF 10 CFR THAT COULD AFFECT THE DESIGN, CONSTRUCTION, OPERATION AND DECOMMISSIONING OF A FUSION POWER REACTOR

PART	TITLE
2	Rules of Practice for Domestic Licensing Proceedings
11	Criteria and Procedures for Determining Eligibility for the Access to or Control Over Special Nuclear Material*
19	Notices, Instructions and Reports to Workers: Inspection and Investigations
20	Standards for Protection Against Radiation
21	Reporting of Defects and Noncompliance
25	Access Authorization for Licensee Personnel
26	Fitness for Duty Programs
50	Domestic Licensing of Production and Utilization Facilities
51	Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
52	Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants
55	Operators' Licenses
70	Domestic Licensing of Special Nuclear Material*
71	Packaging and Transportation of Radioactive Material
73	Physical Protection of Plants and Materials
74	Material Control and Accounting of Special Nuclear Material*
75	Safeguards on Nuclear Material — Implementation of US/IAEA Agreement*
95	Security Facility Approval and Safeguarding of National Security Information and Restricted Data
100	Reactor Site Criteria
140	Financial Protection Requirements and Indemnity Agreements
170	Fees for Facilities and Material Licenses and Other Regulatory Services Under the Atomic Energy Act of 1954, as Amended
171	Annual Fee for Power Reactor Operating Licenses

Note: *Would apply if ^3H becomes a Special Nuclear Material and ^6Li becomes a Source Material.

TABLE 2

THE 10 CFR 50, APPENDIX A GENERAL DESIGN CRITERIA (GDCs) WHOSE GENERIC ASPECTS SHOULD BE CONSIDERED IN FUSION POWER REACTOR DESIGN

GDC #	Consider ‡			Remarks
	Yes	No	Modified	
1	X			Overall requirement.
2	X			Overall requirement.
3	X			Overall requirement.
4	X			Overall requirement.
5	X			Overall requirement.
6	[Reserved - not used]			
7	[Reserved - not used]			
8	[Reserved - not used]			
9	[Reserved - not used]			
10		X		Not needed. Don't have fuel that must remain in a coolable geometry in an intact vessel.
11			X	Need a similar GDC that covers appropriate safety system functions.
12			X	Need a similar GDC that covers appropriate control system functions.
13			X	Need a similar GDC that covers appropriate control system functions.
14		X		Not needed. Don't have fuel that must remain in a coolable geometry in an intact vessel.
15		X		Not needed. Don't have fuel that must remain in a coolable geometry in an intact vessel.
16	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.

TABLE 2

THE 10 CFR 50, APPENDIX A GENERAL DESIGN CRITERIA (GDCs) WHOSE GENERIC ASPECTS SHOULD BE CONSIDERED IN FUSION POWER REACTOR DESIGN

GDC #	Consider ‡			Remarks
	Yes	No	Modified	
17	X			General requirements of electrical power systems.
18	X			General requirements of electrical power systems.
19	X			General requirements for the Control Room.
20			X	Need a similar GDC that covers appropriate control system functions.
21	X			General requirements of control systems.
22	X			General requirements of control systems.
23	X			General requirements of control systems.
24	X			General requirements of control systems.
25			X	Need a similar GDC that covers appropriate control system functions.
26			X	Need a similar GDC that covers appropriate control system functions.
27			X	Need a similar GDC that covers appropriate control system functions.
28			X	Need a similar GDC that covers appropriate control system functions.
29			X	Need a similar GDC that covers appropriate control system functions.
30		X		Not needed. Fusion doesn't have fuel that must remain in a coolable geometry in an intact vessel.
31		X		Not needed. Fusion doesn't have fuel that must remain in a coolable geometry in an intact vessel.

TABLE 2

THE 10 CFR 50, APPENDIX A GENERAL DESIGN CRITERIA (GDCs) WHOSE GENERIC ASPECTS SHOULD BE CONSIDERED IN FUSION POWER REACTOR DESIGN

GDC #	Consider ‡			Remarks
	Yes	No	Modified	
32		X		Not needed. Fusion doesn't have fuel that must remain in a coolable geometry in an intact vessel.
33		X		Not needed. Fusion doesn't have fuel that must remain in a coolable geometry in an intact vessel.
34			X	Need a similar GDC that covers appropriate cooling system functions.
35			X	Need a similar GDC that covers appropriate cooling system functions.
36			X	Need a similar GDC that covers appropriate cooling system functions.
37			X	Need a similar GDC that covers appropriate cooling system functions.
38			X	Need a similar GDC that covers appropriate cooling system functions.
39			X	Need a similar GDC that covers appropriate cooling system functions.
40			X	Need a similar GDC that covers appropriate containment or confinement cooling system functions.
41			X	Need a similar GDC that covers appropriate containment or confinement cleanup system functions.
42			X	Need a similar GDC that covers appropriate containment or confinement cleanup system functions.

TABLE 2

THE 10 CFR 50, APPENDIX A GENERAL DESIGN CRITERIA (GDCs) WHOSE GENERIC ASPECTS SHOULD BE CONSIDERED IN FUSION POWER REACTOR DESIGN

GDC #	Consider ‡			Remarks
	Yes	No	Modified	
43			X	Need a similar GDC that covers appropriate containment or confinement cleanup system functions.
44	X			General requirements of cooling systems.
45	X			General requirements of cooling systems.
46	X			General requirements of cooling systems.
47	[Reserved - not used]			
48	[Reserved - not used]			
49	[Reserved - not used]			
50	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
51	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
52	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
53	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
54	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.

TABLE 2

THE 10 CFR 50, APPENDIX A GENERAL DESIGN CRITERIA (GDCs) WHOSE GENERIC ASPECTS SHOULD BE CONSIDERED IN FUSION POWER REACTOR DESIGN

GDC #	Consider ‡			Remarks
	Yes	No	Modified	
55	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
56	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
57	X			Whether called containment or confinement, barriers that prevent radioactive release to the environment are necessary.
58	[Reserved - not used]			
59	[Reserved - not used]			
60	X			Radiological requirements.
61			X	Needs to reflect physical properties of tritium.
62		X		Fission criticality not a concern.
63	X			Radiological requirements.
64	X			Radiological requirements.

NOTE: ‡ If the GDC is marked "Yes" it is considered germane as written with the possibility of minor changes (i.e., replacing "containment" with "confinement", etc.). If the GDC is marked "Modified" it is considered germane but the GDC needs to be rewritten to reflect fusion power plant characteristics instead of fission power plant characteristics.

APPENDIX A

10 CFR 50, APPENDIX A

"GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS"

APPENDICES

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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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.

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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]

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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 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.

[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.

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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.

52 FR 41288

Criterion 4—Environmental and dynamic effects 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, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

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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.

36 FR 12733

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.

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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.

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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.

36 FR 12733

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.

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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.

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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.

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

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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.

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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.

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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.

APPENDIX B—QUA[redacted] ASSURANCE CRITERIA FOR NUCL[redacted]OWER PLANTS AND FUEL REPRO[redacted]ING PLANTS

35 FR 10498

Introduction. Eve[redacted]pplicant for a construction permit is r[redacted]d by the provisions of § 50.34 to include [redacted] preliminary safety analysis report a de[redacted]ion of the quality assurance program t[redacted]applied to the design, fabrication, co[redacted]ction, and testing of the structures, sy[redacted], and components of the facility. Every [redacted]cant for an operating license is requ[redacted]o include, in its final safety analysis r[redacted], information pertaining to the manag[redacted]nd administrative controls to be used t[redacted]re safe operation. Nuclear powerplants [redacted]uel reprocessing plants* include struct[redacted] systems, and componen[redacted]t prevent or mitigate the consequence[redacted]ostulated accidents that could cause u[redacted]risk to the health and safety of the pub[redacted]his appendix establishes quality assura[redacted]quirements for the design, construction,[redacted] operation of those structures, systems,[redacted] components. The pertinent requireme[redacted] this appendix apply to all activities[redacted]cting the safety-related functions of[redacted]e structures, systems, and componen[redacted]hese activities include designing, p[redacted]sing, fabricating, handling, shipping, s[redacted]g, cleaning, erecting, installing, inspec[redacted]testing, operating, maintaining, repair[redacted] refueling, and modifying.

As used in this ap[redacted]x, "quality assurance" comprises all [redacted] planned and systematic actions nece[redacted] to provide adequate confidence th[redacted]structure, system, or component will p[redacted]m satisfactorily in service. Quality assu[redacted] includes quality control, which compri[redacted]ose quality assurance actions related [redacted] physical characteristics of a material[redacted]cture, component, or system which prov[redacted] means to control the quality of the m[redacted]l, structure, component, or system to [redacted]termined requirements.

40 FR 3210C

I. ORGA[redacted]ION

The applicant[1] shal[redacted]esponsible for the establishment and ex[redacted]on of the quality assurance program. T[redacted]plicant may delegate to others, such [redacted]ntractors, agents, or consultants, the wo[redacted] establishing and executing the quality [redacted]ance program, or any part thereof, but [redacted] retain responsibility therefor. The a[redacted]ity and duties of persons and organiza[redacted] performing activities affecting the s[redacted]related functions of structures, systems,[redacted]components shall be clearly established [redacted]elineated in writing. These activities [redacted]e both the performing functions of [redacted]ning quality objectives and the qualit[redacted]urance functions. The quality assurance [redacted]ions are those of (a) assuring that a[redacted]propriate quality assurance program [redacted]stablished and effectively executed a[redacted]) verifying, such as by checking, audi[redacted] and inspection, that activities affecti[redacted]e safety-related functions have been [redacted]ectly performed. The persons and or[redacted]ations performing quality assura[redacted]functions shall have sufficient auth[redacted] and organizational freedom to id[redacted] quality problems; to initiate, rec[redacted]end, or provide solutions; and to veri[redacted]ple[illegible] of solutions. Such perso[redacted]d organizations performing quality a[redacted]nce functions shall report to a managem[redacted]evel such that this required authority a[redacted]rganizational freedom, including suffici[redacted]ndependence from cost and schedule w[redacted]opposed to safety considerations, are pr[redacted]d. Because of the many variables involv[redacted]ch as the number of personnel, the type[redacted]ctivity being performed, and the locat[redacted]r locations where activities are perform[redacted]he organizational structure for executin[redacted] quality assurance program may take v[redacted]s forms provided that the persons and [redacted]izations assigned the quality assurance [redacted]ions have this required authority an[redacted]anizational freedom. Irrespective of th[redacted]anizational structure, the individual(s[redacted]igned the responsibility for assuring ef[redacted]e execution of any portion of the quali[redacted]ssurance program at any location wher[redacted]ivities subject to this Appendix are bei[redacted]formed shall have direct access to such [redacted]s of management as may be necessary t[redacted]orm this function.

[1] While the term "a[redacted]ant" is used in these criteria, the [redacted]uirements are, of course, applicable [redacted] such a person has received a licen[redacted] construct and operate a nuclear po[redacted]plant or a fuel reprocessing plant. T[redacted]criteria will also be used for guidance [redacted]aluating the adequacy of quality assu[redacted] programs in use by holders of constru[redacted] permits and operating licenses.

*Amended 36 FR 1830[redacted]

35 FR 10498

II. QUALITY ASS[redacted]CE PROGRAM

The applicant shall [redacted]lish at the earliest practicable time, cons[redacted] with the schedule for accomplishing the [redacted]ities, a quality assurance program wh[redacted]omplies with the requirements of this [redacted]dix. This program shall be documented [redacted]itten policies, procedures, or instructio[redacted]d shall be carried out throughout plant [redacted]n accordance with those policies, proce[redacted], or instructions. The applicant shall i[redacted]fy the structures, systems, and compon[redacted] to be covered by the quality assurance [redacted]am and the major organizations particip[redacted] in the program, together with the de[redacted]ted functions of these organizations. T[redacted]quality assurance program shall provide[redacted]rol over activities affecting the quality o[redacted] identified structures, systems, and co[redacted]ents, to an extent consistent with their [redacted]rtance to safety. Activities affecting qu[redacted] shall be accomplished under suitably[redacted]rolled conditions. Controlled conditions[redacted]lude the use of appropriate equipme[redacted]uitable environmental conditions f[redacted]ccomplishing the activity, such as adequ[redacted]leanness; and assurance that all prere[redacted]tes for the given activity have been s[redacted]d. The program shall take into accoun[redacted] need for special controls, processes, test[redacted]pment, tools, and skills to attain the req[redacted] quality, and the need for verification [redacted]uality by inspection and test. The pro[redacted] shall provide for indoctrination and tra[redacted] of personnel performing activities affe[redacted] quality as necessary to assure that [redacted]ble proficiency is achieved and maintain[redacted]he applicant shall regularly review the s[redacted] and adequacy of the quality assurance [redacted]am. Management of other organization[redacted]ticipating in the quality assurance pr[redacted] shall regularly review the status and [redacted]uacy of that part of the quality assuran[redacted]ogram which they are executing.

III. DESIGN[redacted]TROL

Measures shall be [redacted]lished to assure that applicable regula[redacted]requirements and the design basis, as de[redacted] in § 50.2 and as specified in the license[redacted]ication, for those structures, systems, an[redacted]ponents to which this appendix applies[redacted] correctly translated into spe[redacted]drawings, procedures, and instruction[redacted]se measures shall include provisions to a[redacted] that appropriate quality standards are [redacted]ied and included in design documents [redacted] that deviations from such standards a[redacted]trolled. Measures shall also be establishe[redacted] the selection and review for suitability o[redacted]lication of materials, parts, equipmen[redacted]d processes that are essential to the sa[redacted]related functions of the structures, syst[redacted]and components.

Measures shall be [redacted]blished for the identification and co[redacted] of design interfaces and for coordinat[redacted]mong participat-

APPENDIX B

CLINCH RIVER BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

SECTION 3.1

"CONFORMANCE WITH GENERAL DESIGN CRITERIA"

CLINCH RIVER
BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

PROJECT MANAGEMENT CORPORATION

This chapter identifies and discusses the principal architectural and engineering design criteria for the plant. These criteria are supplemented by more specific criteria discussed in Chapters 4 through 12.

3.1 CONFORMANCE WITH GENERAL DESIGN CRITERIA

3.1.1 Introduction and Scope

3.1.1.1 General

Pursuant to the provisions of Title 10, Part 50, Section 50.34 of the Code of Federal Regulations, an application for a nuclear power plant 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.

General design criteria, which establish the minimum requirements for the principal design criteria for nuclear power plants are identified in the Code of Federal Regulations, Title 10, Part 50, Appendix A (10CFR50A). While these criteria provide guidance for all types of nuclear power plants, they are specifically oriented toward water reactors. This is recognized in the Code of Federal Regulations which states, "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".

As a result of the increased design and development activities directed toward the establishment of commercial liquid metal fast breeder reactor (LMFBR) plants, the need for more specific guidance for the design of these plants was recognized. Consequently, the American Nuclear Society Subcommittee ANS-24 (now ANS-54) was established in 1968 to develop and interpret these criteria for the LMFBR. The subcommittee included representatives from the reactor manufacturers, the architect-engineer, vendors, utilities, and the Atomic Energy Commissions's regulatory and development divisions. The efforts of this group resulted in draft General Safety Design Criteria for an LMFBR Nuclear Power Plant.

3.1.1.2 CRBRP General Design Criteria

The 10CFR50 Appendix A criteria and the draft criteria from ANS-54 were considered in developing the General Design Criteria for the Clinch River Breeder

Reactor Plant (CRBRP). In July 1974, the USAEC Directorate of Licensing issued the "Interim General Design Criteria for the Clinch River Breeder Reactor Nuclear Power Plant". These interim criteria were then also carefully considered in finalizing the CRBRP General Design Criteria, which are individually described in Section 3.1.3.

The CRBRP General Design Criteria focus on safety requirements for the CRBRP and are intended to be used in lieu of the General Design Criteria in 10CFR50A. Other essential criteria in such areas as operability, maintainability, and environmental acceptability are included in the overall plant design bases but are not specifically addressed here.

These General Design Criteria recognize the overall design concept selections for the CRBRP, including a three loop plant having a heat transport system consisting of three flow paths in sequence separated by passive barriers. These sequential flow paths are provided by a reactor coolant system, an intermediate coolant system, and an extraction system for utilization or dissipation of heat. The principal components in the reactor coolant system are protected by guard vessels to limit the consequences of failure of the coolant boundary. The passive barriers, i.e., heat exchanger tube walls, are at the reactor coolant system/intermediate coolant system and the intermediate coolant system/heat extraction system interfaces. A low leakage containment barrier is used as the outermost barrier to limit the release of radioactive materials to the environment.

It is recognized that the development of criteria is an evolutionary process, and the on-going LMFBR development program will provide valuable additional inputs to these criteria. Some of the definitions may need further amplification and some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined but these areas must be considered in the design. These areas 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, Section 3.1.2.)

(2) Consideration of redundancy and diversity requirements for the fluid systems important to safety. (See Heat Transport System Design, Criterion 30a, Section 3.1.3.)

(3) Consideration of the design bases for the reactor containment. (See Containment Building Structure Design Basis, Criterion 50, Section 3.1.3.)

(4) Consideration of isolation requirements for the reactor containment unique to an LMFBR having an intermediate coolant system. (See Containment Penetration Criteria, 54 through 57, Section 3.1.3.)

(5) Consideration of the types and combinations of events in determining design requirements to suitably protect against postulated accidents. (See definition of Extremely Unlikely Faults, Section 3.1.2.)

(6) Consideration of the possibility of systematic, non-random, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Protection and Reactivity Control Systems Criteria, Section 3.1.3.)

It is recognized that highly reliable plant operation is an essential element in assuring safe operation. Accident prevention through the use of reliable designs obtained by rigorous application of codes and standards and quality control applied to all phases of design, construction, testing and operation is first and foremost in providing safe operation. The degree to which various off-normal and accident conditions should be considered in formulating the design bases depends on the specific design features and their effectiveness in preventing the accidents.

Section 3.1.2 defines terms used in the criteria, where some possibility of ambiguity has been foreseen. In Section 3.1.3, each of the criteria is stated, together with a statement of the means by which the design has been responsive to the requirements of that criterion.

3.1.2 Definitions and Explanations

The definitions given below form the bases for requirements placed with the criteria quoted in Section 3.1.3.

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.

Active Component. An active component is one in which mechanical movement must be initiated or electrical power must be provided to accomplish a function of the component.

Active Component Failure. Active component failure means failure of an active component to operate or stop as intended on demand or a change of state when no change is intended.

No Loss of Safety Function. No loss of safety function means that the equipment or component retains its capability of accomplishing its safety function as required to accommodate a postulated event, but following the event repairs or replacements could be required to restore the equipment to its original design conditions.

Single Failure. A single failure means an occurrence which results in loss of capability of a component to perform its safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single 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.[1]

Common Mode Failure. Common mode failure is the simultaneous failure (within a single test interval) of redundant equipment caused by a single phenomenon. In the context of this definition consideration should be given to such items as:

(1) degradation of properties of material at different locations due to the same cause.

(2) a design, fabrication, maintenance, operational, or installation deficiency common to multiple components.

Fuel Damage Limits. Fuel damage limits means those limits such as cladding strain, amount of fuel melting, amount of cladding deformation or melting, and fractional fuel failure beyond which the accident consequences are unacceptable. (Different fuel damage limits may be specified for different postulated accidents.)

REACTOR COOLANT BOUNDARY

Reactor Coolant Boundary means the fluid boundary of those components which are (1) part of the reactor coolant system including the passive barrier between the reactor coolant and the intermediate coolant or (2) connected to the reactor coolant system up to and including any and all of the following:

(a) For those components whose failure could result in fuel design limits being exceeded

(i) In the case of a closed system which is connected to the reactor coolant system at both ends, the entire closed system

(ii) In other cases the second of two valves closed during normal operation or automatically isolable under any off-normal plant condition.

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[1]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.

(b) For those components whose failure could not result in fuel design limits being exceeded, the nozzles that connect the component or its associated piping to the reactor coolant system.

INTERMEDIATE COOLANT BOUNDARY

Intermediate coolant boundary means the fluid boundary of those components such as heat exchangers, piping, pumps, tanks and valves which are (1) part of the intermediate coolant system, (2) part of the passive barrier between the intermediate coolant and the working fluid of the heat extraction system, or (3) connected to the intermediate coolant system up to and including the following:

(i) For systems or components whose failure would impair the capability of the intermediate coolant system to perform its safety function:

(a) The outermost containment isolation valve in piping which penetrates reactor containment.

(b) The first valve normally closed or automatically isolable in piping which does not penetrate reactor containment.

(ii) For systems and components whose failure would not impair the capability of the intermediate coolant system to perform its safety function: the nozzle which connects the system or component to the intermediate coolant boundary.

NORMAL OPERATION

Normal operation means steady state operation and those departures from steady state operation which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. It includes conditions such as startup, normal shutdown, stand-by, load following, limited fuel rod leakage, operation with specific equipment out of service as permitted by Technical Specifications, and routine inspection, testing, and maintenance of components and systems during any of these conditions, if it is consistent with the Technical Specifications.

OFF-NORMAL CONDITIONS

Off-normal conditions means those steady state and transient conditions not part of normal operation which (1) individually may be expected to occur once or more during the plant life time and include but are not limited to an inadvertent control rod withdrawal, tripping of sodium circulating pumps, failure of all offsite power, and tripping of the turbine generator set or (2) which individually is not expected to occur during the plant lifetime; however, when integrated over all plant components and systems, events in this category may be expected to occur a number of times. Class (1) events are termed Anticipated Faults and class (2) events are termed Unlikely Faults.

EXTREMELY UNLIKELY FAULTS

Events of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures which are identified as possible.

These extremely unlikely events, which are design bases, shall encompass a spectrum of events appropriate to the design. These may include, for example a large sodium fire, a large sodium-water reaction, and a rupture of a radwaste system tank.

Inert Atmosphere. Inert atmosphere means a gas or gaseous mixture limited in oxygen and other substances that are chemically reactive with sodium so that chemical reactions will not significantly increase the consequences of contact with sodium.

Heat Transport System. The heat transport system is the series of components containing the heat transport fluids and used for extracting heat from the reactor and transporting it to the equipment used for electrical power conversion during normal operation or, after plant shutdown, to an ultimate heat sink. It does not include systems whose prime function is the cooling of structures or equipment.

Reactor Residual Heat Extraction System. The reactor residual heat extraction system is the portion of the heat transport system which, after plant shutdown, transports reactor residual heat to the ultimate heat sink.

Ultimate Heat Sink. The ultimate heat sink is that heat sink (e.g., a river, pond or local atmosphere) to which the heat transport system rejects its heat.

Fuel Design Limits. Fuel design limits means those limits such as temperature, burnup, fluence, and cladding strain which are specified by the designer for normal operation and anticipated operational occurrences.

3.1.3 Conformance with CRBRP General Design Criteria

3.1.3.1 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 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.

RESPONSE

The design of this plant conforms to the intent of this criterion. The design criteria for structures, systems and components important to safety are stated in this chapter. The codes and standards to be employed are identified in applicable sections of this chapter. The seismic classification of structures and the codes and standards for components are discussed in subsections 3.2.1 and 3.2.2, respectively. Quality assurance plans, designed to conform to Appendix B to 10CFR50, of each of the project participants are given in Chapter 17.0. The record-keeping activities are described in Chapter 17.0.

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.

RESPONSE

The design of this plant conforms to the intent of Criterion 2. The historical record and other information influencing the selection of the design basis natural phenomena are given in Sections 2.3, 2.4 and 2.5.

The design criteria for protection of the plant from the effects of natural phenomena are given in Sections 3.3 through 3.11. The systems, components and structures important to safety have been designed to accommodate, without loss of capability, effects of the design basis natural phenomena along with appropriate combinations of normal and accident conditions.

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. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components important to safety.

RESPONSE

The design of this plant conforms to the intent of Criterion 3. The use of combustible materials will be maintained at a minimum practicable. Fire detection and protection measures of appropriate capabilities and capacities have been incorporated into the design. The fire protection system, described in Section 9.13, will not, by rupture or inadvertent operation, impair plant safety.

Criterion 3.a PROTECTION AGAINST SODIUM REACTIONS

Structures, systems, and components containing sodium shall be designed to limit the consequences of sodium chemical reactions resulting from a sodium spill. Special features such as inerted vaults shall be provided as appropriate for the reactor coolant system. Means to detect sodium chemical reactions and initiate and test fire control systems shall be provided to limit and control the extent of such reactions to assure that the functions of components important to safety are maintained. Means shall be provided to limit the release of sodium reaction products to the environment as necessary to protect plant personnel and to avoid undue risk to the public health and safety. Materials which might come in contact with sodium shall be chosen to minimize the adverse effects of possible chemical reactions. In areas where sodium chemical reactions are possible, structures, systems, and components important to safety, including electrical wiring and components, shall be designed and located so that the potential for damage by sodium chemical reactions is minimized. Means shall be provided as appropriate to minimize possible contact between sodium and water. The effects of possible interactions between sodium and concrete shall be considered in the design.

The sodium-steam generator system shall be designed to detect sodium-water reactions and limit the effects of the energy and reaction products released by such reactions so as to prevent loss of safety functions of the heat transport system.

RESPONSE

Protection against sodium reactions is provided for by:

1. The use of stainless or carbon steel for tanks; components and piping containing sodium or NaK.

2. The use of steel cell liners and drip pans in concrete cells to prevent any concrete-sodium reaction in the event of a spill;

3. The use of insulation approved for sodium systems with an inner and outer sheath of stainless steel to minimize an absorption in the insulation;

4. The use of auxiliary coolant fluid in NaK coolers which will not mix since with NaK, nor produce an exothermic reaction; and

5. The use of suitable instrumentation to detect any sodium reactions.

The instrumentation to detect sodium reactions and to control the reaction suppressant dispensing system is described in Section 9.13.2. The cells are either inerted or are provided with fire control capability, electrical equipment is above the normal expected depth of any sodium spill, and the electrical wiring is so located as to minimize damage from sodium fires.

The Steam Generator System is provided with subsystems to detect sodium-water leakage and to limit any reaction effects. These are discussed in Sections 7.5 and 5.5, respectively.

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, off-normal conditions, and Extremely Unlikely Faults. 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. The effects of possible sodium impingement on equipment, support structures, cell liners, containment, steel and concrete surfaces shall be considered in the design.

RESPONSE

The environmental design of safety-related facilities is discussed in Section 3.8 for the design of structures, and in Section 9.6 for ventilation systems. In-containment postaccident environmental conditions are discussed in Section 6.2. Safety-related systems/components using the input from these sections for design are discussed in Section 3.11.

Conservative design methods, segregated routing of piping, provision of missile barrier walls and engineered pipe hangers and pipe restraints are all used to accommodate dynamic effects of postulated accidents. These same features as well as the strength of the Containment and other Category I structures protect the safety-related equipment from missiles which might be generated either within or outside the plant. Sections 3.5 and 3.6 detail the design assumptions, methods and results for protective design against missile and postulated piping ruptures.

Criterion 5 SODIUM HEATING SYSTEMS

Heating systems shall be provided as necessary for systems and components important to safety which contain, or may be required to contain, sodium. The heating systems and their controls shall be appropriately designed to assure that the temperature distribution and rate of change of temperature in sodium systems and components containing sodium are maintained within design limits.

RESPONSE

Heating systems will be provided for all systems and components important to safety which contain, or may be required to contain, sodium. These systems and components comprise:

Reactor Enclosure

Ex-Vessel Storage Tank

Reactor Heat Transport (Primary and Intermediate) Systems

Steam Generator System

Auxiliary Liquid Metal System

Details of the Heating Systems are discussed in Section 9.4.

Criterion 6 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.

RESPONSE

This criterion is not applicable to the CRBRP.

3.1.3.2 Protection by Multiple Fission Product Barriers

Criterion 10 REACTOR DESIGN

The reactor 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 and anticipated operational occurrences.

RESPONSE

The following two design bases, taken together, satisfy this criterion.

a. Fuel Burnup

   In the first core loading, the fuel rods are designed for a peak pellet burnup of 80,000 megawatt days per metric ton of heavy metal (MWd/T). For later cores the peak burnup increases to 150,000 MWd/T with an average burnup of 100,000 MWd/T (see subsection 4.3.2.1, Nuclear Design Description).

b. Power Distribution Limits

   At full reactor power and at the maximum overpower condition permitted by the protection system, the core power distribution limits are not exceeded. These limits are derived from the maximum allowable peak heat generation rates for nominal and overpower conditions, as discussed in detail in Section 4.3.2.2.

Criterion 11 REACTOR INHERENT PROTECTION

The reactor and associated coolant systems shall be designed so that during normal operation and off-normal conditions, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

RESPONSE

The following design basis satisfies this criterion:

The Doppler effect provides the prompt negative reactivity feedback which is required to mitigate the effects of reactivity transients (rapid power increases). Therefore, the fuel temperature (Doppler) coefficient shall be strongly negative when the reactor is critical. The negative Doppler coefficient is obtained through the inherent use of fuel with a large proportion of U-238 (see subsection 4.3.2.3, Reactivity Coefficients).

Criterion 12 SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor 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.

RESPONSE

The CRBR is neutronically tightly coupled, preventing any possibility of spatial instability. The main stabilizing feedback is due to Doppler and the CRBR is inherently stable in response to reactivity perturbations.

Details of the nuclear design are discussed in Section 4.3.2.

Criterion 13 INSTRUMENTATION AND CONTROL

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, off-normal and Extremely Unlikely Fault conditions to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor, the reactor coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

RESPONSE

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, temperatures, pressures, flows, and levels as necessary to assure that adequate plant safety can be maintained. Instrumentation is provided in the Reactor System, Heat Transport System, Steam and Power Conversion System, the Engineered Safety Features Systems, Radwaste Systems and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room, in proximity with the controls for maintaining the indicated parameter in the proper range.

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 4 through 12. Details of the instrumentation and control systems are discussed in Chapter 7.

Criterion 14 REACTOR COOLANT BOUNDARY

The reactor coolant 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.

RESPONSE

The design, fabrication, erection and testing to be employed on the reactor coolant boundary and the extensive quality control measures to be employed during each of the above phases will ensure that this boundary has extremely low probabilities of abnormal leakage, rapidly propagating failure, and gross rupture.

This extremely low probability is further enhanced by the selection of materials for the coolant boundary, and by the operating temperature and pressure conditions. Austenitic Stainless Steels form the coolant boundary at all points at which significant pressure is encountered and also the major portion of the remainder of the boundary. At the typical operating temperature range (775 to 1015°F) these materials are well beyond the point at which brittle fracture is a consideration. Further, pressures are low (<200 psi) so that thermal rather than mechanical loads are of significance; such loads are not conducive to rapid crack propagation.

Detailed discussions of the properties of the primary boundary materials, reasons for their selection and consideration of coolant compatibility is to be found in Sections 5.2 and 5.3. The leakage detection systems, which assure the detection of a leak far in advance of the point at which rapid propagation must be considered, are discussed in Section 7.5.

Criterion 15 REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated control, protection, auxiliary and sodium heating systems, shall be designed with sufficient margin to assure that the design conditions of the reactor coolant boundary are not exceeded during normal operation and off-normal conditions.

RESPONSE

The reactor coolant system and associated auxiliary, control and protection systems are designed to ensure the integrity of the reactor coolant boundary with adequate margins during normal operation and during transient conditions. The system boundary can accommodate loads due to the operating basis earthquake during Anticipated Faults within upset condition code stress limits. The components of the reactor coolant system and associated auxiliary systems are designed in accordance with appropriate ASME and ANSI Codes. These codes are identified in Chapters 3.0, 5.0 and 9.0.

Criterion 16 REACTOR CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight 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 Extremely Unlikely Fault conditions require.

RESPONSE

The containment design comprises a steel shell, with a design pressure of 10 psig, and leak testable penetrations. This completely encloses the reactor coolant boundary. Except for the Intermediate Heat Transport System Loops, which are built to containment quality standards, all piping systems penetrating containment are provided with containment isolation valves, in compliance with Criteria 54 through 57.

The design criteria and methods of analysis for the containment structure are discussed in Section 3.8.2 and the functional design and testing provisions are described in Section 6.2.

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 boundary are not exceeded as a result of off-normal conditions and (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of Extremely Unlikely Faults.

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 normal operation, off-normal, Extremely Unlikely Fault, 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 boundary are not exceeded. One of these circuits shall be designed to be available following any Extremely Unlikely Fault, which does not involve loss of capability of the switchyard to provide power, in time 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.

RESPONSE

The stand-by (on site) electric power system consists of two automatic, fast start-up diesel generators and their power distribution systems, two redundant a-c sources supplying through inverters, through independent batteries, buses and power distribution systems below the power feed into the inverters. The offsite electric power system consists of four 161 KV transmission lines - two transmission lines connected to the generating yard and the remaining two to the reserve yard. The two lines entering the reserve yard are separate and physically independent and are considered as the two circuits satisfying Criterion 17. As indicated in Figure 8.1-1, these two transmission lines enter the switchyard from different directions to preclude the likelihood of their simultaneous failures.

Upon loss of all 161 KV power sources, the diesel generators start automatically and are capable of accepting the required safety loads. Either diesel or any of the 161 KV power sources are capable of providing sufficient power to safely shutdown the plant during the anticipated operational occurrences and to power the necessary engineered safety features in the event of postulated accidents.

The two diesel generators are redundant and independent including the distribution systems which they supply as described in Section 8.3.1.1.1. Automatic starting and loading of each diesel generator to perform the safety function of the distribution systems they supply can be tested by simulating loss of ac power supply to either 13.8 KV ESF distribution bus that is supplied by a diesel generator. Both diesels will start automatically and, if required, after 10 seconds the diesel generator on the disrupted distribution system will be automatically loaded with engineered safety features equipment in a timed sequence. The battery systems are redundant and independent including the distribution systems which they supply as described in Section 8.3.2. Tests performed on dc batteries are described in Section 16.4.5.

To minimize the probability of losing electric power in the off-site electric power system, the plant is connected to the Tennessee Valley Authority (TVA) grid using two separate and independent switchyards and four connections by 161 KV transmission lines to the grid. The plant generating switchyard is connected to the power grid by two 161 KV transmission lines. The plant reserve switchyard is connected to the grid by two separate and physically indepedent 161 KV transmission lines, either of which is designed to be capable of providing full power to the normal and safety related ac distribution systems via the reserve transformers. The generating yard is capable of supplying power to the plant auxiliary ac power distribution system through the auxiliary transformer. The reserve year is connected to the two 100% capacity reserve transformers. Each reserve transformer is capable of supplying full power required for the auxiliary ac power distribution system. To minimize the probability of losing electric power in the on-site electric power system, two redundant diesel generators with independent distribution systems and battery systems with independent distribution systems are provided.

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 operational 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.

RESPONSE

The 161 KV circuit breakers can be inspected and tested as follows:

a. The 161 KV transmission line circuit breakers can be tested on a routine basis. This can be accomplished without removing the transmission line from service.

b. The 161 KV transmission line circuit breakers can be tested with the generators in service since two breakers, each fully rated, are provided to connect the generator to two buses of the generating switchyard.

Provision is included in the design for testing the transfer of power between the power supplies fed from the auxiliary transformer and the start-up reserve transformers. These tests are performed during prolonged plant shutdown periods by simulating loss of the ac power supply from the auxiliary transformer.

The operability of the circuit breakers carrying load under normal plant operation is demonstrated by their performance in supplying power. In addition, the circuit breakers are tested in "Test" position at regular intervals. During this test, the proper operation of the circuit breaker control circuits are verified.

Testing of the circuit breaker of the standby equipment is performed by racking the circuit breaker in "Test" position. In the "Test" position, the main contacts of the circuit breaker are disconnected but the auxiliary and the control circuits are maintained. This facilities functional tests of the circuit breaker and its control circuit.

Each diesel generator is controlled from a separate panel located in the control room. Provision has been made on the control panels to manually initiate a fast start of either diesel generator and to close the associated air circuit breakers connecting the generator to its medium voltage engineered safety feature bus. Testing of this system may be performed by the control room operator at his convenience any time the units are not otherwise running with due regard for reactor auxiliaries in use.

The operability of the 120 volt vital ac power system can be tested by transferring its load to the standby ac source as described in Section 8.3.1.1.5.

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 operation and off-normal conditions and to maintain it in a safe condition under Extremely Unlikely Fault conditions. Adequate radiation protection shall be provided to permit access and occupancy of the control room under Extremely Unlikely Fault 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 event. The control room shall also provide protection from substances such as sodium oxide which might be released to the local environment under Extremely Unlikely Fault conditions.

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.

RESPONSE

The control room is designed following proven power plant design philosophy. All control stations, switches, controllers, and indicators necessary to operate and shut down the plant and to maintain safe control of the facility will be located in the control room.

The design of the control room will permit safe occupancy during abnormal conditions. Shielding will be designed to maintain tolerable radiation exposure levels in the control room under hypothetical accident conditions. The control room ventilation system will provide the necessary environment for both the operators and the instrumentation. The control room Heating, Ventilating, and Air Conditioning (HVAC) system (See Section 9.6.1) is designed with the capability to provide air filtration, heating, cooling dehumidification and humidification (for normal operation plant condition only) as required to permit continuous occupancy under normal and abnormal conditions. Control room air is partially recirculated through high efficiency filters, and filtered outside air pressurization is provided to reduce the ingress of radioactive particles. The control room will be continuously occupied by qualified operating personnel under all conditions.

Alternate local controls and instrumentation at locations outside the control room are provided to bring the plant to, and maintain it in, a hot shutdown condition. Cold shutdown from outside the control room is not contemplated. The control room has been designed to remain operable and habitable under extremely severe postulated events.

3.1.3.3 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 controls systems, to assure that specified acceptable fuel design limits are not exceeded as a result of off-normal conditions and (2) to sense accident conditions and initiate the operation of systems and components important to safety.

RESPONSE

The operational limits for the reactor protection system are defined by analysis of plant operating and transient conditions requiring rapid rod insertion to prevent or limit core damage. A discussion of the appropriate fuel design limits, which form design basis for the reactor protection system, is given in Chapter 4. The systems activated to prevent exceeding fuel design limits are:

1. Primary reactor shutdown system
2. Secondary reactor shutdown system
3. Stand-by diesel generator starting system
4. Steam Generator Auxiliary Heat Removal System

In addition, the protection system will initiate the following actions:

a. Coastdown of all primary and intermediate system cooling pumps at every reactor trip. This is necessary to minimize the thermal transients experienced by the components, and hence to assure endurance throughout the operating life.

b. Isolation of the containment system in the event of a release of activity into the containment atmosphere.

Full details of the reactor protection system are given in Section 7.2, and of the containment isolation system in Sections 7.3.1 and 6.2.4.

Criterion 21 PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any 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.

RESPONSE

Each of the two shutdown systems is designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

The protection system performs indication and alarm functions in addition to its reactor trip and engineered safety features actuation functions. The design meets the requirements of RDT Standard C-16-1, which meets or exceeds those of IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Each system consists of a large number of input measurement channels, redundant logic trains, and redundant reactor trip breakers. The redundant logic trains, and reactor trip breakers for each system are electrically isolated and physically separated. Furthermore, physical separation of the channels is maintained within the separated trains. The design is such that no single failure results in loss of the protection function during operation or testing. Either of the two systems, which are highly redundant, will perform the shutdown function for all normal conditions. All channels employed in power operation are sufficiently redundant so that individual testing and calibration, without degradation of the shutdown function or violation of the single failure criterion, can be performed with the reactor at power. Such testing will disclose failures or reductions in redundance which may have occurred. Removal from service of any single channel or component does not result in loss of minimum required redundancy. For example, a two-of-three function is placed in a one-of-two mode when one channel is removed.

In addition to this manual testing capability of both the primary and secondary systems, a semiautomatic tester is included to test the logic trains of the primary system. This tester has the capability of testing the major part of the protection system very rapidly with the reactor at power. Between tests, equipment is provided to continuously monitor certain internal protection system points, including train power supply voltages.

The protection system is discussed in Section 7.2 of this PSAR.

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 off-normal, and Extremely Unlikely Fault 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 due to common mode failures.

RESPONSE

The protection system has been designed to provide sufficient resistance to a broad class of accident conditions and postulated events.

The defenses against loss of the protection function through the effects of natural phenomena, such as tornado, flood earthquake, and fire, are location and Category I structures physical separation and electrical isolation of redundant channels and subsystems, functional diversity of subsystems, and safe (i.e., in the direction of reactor trip) component and subsystem failure modes. These defenses have been utilized in the design of the reactor protection system. The redundant logic trains, reactor trip breakers, and engineered safety features actuation devices are physically separated and electrically isolated. Physically separate channel cable trays, conduit, and penetrations are maintained upstream from the logical elements of each train. Functional diversity and physical separation are designed into the system.

The factors associated with normal operation are wear, temperature, humidity, dust or dirt, and vibration. The protection system is tested and qualified under environmental conditions in excess of the extreme normal ranges. In the majority of the system, wear is not a factor. The station test and maintenance procedures will provide adequate measures against simultaneous multiple failures due to wear, dust, or dirt. Furthermore, protection of the equipment from dust or other contaminants is afforded by the cabinets in which the equipment is installed.

The possibility of loss of the protection function through improper or incorrect maintenance is minimized by a number of factors. Among these are administrative controls; functional diversity (a pump speed channel and a flow channel are not likely to be miscalibrated in the same direction, for example); and a comprehensive indication, alarm, and status system.

The protection system has been evaluated with respect to functional diversity and with respect to common mode susceptibility. These studies indicates that the system is designed to a very high probability of performing its function in any postulated occurrence. An extensive reliability program has been initiated which will confirm this very high reliability before submission of the FSAR.

The reactor protection system and the engineered safety features actuation system are discussed in Sections 7.2 and 7.3, respectively.

Criterion 23 PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed such that, in the event of failure, it will fail into a safe state or into a state demonstrated to be acceptable on some other defined basis. 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, sodium, sodium reaction products, and radiation) shall be considered.

RESPONSE

The protection system is designed with due consideration to the most probable failure modes of the components.

Where practical, the channels and logics are designed such that failures which may occur will be in the direction which covers a trip. Channel monitoring is provided to detect either safe or unsafe failures of individual channels. Provision of redundancy within each system assures that, should there be a simple failure in the direction to impede a trip, it will result in no loss of the system capability.

The protection system components will be tested and qualified for the extremes of the normal environment to which they are subjected. In addition, components will be tested and qualified according to individual requirements for the adverse environment specific to their location which might result from postulated accident conditions. Protection against sodium and sodium reaction products is provided by location of the components. To the maximum extent practically all protection system components are located in areas away from sodium containing components. Where this is not practical devices are provided to shield components from sodium impingement.

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

RESPONSE

The failure of a single control system component or channel, or the failure or removal from service of any 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 is limited so as to assure that safety is not significantly impaired.

Most functions performed by the reactor protection and the reactor control systems require the same process information. The design philosophy for these systems is to make maximum use of a wide spectrum of diverse and redundant process measurements. The protection system is separate and distinct from the control system. The control system is dependent on the protection system in that control input signals are derived from protection system measurements where applicable. These control signals are transferred to the control system by isolation amplifiers which are classified as protection system components. No credible failure at the output of an isolation amplifier will prevent the corresponding protection channel from performing its protection function. Such failures include short circuits, open circuits, grounds, and the application of the maximum credible a-c and d-c voltages. The adequacy of system isolation has been verified by testing under these fault conditions. The controls are designed such that a single failure of a sensor will not cause a control system malfunction requiring PPS function. The design meets all requirements of RDT Standard C16-1T, which meets or exceeds those of IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations".

The reactor protection systems and the control systems are discussed in Sections 7.2 and 7.4 respectively.

Criterion 25 PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection systems 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.

RESPONSE

The maximum controlled reactivity insertion rate due to control rod withdrawal at the design speed of 9 inches/minute is 2.4 cents/second. This protection system assures that the peak clad temperature is maintained below the maximum allowable value for a ramp rate of this magnitude. Analysis of this transient is presented in Section 15.2.

Criterion 26 REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems, preferably of different principles, shall be provided. Design features including diversity shall be provided to protect against common mode failure. One control system shall be capable of reliably controlling reactivity to assure that under normal operation and off-normal conditions, and with appropriate margin for malfunctions such as a stuck rod, specified acceptable fuel design limits are not exceeded. The second control system shall be capable of reliably controlling reactivity to assure that under normal operation and off-normal conditions, and with appropriate margin for malfunctions, such as a stuck rod, specified acceptable fuel damage limits are not exceeded. Each system shall have sufficient worth, assuming failure of any single active component, to shut down the reactor from any operating condition to zero power and maintain subcriticality at the hot shutdown temperature of the coolant, with allowance for the maximum reactivity associated with any off-normal condition. One system shall have sufficient worth, assuming failure of any single active component, to maintain the reactor subcritical for any cold shutdown conditions.

RESPONSE

Two independent diverse reactivity control systems are provided; namely, the primary and secondary shutdown systems. The primary control system is designed to meet the fuel burnup and load follow requirements for each cycle as well as to compensate for criticality and refueling uncertainties. The primary system will have sufficient worth at any time in the reactor cycle, assuming the failure of any single active component (i.e., a stuck rod) to shut down the reactor from any operating condition and to maintain subcriticality over the full range of coolant temperatures expected during shutdown. Allowance will also be made for the maximum reactivity fault associated with any anticipated occurrence.

The secondary control system using rods of significantly different design principles will have sufficient worth at any time in the reactor cycle, assuming the failure of any single active component (i.e., a stuck rod), to shut down the reactor from any operating condition to the hot shutdown temperature of the coolant. Allowance will also be made for the maximum reactivity fault associated with any anticipated occurrence. This reactivity fault allowance is included in the requirements on both control systems in place of a specific subcritical shutdown margin. The maximum reactivity fault is postulated to occur upon the accidental uncontrolled withdrawal of the highest worth control rod in the reactor in any critical configuration. Control rod ejection or dropout is specifically excluded from consideration in the General Design Criteria (See Section 3.1). The control requirements for the primary and secondary systems are discussed in detail in subsection 4.3.2.4, Control Requirements.

Criterion 27 COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under Extremely Unlikely Fault conditions, and with appropriate margin for malfunctions, such as a stuck rod(s), specified acceptable fuel damage limits are not exceeded.

RESPONSE

From the Response to Criterion 26 discussed above, each of the control systems individually satisfies GDC 27 for all off-normal conditions. The primary system is capable of terminating all Extremely Unlikely Faults without secondary system action assuming a stuck rod in the primary system. Therefore, when consideration is given to both the primary and secondary control systems simultaneously, it is clear that Criterion 27 is satisfied.

3.1.3.4 Cooling Systems

Criterion 30a HEAT TRANSPORT SYSTEM DESIGN

The heat transport system shall be designed to reliably remove heat from the reactor and transport the heat to the turbine-generator or ultimate heat sinks under all plant conditions including normal operation, off-normal and Extremely Unlikely Fault conditions. Consideration shall be given to provision of independence and diversity to provide adequate protection against common mode failures. The system safety functions for that part of the heat transport system utilized as the Reactor Residual Heat Extraction System (see GDC 27) shall be to:

(1) Provide abundant core cooling to prevent exceeding specified acceptable fuel design limits during normal operation and following off-normal conditions, and

(2) Provide abundant decay heat removal capabilities to prevent exceeding specified acceptable fuel damage limits and to maintain integrity of the reactor vessel following Extremely Unlikely Faults.

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

RESPONSE

The Heat Transport System components will be designed and fabricated in conformance with ASME Boiler and Pressure Vessel Code Section III and the applicable RDT standards.

The system is analyzed and designed in compliance with this Criterion. The details are described in Section 5.3 for the Primary Heat Transport System, in Section 5.4 for the Intermediate Heat Transport System, in Section 5.5 for the Steam Generator System, and in Section 5.6 for the Residual Heat Removal System which comprises the Steam Generator Auxiliary Heat Removal System and the Overflow Heat Removal System.

Criterion 30.b ASSURANCE OF ADEQUATE REACTOR COOLANT INVENTORY

The reactor coolant boundary and associated components, control and protection system shall be designed to limit loss of reactor coolant so that an inventory adequate to perform the safety functions of the heat transport system is maintained under normal operation, off-normal conditions, and Extremely Unlikely Faults.

RESPONSE

The high quality standards applied to the design, fabrication, erection, and testing of the coolant boundary (Criterion 30.c) give considerable protection against loss of reactor coolant. This is enhanced by the coolant volume control, which has a feed and bleed system to the reactor vessel, as described in Section 5.3.2.6.

Other major features of the reactor coolant system design which give assurance of adequate inventory at all times are the elevated pipe concept, and the provision of guard vessels around the major components as described in Section 5.3.2.1.1.

Criterion 30.c QUALITY OF REACTOR COOLANT BOUNDARY

Components which are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.

RESPONSE

The design, fabrication, erection, and testing to be employed on the reactor coolant boundary and the extensive quality control measures to be employed during each of the above phases will ensure that this boundary has extremely low probabilities of abnormal leakage, rapidly propagating failure, and gross rupture. The codes and standards to be observed in the design of the reactor coolant pressure boundary are given in Subsection 3.2.2. The quality control plan is discussed in Chapter 17.0. Further details are also given in the responses to Criteria 14 and 31, and in Section 5.3.3.6.

Criterion 31 FRACTURE PREVENTION OF REACTOR COOLANT BOUNDARY

The reactor coolant boundary shall be designed with sufficient margin to assure that when stressed under normal operation, off-normal conditions, and Extremely Unlikely Faults, (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, and other conditions of the boundary material under normal operation, off-normal conditions, and Extremely Unlikely Faults and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry and irradiation on material properties, (3) residual, steady state, and transient stresses and (4) size of flaws.

RESPONSE

Close control will be maintained over material selection and fabrication for the reactor coolant system to assure that the boundary will behave in a non-brittle manner.

Special requirements will be imposed on the quality control procedure for both the basic material of construction and on various subassemblies and final assembly for the reactor coolant loop components.

The analyses taking into account the service temperatures, service degradation of material properties, creep, and other conditions of the boundary material are given in Section 5.3.

Criterion 32 INSPECTION AND SURVEILLANCE OF REACTOR COOLANT BOUNDARY

Components which are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection of areas and features important to safety, to assess their leaktight integrity, and (2) an appropriate material surveillance program. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

RESPONSE

The design will provide the capability and accessibility for appropriate and practical inspection of the reactor coolant boundary during the service life of the system. This inspection capability will compliment the leak detection capability in assuring the integrity of the reactor coolant boundary, and is discussed in Chapter 5.

Leak detection instrumentation is provided for detecting and identifying the location of sodium leaks in the reactor coolant boundary. This instrumentation is described in Section 7.5.5 of this PSAR.

Criterion 33 REACTOR COOLANT AND COVER GAS PURITY CONTROL

Systems shall be provided to monitor and maintain reactor coolant and cover gas purity within specified design limits. These limits shall be based on consideration of the potential for (1) chemical attack, (2) plugging of passages and (3) increased radioactive contamination or equipment requiring maintenance or replacement.

RESPONSE

Plugging temperature indicators are used to monitor the saturation temperature of the total impurities in the primary sodium, the EVST coolant, and the Intermediate Heat Transfer System (IHTS) Sodium. Additionally, sodium samples are taken from these systems for laboratory analysis of sodium impurities. A continuous gas chromatograph is used to monitor the reactor cover gas impurity level. Laboratory analysis of cover gas samples from the EVST and IHTS provide control in these systems. These monitoring systems are described in Section 9.8.

Reactor coolant (primary sodium) and cover gas processing systems are also provided to maintain the reactor coolant and cover gas design purity. These systems are discussed in Sections 9.3.2 and 11.3.

Crtierion 34 INTERMEDIATE COOLANT SYSTEM

The intermediate coolant system shall be designed to transport heat reliably from the reactor coolant system to the reactor residual heat extraction systems as required for the reactor coolant system to meet its safety functions under all plant conditions including normal operation, off-normal and Extremely Unlikely Fault conditions. The intermediate coolant system shall contain coolant that is not chemically reactive with the reactor coolant. A pressure differential shall be maintained across a passive boundary between the reactor coolant system and the intermediate coolant system such that any leakage would flow from the intermediate cooling system to the reactor coolant system unless other provisions can be shown to be acceptable on some defined basis.

RESPONSE

The intermediate coolant system will use sodium coolant, as will the reactor coolant. A nominal positive pressure differential will be maintained, across the passive boundary inside the IHX, from the intermediate coolant side (tube side) to the reactor coolant side. The intermediate coolant system will be designed to adequately and reliably transfer heat, under all plant conditions, from the reactor coolant system by circulating non-radioactive sodium from the IHX tube side to the steam generators. These considerations and other details of the design of the intermediate coolant system is given in Section 5.4.

Criterion 35 FRACTURE PREVENTION OF INTERMEDIATE COOLANT BOUNDARY

Those portions of the intermediate coolant boundary which are safety related shall be designed with sufficient margin to assure that when stressed under normal operation, off-normal and Extremely Unlikely Fault conditions, (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, and other conditions of the boundary material under normal operation, off-normal and Extremely Unlikely Fault conditions and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry, (3) residual, steady state and transient stresses, and (4) size of flaws.

RESPONSE

Similar considerations as described in "Response" to Criterion 31 for the Reactor Coolant Boundary will apply to the intermediate coolant boundary. The details of the related design analyses are given in Section 5.4.

Criterion 36 INSPECTION AND SURVEILLANCE OF INTERMEDIATE COOLANT BOUNDARY

Components which are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection of areas and features important to safety, to assess their leaktight integrity, and (2) an appropriate material surveillance program for the intermediate coolant boundary. Means shall be provided for detecting intermediate coolant leakage.

RESPONSE

A sodium leak detection system is provided for in the intermediate coolant system for detecting sodium to gas leaks and this is described in Sections 5.3.2.5 and 5.4.2.5. The major portion of the intermediate boundary is in readily accessible areas, facilitating in-service inspection by visual and other methods. The materials surveillance program for the intermediate coolant boundary is discussed in Section 5.4.1.3.

Criterion 37 REACTOR RESIDUAL HEAT EXTRACTION SYSTEMS

The reactor residual heat extraction systems shall be provided to transfer residual heat from the reactor heat transport systems to ultimate heat sinks under all plant shutdown conditions following normal operation, off-normal and Extremely Unlikely Fault conditions. A passive boundary shall normally separate heat transport system coolant from the working fluids of the heat extraction systems. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

RESPONSE

The reactor residual heat extraction system will include adequate redundancy and diversity and have a combined heat removal capability sufficient to ensure fuel protection. The system will be designed to Seismic Category I and provided electric power by either the preferred power system or the standby diesel generators. The auxiliary feedwater subsystem part will be used for the first stage cooldown, consisting of largely sensible heat. The other part of the system will provide guaranteed residual heat extraction following the first stage cooldown.

The details of the design of the system are described in Section 5.6. Performance of the system under accident conditions is analyzed and given in Chapter 15.

Criterion 38 INSPECTION OF REACTOR RESIDUAL HEAT EXTRACTION SYSTEMS

The reactor residual heat extraction system shall be designed to permit appropriate periodic inspection of important components, such as heat exchanger and piping, to assure integrity and capability of the system.

RESPONSE

The design of the residual heat extraction system will provide the capability and accessibility for appropriate inspection during the service life of the system. The inspection capability complements the leak detection capability in assuring the integrity of the systems. Details of design of the system are given in Section 5.6.

Criterion 39 TESTING OF REACTOR RESIDUAL HEAT EXTRACTION SYSTEMS

The reactor residual heat extraction systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of their components, (2) the operability and the performance of the active components of the systems, and (3) the operability of each complete system, and under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation for reactor shutdown and following off-normal and Extremely Unlikely Faults conditions, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

RESPONSE

The design of the system will provide the capability for periodic and practical testing of both active and passive components of the system. Design provisions will also include special instrumentation and other facilities to perform system functional test during plant shutdown. Details are provided in Section 5.6.

Criterion 40 ADDITIONAL COOLING SYSTEMS

In addition to the heat rejection capability provided by the reactor residual heat extraction systems, systems to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided, as necessary. The system safety function shall be to transfer the combined heat load of these structures, systems, and components as required for safety 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.

RESPONSE

The Ex-Vessel Storage Tank (EVST) for the fuel is cooled by two completely redundant systems which are connected to the emergency power bus. No single failure of a component or loss of power from a single source will cause loss of cooling to the EVST. Other safety related systems are the cooling systems for EVST heat removal equipment and for the fuel storage tank in the fuel handling cell (FHC). These systems transfer heat from the equipment and cells to the Cooling System by means of two gas coolers for the EVST equipment cooling and by another cooler for the FHC. These cooling systems are designed to be capable of accomplishing the required safety function assuming a single failure.

The Overflow Heat Removal Service (OHRS) provides the capability to remove reactor decay heat 24 hours after shutdown in the event that all normal heat removal paths are unavailable. The makeup pumps of OHRS are cooled by the cooling system. This portion of the cooling system is designed consistent with the safety function of the OHRS.

The CRBRP is provided with cooling system to remove heat from all the inerted cell structures during normal plant operation. However, this cooling function is not safety related. During accident conditions, no cooling to any of these cell structures is necessary.

The Auxiliary Liquid Metal System is described in Section 9.3; the Recirculating Gas Cooling System in Section 3A.1.3; and the Auxiliary Cooling System in 9.7.

Criterion 41 INSPECTION OF ADDITIONAL COOLING SYSTEMS

The additional cooling 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 systems.

RESPONSE

The Safety-related portions of these Cooling Systems piping and equipment are located in accessible areas and may be periodically inspected. The sodium is radioactive, but the operation of the system and the leak detection equipment supplies the necessary information on the system integrity.

Criterion 42 TESTING OF ADDITIONAL COOLING SYSTEMS

The additional cooling systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of their components, (2) the operability and the performance of the active components of the systems, and (3) the operability of the complete systems 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 and the transfer between normal and emergency power sources.

RESPONSE

The safety-related portions of these cooling systems are designed such that they may be tested for integrity, operability, and perfomance on a periodic basis as required.

Details on testing of these systems are described in sections 3A.1, 9.7 and 9.9.

3.1.3.5 Reactor Containment

Criterion 50 DESIGN OF CONTAINMENT BUILDING STRUCTURE

The reactor containment structure, including access openings and penetrations, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions resulting from normal operation, off-normal conditions and any of the Extremely Unlikely Faults. The analyses shall consider the effects of all potential energy sources. The uncertainties in the calculated models and input parameters shall be conservatively treated:

RESPONSE

The containment structure, including access openings and penetrations, will be designed with sufficient conservatism to accommodate, without exceeding the design leakage rate, the peak pressure and temperature associated with conservatively postulated accident conditions.

The containment design consists of a free-standing, all welded steel vessel with a steel lined concrete bottom. The cylindrical shell will be embedded in concrete up to the level of the operating floor. Details of the design and analyses are given in Sections 3.8.2 and 6.2.

The ability of the containment to function as an effective enclosure in the event of sodium fires or radioactive releases is demonstrated in Sections 6.2 and 15.6. Third level design margin requirements are covered in Section 15.1.

Criterion 51 FRACTURE PREVENTION OF REACTOR CONTAINMENT BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under normal operation, off-normal and Extremely Unlikely Fault conditions (1) its metallic materials behave in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperature and other conditions of the containment boundary material during normal operation, off-normal and Extremely Unlikely Fault conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

RESPONSE

The containment vessel and its penetration sleeves will meet the material, design and technical process requirements of ASME-III subsection NE. Charpy V-north impact tests requirements will be in conformance with ASME-III Code, employing a lowest service metal temperature of +15°F. The design will consider uncertainties in material properties, residual, steady-state, and transient stresses, and material flaws, in addition to conservative allowable stress levels for all stressed elements of the containment boundary. Details of the containment design are given in Sections 3.8.2 and 6.2.

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 to verify the design leak rate.

RESPONSE

The reactor containment design will permit overpressure strength testing during construction and permit preoperational integrated leakage rate testing in accordance with Appendix J of 10CFR50. All equipment which may be subjected to the test pressure will be designed or arranged with suitable provisions so that periodic integrated leakage rate testing can be conducted. Further details are provided in Section 3.8.2 and 6.2.

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.

RESPONSE

The reactor containment and the containment isolation system will be designed so that appropriate periodic inspection of all important areas such as penetrations can be made. The design will also be such that an appropriate surveillance program can be maintained. The design will permit periodic testing at containment design pressure of the leaktightness of isolation valves and penetrations having resilient seals and expansion bellows. It will also permit demonstrating periodically the operability of the containment isolation system. Further information is given in Section 6.2.

Criterion 54 PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities in accordance with Criteria 55, 56, and 57, which follow. 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.

RESPONSE

The design of piping systems penetrating reactor containment will conform to this criterion in accordance with Criteria 55, 56, and 57.

Details of the isolation features provided are discussed in Section 6.2.4.

Criterion 55 REACTOR COOLANT BOUNDARY PENETRATING CONTAINMENT

Each line that is part of or directly connected to the reactor coolant boundary and that penetrates 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 will 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 in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment shall include consideration of the population density, use characteristics, and physical characteristics, of the site environs.

RESPONSE

The lines that penetrate the reactor containment and that are connected to the reactor coolant boundary, primary cover gas spaces, or inerted cell atmospheres are the argon and nitrogen supply lines, and the exhaust lines to RAPS and CAPS. While later evaluation may show that adequate protection for the health and safety of the public can be provided without two valves at the containment penetration, the present design includes two valves for prudence. Automatic action is specified if response is necessary within 10 minutes. Otherwise, remote manual initiation is provided. The provisions for containment isolation for these lines meet the requirements of GDC 54 and 55 and are described in detail in Section 6.2.

Criterion 56 REACTOR CONTAINMENT ISOLATION

Each line that connects directly to the containment atomsphere and penetrates 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.

RESPONSE

The following lines penetrate the reactor containment and are directly connected to the containment atmosphere:

Containment Ventilation Air Supply Line	(Section 9.6)
Containment Ventilation Air Exhaust Line	(Section 9.6)
Containment Vacuum Breakers	(If provided)

Each of these lines is provided with two automatic isolation valves, one inside containment and one outside. The design is in conformance with Criterion 56. Design details are provided in Section 6.2.

Criterion 57 CLOSED SYSTEMS PENETRATING CONTAINMENT

Each line of a closed system that penetrates reactor containment and is neither part of the reactor coolant boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve unless the system boundary is protected against accidents, extreme environmental conditions and natural phenomena, or unless it can be demonstrated that containment isolation provisions for a specific class of lines are acceptable on some other defined basis. The isolation valve, if required, 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.

RESPONSE

Each of the following lines of closed systems penetrates the reactor containment and is neither part of the reactor coolant boundary nor connected directly to the containment atmosphere:

Sodium Transfer Line Between Storage Tanks	(Section 9.3)
Sodium Transfer Line from EVST	(Section 9.3)
Auxiliary Coolant Fluid to Containment	(Section 9.7)

Auxiliary Coolant Fluid from Containment	(Section 9.7)
OHRS HX NaK Line to Containment	(Section 9.3)
OHRS HX NaK Line from Containment	(Section 9.3)

Each of these lines has at least one containment isolation valve capable of remote manual operation and located outside and as close to containment as practical. These lines and the associated containment isolation valve designs are discussed in Section 6.2.4.

The design of the IHTS lines meet the requirements of GDC 57 and are therefore not provided with isolation valves.

Criterion 58.a CONTAINMENT ATMOSPHERE CONTROL

Systems to control fission products and other radioactive 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 quantity of these substances released to the site environment following postulated accidents.

Systems shall be provided as necessary to prevent and control the effects of potential chemical reactions which would threaten the integrity of the reactor containment. The necessity of such systems should consider the effects of sodium leakage and its potential reaction with oxygen and its potential for hydrogen generation when in contact with concrete.

The above systems 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 off site 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.

RESPONSE

The Heating, Ventilating, and Air Conditioning (HVAC) system provides sufficient outside ventilation air to dilute radioactive gas leakages to levels below the limit specified in 10CFR20. Double containment isolation valves (butterfly type) are provided, where the duct system penetrates the containment boundary, to permit isolation of the containment atmosphere under accident conditions.

The design measures are described in Sections 9.6.2 and 12.2. Considerations for accident conditions are discussed in Section 6.2.3.

Criterion 58.b INSPECTION OF CONTAINMENT ATMOSPHERE CONTROL SYSTEMS

The containment atmosphere control 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.

RESPONSE

Control systems on the Inert Gas Receiving and Processing and the Heating, Ventilating and Air Conditioning System will be designed to permit appropriate periodic inspection of important control components. These components will, in general, be located in areas where periodic inspections can be performed on a schedule consistent with the anticipated component life and available redundancy.

Detailed information is provided in Sections 9.5 and 9.6.

Criterion 58.c TESTING OF CONTAINMENT ATMOSPHERE CONTROL SYSTEMS

The containment atmosphere control systems shall be designed to permit appropriate periodic 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.

RESPONSE

Both pre-operational and periodic testing of control systems will be performed to assure conformance with Criterion 58.c. The inspection and test requirements for the Heating, Ventilating, and Air Conditioning System are given in Sections 9.6 and Chapter 16.

3.1.3.6 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 and off-normal conditions. 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.

RESPONSE

An extensive system using evaporators and demineralizers has been designed for liquid waste treatment and disposal. Practically all of the intermediate activity level liquid radwaste is reused after decontamination while the low activity liquid radwaste is continually released after decontamination. A detailed description of the Liquid Radwaste System design and design basis is given in Section 11.2.

The CAPS and RAPS portions of the Inert Gas Receiving and Processing System provide means to control the release of radioactive gases during normal reactor operations and off-normal conditions. The design and design basis for these systems is described in Section 11.3.

Solid wastes are solidified in cement (except for clothing, paper, etc.) and processed in 55-gallon drums for eventual disposal in licensed burial grounds. This system is described in Section 11.5.

Criterion 61 FUEL STORAGE FOR 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 operating conditions, off-normal conditions and Extremely Unlikely Fault 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 heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

RESPONSE

Fuel storage facilities and fuel handling equipment important to safety will be designed to provide accessibility for performing inspection, maintenance and testing activities. All fuel storage facilities and fuel handling equipment will be shielded for radiation protection to meet the requirements specified in 10CFR20, 50 and 100. Containment, confinement, and filtering will be provided for all fuel storage facilities and fuel handling equipment containing radioactive material to limit any radioactive releases below those radiation doses specified in 10CFR20 and 100 as appropriate. Adequate cooling capability will be provided for spent fuel storage and spent fuel handling equipment to assure decay heat removal with enough reliability, independence and redundancy to accommodate all plant conditions. A significant reduction of sodium coolant inventory in the spent fuel storage facilities under accident conditions will be prevented by employing high quality design and construction standards to the spent fuel storage vessels, by guard jackets surrounding the storage vessels and by anti-syphon features. The design measures necessary to meet this criterion are described in Section 9.1 for the fuel storage and handling system.

The radwaste system will also be designed to conform to this criterion. Surveillance of safety related items is accomplished by virtue of routine monitoring of the day-to-day operations of the systems. Appropriate shielding filtration, heat removal, and inventory control will be provided. The concentrated liquid radwaste is solidified and shipped to licensed burial sites. The design and operating procedure will preclude excessive release of contaminated water in any postulated accident. The design measures necessary to meet this criterion are described in Section 11.

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.

RESPONSE

Geometrically safe configurations are employed to preclude criticality in new and spent fuel storage facilities and in fuel handling equipment. The appropriate safety measures and the design features necessary to meet this criterion are described in Section 9.1 for the fuel storage and handling system.

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.

RESPONSE

Monitoring systems are provided to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels. Appropriate local alarms will be set off and annunciated in the control room to warn personnel of potential safety problems.

Systems which require heat removal capabilities are provided with temperature and liquid level monitoring instrumentation. The instrumentation is connected to suitable safety systems which initiate appropriate actions to assure sufficient heat removal capabilities.

Radiation monitoring systems are provided to alarm on excessive radiation levels in all equipment cells and operating areas. Personnel will be warned of the excessive radiation levels through the alarm systems while appropriate action will be taken to control the release of radioactive materials to the environment.

The above monitoring and safety systems are described in Sections 9.1, 11.2, 11.3, 11.4, and 12.2.

Criterion 64 MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmospheres, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, off-normal conditions, and Extremely Unlikely Faults.

RESPONSE

The containment atmosphere will be continuously monitored during normal and off-normal conditions, using the containment particulate, gas, and iodine monitors, which will be located in the ventilation exhaust downstream of the containment isolation valves in the Intermediate Bay of the Steam Generator Building. In the event of Extremely Unlikely Faults, samples of the containment atmosphere will be obtained via a bypass sample line arrangement to provide data on existing airborne radioactivity concentrations within the containment. Fixed continuous airborne radioactivity monitors will be provided in frequently occupied work areas. The presence of radioactivity in the normal plant effluent discharge paths and in the site environs will be continuously monitored during normal, off-normal and Extremely Unlikely Faults by the plant radiation monitoring systems and by the off site radiological monitoring program for this plant. These systems are described in detail in Sections 11.4 and 11.6, and 12.2.

APPENDIX C

GENERAL ATOMIC STANDARD SAFETY ANALYSIS REPORT

SECTION 3.1

"CONFORMANCE WITH AEC GENERAL DESIGN CRITERIA"

DOCKET-STN-50535--1

GA-A13200

GASSAR 6

GENERAL ATOMIC
STANDARD SAFETY
ANALYSIS REPORT

MASTER

CONTROL NO. 030

GENERAL ATOMIC

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH AEC GENERAL DESIGN CRITERIA

A summary explaining how the principal design features of safety-related structures, components, equipment, and systems of the HTGR NSS meet the intent of Appendix A to 10CFR50 is provided below. A summary is also given for each criterion, and applicable GASSAR sections which present more detailed information are referenced.

3.1.1 Overall Requirements

3.1.1.1 Criterion 1: Quality Standards and Records

To be provided by applicant.

3.1.1.2 Criterion 2: Dosign Bases for Protection Against Natural Phenomena

To be provided by applicant.

3.1.1.3 Criterion 3: Fire Protection

To be provided by applicant.

3.1.1.4 Criterion 4: Environmental and Missile Design Bases

To be provided by applicant.

3.1.1.5 Criterion 5: Sharing of Structures, Systems, and Components

To be provided by applicant.

3.1.2 Protection by Multiple Fission Product Barriers

3.1.2.1 Criterion 10: Reactor Design

a. Criterion

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

b. Discussion

On the basis of the following information, it is concluded that Criterion 10 is satisfied.

The fuel for this reactor is in the form of coated uranium carbide and thorium oxide kernels. Each kernel is coated with pyrolytic carbon; those particles which contain the uranium carbide will have an additional layer of silicon carbide. These coatings form a highly reliable barrier for fission products. The kernels are designed to withstand normal operating temperatures without exceeding failure limits over the design life and can also withstand temperatures well above normal operating temperatures for short periods without rapid deterioration of the fission product barrier (see Section 4.4).

The graphite fuel elements which contain the fuel particles are known to increase in strength up to temperatures in the range of 4000° to 5000°F. The fuel particle and graphite temperature history, including hot-spot factors, is discussed under core thermal design bases in Section 4.4.

The design of the plant control and plant protection systems has been based upon maintaining plant parameters within design limits both during normal operation and in cases of anticipated transient events. For those situations that can be anticipated, the transient response of the plant has been calculated by means of a digital computer program which simulates the basic features of the plant design. The analyses show no deviations of plant parameters sufficient to cause core overheating or primary coolant system damage for anticipated occurrences, including the following:

1. Step-load changes of ±10% of rated load (see Section 7.7).

2. Normal load change at the fastest permissible rate (see Section 7.7).

3. Trip of a main turbine-generator (see Section 7.7).

4. Sudden shutdown of one primary coolant loop (see Section 7.7).

5. Reactor trip (see Section 7.7).

6. Rod withdrawal terminated by rod withdrawal prohibit (see Chapter 15).

These results have been obtained by simulated operation of the plant control system in its normal mode (except in the cases of reactor trip and turbine trip).

Protective features have been incorporated in the plant design to protect against or minimize the effects of credible failures. For example, to minimize core damage from a steam leak, a steam generator isolation and dump system is included in the design. The system is actuated by a redundant moisture monitoring system. Upon receipt of a signal from the monitors indicating a high primary coolant moisture level, the reactor is tripped, and the leaking loop is shut down and isolated, and its contents are dumped to the steam/water dump tank (see Section 7.6).

The main cooling loops provide core cooling under normal and anticipated transient conditions. If for any reason all main cooling loops become inoperable, core cooling can be provided by the three auxiliary cooling loops which are independent of the main loops and of each other. Safe cooling of the core is provided under all pressurized and depressurized conditions even if one of the loops fails to function.

Following a postulated loss of all off-site power and the occurrence of the SSE, redundant standby on-site power sources provide adequate power to facilitate a safe shutdown of the plant using Seismic Category I equipment, even if one of the power sources fails to function.

3.1.2.2 Criterion 11: Reactor Inherent Protection

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 11 is satisfied.

Since the HTGR has a negligible coolant density feedback effect and the moderator density is insensitive to change in the temperature range of interest, the power coefficient is closely related to the reactor core temperature coefficient. As discussed in Section 4.3, the total temperature coefficient and, in particular, its prompt component are negative from room temperature to temperatures above 5000°F.

The worst reactivity insertion arises from the uncontrolled withdrawal of one control rod pair. Any uncontrolled increase in reactivity results in an increase in power, which in turn produces a prompt increase in fuel temperature. Due to Doppler broadening of the thorium resonances, there is a prompt, negative reactivity feedback with temperature, which tends to compensate for the reactivity increase.

During a load increase, the coolant flow, power-regulating rod pair, and feedwater flow are automatically regulated as follows:

1. An increase in power initiated by opening of the turbine valves causes an increased steam flow and a decreased throttle pressure, both of which lead to an increase in the feedwater flow set point.

2. The measured increase in feedwater flow and the subsequent reduction in main steam temperature act to increase the main circulator speed set point.

3. The increased steam flow and the resulting decrease in the reheat steam temperature cause the

regulating rod pair to be moved out and reactor power to be increased.

The net effect is a higher steady-state power level and a higher average core temperature. An increase in power from 25% to 100% rated power, for example, causes an increase in the average core temperature of about 300°F and a decrease in reactivity between 0.01Δk and 0.005 Δk, depending on time in cycle and type of cycle (initial core, non-recycle, equilibrium, etc.).

3.1.2.3 Criterion 12: Suppression of Reactor Power Oscillations

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 12 is satisfied.

The stability of the total core power level and its spatial distribution is enhanced by the stability of the coolant and moderator at all operating temperatures. Subsection 3.1.2.1 discussed the capability of the fuel and graphite fuel elements to withstand high temperatures, i.e., temperatures well above those that would be experienced during normal operation.

The nuclear stability of the reactor with respect to the various process variables is assured by negative prompt and total temperature coefficients from room temperature to beyond 5000°F (Section 4.3). Thus, the reactor tends to load-follow even if no automatic or operator action is taken. Similarly, through the use of series steam turbine circulator drives, the coolant flow in the primary loop tends to follow the load on the system because of the direct relationship between load and steam flow.

Calculations based on conservative assumptions, which included neglecting the stabilizing effect of the negative temperature coefficient, show that the core is stable against axial power oscillations such that power perturbations (due to rod motion, etc.) are highly damped and do not lead to sustained xenon-induced power oscillations (see Section 4.3). In addition, calculations show that if temperature feedback effects are included, the core is stable against radial and azimuthal power oscillations, and any perturbation is highly damped. However, the design assumes that radial and azimuthal oscillations can occur, and instrumentation, including thermocouples and in-core neutron detectors, is provided to detect them. These assumed oscillations are slow and can be effectively controlled by operator action. The in-core instrumentation is also used to monitor the stability of the axial power shape during reactor operation.

3.1.2.4 Criterion 13: Instrumentation and Control

a. Criterion

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions. These variables and systems include those 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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 13 is satisfied.

(1) General

The automatic control systems (see Section 7.7) are designed to maintain plant variables within prescribed limits. Automatic corrective actions are made by the plant control system (see Section 7.7) when variables exceed preset limits. This includes situations which, if uncorrected, could reduce reactor cooling capability or result in major equipment damage. Manual corrective actions can also be made by the operator when alarms have indicated that variables have exceeded preset limits. Instrumentation is provided in the control room to alarm those conditions which deviate from normal by an amount sufficient to require operator actions.

(2) Nuclear Instrumentation

The redundant nuclear instrumentation monitors neutron flux from a reactor shutdown condition to a level greater than design power (see Section 7.7). The information is displayed in the control room. Loss of a single instrument power source does not result in loss of neutron monitoring indication because of the redundancy and separation of channels on the various instrumentation power sources.

(3) Reactor Control

Reactor control is maintained either automatically or manually with the rod control system (see Section 7.7). Reactor shutdown can be accomplished with the rod control system or, alternately, by means of the manual or automatic reactor trip system. A separate diverse shutdown means is provided by the reserve shutdown system (see Section 4.2).

(4) Control Rod Position

Each of the control rod drives is equipped with redundant rod position potentiometers. Rod position is displayed on redundant cathode ray tube (CRT) terminals in the control room through the DAP system. In addition, the position of each control rod pair is continuously logged and stored in the protected mass memory of the DAP system (see Section 7.7). A printout of the positions of all rods is readily available from the DAP. Interlocks permit only one pair to be in motion outward at one time; however, several pairs may be in motion inward at a time (see Section 7.6).

Each control rod drive control unit contains two redundant voltage comparators which detect rod "in"/"out" position limits from the redundant position potentiometers. "In"/"out" indication is also displayed via the DAP system. Additionally, rod "in" position indicating lights are provided in the control room and in the safe-shutdown room.

(5) Reserve Shutdown System

The reserve shutdown system provides shutdown capability independent of the normal control rod system. Boronated graphite pellets are stored in a hopper in each refueling penetration, from which they can be released by operator action into the core via cylindrical closed-end guide tubes, if required (see Section 4.2).

(6) Primary Coolant Pressure Boundary Protection

To protect the PCRV from possible overpressure resulting from the inleakage of water/steam from a steam generator failure, the primary system is protected by a moisture monitoring system which actuates a reactor trip and the isolation and dump of the faulty steam generator (see Sections 7.2 and 7.6).

A reactor trip signal is also generated on high primary system pressure and results in cooling of the helium and subsequent reduction in the PCRV pressure.

(7) Containment Pressure Protection

Steam line isolation is provided to protect the reactor containment from overpressure as a result of steam line rupture. A containment high-pressure signal coupled with a normal PCRV pressure initiates closure of the isolation valves in the feedwater, the main steam, and the reheat lines. This function is performed by the plant protection system (see Section 7.6).

3.1.2.5 Criterion 14: Reactor Coolant Pressure Boundary (Primary Coolant System Boundary)

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 14 is satisfied.

The main structual component of the reactor coolant pressure boundary is the PCRV. The PCRV is designed, fabricated, erected, and tested according to the requirements of Section III, Division 2, of the ASME Boiler and Pressure Vessel Code. The structural strength of the PCRV is provided by reinforced concrete coupled with linear and circumferential prestressing systems. The redundancy provided in the structural elements reduces the probability of a gross failure of the PCRV to extremely low values such that gross failures of the PCRV are not considered credible. In addition to extensive analysis required by Section III, Division 2, the PCRV design has been verified by model testing. Such tests have demonstrated that even in the presence of pressures which are a factor of two higher than maximum cavity pressure, gross structural failure of the PCRV is virtually impossible to achieve.

All internal cavities of the PCRV are lined with steel liners which act as concrete forms during construction and create a leak-tight membrane during reactor operation. Liner design, fabrication, erection, and testing are done according to the requirements of Section III, Division 2. Following prestressing of the PCRV, and throughout reactor operating life, the liners are continuously loaded in a state of biaxial compression owing to inward movement of the concrete. Anchor studs are provided in critical liner regions to assure that liner movement conforms to that of the concrete. Therefore, gross structural failure of the liners is not a serious consideration.

In accordance with the requirements of Section III, Division 2, PCRV penetrations and closures which are unbacked by concrete for load-carrying purposes are designed, fabricated, erected, and inspected in accordance with the requirements of Section III, Division 1, of the ASME Boiler and Pressure Vessel Code. Thus, the required high integrity of the structure is obtained. This includes selection and certification of materials with impact properties that assure a low probability of a rapidly propagating failure. The penetrations are provided with redundant means of transferring loads from the penetration assembly to the concrete. Each penetration with a removable closure is fitted with a gas-tight bolted closure with either a seal-welded or bolted and double-gasketed design. On the gasketed closures, purified helium is supplied from the helium purification system to the space between the penetration closure gaskets at a pressure at least 2% higher than the primary system pressure so that any leakage of the cavity-side gasket is of purified helium into the reactor cavity. Any outward leakage into the containment is also purified helium since the interspace pressure is above containment pressure.

The in-service leakage monitoring program, combined with the high level of reliability provided with the initial design, assures that the probabilities of abnormal leakage, rapidly propagating failure, and gross rupture of the reactor coolant pressure boundary are so low that they are practically negligible.

3.1.2.6 Criterion 15: Reactor Coolant System Design

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 15 is satisfied.

The only potential sources of overpressure for the reactor coolant system are moisture inleakage or temperature rise resulting from a reactivity accident.

Several levels of overpressure protection are provided, including (1) instrumentation which protects against excessive coolant temperature rise resulting from excess reactivity; (2) instrumentation which automatically trips the reactor if the coolant pressure rises above the normal operating range; (3) the moisture monitor/steam generator dump system which protects against pressure rise caused by moisture inleakage; and (4) the PCRV pressure relief system which provides a final backup level of protection in the case of failure of all other protective systems. A description of the overpressure protection systems is given in Section 5.2. Reactivity and overpressure protection instrumentation is described in detail in Chapter 7, and the moisture monitor/dump system is described in Sections 7.2 and 7.6 and Chapter 10.

The systems for reactivity and overpressure monitoring, moisture monitoring, and steam generator dump provide reliable and effective overpressure protection for the reactor coolant system. The final level of protection provided by the PCRV pressure relief system gives ultimate assurance against the possibility that the pressure boundary design conditions will be exceeded.

3.1.2.7 Criterion 16: Containment Design

To be provided by applicant.

3.1.2.8 Criterion 17: Electrical Power Systems

To be provided by applicant.

3.1.2.9 Criterion 18: Inspection and Testing of Electrical Power Systems

To be provided by applicant.

3.1.2.10 Criterion 19: Control Room

To be provided by applicant.

3.1.3 Protection and Reactivity Control Systems

3.1.3.1 Criterion 20: Protection System Functions

a. Criterion

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 initiate the operation of systems and components important to safety.

b. Discussion

On the basis of the following information, it is concluded that Criterion 20 is satisfied.

The reactor plant protection system (PPS) (see Chapter 7) is designed to prevent or suppress conditions that might result in exceeding of acceptable fuel design limits. The core safety limit is discussed in Section 4.4.

In general, the limitations for fuel are less stringent than those for other components. Therefore, by minimizing transients for the protection of other equipment, additional protection over that which would otherwise be needed is provided for fuel. The major portions of the PPS that contribute to limiting fuel damage and to operation within acceptable limits are automatic reactor trip (see Section 7.2), single rod withdrawal interlock (see Section 7.6), and the steam generator isolation and dump system (see Section 7.6).

The effectiveness of the plant protection system in limiting fuel damage is shown in the safety analysis (see Chapter 15). Pertinent portions are rod withdrawal accidents and steam/water leaks into the primary coolant system.

3.1.3.2 Criterion 21: Protection System Reliability and Testability

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 21 is satisfied.

The PPS encompasses those devices whose failure to provide protection when needed could result in unacceptable consequences for the public. The PPS meets IEEE Standard 279-1971. For the systems requiring automatic initiation, redundant independent channels monitor each initiating parameter during power operation. The physical construction of the instrumentation is designed to ensure system safety, reliability, testability, and maintainability. In accordance with IEEE Standard 279-1971, a high-quality level is maintained throughout construction (see Section 7.1).

Since the basic PPS logic is redundant, it follows that the failure of an entire channel will not preclude obtaining the desired protective action. The system is designed and built so that no single failure prevents a protective action.

The primary coolant boundary is protected by the reactor main-loop shutdown system, steam generator isolation and dump, and the main loop shutdown system, all of which are automatically initiated.

In general, three independent sensing circuits are provided for each PPS input parameter.

The basic input channel logic system for the reactor trip system is a general two-out-of-three system. The output power stage is one-of-two, twice. Independence is maintained on the redundant channels and includes the use of independent power sources. Sensor diversity is provided for major protective functions. Details on the reactor trip system are given in Section 7.2.

The steam generator isolation and dump system has redundant steam/water dump valves and circuitry, either one of which is adequate to dump a steam generator. The associated feedwater shutoff can be accomplished by the closing of either the feedwater block or trim valve. Reverse superheated steam flow is prevented by either the powered stop valve or the check valve on the main superheated steam line. Further information on the steam generator isolation and dump system is given in Section 5.2 and Chapter 10.

The containment pressure protection system (see Section 7.6) utilizes two of three sensing channels and redundant logic networks to operate the various isolation valves.

The main loop shutdown system utilizes two of three sensing channels and redundant logic networks and isolation valves for each steam generator loop (see Section 7.6).

The single rod withdrawal interlock provides redundant means of assuring that only one control rod pair can be withdrawn at a time.

Redundant means of closing the primary coolant shutoff valves are provided to ensure that core bypasses by the primary coolant can be terminated on loss of a loop (see Section 5.5).

Redundancy and single failure protection in the CACS are achieved by the use of independent auxiliary cooling loops. Adequate core cooling is provided even if one of the loops fails to function.

The principal method for checking sensors is by cross-checking between redundant channels. The moisture monitors can be checked via moisture injection into the system through the sampling rake test line. The outputs of the sensors are continually scanned by the DAP system to detect sensor malfunctions. The outputs may also be visually compared by the operator to verify their operation.

Each channel, exclusive of the sensor, is capable of being independently tripped by simulated test signals during operation to verify its ability to act through to the final output stage. Thus, by successively observing the actions of the channel outputs, the operator can positively determine the functional operability of the system. Additional facilities are provided for setting and testing the trip levels of bistable amplifiers.

The reactor trip system can be checked through to and including output power contactors. The steam/water dump system can be tested through to and including actuation of the steam/water dump valves. Steam generator isolation can be tested by partial stroking of the steam and feedwater line isolation valves.

The main loop shutdown system can be tested through partial stroking of the isolation valves. Similarly, the containment pressure protection system can be checked from sensor through to partial stroking of the associated isolation valves.

A more detailed description of the testing provisions of the plant protection system may be found in Chapter 7.

3.1.3.3 Criterion 22: Protection System Independence

a. Criterion

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 the system 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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 22 is satisfied.

The plant protection system, including necessary wiring, is designed and qualified to operate in the most severe environments (nuclear radiation, temperature, pressure, and humidity) expected during normal plant operation and accident conditions for which it is required to remain in operation. The system is also Seismic Category I.

Diversity has been employed within the protective system design. Functional diversity is provided for major protective functions. The reactor trip system power control reflects diversity in the method of power disconnect to the control rod drives. The reserve shutdown system, though not a part of the plant protection system, is Seismic Category I and is a diverse backup for the reactor trip system.

3.1.3.4 Criterion 23: Protection System Failure Modes

a. Criterion

The protection system shall be designed to fail into a safe state or 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, radiation) are experienced.

b. Discussion

On the basis of the following information, it is concluded that Criterion 23 is satisfied.

The reactor trip system, including the final control actuators (drive motors), is designed to release the control rods, allowing them to drop into the core by gravity upon loss of power (see Section 7.2).

The remaining protection systems are normally de-energized and must "turn on" to produce an output action. Redundant outputs independently powered by support systems designed to single failure criteria are provided to ensure action when required.

Some process input channels to the reactor trip system or other circuitry receive their signals from process transmitters that do not fail safely on loss of power. It is recognized that these devices can fail in either a safe or unsafe direction. Failures are alarmed in the control room. However, as previously mentioned, these input channels are redundant, and loss of one channel will not prevent a required safety action or initiate an unnecessary action. Physical separation of instruments and circuitry is provided to prevent simultaneous damage of redundant instrumentation or circuitry. The DAP system continuously monitors the status of each instrument channel and alarms excessive deviations. Equipment qualification to adverse environments has been previously discussed in Criterion 22.

3.1.3.5 Criterion 24: Separation of Protection and Control Systems

a. Criterion

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 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 to assure that safety is not significantly impaired.

b. Discussion

On the basis of the following information, it is concluded that Criterion 24 is satisfied.

The control instrumentation is separate from the protection instrumentation. There are no instances in which control and protection channels are combined, whereby failure or removal from service of any control instrumentation system component or channel can result in a system that does not satisfy all requirements for the protection channels.

Signals from the protection system that are used for other purposes, such as monitoring or alarm by the DAP system, are buffered in the PPS to prevent feedback into the protection system. Therefore, a failure in the external system does not adversely affect the operation of the protection system.

3.1.3.6 Criterion 25: Protection System Requirements for Reactivity Control Malfunctions

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 25 is satisfied.

All credible reactivity transients are considered in Chapter 15. It is shown that acceptable fuel design limits are not exceeded for the worst single malfunction of the reactivity control systems. This analysis indicates that the worst malfunction possible is the continous uncontrolled withdrawal of the most reactive rod pair at any time in life.

The analysis, which was based on conservative assumptions regarding rod pair worth, reactivity feedback, etc., shows that the protective systems provide the necessary protection of the core.

3.1.3.7 Criterion 26: Reactivity Control System Redundancy and Capability

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 26 is satisfied.

Normal reactivity control is provided by the control rods described in Section 4.2. A second system, the reserve shutdown system described in Section 4.2, provides an independent means of reactivity control which is based on a design principle which is different from that of the primary control system.

Both control systems can shut down the reactor from any normal or abnormal hot condition and indefinitely hold it subcritical under cold conditions, as documented in Section 4.3. The normal control system can also indefinitely maintain core subcriticality under cold conditions with the most reactive rod pair (or any combination of rod pairs of equal worth) fully withdrawn and all other rod pairs inserted. Shutdown margins with the two most reactive rod pairs withdrawn and the margins for the reserve shutdown system under similar conditions are documented in Section 4.3.

The control system can prevent fuel design limits from being exceeded during all normal and abnormal transients, and it is capable of producing safe core subcriticality. This analysis assumed the worst possible combination of temperature coefficient of reactivity, stuck rod worth, and control rod bank worth.

The flexible control system design for the reserve shutdown system, described in Section 4.2, ensures that this system is also capable of preventing fuel design limits from being exceeded during power transients if its use should be required. It is not designed for plant control under normal operating conditions; rather, it is designed as a backup system and is based on a different design principle than the primary control system. As documented in Section 4.3, it is capable of independently and safely shutting down the reactor from any normal or upset condition without exceeding fuel design limits.

3.1.3.8 Criterion 27: Combined Reactivity Control System Capability

a. Criterion

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 an appropriate margin for stuck rods the capability to cool the core is maintained.

b. Discussion

On the basis of the following information, it is concluded that Criterion 27 is satisfied.

It is demonstrated in Sections 4.2 and 4.3 and Chapter 15 that the control systems have the following capabilities:

1. Both the control system and the Reserve Shutdown System (RSS) are capable of independently making and maintaining the reactor subcritical at any time in life, at any temperature, with the most reactive unit (rod pair or RSS hoppers) inoperative.

2. The control system and the RSS are capable of independently producing and maintaining subcriticality at refueling temperature for at least two weeks with the two most reactive adjacent units (either rod pairs or RSS hoppers) inoperative.

3. In both systems, any single unit, any combination of units, or all units can be independently inserted into the reactor core at any time.

Thus, either reactivity control system has the capability of shutting down the reactor and maintaining the core subcritical under all conditions. Poison addition by means of the core cooling systems is not utilized.

3.1.3.9 Criterion 28: Reactivity Limits

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 28 is satisfied.

(1) Coolant Pressure Boundary

HTGRs utilize a single-phase primary coolant and solid neutron moderating material. Therefore, a large change in reactivity cannot lead to phase changes in either the moderator or coolant, and positive pressure pulses resulting from such an accident are negligible. The reactivity addition from a rod withdrawal is limited to that of a single rod pair and will not damage the pressure barrier, disrupt the core or PCRV internals, or impair the core cooling capability.

The PCRV liner and the penetration primary closures are capable of accommodating, without rupture, any static and dynamic loads imposed on them as a result of any sudden release of energy to the coolant, since all possible energy releases result in pressure levels lower than the design pressure for those components. The phenomenon of a sudden large release of energy in a coolant is restricted to two-phase coolants in which a small change in heat or energy balance could cause an extremely large change in coolant density and therefore pressure. The rapidity of coolant energy changes in the HTGR is restricted by the use of helium, a single-phase primary coolant.

(2) Core Reactivity Increases

Core reactivity is primarily controlled by the control rod system consisting of individually driven control rod pairs. Each control rod channel ends at the graphite reflector block. In the unlikely event of the failure of a supporting cable, the reflector block prevents control rods from dropping out of the core.

Each control rod pair is suspended from its control rod drive, which is housed in one of the refueling penetrations set in the PCRV top head. The design pressure for the penetration closures is equal to the PCRV relief valve set pressure. However, even if a closure in a refueling penetration failed, a set of thick steel hold-down plates covering the PCRV top head would limit the movement of a refueling penetration closure to less than 1 in. Therefore, ejection of control rods from the core due to coolant pressure is prevented.

Control rod withdrawal is limited to the maximum speed of the control rod drives. Interlocks on the control rod drives are such that only one rod pair can be withdrawn at a time. The maximum reactivity insertion rate due to an uncontrolled withdrawal of the highest-worth rod pair during normal operation at full power is discussed in Section 4.3 and Chapter 15. Treatment of such a condition is given in Chapter 15 and demonstrates the effectiveness of the protection system in limiting the reactivity addition such that the increase in vessel pressure is negligible.

Another possible method of introducing positive reactivity into the core is a sudden decrease in core temperatures. However, the high thermal capacity of the core makes large changes of this type impossible under almost all conditions. Significant changes in core temperature over relatively short periods of time can only occur after a fast reduction in power to a very low level with the coolant flow rate kept high. Even in such cases, the maximum rate of reactivity addition is less than that corresponding to the uncontrolled rod withdrawal accident (see Chapter 15). Hence, the consequences are also less. In a similar manner, the effect of sudden change in coolant temperature ("cold water accident") is negligible (see Chapter 15).

A steam generator rupture results in the addition of hydrogen to the core, which increases core reactivity by reducing neutron leakage, reducing the thorium-232 capture rate and lessening the effectiveness of the control poison. The maximum inleakage rate and the protective-action-limited inleakage are discussed in Chapter 15. The resulting reactivity addition rate and total reactivity are less than those experienced in the rod withdrawal accident.

Since the reactivity coefficient due to pressure change is extremely small, the reactivity effect of any change in coolant pressure is negligible.

3.1.3.10 Criterion 29: Protection Against Anticipated Operational Occurrences

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 29 is satisfied.

The design of the protection and reactivity control systems is discussed in detail in Chapter 7. The discussion in this chapter demonstrates that the systems can accomplish their safety functions with high reliability.

3.1.4 Fluid Systems

3.1.4.1 Criterion 30: Quality of Reactor Coolant Pressure Boundary (Primary Coolant System Boundary)

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 30 is satisfied.

As stated in the response to Criterion 14, the structural strength of the PCRV is provided by a multiplicity of redundant reinforcing and prestressing elements. This redundancy minimizes the probability of failure of the PCRV.

The PCRV cavity and penetration liners are designed, fabricated, and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 2. The above requirements ensure that the highest quality standards available are used.

Significant and uncontrolled leakage of reactor coolant through the pressure boundary to the reactor containment atmosphere will be detected by radiation monitors in the reactor containment. It is expected that operational leakage will be small since the pressure boundary is designed to ensure the integrity of the system (see Chapter 5).

The designs of the bolted penetration closures and their sealing arrangements are such that helium leakage tests can be conducted at the time of assembly of the closures and at any subsequent time during operation of the reactor. All bolted penetration closures are sealed either by concentric, double-gasketed joints or by seal welding.

Purified helium is supplied from the helium purification system to the spaces between the primary and secondary gaskets on the gasketed closures. The purified helium is at a constant pressure at least 2% higher than the primary system pressure so that any primary gasket leakage will be of purified helium into the primary coolant. Outward leakage from the secondary gasket into the containment will also be purified helium.

The pressurizing line is monitored for flow. Such flow will indicate leakage at one or both gaskets. If a large leakage is observed, that particular section will be isolated at a pressure below the reactor primary coolant pressure. An increase in pressure in the isolated section indicates a leak in the primary gasket, and a decrease indicates a leak in the secondary gasket.

3.1.4.2 Criterion 31: Fracture Prevention of Reactor Coolant Pressure Boundary (Primary Coolant System Boundary)

a. Criterion

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and 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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 31 is satisfied.

The PCRV makes up the major portion of the reactor coolant pressure boundary. The redundant structural members of the PCRV (rebar and linear and circumferential prestressing elements) provide assurance against rapidly propagating fracture of the PCRV itself.

The steel cavity liners which serve as a leak-tight membrane for the PCRV are backed by concrete and are under general biaxial compression during reactor operation. Therefore, rapidly propagating fracture of these liners is not considered credible. Ductile behavior of these liners is further ensured by satisfying the material impact requirements of Section III, Division 2, of the ASME Boiler and Pressure Vessel Code. According to this code, the material nil-ductility transition temperature (T_{NDT}) must be at least 60°F below the lowest metal service temperature during reactor operation. Furthermore, for irradiated regions of the liner, the shift in T_{NDT} due to irradiation is considered.

The ductile behavior of the steel penetrations and closures of the PCRV that are not backed by concrete for load-carrying purposes is ensured by satisfying the fracture toughness requirements of Section III, Division 1, of the ASME Boiler and Pressure Vessel Code. Compliance with this code requires that protection against nonductile fracture be provided by assuring that the lowest metal service temperature is sufficiently above the material reference nil-ductility transition temperature (RT_{NDT}), considering material properties, loading conditions, and presence of flaws.

The materials for the fabrication of all pressure boundary components are subjected to the appropriate inspection and quality control requirements specified by the ASME Boiler and Pressure Vessel Code, Section III, Divisions 1 and 2, in order to control defects within accepted levels.

For additional details, see Section 5.2.

3.1.4.3 Criterion 32: Inspection of Reactor Coolant Pressure Boundary (Primary Coolant System Boundary)

a. Criterion

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 leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

b. Discussion

On the basis of the following information, it is concluded that Criterion 32 is satisfied.

The PCRV is composed of a multiplicity of structural elements consisting of reinforcing bars, unbonded tendons, and circumferential prestressing. Selected prestressing elements will be continuously monitored for load changes. The failure of individual prestressing elements does not affect the structural integrity of the vessel.

The gasketed primary closures will be continuously monitored for leakage, as discussed in the response to Criterion 30. In addition, added assurance regarding pressure boundary integrity will be obtained through an inspection program which meets the intent of the ASME Boiler and Pressure Vessel Code, Section XI (see Section 5.2).

A material surveillance program of the materials used on the coolant pressure boundary is planned. In accordance with ASTM-E185-73, it provides for Charpy V surveillance specimens of the liner material to be irradiated just outside the liner at positions where the neutron flux is maximum. The specimens will be periodically removed and tested to determine the shift in the NDT as well as changes in the impact energy absorption characteristics. The program does not provide for tensile specimens since the changes due to irradiation are expected to be negligible (see Section 5.2).

The cavity liner of the PCRV is not accessible from either side for inspection, since the outer surface is supported by concrete and the inner surface is covered with thermal barrier. It must be noted that the liner is not the pressure boundary; it is the fission product boundary. Thus, its failure, resulting in a liner leak, will neither lead to rapid depressurization of the primary coolant system nor affect the structural capability of the vessel. Provisions are made for in-service inspection of projecting parts of the penetration liner.

3.1.4.4 Criterion 33: Reactor Coolant Makeup

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 33 is satisfied.

The reactor coolant makeup system serves no safety function for the HTGR. Operational leakage from the PCRV is made up through the helium storage system. The reactor coolant makeup system is not relied upon nor is it required to maintain the fuel within design limits in the event of leakage or accidental loss of reactor coolant.

3.1.4.5 Criterion 34: Residual Heat Removal

a. Criterion

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 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 electrical power systems operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

b. Discussion

On the basis of the HTGR's inherent capabilities coupled with the core cooling systems available for residual heat removal as outlined below, it is concluded that Criterion 34 is satisfied.

The plant has two systems for removing the nuclear heat from the core and transferring it to a secondary coolant system. The first of these systems consists of the main loops which perform the dual functions of generating steam from the nuclear heat source during normal power operation and removing the residual decay heat during shutdown conditions. The second system consists of the CACS, which is an engineered safeguard system designed to Seismic Category I and capable of cooling the core under all accident conditions.

The CACS is designed to remove the stored heat and decay heat from the reactor core following a reactor trip accompanied by the loss of main loop cooling. In the event of such a condition, the CACS automatically starts within 5 min and provides sufficient heat removal such that no safety limits are exceeded. Sufficient redundancy is provided to maintain core cooling following a single failure (active or passive). The system is powered by either off-site power or the standby diesels such that a loss of either does not impair the cooling ability of the system.

3.1.4.6 Criterion 35: Emergency Core Cooling (Core Auxiliary Cooling)

a. Criterion

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 on-site electric power system operation (assuming off-site power is not available) and for off-site power is not available) the operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

b. **Discussion**

On the basis of the following information, it is concluded that Criterion 35 is satisfied.

In the HTGR, the points related to fuel clad damage and clad metal-water reactions do not apply. The analogous points relate to internal fuel and vessel damage which could interfere with cooling.

In the event that for any reason adequate core cooling from the main coolant loops is not available, abundant cooling is provided by the three auxiliary cooling loops. Sufficient cooling is provided to effect a rapid core cool-down and avoid damage to either fuel or reactor vessel internal components and structures even if one of the independent auxiliary cooling loops fails to operate.

Even in the event that coolant flow is completely interrupted and the operation of the auxiliary loops is delayed, the core thermal capacity is sufficient to prevent significant fuel overtemperature and avoid any internal vessel damage that would interfere with continued effective core cooling. Likewise, in the event of an accident in which the PCRV is depressurized to the containment, sufficient cooling is provided by the core auxiliary cooling loops to avoid damage that would prevent continuation of effective core cooling (see Chapter 15).

The loops are completely independent and redundant, having separate circulators, heat exchangers, and water supplies, and are operable with either off-site or on-site electric power. Single failures, including those in the on-site power system, will not prevent effective core cooling. Reliable leak detection and automatic isolation to limit leakage is provided for each heat exchanger (see Section 6.3).

3.1.4.7 Criterion 36: Inspection of Emergency Core Cooling System (Core Auxiliary Cooling System)

a. Criterion

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 36 is satisfied.

The portion of the auxiliary loops outside the PCRV can be readily inspected. The equipment inside the PCRV can be removed for inspection during shutdown, although it is not planned to do so. Successful operation of the loops is most readily determined by operational testing, as discussed in Criterion 37.

Physical inspection of components inside the PCRV is not practical as a routine procedure. Operational testing is a better assurance of operability.

Portions of the system which form the pressure-retaining boundary for the primary coolant system shall meet the intent of the ASME Boiler and Pressure Vessel Code, Section XI, as discussed in Section 5.2.

3.1.4.8 Criterion 37: Testing of Emergency Core Cooling System (Core Auxiliary Cooling System)

a. Criterion

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the 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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 37 is satisfied.

In the auxiliary loops, water circulation through the heat exchangers is continuously maintained by a small auxiliary water pump. Performance and leak-tight integrity of the heat rejection system are therefore demonstrated. The large main water pump will be periodically started, and the auxiliary circulator with its related equipment will be run at scheduled intervals (with its helium valve closed) to check operability. Testing of the helium valve will be accomplished by operation of the auxiliary loops during the refueling period.

In addition, the operation of the auxiliary loops as a system can be tested during shutdown by using them for decay heat removal while holding the primary loops in reserve. With this procedure, the capability and availability of the auxiliary loops are regularly assured.

Demonstration of sequential programming of essential electrical loads, including auxiliary loops to the diesel generators, may be done with the plant shut down.

3.1.4.9 Criterion 38: Containment Heat Removal

Not applicable to the HTGR.

3.1.4.10 Criterion 39: Inspection of Containment Heat Removal System

Not applicable to the HTGR.

3.1.4.11 Criterion 40: Testing of Containment Heat Removal System

Not applicable to the HTGR.

3.1.4.12 Criterion 41: Containment Atmosphere Cleanup

To be provided by applicant.

3.1.4.13 Criterion 42: Inspection of Containment Atmosphere Cleanup System

To be provided by applicant.

3.1.4.14 Criterion 43: Testing of Containment Atmosphere Cleanup Systems

To be provided by applicant.

3.1.4.15 <u>Criterion 44: Cooling Water</u>

To be provided by applicant.

3.1.4.16 Criterion 45: Inspection of Cooling Water System

To be provided by applicant.

3.1.4.17 Criterion 46: Testing of Cooling Water System

To be provided by applicant.

3.1.5 Reactor Containment

3.1.5.1 Criterion 50: Containment Design Basis

To be provided by applicant.

3.1.5.2 Criterion 51: Fracture Prevention of Containment Pressure Boundary

To be provided by applicant.

3.1.5.3 Criterion 52: Capability for Containment Leakage Rate Testing

To be provided by applicant.

3.1.5.4 Criterion 53: Provisions for Containment Inspection and Testing

To be provided by applicant.

3.1.5.5 Criterion 54: Systems Penetrating Containment

To be provided by applicant.

3.1.5.6 Criterion 55: Reactor Coolant Pressure Boundary Penetrating Containment (Primary Coolant System Boundary)

a. Criterion

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 the 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 the containment.

Isolation valves outside the 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.

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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 55 is satisfied.

The primary coolant system boundary is totally enclosed by the containment structure. However, there are very small instrument lines which are used for primary coolant sampling that penetrate the containment.

The only lines containing primary coolant helium penetrating the containment are the instrument sampling lines leading to the analytical instrumentation. These lines are

1. Provided with two automatic isolation valves, one inside the containment and the other outside the containment; the latter is as close to the containment boundary as possible.

2. Located such that the probability of their failure induced by conditions arising from natural phenomena, dynamic effects of secondary coolant line breaks, etc., will be minimized.

3. Designed and fabricated to Seismic Category I, Safety Class 3, from the PCRV to the second isolation valve.

4. Sized small to limit any accidental release if they break.

5. Provided with low-flow switches and alarms in the control room to safeguard against a line break.

Analysis of the off-site consequences from the failure of these lines is presented in Chapter 15.

3.1.5.7 Criterion 56: Primary Containment Isolation

To be provided by applicant.

3.1.5.8 Criterion 57: Closed System Isolation Valves

To be provided by applicant.

3.1.6 Fuel and Radioactivity Control

3.1.6.1 Criterion 60: Control of Releases of Radioactive Materials to the Environment

To be provided by applicant.

3.1.6.2 Criterion 61: Fuel Storage and Handling and Radioactivity Control

a. Criterion

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 a reliability and testability that reflect 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.

b. Discussion

On the basis of the following information, it is concluded that Criterion 61 is satisfied.

(1) Fuel Handling

All handling of fuel is done with the pressure of the PCRV, storage wells, transfer cask, and fuel handling machine maintained slightly subatmospheric. This assures that any leakage would be into rather than out of the equipment. Cooling of the fuel handling machine or the transfer cask is not required in the refueling operation to keep equipment temperatures at levels below that at which significant fission product release occurs. In addition, if it becomes necessary, the fuel handling machine and transfer cask can be connected to the gas waste system.

(2) Fuel Storage Facility

The fuel storage facility is a Seismic Category I structure, and it is enclosed in a Seismic Category I Building.

To assure a continuous supply of cooling water to the fuel storage facility, two completely independent external circuits furnish water to alternate cooling tubes attached to the outside of the storage wells. Adequate cooling can be provided by either system to prevent the fuel from reaching temperatures at which self-sustained burning of the graphite would occur and temperatures far below those which would result in significant fission product release from the fuel elements. This arrangement provides a highly reliable cooling system. There cannot be an uncontrolled release of activity to the atmosphere since the wells are vented to the gas waste system via the fuel handling purge and evacuation system.

The seals and closure mechanisms on the fuel storage wells can be periodically tested. Any corrections to the seals, closure mechanisms, or sealing surfaces can be performed at the most opportune time after refueling.

(3) Radioactive Waste Systems

Major components of the radioactive liquid and gas waste systems are located in compartments below the fuel monolith. The concrete walls of the structure provide shielding for areas immediately adjacent to the monolith. Local shielding within each compartment is provided as necessary to permit entry for periodic maintenance, routine operations, and required tests and inspections. The radwaste compartments are served by an independent ventilation system and are equipped with an emergency exhaust system for abnormal leakages which might occur within the compartment. Radiation monitors in each area will automatically isolate the vent duct to and from any area if high airborne activity is detected. The compartment can then be ventilated through roughing or by the high-efficiency particulate adsorber (HEPA) and charcoal filters to the plant vent.

For additional details concerning radioactive waste management and radiation protection, see Chapters 11 and 12.

3.1.6.3 Criterion 62: Prevention of Criticality in Fuel Storage and Handling

a. Criterion

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes preferably having geometrically safe configurations.

b. **Discussion**

On the basis of the following information, it is concluded that Criterion 62 is satisfied.

The fuel storage facilities have been designed so that the fuel contained therein will be subcritical even if all the following improbable events occurred simultaneously:

1. All locations in the fuel storage facility were occupied by the most reactive fuel elements.

2. None of the fuel elements contained their design burnable poison load, and they all contained the most reactive fuel.

3. The most reactive conditions of water flooding and reflection existed.

4. The storage containers were assumed to form an infinite array, and each container was assumed to be infinite in height.

Comparison with measurements from actual critical assemblies shows that the calculation techniques used in the design of these storage facilities are accurate to within $\pm 0.02\ \Delta k/k$. In addition, the design of these storage facilities is conservatively based on a calculated $k_{eff} \leq 0.90$ under the worst (i.e., most reactive) conditions.

"Safe geometries" are used in shipping, storing, and handling new fuel prior to insertion into normal storage facilities. The design of these facilities is also based upon the conservative conditions listed above.

3.1.6.4 Criterion 63: Monitoring Fuel and Waste Storage

To be provided by applicant.

3.1.6.5 Criterion 64: Monitoring Radioactivity Releases

To be provided by applicant.

APPENDIX D

UNITED STATES NUCLEAR REGULATORY COMMISSION

"GUIDE TO TYPES OF NRC FORMAL DOCUMENTS AND THEIR USES"

NUREG/BR-0070

MAY 1984

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United States
Nuclear Regulatory Commission

Guide to Types of NRC Formal Documents and Their Uses

May 1984

Division of Technical Information
and Document Control
Office of Administration

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GUIDE TO TYPES OF NRC FORMAL DOCUMENTS AND THEIR USES

Introduction

This guide is designed to serve two purposes: (1) to provide information for NRC employees to use in determining what documents to prepare to accomplish their licensing, regulatory, administrative, or management goals, and (2) to provide information for the public to use in determining which types of NRC documents to request to obtain the information they seek. This guidance is for all employees, but it will be particularly useful to new or reassigned employees, and to those who must review and approve the work of others in determining whether the appropriate type of document is being using to record and disseminate information.

Requirements for retention of the types of documents described here have been established and are set forth in NUREG-0910, "NRC Comprehensive Records Disposition Schedule." Most of the documents referenced in this guide for further information are available from the Division of Technical Information and Document Control (TIDC) or are available for inspection and copying for a fee in the NRC Public Document Room. If not, the availability is indicated.

For each type of formal document, the following information is provided if it is applicable:

Purpose - a description of the type of information provided in the document, and the intended use, with historical precedence cited where pertinent.

Criteria for Use - the principles used for determining when to use the particular type of document described.

Sources - the NRC or contractor organization that normally originates the type of document described.

Concurrences Required - the NRC organizational entities that must be considered in the approval chain for the document. Information is included to provide for Office of Management and Budget (OMB) clearance prior to dissemination, if required.

Distribution Method - the normal means for disseminating the information, along with exceptions.

Availability - the minimum requirement for availability of the document to staff and the public.

References - sources of other pertinent information on the type of document described.

ACRONYMS AND ABBREVIATIONS USED IN THIS GUIDE

ACRS	Advisory Committee on Reactor Safeguards
AEOD	Office for Analysis and Evaluation of Operational Data (NRC)
CFR	Code of Federal Regulations (e.g., 10 CFR 2.202, where 10 refers to Title 10, 2 refers to Part 2, and 202 refers to Section 202)
CRGR	Committee for Review of Generic Requirements (NRC)
DES	Draft Environmental Statement (a licensing document)
DOE	Department of Energy, U.S.
EDO	Executive Director for Operations (NRC), or Office of
ELD	Office of Executive Legal Director (NRC)
EPRI	Electric Power Research Institute
ES	Final Environmental Statement (a licensing document)
FOIA	Freedom on Information Act
GPO	Government Printing Office
GRA	Government Research Abstracts published by the National Technical Information Service
GRI	Government Research Index published by the National Technical Information Service
IE	Office of Inspection and Enforcement (NRC)
INPO	Institute for Nuclear Power Operations
NMSS	Office of Nuclear Material Safety and Safeguards (NRC)
NRC	Nuclear Regulatory Commission, U.S.
NRCI	Nuclear Regulatory Commission Issuance
NRR	Office of Nuclear Reactor Regulation (NRC)
NSAC	Nuclear Safety Analysis Center (EPRI)
NTIS	National Technical Information Service
NUREG	A formal NRC staff publication
NUREG/BR	An NRC Brochure
NUREG/CP	Conference proceedings published by NRC
NUREG/CR	Formal contractor and grantee reports
OGC	Office of the General Counsel (NRC)
OMB	Office of Management and Budget, U.S.
RES	Office of Nuclear Reactor Research (NRC)
RIL	Research Information Letter
RM	Office of Resource Management (NRC)
SALP	Systematic Assessment of Licensee Performance
SECY	Office of the Secretary (NRC)
SER	Safety Evaluation Report (a licensing document)
SRP	Standard Review Plan (a licensing document)
TIDC	Division of Technical Information and Document Control in the Office of Administration (NRC)

COMMISSION PAPERS (SECY PAPERS)

Purpose: Commission papers, also referred to as SECY papers because the Secretary of the Commission numbers them, and controls their issuance, are used for three purposes: (1) to respond to questions raised by the Commission, the Chairman, and/or the Commissioners individually, (2) to propose rulemaking or respond to petitions for rulemaking, and (3) to bring a matter to the attention of the Commission for information or action.

Commission papers may be of the following types:

1. Commission Meeting Papers, which present major policy issues for discussion and decision by the Commission, usually at a scheduled meeting.

2. Affirmation Papers, which present minor policy issues and usually concern rules and regulations.

3. Notation Vote Papers, which concern matters that do not require a Commission decision at a meeting, but which do require Commissioner concurrence and/or comment.

4. Negative Consent Papers, which concern matters that Commissioners would like brought to their attention before action is taken but which do not require the formality of a Commission vote.

5. Information Papers, which provide the Commission with information on significant matters. Commission action is not normally requested or required.

Criteria for Use: Commission papers are prepared when it is necessary or desirable to inform the Chairman and the Commissioners or to request a decision by them.

Sources: Commission papers may be prepared by NRC staff reporting to the Executive Director for Operations (EDO). Those papers are signed by the EDO. Commission papers may be prepared also by Commission staff. Those papers are signed by the appropriate Commission official.

Concurrences Required: The normal Office concurrences are required before the paper is presented for signature. This includes concurrences of the Office Director and the Directors of other Offices and Regional Administrators affected. If appropriate, the Office of the Executive Legal Director (ELD) or the Office of the General Counsel (OGC) is consulted.

Distribution Method: Distribution is made by the Office of the Secretary and the EDO.

Availability Required: Publicly available SECY Papers can be found in the NRC Public Document Room.

References: Format and details of content and procedures are presented in the "EDO Procedures Manual," NUREG/BR-0016. See also NRC Chapter 0240, "Correspondence Management," and NUREG/BR-0053, "NRC Regulations Handbook."

ENFORCEMENT DOCUMENTS

Purpose: The enforcement documents are means of directing or requiring specific licensee or applicant actions and for applying the sanctions needed to enforce NRC policy (10 CFR 2, App. C). The basic enforcement documents are notices of violation and orders of various types. These are defined below:

A Notice of Violation is a written notice setting forth one or more violations of a legally binding requirement. The notice normally requires the licensee to provide a written statement describing (1) corrective steps which have been taken by the licensee and the results achieved, (2) corrective steps which will be taken to prevent recurrence, and (3) the date full compliance will be achieved. If the Notice of Violation includes a proposed imposition of a civil penalty, the licensee will be required also to (4) admit or deny the violation and (5) state the reason for the violation. A civil penalty is a monetary penalty that may be imposed for violation of (1) certain specified licensing provisions of the Atomic Energy Act or supplementary NRC rules or orders, (2) any requirement for which a license may be revoked, or (3) reporting requirements under Section 206 of the Energy Reorganization Act.

An Order is a written NRC directive to modify, suspend, or revoke a license; to cease and desist from a given practice or activity; or to take such other action as may be proper (see 10 CFR 2.202 and 2.204). The following types of orders are issued:

1. License Modification Orders are issued when some change in licensee equipment, procedures, or management controls is necessary.

2. Suspension Orders may be used (a) to remove a threat to the public health and safety, common defense and security, or the environment; (b) to stop facility construction; (c) when the licensee has not responded adequately to other enforcement action; (d) when the licensee interferes with the conduct of an inspection or investigation.

3. Revocation Orders may be used (a) when a licensee is unable or unwilling to comply with NRC requirements, (b) when a licensee refuses to correct a violation, (c) when a licensee does not respond to a notice of violation where a response was required, (d) when a licensee refuses to pay a fee required by 10 CFR 170, or (e) for any other reason for which revocation is authorized under Section 186 of the Atomic Energy Act.

4. Cease and Desist Orders are used to stop an unauthorized activity that has continued after notification by NRC that such activity is unauthorized.

Documents Relative to Enforcement Mechanisms, such as the following, are also used by NRC:

1. Minutes of Enforcement Conferences held by NRC with licensee management to discuss safety, safeguards or environmental problems, licensee's compliance with regulatory requirements, a licensee's proposed corrective

measures (including schedules for implementation) and enforcement options available to NRC.

2. Bulletins and Information Notices, which are written notifications to groups of licensees identifying specific problems and recommending specific actions (see IE Bulletins and IE Information Notices).

3. Notices of Deviation, which are written notices describing a licensee's or a vendor's failure to satisfy a commitment, when the commitment involved is not a legally binding requirement. The notice of deviation requests the licensee or vendor to provide a written explanation or statement describing corrective steps taken (or planned), the results achieved, and the date when corrective action will be completed.

4. Confirmatory Action Letters, which are letters confirming a licensee's agreement to take certain actions to remove significant concerns about health and safety, safeguards, or the environment.

Criteria for Use: The purpose statements provided above contain the criteria for use of these enforcement documents.

Source: Enforcement documents are prepared by NRC staff.

Concurrences: Explicit instructions regarding approvals and concurrence signatures are provided in the Inspection and Enforcement Manual and in 10 CFR 2.

Distribution Method: Distribution is made by the office of the signator.

Availability: The documents and all related correspondence are submitted to the NRC Public Document Room and the appropriate Local Public Document Room(s) through the Document Control System.

References: (1) Inspection and Enforcement Manual. (2) Title 10, Code of Federal Regulations, Part 2, Appendix C, and other cited Parts.

GENERIC LETTERS

Purpose: Generic letters are prepared primarily to inform applicants and licensees of regulatory requirements related to licensing matters and schedules for compliance. These letters are used also to clarify NRC policy, to request information, and to transmit information.

Criteria for Use: Generic letters are used when the information being requested or disseminated is pertinent to all applicants and licensees or selected groups of them.

Sources: Generic letters are prepared by staff of the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Material Safety and Safeguards (NMSS).

Concurrences Required: The normal Division and Office concurrences are required. Depending on the content of the letter, review by the Committee for Review of Generic Requirements (CRGR) may be necessary. In addition, TIDC review and OMB clearance may have to be obtained if the reporting and recordkeeping requirements are not covered by an OMB generic clearance.

Distribution Method: Distribution is made on standard distribution lists by TIDC.

Availability: The letters and all attachments are submitted to the NRC Public Document Room and the Local Public Document Rooms through the Document Control System.

References: NRR and NMSS procedures and directives.

IE BULLETINS

Purpose: An Office of Inspection and Enforcement (IE) Bulletin is used to transmit information to and to request action and/or a written response from licensees and permit holders regarding matters of health and safety, safeguards, or environmental significance. Bulletins may be used also to obtain specific actions on a one-time basis; i.e., special inspections, surveys, or checks to determine whether certain events and/or conditions may have generic applicability. IE Bulletins are not intended to substitute for new or revised license conditions or requirements.

Criteria for Use: A bulletin may be issued when timely action is necessary by licensees or permit holders, or timely information is needed by NRC for assessment of a particular situation.

Sources: Recommendations for IE Bulletins originate primarily within IE staff, but they may come from staff of NRR, NMSS or Regional Offices.

Concurrences Required: All affected NRC organizations are contacted for approval or information. A preapproved OMB clearance number must be referenced. Specific clearance numbers have been granted for data collection associated with each part of 10 CFR, and a separate clearance number has been granted for emergency actions. Review and approval of the CRGR is required.

Distribution Method: Distribution to licensees or permit holders and other groups, including NRC staff, is made on established mailing lists by TIDC.

Availability Required: The documents and all related correspondence are submitted to the NRC Public Document Room and the Local Public Document Rooms through the Document Control System.

References: IE Inspection and Enforcement Manual.

IE INFORMATION NOTICES

Purpose: An Information Notice is issued to licensees or permit holders to give preliminary information on an event or condition (essentially unevaluated by NRC) that NRC believes may be relevant to health and safety, safeguards, or protection of the environment. Information notices may be issued also to inform licensees and permit holders promptly of changes in NRC procedures and the implementation of new rules and regulations. Replies are not required.

Criteria for Use: An Information Notice may be issued when, based on the information available at the time, the event or condition does not meet the criteria for issuance of a Bulletin, but licensees or permit holders should be notified promptly.

Sources: Recommendations for IE Information Notices originate primarily within IE staff, but they may come from staff of NRR, NMSS or Regional Offices.

Concurrences Required: Formal approval or concurrence is required by the Director, Division of Emergency Preparedness and Engineering Response (IE) for all Information Notices and by the Director, Division of Quality Assurance, Safeguards, and Inspection Programs (IE) for Information Notices sent to licensed nonreactor facilities. Review and approval of the CRCR is required.

Distribution Method: Distribution to licensees or permit holders and other groups, including NRC staff, is made on established mailing lists by TIDC.

Availability Required: The documents and related correspondence are submitted to the NRC Public Document Room and the Local Public Document Rooms through the Document Control System.

References: IE Inspection and Enforcement Manual.

INSPECTION REPORTS

Purpose: Inspection reports document inspection activities and findings. Inspections are conducted to evaluate compliance with specific requirements and commitments, Regulatory Guides, staff positions and interpretations, and consensus standards. Inspection findings are the basis for enforcement actions. The inspection report may also present observations of the strengths and weaknesses of an applicant or licensee. (See also Systematic Assessment of Licensee Performance (SALP) Reports).

Criteria for Use: An inspection report is a necessary component of an inspection. It must be part of the record of any inspection.

Sources: Inspection reports are prepared by the assigned inspector or inspection team.

Concurrence Required: Approvals vary with the type of inspection.

Distribution Method: Inspection reports are transmitted to the applicant or licensee by letter. Copies are distributed to Regional management by the responsible inspector or team leader. Inspection reports are made publicly available unless they contain security or proprietary information.

Availability Required: The reports and related correspondence and documents are submitted to the NRC Public Document Room and the appropriate Local Public Document Rooms(s) through the Document Control System.

References: IE Inspection and Enforcement Manual.

LICENSING DOCUMENTS FOR REACTORS

Purpose: Reactor licensing documents are (1) analysis and evaluation reports that develop the background for decisions on issuing licenses and (2) the actual licenses. The documents normally prepared for power plant and research reactor licensing are:

Safety Evaluation Report and Supplements (published in the NUREG series)

Draft Environmental Statement (published in the NUREG series)

Final Environmental Statement (published in the NUREG series)

Plant Technical Specifications (may be issued as part of License package or published in the NUREG series)

Construction Permit

Operating License and Environmental Protection Plan

License Amendments

Operator's License

Safety evaluation reports give details of NRC staff review of design, construction, and operating features of the facility. The draft and final environmental statements are required by the National Environmental Policy Act (NEPA) of 1969. These statements report NRC examination of the affected environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. The plant technical specifications are part of the licensing packages and are the operating requirements.

Criteria for Use: These documents are required for the issuance and maintenance of licenses.

Source: Office of Nuclear Reactor Regulation (NRR).

Concurrences Required: The approvals required vary with the type of document. See NRR procedures.

Distribution Method: Distribution is made on standard distribution lists by TIDC, except for licenses and their attachments, which are distributed by NRR Branches.

Availability Required: The documents and all related correspondence are submitted to the NRC Public Document Room and the appropriate Local Public Document Rooms through the Document Control System.

References: Title 10, Code of Federal Regulations. NRR procedures.

LICENSING DOCUMENTS FOR NUCLEAR MATERIALS AND FACILITIES USING NUCLEAR MATERIALS

Purpose: Licenses and certificates of compliance are issued for facilities and materials associated with the processing, transport and handling of nuclear materials and the disposal of nuclear waste. The issue of a license or certificate follows detailed safety and quality assurance analyses and the evaluation of environmental effects.

Licenses are issued for fuel-cycle functions and the possession of special nuclear materials (SNM).

Fuel-cycle licenses are for

1. Fresh fuel storage prior to issue of an operating license
2. Production of UF_6 for enrichment
3. Milling of yellow cake
4. Enriched uranium fuel processing and fabrication
5. Possession and use in the fuel cycle of critical amounts of special nuclear materials
6. Uranium fuel research and development and pilot plants
7. Source material
8. Away-from-reactor spent fuel storage

Materials licenses, other than for the fuel cycle, are for

1. Waste management facilities
2. Hot-cell facilities for examining irradiated fuel elements and for producing radioisotopes for medical uses
3. Possession and use of byproducts
4. Possession and use of radioisotopes

Certificates of Compliance are issued for radioactive materials packages for shipment to certify that the packaging and contents meet safety standards. A "Directory of Certificates of Compliance for Radioactive Materials Packages" (NUREG-0383) is issued annually that includes a summary report of NRC quality-assurance programs for radioactive materials packages.

Criteria for Use: These documents are issued following review and analysis of applications.

Source: Office of Nuclear Material Safety and Safeguards (NMSS).

Concurrences Required: The approvals required vary with the type of document.

Distribution Method: Licenses are issued to applicants by the NMSS Branch responsible for review and analysis of the application. Certificates of Compliance for Radioactive Materials Packages are issued to the applicants and to registered users of the packages.

Availability Required: Copies of the licenses and certificates of compliance, along with the related correspondence, are submitted to the NRC Public Document Room through the Document Control System.

NRC BROCHURES (NUREG/BR)

Purpose: Brochures (pamphlets, booklets) provide brief treatment of a specific subject. Brochures are identified as NUREG/BR-XXXX.

Criteria for Use: Brochures are single-purpose documents that may or may not be subject to updating. They may be for internal use or they may provide information or guidance to licensees and interested public organizations and individuals. They may present technical, regulatory, management, administrative, and procedural guidance, as well as statistical and other limited-subject information.

The following titles of existing NRC brochures illustrate the wide scope of these documents:

1. NRC Recruitment
2. Public Document Room User's Guide
3. Instructions for Completing Nuclear Material Transaction Reports
4. Handbook for Preparing for and Holding Public Meetings
5. Document Control System Newsletter
6. The Honor Law Graduate Program
7. Information Report on State Legislation
8. MATS User's Guide
9. Doing Business with the U.S. Nuclear Regulatory Commission
10. Power Reactor Events
11. Employee Handbook

Sources: Brochures may be originated in any NRC Office.

Concurrences Required: The normal office concurrences are required. Final clearance of the brochure for publication is controlled by TIDC in accordance with the guidance of OMB.

Distribution Method: Brochures may be distributed on standard distribution lists, and additional specialized distribution may be made.

Availability Required: The originating office retains the development record. TIDC retains the control information required by OMB. Brochures with external distribution are available for purchase from the NRC/GPO Sales Program. All brochures are maintained in NRC inventory and in the Public Document Room. They are available on request in Local Public Document Rooms.

References: Manual Chapter 3212, "Control of Production and Distribution of Periodicals and Pamphlets."

NRC MANAGEMENT DIRECTIVES - MANUAL CHAPTERS, BULLETINS AND ANNOUNCEMENTS

Purpose: NRC Manual Chapters are issued to communicate to NRC employees (1) basic NRC policies, requirements, procedures, and management information of overall applicability; and (2) detail on the manner of compliance with pertinent laws, Executive Orders, regulations, and directives of other agencies.

NRC Bulletins contain urgently needed interim directive material that will be published later in the appropriate chapter.

NRC Announcements contain information of a non-permanent nature needed by all employees or segments of employees.

Criteria for Use: NRC Management Directives are prepared to publish directives and information concerning functions performed by the agency in a controlled system of permanent records that can be maintained current.

Sources: NRC Management Directives may be initiated by any NRC organization with information that meets the above criteria and is within the purposes stated.

Concurrences Required: Draft Manual Chapters and Bulletins are distributed to the Directors of interested NRC Offices and to the Regional Administrators for comment. Drafts incorporating comments are submitted for concurrence. The final draft indicating concurrences and/or unresolved issues is presented to the approving authority. Approval authorities vary depending on the scope and content of the issuance. The Chairman approves Chapters covering organization and functions for offices reporting directly to the Chairman and to the Commission. Generally, NRC Chapters containing new policy or significant revisions of policy are approved by the EDO. Revisions of existing directives are approved by the Director, RM. Bulletins may be approved by Office or Division Directors. In some cases drafts must be reviewed/approved by another Government agency, such as the Office of Personnel Management. Further, issuances may be forwarded to the exclusive representative of the employees (union) for impact and implementation bargaining, if appropriate. NRC announcements are approved by Division Directors or comparable authority.

Distribution Method: Distribution is based on standard distribution lists.

Availability Required: The official records are maintained by the Office of Resource Management. Copies are available for inspection and use in the NRC Public Document Room. Inventory copies are maintained by TIDC.

References: NRC Manual Chapter 0201, "NRC Management Directive System."

NRC RULES AND REGULATIONS

Purpose: NRC rules and regulations are prepared to codify NRC action. They normally arise from (1) Congressional promulgation of a new statute specifying new regulatory requirements; (2) Commission or staff initiatives indicating a need for further regulation to resolve a safety, safeguards, or environmental problem; or (3) Commission receipt of a petition for rulemaking. Proposed rules and regulations are published in the Federal Register for a comment period to enable citizen participation in the decision-making process.

The following types of documents may be prepared:

1. Petitions for Rulemaking
2. Advance Notices of Proposed Rulemaking
3. Proposed Rules
4. Final Rules
5. Policy Statements
6. Memorandums of Understanding
7. Systems of Records

Criteria for Use: The rules and regulations under which NRC conducts its licensing and regulatory activities are issued in Title 10, Chapter 1, of the Code of Federal Regulations, and a rule or regulation published in the Code of Federal Regulations has the force of law. Therefore, this method of publishing is used when enforceability is a requirement.

Source: Action on a regulation is normally assigned to a member of the technical, administrative or legal staff who is familiar with the subject area.

Concurrences Required: The approvals required vary with the subject matter. For precise information see NUREG/BR-0053, "NRC Regulations Handbook," or contact the Rules and Procedures Branch, Division of Rules and Records, ADM.

Distribution Method: All materials for publishing in the Federal Register are transmitted through the Office of the Secretary. Internal NRC staff distribution is the responsibility of the originator and should include all involved and interested persons.

NRC Federal Register notices are also distributed automatically to affected licensees and parties to proceedings and to groups and organizations who have requested this information.

Availability Required: The documents and related correspondence are submitted to the Public Document Room through the Document Control System. The documents are available in microfiche in the Local Public Document Rooms.

References: Detailed instructions for preparation and approvals are contained in NUREG/BR-0053, "NRC Regulations Handbook," and NUREG/BR-0055, "Checklist for Preparation and Review of Federal Register Rulemaking Documents." Guidance is also available in the "EDO Procedures Manual," NUREG/BR-0016.

NRC-SPONSORED BOOKS

Purpose: Books are prepared to record and disseminate information for use as permanent reference material, teaching aids, or major critical reviews of technical or regulatory topics.

Criteria for Use: Proposals by staff, contractors, or grantees for the sponsorship of the preparation and distribution of books are evaluated to ensure that the book proposed fills a unique need and serves an industry-wide purpose. The information to be presented must be considered to be of permanent value (i.e., have a life of at least ten years before becoming obsolete) and must have a sufficiently large potential audience to justify the extra cost compared with that of a report. It must be possible to obtain validation of the information presented by peer review.

Sources: NRC staff, contractors and grantees.

Concurrences Required: For books prepared by NRC staff, the internal approvals required will vary according to the organization to which the author(s) belongs.

Distribution Methods: Books may be published and distributed by NRC, GPO, or a commercial publisher. They will be available for purchase from the NRC/GPO sales program, GPO, or the commercial publisher. The author(s) will receive a limited number of free copies and limited free distribution may be approved by a Division Director or comparable authority for NRC official business.

Availability Required: The book and related correspondence must be available in the NRC Public Document Room.

References: Manual Chapter 3210, "Book Writing and Publishing."

NUCLEAR REGULATORY COMMISSION ISSUANCES (NRCIs)

Purpose: The Nuclear Regulatory Commission Issuances set forth formal orders, opinions, and decisions on regulatory proceedings by the Commission, the Atomic Safety and Licensing Appeal Board, the Atomic Safety and Licensing Boards, and the Administrative Law Judge. Directors' Decisions and Denials of Petitions for Rulemaking are included. They are published monthly, indexed quarterly, and compiled semiannually.

Criteria for Use: These issuances are used only for the purposes described above.

Sources: Issuances are prepared by the groups referred to above.

Distribution Method: The NRCIs are distributed on standard internal and external distribution lists.

Availability Required: NRCIs are available to the public for inspection and use in the NRC Public Document Room. They are also available for purchase from the NRC/GPO Sales Program. All NRCIs are entered in the Document Control System. They are available in microfiche in the Local Public Document Rooms.

References: See any monthly issuance, quarterly index or semiannual cumulation. The NRCIs carry the identification NUREG-0750.

REGULATORY GUIDES

Purpose: Regulatory guides present methods acceptable to the NRC staff of implementing specific parts of NRC regulations, delineate techniques used by the staff in evaluating specific problems or postulated accidents, and provide guidance to applicants and licensees. Where possible, national standards are endorsed, with or without exceptions or additions.

Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions different from those set out in the guides are acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

These guides are useful to applicants, licensees, and permit holders in preparing the documentation required by NRC. They are a basis for the reviews specified in the Standard Review Plan (NUREG-0800) and the preparation of environmental impact statements and safety evaluation reports.

Regulatory guides are issued in ten divisions that cover the following major subjects:

1. Nuclear Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environment and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust and Financial Review
10. General

Criteria for Use: Regulatory guides are prepared when detailed guidance is needed for implementing NRC requirements.

Sources: Office of Nuclear Regulatory Research (RES).

Concurrences Required: Draft regulatory guides are issued for comment to interested members of the public after Division review and approval and, if appropriate, CRGR and ACRS review. After the incorporation of comments, active regulatory guides are reviewed by ELD and, if appropriate, CRGR and ACRS. Office review and approval are required for issuance.

Distribution Method: Draft guides are made available to interested parties at no cost on standard distribution lists through TIDC. Special internal NRC distribution is made by the originator. Active regulatory guides are provided at no cost to affected licensees and may be purchased on subscription or as individual copies from the Superintendent of Documents, GPO.

Availability Required: Regulatory Guides are entered into the Document Control System and the NRC Public Document Room. They are available on microfiche in the Local Public Document Rooms. Most program offices and divisions maintain collections of Regulatory Guides. Individual copies are maintained in inventory by TIDC.

Reference: Manual Chapter NRC-3201 "Publication of NRC Staff-Generated Regulatory and Technical Reports."

RESEARCH AND TECHNICAL ASSISTANCE REPORTS PREPARED BY CONTRACTORS, GRANTEES AND OTHER GOVERNMENT AGENCIES

Purpose: Research and technical assistance reports may be of the following types:

1. Technical letter reports (interim or final)
2. Formal technical reports (NUREG/CR)
3. Monthly and final letter status reports (business letter reports)
4. Research Information Letters (RILs)
5. Conference Proceedings (NUREG/CP)

Technical letter reports provide information on the technical aspects of work by contractors or other government agencies and their contractors. These letter reports may consist of transmittal letters with interim or final informal reports attached, or the technical information may be incorporated in the body of the letter. The form and frequency of reporting is specified in the Contract Statement of Work. These reports receive limited internal and external distribution and are publicly available in the NRC Public Document Room.

Interim technical letter reports provide information on the technical aspects of the work at various stages and form a basis for development of the formal reports required by the contract. Interim technical letter reports may include, but are not limited to, informal (interim) progress reports, "quick-look" reports, data reports, project descriptions, pretest predictions, model verifications, experiment safety analyses, experiment operating procedures, facility certification reports and test results.

Final technical letter reports are prepared to record the results of contract work that comprises review and evaluation of the work of others or work to be used by the staff in the licensing and regulation process. These technical letter reports are not followed by a formal report.

Formal technical reports, which may be periodic progress reports on long-term projects or final reports, are published in the NUREG/CR series. These reports are final products of research, original investigations, or significant compilations of information, or they may be progress reports. They meet the requirements of the Energy Reorganization Act of 1974 for production and dissemination of information and reports on the regulatory process.

Monthly and final letter status reports (business letter reports) provide administrative and contractual information, including personnel time expenditures, costs incurred and obligated funds. These reports are specified in the Statement of Work.

Research Information Letters (RILs) summarize research information for staff use. They are prepared in the NRC Office of Nuclear Regulatory Research (RES).

Conference Proceedings report the information presented at conferences sponsored or cosponsored by NRC. Sponsorship or cosponsorship by NRC requires that the subjects covered be of interest and value to NRC. Such proceedings are published in the NUREG/CP series.

Criteria for Use: Except for RILs, the criteria for use of these documents are provided in the Statement of Work of the contract or interagency agreement covering the work. RILs are prepared as deemed necessary by the staff of RES to summarize or correlate information.

Sources: Contractors and other governmental agencies and their contractors, primarily the Department of Energy (DOE) and the National Laboratories operated for DOE.

Concurrences Required: Prepublication approvals are specified in the Stateement of Work, including review by the Committee for Review of Generic Requirements, if appropriate. Requests for Publication (NRC Form 426A) of formal reports (NUREG/CRs) must be signed by the responsible Division Director or comparable authority unless such authority is delegated to a DOE contractor by the Division Director or comparable authority responsible for the work. The completed Form NRC 426A signifies that all pertinent technical and management reviews have been completed and that the document is approved for public dissemination.

Distribution Method: Technical letter reports are distributed by or for the NRC Project Manager in accordance with the Statement of Work.

Formal reports are distributed by NRC, even if printed by a DOE contractor, on standard distribution lists by TIDC, and additional specialized distribution may be made. All formal reports in the NUREG/CR and -CP series are available for purchase from the NRC/GPO Sales Program. They are announced in the NRC monthly "Title List of Documents Made Publicly Available" (NUREG-0540) and the NRC quarterly "Regulatory and Technical Reports" compilation (an abstract/index), NUREG-0304. They are available in hard copy for review and copying and are indexed by title in the NRC Public Document Room.

Formal reports (NUREG/CR and -CP series) are announced in the GPO monthly catalog and the National Technical Information Service (NTIS) Government Research Abstracts (GRA) and Government Research Index (GRI). Microfiche copy is also provided to U.S. and foreign Depository Libraries on a selective or exchange basis.

Availability Required: Standard distribution lists are used that include the Document Control System and the Public Document Room. Selected reports are available in the Local Public Document Rooms, as well as the abstract/index (NUREG-0304) of all NUREG-series reports.

References: Manual Chapter NRC-3202, "Publication of Technical Reports Prepared by NRC Contractors." Manual Chapter 1102, "Procedures for Placement of Work with the Department of Energy."

STAFF REPORTS ON REGULATORY, TECHNICAL AND ADMINISTRATIVE ISSUES (NUREG SERIES)

Purpose: Formal reports on regulatory, technical, and administrative issues of interest to staff, industry, other governmental entities, and the public are published in the NUREG series. They present:

1. results of licensing studies of specific plants or facilities preliminary to licensing actions.

2. results of analyses of general or specific problems of a regulatory or technical nature that are of interest to a major segment of the industry.

3. action and review plans, as well as guidance, for meeting NRC requirements.

4. task force reports on specific topics.

5. proceedings of conferences and workshops.

6. management and program analysis reports.

7. statistical analyses that are of interest to the staff, the industry and the public.

8. administrative reports that are of interest to the staff, the industry and the public.

Publication in the NUREG series assures announcement in (1) the Government Printing Office (GPO) monthly catalog, (2) the National Technical Information Service (NTIS) Government Research Abstracts (GRA) and Government Research Index (GRI), and (3) the provision of microfiche copy to U.S. and foreign Depository Libraries on a selective or exchange basis. NUREG-series reports are announced also in the NRC monthly "Title List of Documents Made Publicly Available" (NUREG-0540) and in the quarterly abstract/index journal entitled "Regulatory and Technical Reports" (NUREG-0304).

Criteria for Use: Textual and statistical information needed by the industry and the public in report format is prepared for publishing in the NUREG series. Such reports satisfy the Freedom of Information Act (FOIA) requirements for public availability and the Atomic Energy Act of 1954 (as amended) requirements for production and dissemination of information and reports on the regulatory process. NUREG-series reports do not, in themselves, constitute regulatory requirements. Reports in the NUREG series include:

1. Regulatory and Technical Reports (see also licensing documents)

Draft and Final Environmental Statements, DES and FES (plant or facility specific)

Safety Evaluation Reports, SER (plant or facility specific)

Standard Format and Content Guides

Standard Review Plan, SRP (including Branch technical positions)

Task Action Plans

Task Force Reports

2. Management Information Reports ("Rainbow Books")

Licensed Operating Reactors Report (Gray Book), NUREG-0020

Licensee, Contractor and Vendor Inspection Status Report (White Book), NUREG-0040

Topical Review Status, NUREG-0390

Systematic Evaluation Program Status Summary Report, NUREG-0485

Safeguards Summary Event List, NUREG-0525

Standards Development Status Summary (Green Book), NUREG-0566

Unresolved Safety Issues (Aqua Book), NUREG-0606

Operating Reactors Licensing Actions Summary, NUREG-0748

Summary Information Report, NUREG-0871

3. Reference Reports

NRC Annual Report

Handbook of Acronyms and Initialisms, NUREG-0544

NRC Organization Charts, NUREG-0325

Regulatory and Technical Reports, an abstract/index of NUREG, NUREG/CR, and NUREG/CP issuances, issued quarterly and compiled annually, NUREG-0304

Report to Congress on Abnormal Occurrences, NUREG-0090

Staff Practices and Procedures Digest, NUREG-0386

Standard Distribution for Unclassified U.S. Nuclear Regulatory Commission Reports, NUREG-0550

Title List of Documents Made Publicly Available, NUREG-0540

Sources: NUREG-series reports may be originated in any NRC office.

Concurrences Required: Requests for Publication (NRC Form 426) must be signed by the responsible Division Director or comparable authority. This signature signifies that all pertinent technical and management reviews, including, as appropriate, review by the Committee for Review of Generic Requirements, have been completed and that the document is approved for public dissemination.

Distribution Methods: These formal reports are distributed on standard distribution lists, and additional specialized distribution may be made. They are available for purchase from the NRC/GPO Sales Program. They are also available in hard copy for review and copying and are indexed by title in the NRC Public Document Room.

Availability Required: Standard distribution lists are used that include the Document Control System and the Public Document Room. Selected reports are available in the Local Public Document Rooms, as well as an abstract/index (NUREG-0304) of all NUREG-series reports.

References: Manual Chapter 3201, "Publication of NRC Staff-Generated Regulatory and Technical Reports."

References pertinent to specific types of NUREG-series reports:

1. NRR Office Letter No. 2, Rev. 2 - NUREG-0800 - Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.

2. NRR Office Letter No. 3, Standard Review Plans for Environmental and Antitrust Reviews.

3. Project Manager's Handbook, Division of Licensing, Office of Nuclear Reactor Regulation.

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP) REPORTS

Purpose: Systematic Assessment of Licensee Performance (SALP) Reports are prepared annually for each power reactor facility by an SALP Board. These assessments are prepared by inspectors and NRC managers knowledgeable of the inspection findings and of any regulatory issues of significance that developed during the designated assessment period. The SALP Board reports are discussed with the licensee management in a management meeting, and the licensees prepare written responses on identified topics.

Criteria for Use: The SALP program is used to identify utility organizational strengths and problem areas and to discuss the identified problems with the utility's management to reach an understanding of proposed corrective action. The SALP reports are prepared to record the results of the assessments.

Sources: SALP packages, which include the report, are prepared in the Regional Offices.

Concurrences Required: The SALP package is transmitted to the licensee by letter signed by the Regional Administrator upon concurrence of those who participated in the evaluation.

Distribution Method: The SALP packages are distributed by the Regional Office.

Availability Required: The SALP package consists of the transmittal letter to the licensee, the SALP Board report, and the licensee response to the report and/or the management meeting. This package is submitted to the NRC Public Document Room through the Document Control System. The packages are also sent to the appropriate Local Public Document Rooms.

References: Manual Chapter 0516, "Systematic Assessment of Licensee Performance."

TECHNICAL REPORTS OF THE OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

Purpose: Three types of technical reports (i.e., technical reviews, engineering evaluations, and case studies) are prepared by the Office for Analysis and Evaluation of Operational Data (AEOD) based on a review of operational experience as obtained in Licensee Event Reports (LERs). The AEOD reports are prepared to provide for understanding of (1) the event or situation, (2) the range and seriousness of the safety implications and lessons to be learned from the event, and (3) any actions that should be taken to minimize the possibility of recurrence and avoid even more serious events.

Criteria for Use: AEOD technical reports are used to formally document various levels of studies performed by AEOD on specific events or situations. A Technical Review is a preliminary evaluation of a potentially significant event, while an Engineering Evaluation or Case Study is a more substantial evaluation of a more substantive event or situation. While a Technical Review may be used to support a recommendation for issuance of an IE Information Notice, it generally does not contain recommendations for actions outside AEOD. Engineering Evaluations and Case Studies may include recommendations for action by other NRC offices.

Sources: AEOD staff.

Concurrences Required: Formal approval or concurrence for technical reviews and engineering evaluations is obtained within AEOD. Case Studies receive peer review from major program offices prior to formal approval from the Director, AEOD.

Distribution Method: Distribution is made by AEOD to the Commission and other NRC offices, the Institute for Nuclear Power Operations (INPO), the Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Institute (EPRI), and the licensee involved, as appropriate, based on the type and significance of the report. Selected reports are summarized in the AEOD publication titled, Power Reactor Events, NUREG/BR-0051, which is published monthly.

Availability Required: The documents and related correspondence are submitted to the NRC Public Document Room through the Document Control System.

References: AEOD Procedure No. 5, Documentation of AEOD Reports. Available from AEOD.

APPENDIX E

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF THE SECRETARY

"DESCRIPTIONS OF STANDING ORDER ITEMS"

SEPTEMBER 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

DESCRIPTIONS OF STANDING ORDER ITEMS

The following are descriptions of standing order items available from the U.S. Nuclear Regulatory Commission (NRC) Public Document Room (PDR). Users need to fill out a "Request for Standing Order" form in order to receive the documents desired on a regular basis. An estimated number of documents each item generated for the last year and the total number of pages is stated in parentheses at the end of each description. These estimates are provided for guidance only. The amount of documents and pages for each document type may vary from year to year. The current cost for copying documents is $0.09 per page. Postage and applicable sales tax are added to the bill. Any addition or cancellation of a standing order item, or change to a recipient or billing address needs to be submitted in writing to the PDR standing order administrator. Users will be billed for standing orders through the PDR copy service contractor.

NRC ADMINISTRATIVE LETTERS (SOALTR): Administrative letters are new generic communications that commenced in July 1993. These letters inform addressees of: administrative procedure changes being made to implement new regulations, issuances of topical report evaluations or NUREGs or changes in NRC internal procedures or organizations. The letters may request submittal of voluntary information that is administrative in nature; announce events of interest such as workshops or conferences; or be used for other purposes that are strictly administrative in nature. (1 Document/ 2 Pages)

BEVILL REPORTS (SOBEV): Bevill reports are annual reports of licensing schedules and activities for pending reactor operating license applications. The reports were formerly issued quarterly. They are addressed to Tom Bevill, Chairman of the Subcommittee on Energy and Water Development, House of Representatives, hence the name "Bevill Reports." (1 Document/4 Pages)

NRC BULLETINS (SONRCB): Formerly known as I&E (Inspection and Enforcement) bulletins, these documents are used to transmit information and request action and/or a written response from licensees and permit holders regarding matters of health and safety, safeguards or environmental significance. Bulletins also may be used to obtain specific actions on a one-time basis. (3 Documents/33 Pages)

CIVIL PENALTIES (SOCP): Civil penalties are monetary penalties that may be imposed for violation of either certain specified licensing provisions of the Atomic Energy Act or supplementary NRC rules or orders; or any requirements under Section 206 of the Energy Reorganization Act. Civil penalty packages contain: an NRC Office of Enforcement notification (EN) of significant enforcement action; and a letter to the licensee forwarding the notice of violation and proposed imposition of civil penalty. (118 Documents/506 Pages)

CRGR MEETING MINUTES (SOCRGR): All generic requirements proposed by the NRC staff related to one or more classes of reactors, including backfit requirements, must be reviewed by the Committee to Review Generic Requirements (CRGR). The CRGR is composed of senior NRC managers who review the proposed requirements and recommend to the NRC Executive Director for Operations to approve, disapprove, modify or provide conditions to the requirements. (14 Documents/65 Pages)

ENFORCEMENT MANUAL UPDATES (SOEM): The NRC Enforcement Manual provides detailed guidance and procedures to implement the general statement of policy and procedure for NRC enforcement actions described in 10CFR Part 2, Appendix C. Enforcement sanctions are used in accordance with NRC's enforcement policy for the purpose of ensuring the public health and safety. The Enforcement Manual is one volume with approximately 350 pages. Users will need to obtain a copy of the manual in order to insert the updated pages. (0 Documents/0 Pages)

EXPORT LICENSING ACTIVITIES (SOEXPT): Export licensing activity reports list all export actions completed by the NRC for each month. Actions include all new export applications received and cases pending. (12 Documents/60 Pages)

GENERIC COMMUNICATIONS (SOGCOM): Generic communications is a monthly listing of NRC guidance documents and their subjects being considered or under development by the NRC staff. NRC guidance documents include generic letters, NRC bulletins, information notices, temporary instructions and administrative letters. Prior to 1993 the listing was issued bimonthly. (11 Documents/64 Pages)

GENERIC ISSUE MANAGEMENT CONTROL SYSTEM REPORTS (SOGIMC): Generic Issue Management Control System (GIMCS) reports provide information necessary to manage the resolution of safety-related and nonsafety-related generic issues. The GIMCS is part of an integrated system of reports and procedures that is designed to manage generic safety issues through the stages of prioritization and resolution. The report is issued on a quarterly basis. The priority evaluation for each issue is contained in NUREG-933, "A Prioritization of Generic Safety Issues." (4 Documents/520 Pages)

GENERIC LETTERS (SOGENL): Generic letters are prepared primarily to inform applicants and licensees of regulatory requirements related to licensing matters and schedules for compliance. Generic letters are also used to clarify NRC policy and to request and transmit information. (12 Documents/144 Pages)

HEALTH PHYSICS POSITION PAPERS (SOHPPO): HPPOS are a compilation of NRC branch position papers, internal memoranda and letters to licensees, all presenting NRC policy on health physics issues. The papers give guidance on various topics, including release from radiological controlled areas, placement of dosimetry and control of access to high radiation areas. Some HPPOS attachments may have poor copy quality due to their age. NUREG/CR-5569, "Health Physics Positions Data Base," summarizes 247 papers. (74 Documents/408 Pages)

NRC INFORMATION NOTICES (SONIN): Information notices are issued to licensees or permit holders to give preliminary information on an event or condition that the NRC believes may be relevant to health and safety, safeguards, or protection of the environment. Notices may also be issued to inform licensees and permit holders promptly of changes in NRC procedures and the implementation of new rules and regulations. (91 Documents/594 Pages)

NRC INSPECTION MANUAL UPDATES (SONIMU): The NRC Inspection Manual provides policy, guidance and procedure of the NRC inspection program to ensure that planned and existing licensed operations can be and are being conducted without undue risk to the public. The manual is a multivolume looseleaf set. Updates to sections and pages are issued through change notices which are to be interfiled in the affected sections. Cost for copying the entire manual is estimated at $600. (13 Documents/600 Pages)

LICENSEE EVENT REPORTS (SOLER): LERs provide information on reportable events at nuclear power plants. Licensees are required to submit reports within 30 days after the discovery of an event as defined in 10 CFR Part 50.73. (2,269 Documents/18,152 Pages)

LOW-LEVEL WASTE TOPICAL REPORT STATUS (SOLLWT): Low-level waste topical report tracking system updates are composed of a summary listing of all active and non-active topical reports and their disposition; a disposition/status summary showing submittal and completion dates; and past and current reviewers. (12 Documents/204 Pages)

NRC MANAGEMENT DIRECTIVES (SONMGT): NRC Management Directives consist of 21 volumes of information that communicate: 1) basic NRC policies, requirements, procedures and management information of overall applicability; and 2) information on the manner of compliance with pertinent laws, Executive Orders, regulations and the directives of other Federal agencies, to NRC employees. Updates are issued monthly. NRC Management Directives supersede the NRC Manual. (32 Documents/1,500 Pages)

NMSS TECHNICAL NEWSLETTER (SONMSS): Also known as NUREG/BR-0017, this quarterly nureg is issued by the NRC Office of Nuclear Material Safety and Safeguards. The newsletter provides current licensing, inspection and other regulatory information regarding fuel cycle and material safety, materials and facilities safeguards and waste management. (4 Documents/52 Pages)

NRR OFFICE LETTERS (SONOFL): NRR office letters provide policy; define responsibilities, authorities and procedures; and establish requirements for managing and performing functions in the Office of Nuclear Reactor Regulation. (5 Documents/69 Pages)

NRR TECHNICAL NEWSLETTER (SONRR): The NRC Office of Nuclear Reactor Regulation's newsletter (NUREG-BR-125) provides information on activities for nuclear power reactors and non-power research reactors, such as those operated by universities. Topics include: status of licensing, plant license renewal, inspection programs, quality assurance, operator licensing, decommissioning and antitrust considerations. The newsletter is issued approximately twice per year. (2 Documents/25 Pages)

NUREG-800 (STANDARD REVIEW PLAN) UPDATES (SOSRP): The SRP provides guidance for the Office of Nuclear Reactor Regulation staff when performing safety reviews of applications to construct or operate nuclear power plants. The complete SRP consists of 3 volumes and costs approximately $215. (1 Document/19 Pages)

OPERATING REACTOR EVENTS MEETING MINUTES (SOREMI): Informs senior managers from the offices of the Commission, Advisory Committee on Reactor Safeguards, Enforcement, Nuclear Reactor Regulation and regional offices of selected events that have occurred for the time period indicated. A list of attendees, significant elements of the events discussed and reactor scram statistics are provided. (19 Documents/365 Pages)

NRC ORDERS AND ISSUANCES (SONOI): Orders and issuances are legal documents issued by the Commission, Atomic Safety and Licensing Boards, Administrative Law Judges and NRC Directors and served by the NRC Docketing and Services Branch. These documents include: memoranda and orders, Board notifications and correspondence with parties. (132 Documents/2,640 Pages)

10 C.F.R. PART 21 REPORTS (SOPT21): Part 21 or deficiency reports provide NRC with information, from firms constructing, owning, operating or supplying components for NRC-licensed facilities or activities, on substantial safety hazards, failures or defects. (182 Documents/1,600 Pages)

PERFORMANCE INDICATORS REPORTS (SOPERF): Performance indicator reports provide data for commercial power reactors regarding automatic reactor scrams while critical, significant events, forced outage rates, collective radiation exposure, safety system actuations and failures and equipment forced outages. Formerly released on quarterly basis, the reports are now issued semiannually. (4 Documents/1,952 Pages)

PRELIMINARY NOTIFICATIONS OF OCCURRENCE (SOPNO): PNOs are early notices of possible safety significance or of public interest. The information is received without verification or evaluation and is basically all the information known by the NRC staff at that time. (301 Documents/553 Pages)

PRESS RELEASES (SOPRES): Issued by the NRC Office of Public Affairs, press releases disseminate information on NRC policy, programs and activities to the public and news media. (197 Documents/425 Pages)

REGULATORY GUIDES (FINAL AND DRAFT) (SORGGD): Regulatory guides describe methods acceptable to the NRC staff for implementing specific portions of NRC regulations. Some regulatory guides lay out steps taken by the staff in evaluating specific situations. Others provide guidance to applicants concerning information needed by staff in its review of applications for permits and licenses or refer to or endorse national standards. (38 Documents/650 Pages)

RFPS (SORFP): RFPs are request for proposals for work to be done for the NRC under the terms of a contract. The solicitations are distributed by the NRC Division of Contracts. (26 Documents/520 Pages)

PROPOSED AND FINAL RULES (SORULE): Proposed and final rules are newly proposed regulations or proposed amendments to existing regulations. Rulemakings may be initiated by the Commission, on the recommendation of another agency of the United States, or on the petition of any other interested person. The regulations are codified and annually incorporated into the Code of Federal Regulations. (50 Documents/2,500 Pages)

ROSTER OF UTILITIES (SOROST): The roster of utlities lists the addresses and telephone numbers of utilities who operate commercial nuclear power plants, including: the utility chief executive officer and vice president; management, technical and environmental and legal contacts, a local official, the nearest local Public Document Room, resident inspectors, project managers and the switchboard telephone number for licensee communications. General information on the plant is also provided. (1 Document/128 Pages)

SECY PAPERS (SOSECY): These documents are called SECY papers because the Secretary of the Commission numbers them and controls their issuance. SECY papers respond to questions raised by the Commission, the Chairman and/or the Commissioners; propose rulemaking or respond to petitions for rulemaking; and bring matters to the attention of the Commission for information or action. (296 Documents/13,772 Pages)

SELECTED NRC PRODUCTS-HIGH-LEVEL WASTE PROGRAM (SOWAST): These documents identify selected NRC high-level waste program products and provide a summary of current and planned documents issued by the NRC staff. The summary is divided into 3 major components: regulations, licensing guidance process and NRC review plans. (1 Document/41 Pages)

SPEECHES (SOSPCH): Speeches are issued by the Office of Public Affairs and cover the NRC Chairman, Commissioners and occassionally the NRC staff. (41 Documents/375 Pages)

STAFF REQUIREMENTS MEMOS (SOSRM): SRMs provide the disposition on: Commission SECY papers (see description above) and Commission meetings. The memos may require the NRC staff to provide information to the Commissioners or may approve or disapprove of the stated issue. (198 Documents/667 Pages)

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE REPORTS (SOSALP): SALP reports identify utility organizational strengths and problem areas for a 1-2 year time period. The assessments are discussed with licensee management in a management meeting and the licensee prepares a written response on identified topics. SALP packages consist of the transmittal letter to the licensee and the SALP Board report. (93 Documents/2,680 Pages)

NRC TELEPHONE BOOK AND UPDATES (SOTELE): The NRC telephone book (NUREG-BR-0046) provides telephone and facsimile numbers for the entire NRC staff, including the Commission offices, regional staff and Advisory Committees and Boards. (3 Documents/480 Pages)

COMMISSION TRANSCRIPTS (SOCOMT): Transcripts are the corrected verbatim transcripts of meetings/briefings held before the NRC Commissioners. The PDR receives the transcripts 3-5 days after the meeting/briefing was held. (68 Documents/5,976 Pages)

WEEKLY INFORMATION REPORTS (SOWIR): Weekly information reports provide weekly highlights of significant matters conducted in all offices of the NRC. Meeting notices for the NRC Office of Nuclear Reactor Regulation, Nuclear Material Safety and Safeguards, Research, State Programs and the NRC Regions are included. (59 Documents/2,179 Pages)