NRC FORM 464 Part I (6-1998) NUDOCS

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA	RESPONSE NUMBER
99-373	3

RESPONSE TYPE: [ ] FINAL [✓] PARTIAL

REQUESTER: Diane Curran

DATE: JAN 27 2000

PART I. -- INFORMATION RELEASED

- [ ] No additional agency records subject to the request have been located.
- [ ] Requested records are available through another public distribution program. See Comments section.
- [ ] APPENDICES: Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.
- [✓] APPENDICES: D — Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.
- [ ] Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, NW, Washington, DC.
- [✓] APPENDICES: D — Agency records subject to the request are enclosed.
- [ ] Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.
- [✓] We are continuing to process your request.
- [ ] See Comments.

PART I.A -- FEES

AMOUNT * $

- [ ] You will be billed by NRC for the amount listed.
- [ ] You will receive a refund for the amount listed.
- [ ] None. Minimum fee threshold not met.
- [ ] Fees waived.

* See comments for details

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

- [ ] No agency records subject to the request have been located.
- [ ] Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.
- [ ] This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Carol Ann Reed Carol Ann Reed

FOIA-99-373

APPENDIX D
RECORDS BEING RELEASED IN THEIR ENTIRETY

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	Undated	Appendix "D" Draft Public Announcement, AEC Publishes General Design Criteria for Nuclear Power Plant Construction Permits (2 pages)
2.	Undated	Atomic Energy Commission, Licensing of Production and Utilization Facilities, General Design Criteria for Nuclear Power Plant Construction Permits (5 pages)
3.	Undated	Proposed Amendment to 10 CFR 50 General Design Criteria for Nuclear Power Plants, Report to the Director of Regulation (45 pages)
4.	Undated	Questionnaire to USAEC Div of Legal (1 page)
5.	12/12/66	General Design Criteria for Nuclear Power Plant Construction Permits (3 pages)

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APPENDIX "D"

DRAFT PUBLIC ANNOUNCEMENT

AEC PUBLISHES GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The AEC is publishing for public comment a revised set of proposed General Design Criteria which have been developed to assist in the preparation of applications for nuclear power plant construction permits.

In November 1965, the AEC issued an announcement requesting comments on General Design Criteria developed by its regulatory staff. These criteria were statements of design principles and objectives which have evolved over the years in licensing nuclear power plants by the AEC.

It was recognized at the time the criteria were first issued for comment that further efforts were needed to develop them more fully. The revision being published today reflects comments received following the 1965 announcement from 20 groups or individuals, suggestions made at meetings with the Atomic Industrial Forum, and review within the AEC.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment.

The General Design Criteria reflect the predominating experience to date with water reactors, but they are considered to be generally applicable to all power reactors. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant. The framework within which the criteria are presented provides sufficient flexibility for applicants to establish design requirements using alternate and/or additional criteria ~~so long as safety can be assured.~~ In particular, additional criteria will be needed for unusual sites and environmental conditions and for new or advanced types of reactors. In every case, however, the applicant will be required to identify its principal design criteria and provide assurance that they encompass all those facility design features required in the interest of public health and safety.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed

certain of the criteria; those have been

at the construction permit stage for the items listed in Category A than for

designated as category A. — certain items; these items have

Category B.

Development of these criteria is part of a longer-range Commission program to develop criteria, standards, and codes for nuclear reactor plants. This includes codes and standards that industry is developing with AEC participation. The ultimate goal is the evolution of industry codes and standards based on accumulated knowledge and experience as has occurred in various fields of engineering and construction.

The provisions of the proposed amendment relating to General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

The proposed criteria, which would become Appendix A to Part 50 of the AEC's regulations, will be published in the Federal Register on ____________. Interested persons may submit written comments or suggestions to the Secretary, U. S. Atomic Energy Commission, Washington, D.C., 20545, within 60 days. A copy of the proposed "General Design Criteria for Nuclear Power Plant Construction Permits" is attached.

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ATOMIC ENERGY COMMISSION

/10 CFR PART 50/

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria
for Nuclear Power Plant Construction Permits[1/]

The Atomic Energy Commission has under consideration an amendment to its regulation, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits." The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria to be included in applications for Commission construction permits. These General Design Criteria would not add any new requirements, but are intended to describe more clearly present Commission requirements to assist applicants in preparing applications.

The proposed amendment would complement other proposed amendments to Part 50 which were published for public comment in the FEDERAL REGISTER on August 16, 1966 (31 F.R. 10891).

1/ Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

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The proposed amendments to Part 50 reflect a recommendation made by a seven-member Regulatory Review Panel, appointed by the Commission to study: (1) the programs and procedures for the licensing and regulation of reactors and (2) the decision-making process in the Commission's regulatory program. The Panel's report recommended the development, particularly at the construction permit stage of a licensing proceeding, of design criteria for nuclear power plants. Work on the development of such criteria had been in process at the time of the Panel's study.

As a result, preliminary proposed criteria for the design of nuclear power plants were discussed with the Commission's Advisory Committee on Reactor Safeguards and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. In developing the proposed criteria set forth in the proposed amendments to Part 50, the Commission has taken into consideration comments and suggestions from the Advisory Committee on Reactor Safeguards, from members of industry, and from the public.

Section 50.34, paragraph (b), as published for comment in the FEDERAL REGISTER on August 16, 1966, would require that each application for a construction permit include a preliminary safety analysis report. The minimum information to be included in this preliminary safety analysis report is (1) a description and safety assessment of the site, (2) a summary description of the facility, (3) a preliminary design of the facility, (4) a preliminary safety analysis and evaluation of the facility, (5) an identification of subjects expected to be technical specifications, and (6) a preliminary plan for the organization, training, and operation. The following information is specified for inclusion as part of the preliminary design of the facility:

" (i) The principal design criteria for the facility;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;"

The "General Design Criteria for Nuclear Power Plant Construction Permits" proposed to be included as Appendix A to this part are intended to aid the applicant in development item (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

Section 50.34 as published in the FEDERAL REGISTER on August 16, 1966, would be further amended by adding to Part 50 a new Appendix A containing the General Design Criteria applicable to the construction of nuclear power plants and by a specific reference to this Appendix in §50.34, paragraph (b).

The Commission expects that the provisions of the proposed amendments relating to General Design Criteria for Nuclear Power Plant Construction

Permits will be useful as interim guidance until such time as the Commission takes further action on them.

Pursuant to the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act of 1946, as amended, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary, United States Atomic Energy Commission, Washington, D.C. 20545, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1. §50.34(b)(3)(i) of 10 CFR Part 50 is amended to read as follows:

§50.34 Contents of applications; technical information safety analysis report.[2/]

* * * * *

(b) Each application for a construction permit shall include a preliminary safety analysis report. The report shall cover all pertinent

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2/ Inasmuch as the Commission has under consideration other amendments to §50.34 (31 F.R. 10891), the amendment proposed herein would be a further revision of §50.34(b)(3)(i) previously published for comment in the FEDERAL REGISTER.

subjects specified in paragraph (a) of this section as fully as available information permits. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility, including:

(i) The principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.

2. A new Appendix A is added to read as follows:

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OFFICIAL USE ONLY

ATOMIC ENERGY COMMISSION
PROPOSED AMENDMENT TO 10 CFR 50
GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Report to the Director of Regulation
by the
Director, Division of Reactor Standards

THE PROBLEM

1. To consider publication for public comment of a proposed amendment to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plants".

BACKGROUND AND SUMMARY

2. At Regulatory Meeting 255 on June 28, 1967, the Commission approved publication of a Notice of Proposed Rule Making to amend 10 CFR Part 50 by adding an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" (AEC-R 2/57). That proposed amendment was published in the Federal Register on July 11, 1967, with a 60-day comment period.

3. Comments from twenty-one organizations and individuals, as listed in Appendix "B," were received in response to the previously proposed amendment. Because of the volume, the comments are not attached. Copies of all comments received have been placed in the Public Document Room.

4. The general reaction to the proposed criteria was favorable. The published proposed criteria were regarded as a considerable improvement over those originally released in Press Release H-252 dated November 22, 1965.* None of the commentators objected to the issuance of general design criteria. Most of the comments received were in the form of suggested improvements in language to facilitate understanding of the intent of the criteria, with few

*Secretariat Note: A copy of AEC Press Release H-252, November 22, 1965, is on file in the Office of the Secretary.

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suggestions to change or delete many requirements. The more significant comments and our resolution of them were:

a. Published Criterion 1 - Quality Standards

Public Comment - A showing of sufficiency should not be required where applicable codes and standards exist.

Resolution - This criterion has been modified to provide that a showing of sufficiency is not necessarily required, but an evaluation by the applicant of the applicable codes and standards to determine sufficiency is necessary (see New Criterion 1). Nuclear codes and standards have not been developed to the degree where it can be assumed that they are sufficient. The number of codes that have an "Issued for Trial Use and Comment" status and remain in this status for long periods of time and the additional requirements contained in the addenda to accepted codes indicate the need for an applicant to evaluate the codes and standards to assure itself of their sufficiency.

b. Published Criterion 11 - Control Room

Public Comments - (1) The criterion as published could be interpreted to require two control rooms and (2) Part 20 is not applicable to accidents.

Resolution - The criterion has been rewritten to make it clear that only one control room is required and reference to Part 20 has been deleted (see New Criterion 19). It should be noted that we have discussed control room requirements with industry representatives in order to understand better their views. One reactor manufacturer, supported by several utilities, made a presentation to the regulatory staff on this subject. The new wording of the criterion is in agreement with the industry position expressed in these discussions.

c. Published Criterion 28 - Reactivity Hot Shutdown Capability

Public Comment - The criterion can be interpreted to require two reactivity control systems capable of fast shutdown.

Resolution - The criterion has been rewritten to make it clear that only one system must be capable of fast shutdown (see New Criterion 26).

d. Published Criterion 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention

Public Comment - The requirements of this criterion are too specific and should be deleted.

Resolution - The criterion has been rewritten in a more general form. All references to specific margins above NDT temperature have been deleted (see New Criterion 31). It should be noted that interim revisions of the criterion on fracture prevention were discussed with the major reactor manufacturers. This resulted in a change in their position from recommending that the criterion be deleted to recommending that it be retained in the revised form.

e. Published Criterion 39 - Emergency Power for Engineered Safety Features

Public Comments - (1) The requirement that offsite power must satisfy the "single failure criterion" is impractical and (2) eliminate all reference to offsite power.

Resolution - The criterion has been rewritten to make it clear that the offsite power system need not meet the "single failure criterion." Reference to offsite power has not been deleted because we believe that offsite power is required to provide adequate assurance of safety (see New Criterion 17). It should be noted that we have discussed new Criterion 17 with the IEEE Subcommittee which is developing criteria for power requirements for nuclear power units. The members of the subcommittee indicated that the new criterion is acceptable and consistent with their requirements.

f. Published Criterion 44 - Emergency Core Cooling Systems Capability

Public Comment - Two independent emergency core cooling systems are not necessary.

Resolution - The criterion has been rewritten so that one system with sufficient redundancy is acceptable (see New Criterion 35). It should be noted that an interim version of the revised criterion for emergency core cooling was discussed with the ANS Systems Engineering Subcommittee. This subcommittee is in the process of developing criteria applicable to pressurized-water reactors. This interim version, which presented the one system concept, was acceptable to the ANS group with minor suggestions for changes in wording.

g. Published Criterion 49 - Containment Design Basis

Public Comment - Functioning of the emergency core cooling system is required for containment integrity; therefore, it is inconsistent to require that the containment design be based on the assumed failure of emergency core cooling systems.

Resolution - The criterion has been rewritten so that a design margin which reflects consideration of the possible effects of degraded emergency core cooling performance is required (see New Criterion 50).

5. The staff recently met with an ad hoc industry group, which included representatives of reactor manufacturers, utilities, architect engineers, and the Atomic Industrial Forum to discuss the revised General Design Criteria. Although the reaction of individual industry members was mixed, the Forum representative stated that he believed the criteria should be published for public comment after taking into consideration comments made at the meeting. These comments have been reflected in the proposed General Design Criteria, Appendix "A."

6. The amendment now proposed by the staff, which is attached as Appendix "A," would establish "minimum requirements" for water-cooled nuclear power units whereas the previously proposed amendment would have provided "guidance" for applicants for construction permits for all types of nuclear power plants.

7. The proposed amendment in Appendix "A" includes a section of definitions in accordance with comments received from industry that certain crucial terms should be defined. In addition, the criteria have been rearranged to increase their usefulness to designers and evaluators.

8. The Category A or B designation for each criterion which was included in the previously proposed amendment has been deleted. These categories had been included to provide guidance on the quantity and detail of information required for individual items at the construction permit stage. The amendment to § 50.34 of 10 CFR Part 50, published December 17, 1968, gives sufficient guidance in this area.

9. The proposed amendment does not include the term "engineered safety features." The requirements in the previously proposed amendment for "engineered safety features" have been incorporated in the proposed amendment by including them in the criteria for the individual systems.

10. There are criteria in the proposed amendment which do not have direct counterparts in the previously proposed amendment. Most of these are not new requirements but represent more specific guidance on requirements that were included in the previously proposed amendment in a more general form.

11. The regulatory staff has considered all comments received in revising the criteria and has worked closely with the Advisory Committee on Reactor Safeguards in the development of the criteria. The proposed criteria in Appendix "A" reflect ACRS review and comments.

STAFF JUDGMENTS

12. The Office of the General Counsel and the Divisions of Reactor Licensing and Compliance concur in the recommendations of this paper. The Office of Congressional Relations concurs in Appendix "C". The Division of Public Information has prepared Appendix "D."

RECOMMENDATION

13. The Director of Regulation recommends that the Atomic Energy Commission:

a. Approve publication of a proposed amendment to 10 CFR Part 50 which would add an Appendix A, "General Design Criteria for Nuclear Power Plants" establishing minimum requirements for water-cooled nuclear power units similar in design and location to units for which construction permits have been previously issued by the Commission and providing guidance to the applicants for construction permits for establishing the principal design criteria for other types of nuclear power units;

b. Note that the proposed amendment to 10 CFR Part 50 contained in Appendix "A" will be published in the Federal Register allowing 60 days for public comment;

c. Note that the Joint Committee on Atomic Energy will be informed by letter such as Appendix "C";

d. Note that a public announcement such as Appendix "D" be issued on filing the notice on proposed rule making with the Federal Register.

e. Note that, if after expiration of the comment period no adverse comments or significant questions have been received and no substantial changes in the text of the rule are indicated, the Director of Regulation will arrange for publication of the amendment in final form. If adverse comments or significant questions have been received or substantial changes in the text of the rule are indicated, the revised amendment will be submitted to the Commission for approval.

LIST OF ENCLOSURES

APPENDIX "A"

[10 CFR Part 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

The Atomic Energy Commission has under consideration an amendment to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plants."

Paragraph 50.34(a) of Part 50 requires that each application for a construction permit include the preliminary design of the facility. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility

(ii) The design basis and the relation of the design basis to the principal design criteria

(iii) Information relative to materials of construction, general arrangement, and the approximate dimensions, sufficient to provide reasonable assurance that the final design will conform with the design bases with adequate margin for safety.

The "General Design Criteria for Nuclear Power Plants" proposed to be added as Appendix A to Part 50 would establish the minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units for which construction permits have been issued by the Commission and would provide guidance in establishing the principal design criteria for other types of nuclear power units. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission would require that the criteria have been satisfied in the detailed design and construction of the facility, or that any changes in such criteria are justified.

A previously proposed Appendix A "General Design Criteria for Nuclear Plant Construction Permits" to 10 CFR Part 50 was published in the FEDERAL REGISTER (32 FR 10213) on July 11, 1967. The comments and suggestions received in response to that notice of proposed rule making have been considered in the revised proposed criteria which follow.

The revised proposed criteria would establish minimum requirements for water-cooled nuclear power units similar in design and location to units for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised proposed criteria have been reduced to 56 in number, include definitions of important terms, and have been rearranged to increase their usefulness to facility designers. Additional criteria describing specific requirements on matters covered in the previously proposed criteria in more general terms have been added to the revised proposed criteria. The Categories A and B used to characterize each criterion in the previously proposed criteria have been eliminated. The categories were intended to indicate the definitiveness of information required for each criterion. Since additional guidance on the amount and detail of information required to be submitted by applicants for facility licenses at the construction permit stage has since been included in § 50.34 of Part 50, such categorization is no longer necessary. The term "engineered safety features" has been eliminated and the requirements in the previously proposed criteria for "engineered safety features" incorporated in the revised proposed criteria for individual systems.

Pursuant to the Atomic Energy Act of 1954, as amended, and section 553 of Title 5 of the United States Code, notice is hereby given that adoption of the following amendment to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendment should send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1. Subdivision 50.34(a)(3)(i) is amended to read as follows:

§ 50.34 Contents of applications; technical information.

(a) Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility. Appendix A, General Design Criteria for Nuclear Power Plants, establishes mimimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units.

* * * * *

2. Footnote[2] to § 50.34 is amended to read as follows:

[2]General design criteria for chemical processing facilities are being developed.

* * * * *

3. A new Appendix A is added to read as follows:

(See Attachment)

APPENDIX A

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

INTRODUCTION

Pursuant to the provisions of §50.34, an application for a construction permit must include the principal design criteria for a proposed facility. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units 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 be used for guidance in establishing the principal design criteria for these other units.

The principal design criteria for a nuclear power unit establish necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that prevent or mitigate the consequences of accidents which could cause undue risk to the health and safety of the public. There will be some water-cooled nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be satisfied by the design in the interest of public safety. 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 units such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to prevent or mitigate the consequences of accidents which could cause 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 any size break in the piping, pressure vessels, pumps, and valves connected to the reactor pressure vessel and which are part of the reactor coolant pressure boundary, up to and including a break in these components equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

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.

Mechanical 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 any passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions. The failure of a passive component need not be considered in the design of mechanical systems if it can be demonstrated that the design is acceptable on some other defined basis, such as an appropriate combination of unusually high quality, high strength or low stress, inspectability, repairability, or short-term use.

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 the 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.

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, (2) sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (3) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (4) 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. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the capability of these structures, systems, and components.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. 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 sources outside the nuclear power unit.

CRITERION 5 - PROTECTION AGAINST INDUSTRIAL SABOTAGE

Structures, systems, and components important to safety shall be physically protected to minimize, consistent with other safety requirements, the probability and effects of industrial sabotage.

CRITERION 6 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

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 for all conditions 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 of specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

CRITERION 13 - REACTOR INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor and to maintain variables within prescribed operating ranges, including those variables and systems which can affect the fission process and the integrity of the reactor core.

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 all conditions of normal operation, including anticipated operational occurrences.

CRITERION 16 - 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 postulated accident conditions require.

CRITERION 17 - ELECTRICAL POWER SYSTEMS

An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system alone 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 electrical power sources, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Two physically independent transmission lines, each with the capability of supplying electrical power from the transmission network to the switchyard, and two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of electrical power from all other alternating current sources, including onsite electrical sources, 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 immediately 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 electrical power via any of the remaining circuits 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 electrical power sources.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems required for safety shall be designed to permit 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 active 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 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) having 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 emergency 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 can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functional performance 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, 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 in the event of systematic, nonrandom, concurrent failures of redundant elements.

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, considering the possibility of systematic, nonrandom, concurrent failures of control system components or channels, or of those common to the control and protection systems.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that 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 or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems, preferably of different design principles and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control 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 reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability in conjunction with 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 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. Their design shall reflect consideration of systematic, nonrandom, concurrent failures of redundant elements.

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 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.

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

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 and for offsite electrical power system operation 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, interconnections, and leak detection and isolation capabilities shall be provide to assure that for onsite and for offsite electrical power system operation 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-coolant accident 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. The performance of the system shall be evaluated conservatively.

Suitable redundancy in components and features, interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM COMPONENTS

Components of the emergency core cooling system shall be designed to permit periodic inspection and appropriate pressure testing of important areas and features, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic functional testing of (1) the operability and performance of the active components of the system, such as pumps and valves, and (2) the operability of the system as a whole and, under conditions as close to design as practical, the full operational sequence that brings the system into operation, including operation of the protection system, the transfer between normal and emergency power sources, and 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 low levels.

Suitable redundancy in components and features, interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM COMPONENTS

Components of the containment heat removal system shall be designed to permit periodic inspection and appropriate pressure testing of important areas and features, such as the torus, sumps, spray nozzles, and piping, to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic functional testing of (1) the operability and performance of the active components of the system, such as pumps and valves and (2) the operability of the system as a whole, and, under conditions as close to the design as practical, the full operational sequence that brings the system into operation, including operation of the protection system, the transfer between normal and emergency power sources, and 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 quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, interconnections, and leak detection and isolation capabilities to assure that for onsite and for offsite electrical power system operation its safety function can be accomplished assuming a single failure.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS COMPONENTS

Components of the containment atmosphere cleanup systems shall be designed to permit periodic inspection and appropriate pressure testing of important areas and features such as filter frames, ducts, and piping to assure their structural and leaktight integrity and the full design capability of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit periodic functional testing of (1) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves 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 the protection system, the transfer between normal and emergency power sources, and 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, interconnections, and leak detection and isolation capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM COMPONENTS

Components of the cooling water system shall be designed to permit periodic inspection and appropriate pressure testing of important areas and features, such as heat exchangers and piping, to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic functional testing of (1) the operability and performance of the active components of the system, such as pumps and valves, and (2) the operability of the system as a whole and, under conditions as close to design as practical, the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation 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 energy from metal-water and other chemical reactions that may result from degraded 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 necessarily 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) inspection of all important areas, such as penetrations, (2) an appropriate materials surveillance program, and (3) periodic testing of the leaktightness of penetrations which have resilient seals and expansion bellows at containment design pressure.

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 which is part of the reactor coolant pressure boundary and which penetrates primary reactor containment shall be provided with one automatic isolation valve inside and one automatic isolation valve, other than a simple check valve, outside of containment, unless it can be demonstrated that the design is acceptable on some other defined basis. The valve outside of containment shall be located as close to containment as practical and upon loss of actuating power the 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 - CONTAINMENT PRESSURE BOUNDARY ISOLATION VALVES

Each line which connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with one automatic isolation valve inside and one automatic isolation valve, other than a simple check valve, outside of containment, unless it can be demonstrated that the design is acceptable on some other defined basis. The valve outside of containment shall be located as close to containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 57 - CLOSED SYSTEMS ISOLATION VALVES

Each line which 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 isolation valve, other than a simple check valve. This valve shall be outside of containment and shall be located as close to containment as practical.

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 maintain suitable control over radioactive materials in gaseous and liquid effluents and in 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 their release to the environment.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling and radioactive waste systems 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 inspection and testing of important areas and features of the components of these systems, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

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

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

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

(Sec. 161, 68 Stat. 948; 42 U.S.C.2201)

Dated at ______________________________ this ______________

day of ____________________ 1970.

For the Atomic Energy Commission

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W. B. McCool
Secretary

LIST OF COMMENTS ON
PREVIOUS NOTICE OF PROPOSED RULE MAKING (32 FR 10213)
PUBLISHED IN THE FEDERAL REGISTER, JULY 11, 1967

1. H. C. Paxton, Los Alamos Scientific Laboratory, Member ASLB Panel, 7/25/67.
2. Eugene Greuling, Duke University Member, ASLB Panel, 7/26/67.
3. Stuart McLain, McLain Associates, 8/22/67.
4. Einar Swanson, Black and Veatch, 8/25/67.
5. G. J. Stathakis, General Electric Company, 9/5/67.
6. William B. Cottrell, Oak Ridge National Laboratory, 9/6/67.
7. J. M. Gallagher, Jr., IEEE, Nuclear Science Group, Reactor Instrumentation and Controls Standards Subcommittee, 9/6/67.
8. David N. Barry, III, Southern California Edison Company, 9/7/67.
9. J. C. Rengel, Westinghouse Electric Corporation, 9/8/67.
10. W. B. Behnke Jr., Commonwealth Edison Company, 9/8/67.
11. Sol Burstein, Wisconsin Electric Power Company, 9/8/67.
12. L. E. Minnick, Yankee Atomic Electric Company, 9/8/67.
13. D. M. Leppke, Pioneer Service and Engineering Company, 9/19/67.
14. W. R. Cooper, Tennessee Valley Authority, 9/20/67.
15. R. E. Wascher, Babcock & Wilcox, 9/20/67.
16. J. J. Flaherty, Atomics International, 9/25/67.
17. Edwin A. Wiggin, Atomics Industrial Forum, Inc., 10/2/67.
18. William S. Lee, Duke Power Company 11/2/67.
19. Charles O'D. Lee, Jr., Specifications Engineer, California, 12/20/67.
20. H. B. Stewart, Gulf General Atomic, Inc., 2/15/68.
21. J. M. West, Combustion Engineering, Inc., 2/21/68.

APPENDIX "C"

DRAFT LETTER TO THE JOINT COMMITTEE ON ATOMIC ENERGY

1. Enclosed for the information of the Joint Committee is a copy of a Notice of Proposed Rule Making to amend the Commission's regulation "Licensing of Production and Utilization Facilities," 10 CFR Part 50 which would add an Appendix A, General Design Criteria for Nuclear Power Plants. The proposed criteria are a revision of the criteria published for comment on July 11, 1967.

2. The proposed criteria established minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units for which construction permits have previously been issued by the Commission and provide guidance to applicants for construction permits for establishing the principal design criteria for other types of nuclear power units.

3. The notice has been transmitted to the Office of the Federal Register and will allow 60 days for public comment.

4. Enclosed also is a copy of a public announcement we plan to issue on this matter in the next few days.

APPENDIX "D"

DRAFT PUBLIC ANNOUNCEMENT

AEC PUBLISHES GENERAL DEISGN CRITERIA FOR NUCLEAR POWER PLANTS

The AEC is publishing for public comment a revised set of proposed general design criteria developed to assist applicants in establishing the principal design criteria for nuclear power units.

In July 1967 AEC published in the Federal Register for public comment "General Design Criteria for Nuclear Power Plant Construction Permits" developed by its regulatory staff. The revision published today reflects extensive comment received from 21 groups or individuals, review within the AEC, and developments that have occurred in the nuclear industry since publication of the criteria in 1967.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards in developing the revised criteria.

The revised criteria fix minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units previously approved by the Commission for construction. They provide guidance, also, for establishing the principal design criteria for other types of nuclear power units. Additional or different criteria are expected to be needed for unusual sites and environmental conditions, and for nuclear power units of advanced design. Development of these criteria is part of a longer range Commission program to develop criteria, codes, and standards applicable to nuclear power units. This includes criteria, codes, and standards that industry is developing with AEC participation. The ultimate goal is the evolution of industry criteria, codes, and standards based on accumulated knowledge and experience in various fields of engineering and industry.

The proposed criteria, which will become Appendix A to Part 50 of AEC's regulation, will be published in the Federal Register on _______________. Interested persons may submit comments or suggestions to the Office of the Secretary, U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Chief, Public Proceedings Branch, within 60 days. A copy of the proposed "General Design Criteria for Nuclear Power Plants" is attached.

4

QUESTIONNAIRE TO USAEC DIV OF LEGAL

(The Nuclear Power Plant Design Criteria Study Team)

(1) What is the formal situation of the General Design Criteria in the legal system?

And how are you intending to control and manage it?

(2) We understand that you have already received considerable comments on the General Design Criteria from various organizations.

What is your position of reaction for those comments?

Do you have under consideration an amendment to G.D.C. in near future? And also we would like to know the time schedule of prospective treatment.

(3) We understand that such relating supplemental criteria and standards, as for ASME Code constructed nuclear pressure vessels, reactor vessel material surveillance program (ASTM-E-185), reactor protection system (IEEE Standard) etc., have already proposed, however they are still of proposed, What is your present intention to control or manage them in the legal system?

(4) In the application of 10CFR50 and 10CFR20, especially for the cases of Construction Permit and Operating License for nuclear power plant, what kind of interim guides are you using within the Atomic Energy Commission?

(For example, we know typical one, Reactor Containment Leakage Testing and Surveillance Requirements, in Technical Safety Guide, titled SAFETY STANDARDS, CRITERIA, AND GUIDES FOR THE DESIGN, LOCATION, CONSTRUCTION, AND OPERATION OF REACTORS.)

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SS files

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December 12, 1966

GENERAL DESIGN CRITERIA
FOR
NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Introduction

We appreciate the opportunity to comment on the October 20, 1966, revised draft. We offer the following comments resulting from reviews by our staff and the safety staffs of a number of field offices.

General Comments

1. "Guidelines" rather than "criteria" are being presented. "Criteria" implies a rigorous set of standards to which a design can be compared. These design criteria do not represent standards for judging the acceptability of a design, but rather are guides to aid the designer in achieving a design with sufficient safety features to the Commission. This is supported in the introduction by: "...the applicant is free to establish the safety of his design by alternate criteria." We suggest the title "Design Guidelines for Nuclear Power Plant Construction Permits".

2. Many statements are complex, and this complexity tends to obstruct the meaning intended. Shorter sentences should be adopted wherever possible. There is also a general tendency to become too specific, and to state the proposed guidance in terms of specific examples rather than general philosophy. Examples are fine, but they should not replace statements of general position. For example, 9.0 could be restated as follows: "In determining the suitability of a facility for a proposed site, the reliance permitted to be placed upon the inherent and engineered safeguards must be conservatively related to their demonstrated capability and reliability, and the extent to which they can be inspected and tested during the life of the plant." The remainder of criterion 9 amplifies this statement adequately through examples.

3. The "criteria" seem to be slanted heavily, if not completely, toward high-power, thermal, light water cooled and moderated reactors. This should be recognized in the introduction. To further emphasize this characteristic, the second sentence of the second paragraph on page 1, which states that one or more of the criteria may be unnecessary or insufficient, should be given stronger emphasis.

4. Although there is considerable improvement over previous drafts, there is still unnecessary usage of qualifiers which tend to "muddy the waters" (such as "exceedingly low", "appropriate", "as necessary", and "as required").

5. The format of the draft is not consistent. There is no superheading for criteria 1, 2, and 3, while the remaining all have superheadings.

Specific Comments

2.1: We would assume that the first sentence applys under normal operating conditions. Perhaps this should be stated.

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2.3: The second sentence should be omitted. It is the subject of criterion 8.

3 discusses more than nuclear and radiation process controls. Change the title to "INSTRUMENTATION AND CONTROLS".

3.1: Reword the first sentence to read: "A control center from which the operational status of the plant can be regulated..." On line 3 omit ", even". On line 4, change "room or" to "center and".

3.2: Replace the remainder of the sentence starting with "and to prevent" by "to avoid damage to the fuel and other essential components of the plant".

3.3: The use of the word "rods" should be avoided wherever it appears. "Control element" has a more general connotation.

3.4: Delete "of reactivity controls,".

3.7: Is it the intent to require a capability to monitor for "conditions that might contribute to inadvertent criticality"? The requirements of 10 CFR 70 appear to be a more realistic requirement.

4 is more extensive than just "CORE PROTECTION", since protective instrumentation may initiate closure of containment or operation of engineered safeguards. Criterion 4 deals strictly with instrumentation. Change the title to "RELIABILITY AND TESTABILITY OF PROTECTIVE INSTRUMENTATION". "Protective systems" implies such things as engineered safeguards that are covered elsewhere.

4.1.2 and 4.1.3 clarify the intent of 4.1.1, and should either be incorporated into 4.1.1 or be eliminated.

4.3: Replace "state or a state established as tolerable on some other basis" with "or tolerable state".

5.2: The secondary reactivity control means should be required to hold the reactor subcritical as well as shut it down initially.

5.4: Here is one of the few opportunities to present a real criterion if one is wanted. Consider the merits of specifying values for maximum reactivity worths and reactivity addition rates. As a minimum, this section should be rewritten, as it is presently unclear.

6.1: Is it required that the coolant boundary absorb the energy released from a sudden reactivity insertion without protective system action, or can credit be permitted for protective systems which could reduce the effects?

7.2 is confusing. Perhaps it can be considered superfluous, since criterion 8.1 seems to cover the same requirement.

8.0: A number of our field offices are concerned over the lack of recognition of confinement systems. However, if it is true that the guidance offered applies primarily to high-power, light water reactors (see General Comment No. 3), OS is not bothered by this omittance.

8.1: Change the ending to "absence of operability of core quenching systems."

8.2: Delete the second sentence. Change the ending of the first sentence to "above NDT + 30°F."

8.3: Delete "as necessary".

9 should specifically require the operability of engineered safeguards equipment under abnormal and accident conditions.

9.1: This requirement is unclear. The phrase "where importance of the safety function requires" provides no guidance to the designer.

9.1.5: The intent of this requirement is not clear, particularly as it relates to the phrase "partial loss of installed capacity".

9.2.1.2 is too restrictive. It excludes any type of non-integrated leak test program designed to provide the same assurance as an integrated test.

9.2.3.1: It is not clear whether physical inspection must be possible after completion of construction only, or periodically, or as necessary.

10.1: The word "prevented" implies a state of the art which will probably never be achieved. "Minimized" is suggested in its place.

11.0 is not adequately restrictive. As we interpret this criterion, the licensee could, for example, release liquid effluents which exceed 10 CFR 20 limits to the soil within his site boundary, if he could show that it would be within the limits when it leaves the site due to percolation, ion exchange, etc.