January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400
(Shearon Harris Nuclear	)	
Power Plant)	)	
	)	

DECLARATION OF DR. GORDON THOMPSON IN SUPPORT OF ORANGE COUNTY'S SUMMARY AND SWORN SUBMISSION REGARDING CONTENTION TC-2 (INADEQUATE PREVENTION OF CRITICALITY)

I, Gordon Thompson, declare as follows:

1. I am the executive director of the Institute for Resource and Security Studies (IRSS), a nonprofit, tax-exempt corporation based in Massachusetts. Our office is located at 27 Ellsworth Avenue, Cambridge, MA 02139. IRSS was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace and international security, efficient use of natural resources, and protection of the environment.

2. I received an undergraduate education in science and mechanical engineering at the University of New South Wales, in Australia. Subsequently, I pursued graduate studies at Oxford University and received from that institution a Doctorate of Philosophy in mathematics in 1973, for analyses of plasmas undergoing thermonuclear fusion. During my graduate studies I was associated with the fusion research program of the UK Atomic Energy Authority.

3. During my professional career, I have performed technical and policy analyses on a range of issues related to international security, energy supply, environmental protection, and sustainable use of natural resources. Since 1977, a significant part of my work has consisted of technical analyses of safety and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in North America and Western Europe. Drawing upon these analyses, I have provided expert testimony in legal and regulatory proceedings, and have served on committees advising US government agencies. A copy of my resume is appended as Attachment A to the Declaration of Dr. Gordon Thompson (February 12, 1999), which is attached as Exhibit 2 to Orange County's Supplemental Petition to Intervene (April 5, 1999).

4. I have reviewed the December 23, 1998, license amendment application filed by Carolina Power and Light (CP&L) for an amendment to Facility Operating License No. NPF-63, which seeks permission to activate spent fuel storage pools C and D at the Shearon Harris nuclear power plant. I have also reviewed the NRC's Federal Register notice for the proposed license amendment, the Final Safety Analysis Report for the Shearon Harris Nuclear Power Plant, and the Final Environmental Statement related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2 (NUREG-0972, October 1983). In addition, I reviewed various correspondence and technical documents relating to the proposed license amendment and to risks of spent fuel storage, which are identified in Orange County's contentions.

5. I participated in the preparation of Orange County's contentions regarding the proposed license amendment. Following admission of Contention TC-2, Inadequate Criticality Prevention, I was principally responsible for evaluating whether CP&L's License Amendment Application conforms to the requirements of General Design Criterion 62 and applicable NRC Staff guidance.

6. In making my evaluation, I conducted an extensive review of documents related to criticality prevention at Harris and in general, including correspondence between CP&L and the NRC Staff, criticality studies performed by or for CP&L, NRC Staff and licensee documents regarding proposed spent fuel storage pool expansion applications, Licensee Event Reports of criticality-related occurrences, NRC Staff and industry guidance documents and related correspondence, the rulemaking history of GDC 62, and other publicly available information regarding spent fuel storage and criticality prevention. I also participated in preparing for depositions of CP&L and NRC Staff witnesses regarding contention TC-2, and in reviewing the deposition testimony of these witnesses. In addition, I was deposed by both CP&L and the NRC Staff.

7. I am responsible for all of the technical factual assertions contained in Orange County's Detailed Summary Of Facts, Data And Arguments On Which Orange County Intends To Rely At Oral Argument To Demonstrate The Existence Of A Genuine And Substantial Dispute Of Fact With The Licensee Regarding The Proposed Expansion Of Spent Fuel Storage Capacity At The Harris Nuclear Power Plant, With Respect To Criticality Prevention Issues (Contention TC-2), including Appendices A, B, and C, submitted to the Licensing Board on January 4, 2000 (hereinafter "Summary"). As I have attested in signing the Summary, the technical factual assertions therein are true and correct to the best of my knowledge, and all expressions of technical opinion therein are based on my best professional judgment.

I declare, under penalty of perjury, that the foregoing is true and correct.

Executed on January 4, 2000.

G. R. Thompson

Gordon Thompson

CONTENTION TC-2: EXHIBIT 2

Letter from Brian K. Grimes of the NRC Staff to All Power Reactor Licensees (April 14, 1978)

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorization to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

OT POSITION FOR REVIEW AND ACCEPTANCE OF
SPENT FUEL STORAGE AND HANDLING APPLICATIONS

I. BACKGROUND

Prior to 1975, low density spent fuel storage racks were designed with a large pitch, to prevent fuel pool criticality even if the pool contained the highest enrichment uranium in the light water reactor fuel assemblies. Due to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better utilize available space. In the case of operating plants the new rack system interfaces with the old fuel pool structure. A proposal for installation of high density storage racks may involve a plant in the licensing stage or an operating plant. The requirements of this position do not apply to spent fuel storage and handling facilities away from the nuclear reactor complex.

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, an environmental impact statement or environmental impact appraisal shall be prepared in which five specific factors in addition to the normal cost/benefit balance and environmental stresses should be applied, balanced and weighed.

The overall design objectives of a fuel storage facility at the reactor complex are governed by various Regulatory Guides, the Standard Review Plan, and industry standards which are listed in the reference section. Based on the reviews of such applications to date it is obvious that the staff had to request additional information that could be easily included in an adequately documented initial submittal. It is the intent of this document to provide guidance for the type and extent of information needed to perform the review, and to indicate the acceptance criteria where applicable.

II. REVIEW DISCIPLINES

The objective of the staff review is to prepare (1) Safety Evaluation Report, and (2) Environmental Impact Appraisal. The broad staff disciplines involved are nuclear, mechanical, material, structural, and environmental.

Nuclear and thermal-hydraulic aspects of the review include the potential for inadvertant criticality in the normal storage and handling of the spent fuel, and the consequences of credible accidents with respect to criticality and the ability of the heat removal system to maintain sufficient cooling.

Mechanical, material and structural aspects of the review concern the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand the effects of natural phenomena such as earthquakes, tornadoes, flood, effects of external and internal missiles, thermal loading, and also other service loading conditions.

The environmental aspects of the review concern the increased thermal and radiological releases from the facility under normal as well as accident conditions, the occupational radiation exposures, the generation of radioactive waste, the need for expansion, the commitment of material and nonmaterial resources, realistic accidents, alternatives to the proposed action and the cost-benefit balance.

The information related to nuclear and thermal-hydraulic type of analyses is discussed in Section III.

The mechanical, material, and structural related aspects of information are discussed in Section IV.

The information required to complete an environmental impact assessment, including the five factors specified by the Commission, is provided in Section V.

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

1. Neutron Multiplication Factor

To include all credible conditions, the licensee shall calculate the effective neutron multiplication factor, k_{eff}, in the fuel storage pool under the following sets of assumed conditions:

1.1 Normal Storage

a. The racks shall be designed to contain the most reactive fuel authorized to be stored in the facility without any control rods or any noncontained* burnable poison and the fuel shall be assumed to be at the most reactive point in its life.

b. The moderator shall be assumed to be pure water at the temperature within the fuel pool limits which yields the largest reactivity.

c. The array shall be assumed to be infinite in lateral extent or to be surrounded by an infinitely thick water reflector and thick concrete,** as appropriate to the design.

d. Mechanical uncertainties may be treated by assuming "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

e. Credit may be taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption, provided a means of inspection is established (refer to Section 1.5).

1.2 Postulated Accidents

The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies. The

*"Noncontained" burnable poison is that which is not an integral part of the fuel assembly.

**It should be noted that under certain conditions concrete may be a more effective reflector than water.

postulated accidents shall include: (1) dropping of a fuel element on top of the racks and any other achievable abnormal location of a fuel assembly in the pool; (2) a dropping or tipping of the fuel cask or other heavy objects into the fuel pool; (3) effect of tornado or earthquake on the deformation and relative position of the fuel racks; and (4) loss of all cooling systems or flow under the accident conditions, unless the cooling system is single failure proof.

1.3 Calculation Methods

The calculation method and cross-section values shall be verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. Sufficiently diverse configurations shall be calculated to render improbable the "cancellation of error" in the calculations. So far as practicable the ability to correctly account for heterogeneities (e.g., thin slabs of absorber between storage locations) shall be demonstrated.

A calculational bias, including the effect of wide spacing between assemblies shall be determined from the comparison between calculation and experiment. A calculation uncertainity shall be determined such that the true multiplication factor will be less than the calculated value with a 95 percent probability at a 95 percent confidence level. The total uncertainity factor on k_{eff} shall be obtained by a statistical combination of the calculational and mechanical uncertainties. The k_{eff} value for the racks shall be obtained by summing the calculated value, the calculational bias, and the total uncertainty.

1.4 Rack Modification

For modification to existing racks in operating reactors, the following information should be provided in order to expedite the review:

(a) The overall size of the fuel assembly which is to be stored in the racks and the fraction of the total cell area which represents the overall fuel assembly in the model of the nominal storage lattice cell;

(b) For H_2O + stainless steel flux trap lattices; the nominal thickness and type of stainless steel used in the storage racks and the thermal (.025 ev) macroscopic neutron absorption cross section that is used in the calculation method for this stainless steel;

(c) Also, for the H_2O + stainless steel flux trap lattices, the change of the calculated neutron multiplication factor of infinitely long fuel assemblies in infinitely large arrays in the storage rack (i.e., the k_∞ of the nominal fuel storage lattice cell and the changed k_∞) for:

(1) A change in fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly where it is assumed that this change is made by increasing the enrichment of the U^{235}; and,

(2) A change in the thickness of stainless steel in the storage racks assuming that a decrease in stainless steel thickness is taken up by an increase in water thickness and vice versa;

(d) For lattices which use boron or other strong neutron absorbers provide:

(1) The effective areal density of the boron-ten atoms (i.e., B^{10} atoms/cm^2 or the equivalent number of boron-ten atoms for other neutron absorbers) between fuel assemblies.

(2) Similar to Item C, above, provide the sensitivity of the storage lattice cell k_∞ to:

(a) The fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly,

(b) The storage lattice pitch; and,

(c) The areal density of the boron-ten atoms between fuel assemblies.

1.5 Acceptance Criteria for Criticality

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions

(1) For those facilities which employ a strong neutron absorbing material to reduce the neutron multiplication factor for the storage pool, the licensee shall provide the description of onsite tests which will be performed to confirm the presence and retention of the strong absorber in the racks. The results of an initial, onsite verification test shall show within 95 percent confidence limits that there is a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95. In addition, coupon or other type of surveillance testing shall be performed on a statistically acceptable sample size on a

periodic basis throughout the life of the racks to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95.

(2) Decay Heat Calculations for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 75/087).

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

Conservative methods should be used to calculate the maximum fuel temperature and the increase in temperature of the water in the pool. The maximum void fraction in the fuel assembly and between fuel assemblies should also be calculated.

Ordinarily, in order not to exceed the design heat load for the spent fuel cooling system it will be necessary to do a certain amount of cooling in the reactor vessel after reactor shutdown prior to moving fuel assemblies into the spent fuel pool. The bases for the analyses should include the established cooling times for both the usual refueling case and the full core off load case.

A potential for a large increase in the reactivity in an H_2O flux trap storage lattice exists if, somehow, the water is kept out or forced out of the space between the fuel assemblies, conceivably by trapped air or steam. For this reason, it is necessary to show that the design of the storage rack is such that this will not occur and that these spaces will always have water in them. Also, in some cases, direct gamma heating of the fuel storage cell walls and of the intercell water may be significant. It is necessary to consider direct gamma heating of the fuel storage cell walls and of the intercell water to show that boiling will not occur in the water channels between the fuel assemblies. Under postulated accident conditions where all non-Category I spent fuel pool cooling systems become inoperative, it is necessary to show that there is an alternate method for cooling the spent pool water. When this alternative method requires the installation of alternate components or significant physical alteration of the cooling system, the detailed steps shall be described, along with the time required for each. Also, the average amount of water in the fuel pool and the expected heat up rate of this water assuming loss of all cooling systems shall be specified.

(4) Potential Fuel and Rack Handling Accidents

The method for moving the racks to and from and into and out of the fuel pool, should be described. Also, for plants where the spent fuel pool modification requires different fuel handling procedures than that described in the Final Safety Analysis Report, the differences should be discussed. If potential fuel and rack handling accidents occur, the neutron multiplication factor in the fuel pool shall not exceed 0.95. These postulated accidents shall not be the cause of the loss of cooling for either the spent fuel or the reactor.

(5) Technical Specifications

To insure against criticality, the following technical specifications are needed on fuel storage in high density racks:

1. The neutron multiplication factor in the fuel pool shall be less than or equal to 0.95 at all times.

2. The fuel loading (i.e., grams of uranium-235, or equivalent, per axial centimeter of assembly) in fuel assemblies that are to be loaded into the high density racks should be limited. The number of grams of uranium-235, or equivalent, put in the plant's technical specifications shall preclude criticality in the fuel pool.

Excessive pool water temperatures may lead to excessive loss of water due to evaporation and/or cause fogging. Analyses of thermal load should consider loss of all pool cooling systems. To avoid exceeding the specified spent fuel pool temperatures, consideration shall be given to incorporating a technical specification limit on the pool water temperature that would resolve the concerns described above. For limiting values of pool water temperatures refer to ANSI-N210-1976 entitled, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," except that the requirements of the Section 9.1.3.III.1.d of the Standard Review Plan is applicable for the maximum heat load with normal cooling systems in operation.

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool and the racks. The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed under the environmental discipline. Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. Integrity of the racks and the fuel pool due to a postulated fuel handling accident is reviewed under the mechanical, material, and structural disciplines. Sketches and sufficient details of the fuel handling system should be provided to facilitate this review.

(2) Applicable Codes, Standards and Specifications

Construction materials should conform to Section III, Subsection NF of the ASME* Code. All Materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for stainless steel base metal may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code.

Other materials, design procedures, and fabrication techniques will be reviewed on a case by case basis.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61 respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guide 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.

**American Institute of Steel Construction, Latest Edition.

The peak response from each direction should be combined by square root of the sum of the squares. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

The effect of submergence of the rack system on the damping and the mass of the fuel racks has been under study by the NRC. Submergence in water may introduce damping from two sources, (a) viscous drag, and (b) radiation of energy away from the submerged body in those cases where the confining boundaries are far enough away to prevent reflection of waves at the boundaries. Viscous damping is generally negligible. Based upon the findings of this current study for a typical high density rack configuration, wave reflections occur at the boundaries so that no additional damping should be taken into account.

A report on the NRC study is to be published shortly under the title "Effective Mass and Damping of Submerged Structures (UCRL-52342)," by R. G. Dong. The recommendations provided in this report on the added mass effect provide an acceptable basis for the staff review. Increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described in the report: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to the applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and base slab. Temperature gradient across the rack structure due to differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of Section 3.8.4-II.3 of the Standard Review Plan.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach, two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and,

(b) A static analysis of the rack system by subjecting it to the maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	
Elastic Analysis	**Acceptance Limit**
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	1.5 times normal limits or the lesser of 2 Sy and Su
D + L + To + E	1.5 times normal limits or the leser of 2 Sy and Su
D + L + Ta + E	1.6 times normal limits or the lesser of 2 Sy or Su
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be ammended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

the construction phase should be provided. Methods for structural qualification of special poison materials utilized to absorb neutron radiation should be described. The material for the fuel rack is reviewed for compatibility inside the fuel pool environment. The quality of the fuel pool water in terms of the pH value and the available chlorides, fluorides, boron, heavy metals should be indicated so that the long-term integrity of the rack structure, fuel assembly, and the pool liner can be evaluated.

Acceptance criteria for special materials such as poison materials should be based upon the results of the qualification program supported by test data and/or analytical procedures.

If connections between the rack and the pool liner are made by welding, the welder as well as the welding procedure for the welding assembly shall be qualified in accordance with the applicable code.

If precipitation hardened stainless steel material is used for the construction of the spent fuel pool racks, hardness testing should be performed on each rack component of the subject material to verify that each part is heat treated properly. In addition, the surface film resulting from the heat treatment should be removed from each piece to assure adequate corrosion resistance.

(8) Testing and Inservice Surveillance

Methods for verification of long-term material stability and mechanical integrity of special poison material utilized for neutron absorption should include actual tests.

Inservice surveillance requirements for the fuel racks and the poison material, if applicable, are dependent on specific design features. These features will be reviewed on a case by case basis to determine the type and the extent of inservice surveillance necessary to assure long-term safety and integrity of the pool and the fuel rack system.

V. COST/BENEFIT ASSESSMENT

1. Following is a list of information needed for the environmental Cost/Benefit Assessment:

1.1 What are the specific needs that require increased storage capacity in the spent fuel pool (SFP)? Include in the response:

(a) status of contractual arrangements, if any, with fuel-storage or fuel-reprocessing facilities,

(b) proposed refueling schedule, including the expected number of fuel assemblies that will be transferred into the SFP at each refueling until the total existing capacity is reached,

(c) number of spent fuel assemblies presently stored in the SFP,

(d) control rod assemblies or other components stored in the SFP, and

(e) the additional time period that spent fuel assemblies would be stored onsite as a result of the proposed expansion, and

(f) the estimated date that the SFP will be filled with the proposed increase in storage capacity.

1.2 Discuss the total construction associated with the proposed modification, including engineering, capital costs (direct and indirect) and allowances for funds used during construction.

1.3 Discuss the alternative to increasing the storage capacity of the SFP. The alternatives considered should include:

(a) shipment to a fuel reprocessing facility (if available),

(b) shipment to an independent spent fuel storage facility,

(c) shipment to another reactor site,

(d) shutting down the reactor.

The discussion of options (a), (b) and (c) should include a cost comparison in terms of dollars per KgU stored or cost per assembly. The discussion of (d) should include the cost for providing replacement power either from within or outside the licensee's generating system.

1.4 Discuss whether the commitment of material resources (e.g., stainless steel, boral, B_4C, etc.) would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. Describe the material resources that would be consumed by the proposed modification.

1.5 Discuss the additional heat load and the anticipated maximum temperature of water in the SFP which would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems and whether there will be any significant increase in the amount of heat released to the environment.

V.2. RADIOLOGICAL EVALUATION

2. Following is a list of information needed for radiological evaluation:

2.1 The present annual quantity of solid radioactive wastes generated by the SFP purification system. Discuss the expected increase in solid wastes which will result from the expansion of the capacity of the SFP.

2.2. Data regarding krypton-85 measured from the fuel building ventilation system by year for the last two years. If data are not available from the fuel building ventilation system, provide this data for the ventilation release which includes this system.

2.3 The increases in the doses to personnel from radionuclide concentrations in the SFP due to the expansion of the capacity of the SFP, including the following:

(a) Provide a table showing the most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations.

(b) The models used to determine the external dose equivalent rate from these radionuclides. Consider the dose equivalent rate at some distance above the center and edge of the pool respectively. (Use relevant experience if necessary).

(c) A table of recent analysis performed to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

(d) The model and assumptions used to determine the increase, if any, in dose rate from the radionuclides identified in (c) above in the SFP area and at the site boundary.

(e) An estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter media.

(f) The buildup of crud (e.g., ^{58}Co, ^{60}Co) along the sides of the pool and the removal methods that will be used to reduce radiation levels at the pool edge to as low as reasonably achievable.

(g) The expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (e) and (f) above.

A discussion of the radiation protection program as it affects (a) through (g) should be provided.

2.4 Indicate the weight of the present spent fuel racks that will be removed from the SFP due to the modification and discuss what will be done with these racks.

V.3 ACCIDENT EVALUATION

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

(b) evaluation of the overhead handling system with respect to Regulatory Guide 1.104.

3.2 If the accident aspects of review do not establish acceptability with respect to either (a) or (b) above, then technical specifications may be required that prohibit cask movement in the spent fuel building.

3.3 If the accident review does not establish acceptability with respect to (b) above, then technical specifications may be required that:

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

(2) indicate the minimum age of fuel in pool sections during movement of heavy loads near the pool. In special cases evaluation of consequences-limiting engineered safety features such as isolation systems and filter systems may be required.

3.4 If the cask drop/tip analysis as in 3.1(a) above is promised for future submittal, the staff evaluation will include a conclusion on the feasibility of a specification of minimum age of fuel based on previous evaluations.

3.5 The maximum weight of loads which may be transported over spent fuel may not be substantially in excess of that of a single fuel assembly. A technical specification will be required to this effect.

3.6 Conclusions that determination of previous Safety Evaluation Reports and Final Environmental Statements have not changed significantly or impacts are not significant are made so that a negative declaration with an Environmental Impact Appraisal (rather than a Draft and Final Environmental Statement) can be issued. This will involve checking realistic as well as conservative accident analyses.

VI. REFERENCES

1. Regulatory Guides

   1.13 - Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

   1.29 - Seismic Design Classification

   1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

   1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

   1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

   1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section III, Division 1

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

   4. American Society of Civil Engineers, Suggested Specification for Structures of Aluminium Alloys 6061-T6 and 6067-T6

5. The Aluminium Association, Specification for Aluminium Structures

CONTENTION TC-2: EXHIBIT 3

Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis (December 1981)

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

December 1981
Division 1
Task CE 913-5

Contact: C. Schulten (301)443-5910

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. It also requires that these systems 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. This guide describes a method acceptable to the NRC staff for implementing Criterion 61.

B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard that details minimum design requirements for

*The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by MAR 5 1982

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," by the American National Standards Institute on April 12, 1976.

Primary facility design objectives are:

a. To prevent loss of water from the fuel pool that would uncover fuel,
b. To protect the spent fuel from mechanical damage, and
c. To provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. LOSS OF WATER FROM STORAGE POOL

Unless protective measures are taken to prevent the loss of water from a fuel storage pool, the spent fuel could overheat and cause damage to fuel cladding integrity, which could result in the release of radioactive materials to the environment. Equipment failures in systems connected to the pool could also result in the loss of pool water. A permanent coolant makeup system designed with suitable redundancy or backup would prevent the fuel from being uncovered should pool leaks occur. Further, early detection of pool leakage and fuel damage can be made using pool-water-level monitors and pool radiation monitors that alarm locally and also at a continuously manned location to ensure timely operation of building filtration systems. Natural events such as earthquakes or high winds can damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity will alleviate these concerns.

2. MECHANICAL DAMAGE TO FUEL

The release of radioactive material from fuel may occur as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or objects onto fuel elements during the refueling process and at other times.

Plant arrangements consider low-probability accidents such as the dropping of heavy loads (e.g., a 100-ton fuel cask) where such loads are positioned or moved in or over the spent fuel pool. It is desirable that cranes capable of carrying heavy loads be prevented from moving into the vicinity of the stored fuel.

Missiles generated by high winds also are a potential cause of mechanical damage to fuel. This concern can be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. LIMITING POTENTIAL OFFSITE EXPOSURES

Mechanical damage to the fuel might cause significant offsite doses unless dose reduction features are provided. Dose reduction designs such as negative pressure in the fuel handling building during movement of spent fuel would prevent exfiltration and ensure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment. Even if measures not described are used to maintain the desired negative pressure, small leaks from the building may still occur as a result of structural failure or other unforeseen events.

The staff considers Seismic Category I design assumptions acceptable for the spent fuel pool cooling, makeup, and cleanup systems. Tornado protection requirements are acceptable for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the filtration-ventilation system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

Power Plants," provide guidelines to limit potential offsite exposures through the filtration-ventilation system of the pool building.

Occupational radiation exposure is kept as low as is reasonably achievable (ALARA) in all activities involving personnel, and efforts toward maintaining exposures ALARA are considered in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

The requirements in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations,"* are generally acceptable to the NRC staff as a means for complying with the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 as related to light-water reactors (LWRs), subject to the following clarifications and modifications:

1. In lieu of the example inventory in Section 4.2.4.3(1), the example inventory should be that inventory of radioactive materials that are predicted to leak under the postulated maximum damage conditions resulting from the dropping of a single spent fuel assembly onto a fully loaded spent fuel pool storage rack. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

2. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525 filtration system, the ESF fuel storage building ventilation should provide and maintain a negative pressure of at least 3.2 mm (0.125 in.) water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed so that travel directly over the spent fuel storage pool or safety-related equipment is not possible. This should be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I, and safety-related structures and equipment should be subjected to quality assurance programs that meet the applicable provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Further, these programs should obtain guidance from Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," endorsing ANSI N45.2, and from the applicable provisions of the ANSI N45.2-series standards endorsed by the following regulatory guides:

1.30 (Safety Guide 30) "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (N45.2.4).

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (N45.2.2).

1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (N45.2.6).

1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants" (N45.2.11).

3. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to prevent tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to prevent missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado-generated missiles or from seismic damage to ensure that nothing bypasses the ESF-grade filtration system in the containment building.

4. In addition to meeting the requirements of Section 5.1.5.1, provisions should be made to ensure that nonfuel components in fuel pools are handled below the minimum water shielding depth. A system should be provided that, either through the design of the system or through administrative procedures, would prohibit unknowing retrieval of these components.

5. In addition to meeting the requirements of Section 5.1.12.10, the maximum potential kinetic energy capable of being developed by any object handled above stored spent fuel, if dropped, should not exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface should be provided between the cask venting system and the building ventilation system to minimize personnel exposure to the "vent-gas" generated from filling a dry loaded cask with water.

7. In addition to meeting the requirements of Section 5.3.3, radioactivity released during a Condition IV fuel handling accident should be either contained or removed by filtration so that the dose to an individual is less than the guidelines of 10 CFR Part 100. The calculated offsite dose to an individual from such an event should be well within the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to ensure that released activity does not bypass the

1.74 "Quality Assurance Terms and Definitions" (N45.2.10).

I.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (N45.2.9).

1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (N45.2.5).

1.116 "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (N45.2.8).

1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (N45.2.13).

10. The spent fuel pool water temperatures stated in Section 6.6.1(2) exceed the limits recommended by the NRC staff. For the maximum heat load during Condition I occurrences with normal cooling systems in operation and assuming a single active failure, the pool water temperature should be kept at or below 60°C (140°F). Under abnormal maximum heat load conditions (full core unload) and also for Condition IV occurrences, the pool water temperature should be kept below boiling.

11. A nuclear criticality safety analysis should be performed in accordance with Appendix A to this guide for each system that involves the handling, transfer, or storage of spent fuel assemblies at LWR spent fuel storage facilities.

12. The spent fuel storage facility should be equipped with both electrical interlocks and mechanical stops to keep casks from being transported over the spent fuel pool.

13. Sections 6.4 and 9 of ANS-57.2 list those codes and standards referenced in ANS-57.2. Although this regulatory guide endorses with clarifications and modifications ANS-57.2, a blanket endorsement of those referenced codes and

standards is not intended. (Other regulatory guides may contain some such endorsements.)

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits and operating licenses docketed after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than June 30, 1982.

APPENDIX A

NUCLEAR CRITICALITY SAFETY

1. SCOPE OF NUCLEAR CRITICALITY SAFETY ANALYSIS

1.1 A nuclear criticality safety analysis should be performed for each system that involves the handling, transfer, or storage of spent fuel assemblies at light-water reactor (LWR) spent fuel storage facilities.

1.2 The nuclear criticality safety analysis should demonstrate that each LWR spent fuel storage facility system is subcritical (k_{eff} not to exceed 0.95).

1.3 The nuclear criticality safety analysis should include consideration of all credible normal and abnormal operating occurrences, including:

a. Accidental tipping or falling of a spent fuel assembly,
b. Accidental tipping or falling of a storage rack during transfer,
c. Misplacement of a spent fuel assembly,
d. Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system,
e. Fuel drop accidents,
f. Stuck fuel assembly/crane uplifting forces,
g. Horizontal motion of fuel before complete removal from rack,
h. Placing a fuel assembly along the outside of rack, and
i. Objects that may fall onto the stored spent fuel assemblies.

1.4 At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

1.5 The nuclear criticality safety analysis should explicitly identify spent fuel assembly characteristics upon which subcriticality in the LWR spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis should explicitly identify design limits upon which subcriticality depends that require physical verification at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis should explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

2. CALCULATION METHODS AND CODES

Methods used to calculate subcriticality should be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," which endorses ANSI N16.9-1975.

3. METHOD TO ESTABLISH SUBCRITICALITY

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks, k_s, under normal and credible abnormal conditions should be equal to or less than an established maximum allowable multiplication factor, k_a; i.e.,

$$k_s \leq k_a$$

The factor, k_s, should be evaluated from the expression:

$$k_s = k_{sn} + \Delta k_{sb} + \Delta k_u + \Delta k_{sc}$$

where

k_{sn} = the computed effective multiplication factor; k_{sn} is calculated by the same methods used for benchmark experiments for design storage parameters when the racks are loaded with the most reactive fuel to be stored,

Δk_{sb} = the bias in the calculation procedure as obtained from the comparisons with experiments and including any extrapolation to storage pool conditions,

Δk_u = the uncertainty in the benchmark experiments, and

Δk_{sc} = the combined uncertainties in the parameters listed in paragraph 3.2 below.

3.2 The combined uncertainties, Δk_{sc}, include:

a. Statistical uncertainty in the calculated result if a Monte Carlo calculation is used,

b. Uncertainty resulting from comparison with calculational and experimental results,

c. Uncertainty in the extrapolation from experiment to storage rack conditions, and

d. Uncertainties introduced by the considerations enumerated in paragraphs 4.3 and 4.4 below.

3.3 The various uncertainties may be combined statistically if they are independent. Correlated uncertainties should be combined additively.

3.4 All uncertainty values should be at the 95 percent probability level with a 95 percent confidence value.

3.5 For spent fuel storage pool, the value of k_a should be no greater than 0.95.

4. STORAGE RACK ANALYSIS ASSUMPTIONS

4.1 The spent fuel storage rack module design should be based on one of the following assumptions for the fuel:

a. The most reactive fuel assembly to be stored at the most reactive point in the assembly life, or

b. The most reactive fuel assembly to be stored based on a minimum confirmed burnup (see Section 6 of this appendix).

Both types of rack modules may be present in the same storage pool.

4.2 Determination of the most reactive spent fuel assembly includes consideration of the following parameters:

a. Maximum fissile fuel loading,
b. Fuel rod diameter,
c. Fuel rod cladding material and thickness,
d. Fuel pellet density,
e. Fuel rod pitch and total number of fuel rods within assembly,
f. Absence of fuel rods in certain locations, and
g. Burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design should be the arrangement that results in the highest value of k_s considering:

a. Spacing between assemblies,
b. Moderation between assemblies, and
c. Fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of k_s shall include consideration of the following:

a. Eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,
b. Dimensional tolerances,
c. Construction materials,
d. Fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e. Presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f. Presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Fuel burnup determination should be made for fuel stored in racks where credit is taken for burnup. The following methods are acceptable:

a. A minimum allowable fuel assembly reactivity should be established, and a reactivity measurement should be performed to ensure that each assembly meets this criterion; or

b. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and a measurement should be performed to ensure that each fuel assembly meets the established criterion; or

c. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and an analysis of each fuel assembly's exposure history should be performed to determine its burnup. The analyses should be performed under strict administrative control using approved written procedures. These procedures should provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in paragraph 1.4 above.

The uncertainties in determining fuel assembly storage acceptance criteria should be considered in establishing storage rack reactivity, and auditable records should be kept of the method used to determine the fuel assembly storage acceptance criterion for as long as the fuel assemblies are stored in the racks.

Consideration should be given to the axial distribution of burnup in the fuel assembly, and a limit should be set on the length of the fuel assembly that is permitted to have a lower average burnup than the fuel assembly average.

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a. The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b. Fixed neutron absorbers should be an integral, nonremovable part of the storage rack.

c. When a fixed neutron absorber is used as the primary nuclear criticality safety control, there should be provision to:

(1) Initially confirm absorber presence in the storage rack, and

(2) Periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water should not normally be used in the evaluation of k_s. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

6.1 Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, ^{235}U depletion, amount of burnable poison, plutonium buildup and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildup are not necessarily the same.

6.2 Consideration should be given to the practical implementation of the spent fuel screening process. Factors to be considered in choosing the screening method should include:

a. Accuracy of the method used to determine storage rack reactivity;
b. Reproducibility of the result, i.e., what is the uncertainty in the result?
c. Simplicity of the procedure; i.e., how much disturbance to other operations is involved?
d. Accountability, i.e., ease and completeness of recordkeeping; and
e. Auditability.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Each nuclear power plant has a spent fuel storage facility. General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

1.2 Need for Proposed Action

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested that this guide be updated.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit or operating license application. Therefore, there should be a minimum number of cases where the applicant and the staff radically disagree on the design criteria.

1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also, recommendations made by Task A-36, which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," would be included.

3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

4. STATUTORY CONSIDERATIONS

4.1 NRC AUTHORITY

Authority for this regulatory guide is derived from the safety requirements of the Atomic Energy Act of 1954, as amended, through the Commission's regulations, in particular, General Design Criterion 61 of Appendix A to 10 CFR Part 50.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. CONCLUSION

Regulatory Guide 1.13 should be updated.

CONTENTION TC-2: EXHIBIT 4

Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants (August 19, 1998)

August 19, 1998

MEMORANDUM TO: Timothy Collins, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Laurence Kopp, Sr. Reactor Engineer /s/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

Attached is a copy of guidance concerning regulatory requirements for criticality analysis of new and spent fuel storage at light-water reactor power plants used by the Reactor Systems Branch. The principal objective of this guidance is to clarify and document current and past NRC staff positions that may have been incompletely or ambiguously stated in safety evaluation reports or other NRC documents. It also describes and compiles, in a single document, NRC staff positions on more recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests. This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel.

Attachment:
As stated

DISTRIBUTION

PDR	File Center
SRXB R/F	E. Kendrick
T. Marsh	T. Collins
G. Hubbard	E. Weiss
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

1. INTRODUCTION

This document defines the NRC Reactor Systems Branch guidance for the assurance of criticality safety in the storage of new (unirradiated or fresh) and spent (irradiated) fuel at light-water reactor (LWR) power stations. Safety analyses submitted in support of licensing actions should consider, among other things, normal operation, incidents, and postulated accidents that may occur in the course of handling, transferring, and storing fuel assemblies and should establish that an acceptable margin exists for the prevention of criticality under all credible conditions.

This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel. The guidance considers only the criticality safety aspects of new and spent LWR fuel assemblies and of fuel that has been consolidated; that is, fuel with fuel rods reassembled in a more closely packed array.

The guidance stated here is based, in part, on (a) the criticality positions of Standard Review Plan (SRP) Section 9.1.1 (Ref. 1) and SRP 9.1.2 (Ref. 2), (b) a previous NRC position paper sent to all licensees (Ref. 3), and (c) past and present practices of the staff in its safety evaluation reports (SERs). The guidance also meets General Design Criterion 62 (Ref. 4), which states:

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

The principal objective of this guidance is to clarify and document current and past staff positions that may have been incompletely or ambiguously stated in SERs or other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests (for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for non-removable poison inserts). Although these statements are not new staff positions, this document compiles them in a single paper. In addition, a recently approved staff position for pressurized-water reactors (PWRs) would allow partial credit for soluble boron in the pool water (Ref. 5).

The guidance stated here is applicable to both PWRs and boiling-water reactors (BWRs). The most notable difference between PWR and BWR fuel storage facilities is the larger size of the fuel assemblies and the presence of soluble boron in the spent fuel pool water of PWRs

The determination of the effective multiplication factor, k_{eff}, for the new or spent fuel storage racks should consider and clearly identify the following:

a. fuel rod parameters, including:

   1. rod diameter

   2. cladding material and cladding thickness

   3. fuel rod pellet or stack density and initial uranium-235 (U-235) enrichment of each fuel rod in the assembly (a bounding enrichment is acceptable)

b. fuel assembly parameters, including:

   1. assembly length and planar dimensions

   2. fuel rod pitch

   3. total number of fuel rods in the assembly

   4. locations in the fuel assembly lattice that are empty or contain nonfuel material

   5. integral neutron absorber (burnable poison) content of various fuel rods and locations in fuel assembly

   6. structural materials (e.g., grids) that are an integral part of the fuel assembly

The criticality safety analysis should explicitly address the treatment of axial and planar variations of fuel assembly characteristics such as fuel enrichment and integral neutron absorber (burnable poison), if present (e.g., gadolinia in certain fuel rods of BWR and PWR assemblies or integral fuel burnable absorber (IFBA) coatings in certain fuel rods of PWR assemblies).

Whenever reactivity equivalencing (i.e., burnup credit or credit for imbedded burnable absorbers) is employed, or if a correlation with the reactivity of assemblies in a standard core geometry is used (k_{∞}), such as is typically done for BWR racks, the equivalent reactivities must be evaluated in the storage rack configuration. In this latter approach, sufficient uncertainty should be incorporated into the k_{∞} limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of k_{∞}, and (3) uncertainty in average assembly enrichment.

If various locations in a storage rack are prohibited from containing any fuel, they should be physically or administratively blocked or restricted to non-fuel material. If the criticality safety of the storage racks relies on administrative procedures, these procedures should be explicitly identified and implemented in operating procedures and/or technical specification limits.

2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In general, transport methods of analysis are necessary for acceptable results. Storage rack characteristics such as boron carbide (B_4C) particle size and thin layers of structural and neutron absorbing material (poisons) need to be carefully considered and accurately described in the analytical model. Where possible, the primary method of analysis should be verified by a second, independent method of analysis. Acceptable computer codes include, but are not necessarily limited to, the following:

- CASMO - a multigroup transport theory code in two dimensions
- NITAWL-KENO5a - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- PHOENIX-P - a multigroup transport theory code in two dimensions, using discrete ordinates
- MONK6B - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- DOT - a multigroup transport theory code in two dimensions, using discrete ordinates

Similarly, a variety of cross-section libraries is available. Acceptable cross-section libraries include the 27-group, 123-group, and 218-group libraries from the SCALE system developed by the Oak Ridge National Laboratory and the 8220-group United Kingdom Nuclear Data Library (UKNDL). However, empirical cross-section compilations, such as the Hansen-Roach library, are not acceptable for criticality safety analyses (see NRC Information Notice No. 91-26). Other computer codes and cross-section libraries may be acceptable provided they conform to the requirements of this position statement and are adequately benchmarked

The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments. This qualifies both the ability of the analyst and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The Babcock & Wilcox series of critical experiments (Ref 6) provides an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control. Similarly, the Babcock & Wilcox critical experiments on close-packed arrays of fuel (Ref 7) provide an acceptable experimental basis for benchmark analyses for consolidated fuel arrays. A comparison with methods of analysis of similar sophistication (e g , transport theory) may be used to augment or extend the range of applicable critical experiment data

The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref 8)

The maximum k_{eff} shall be evaluated from the following expression:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(uncert) + \delta k(burnup),$$

where

k(calc)	= calculated nominal value of k_{eff},
δk(bias)	= bias in criticality analysis methods,
δk(uncert)	= manufacturing and calculational uncertainties, and
δk(burnup)	= correction for the effect of the axial distribution in burnup, when credit for burnup is taken.

A bias that reduces the calculated value of k_{eff} should not be applied. Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff} or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

4. NEW FUEL STORAGE FACILITY (VAULT)

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation, Ref. 9). Therefore, criticality safety analyses must address two independent accident conditions, which should be incorporated into plant technical specifications:

a. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with pure water, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

b. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum k_{eff} shall be no greater than than 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the limiting k_{eff} is maintained no greater than 0.95.

At low moderator density, the presence of relatively weak absorber material (for example, stainless steel plates or angle brackets) is often sufficient to preclude neutronic coupling between assemblies, and to significantly reduce the reactivity. For this reason, the phenomenon of low-density (optimum) moderation is not significant in racks in the *spent fuel pool* under the initial conditions before the pool is flooded.

Under low-density moderator conditions, neutron leakage is a very important consideration. The new fuel storage racks should be designed to contain the highest enrichment fuel assembly to be stored without taking credit for any nonintegral neutron absorber. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which subcriticality depends should be explicitly identified (e.g., fuel enrichment and the presence of steel plates or braces).

5. SPENT FUEL STORAGE RACKS

A. Reference Criticality Safety Analysis

1. For BWR pools or for PWR pools where no credit for soluble boron is taken, the criticality safety analyses must address the following condition, which should be incorporated into the plant technical specifications:

a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than or equal to 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level

2. If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:

   a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than 1.0, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

   b. With the spent fuel storage racks loaded with fuel of the maximum permissible ɜctivity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.[1]

3. The reference criticality safety analysis should also include, as a minimum, the following:

   a. If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable poison rods.

   b. For fuel assemblies containing burnable poison, the maximum reactivity should be the peak reactivity over burnup, usually when the burnable poison is nearly depleted.

   c. The spent fuel storage racks should be assumed to be infinite in the lateral dimension or to be surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of a reflector on the top and bottom of the fuel may be evaluated.

   d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1 0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.

4. The fuel assembly arrangement assumed in the criticality safety analysis of the spent fuel storage racks should also consider the following

---

[1] [*] is the boron concentration required to maintain the 0 95k_{eff} limit withcut consideration of accidents

a. the effect of eccentric positioning of fuel assemblies within the storage cells

b. the reactivity consequence of including the flow channel in BWR fuel assemblies

5. If one or more separate regions are designated for the storage of spent fuel, with credit for the reactivity depletion due to fuel burnup, the following applies.

   a. The minimum required fuel burnup should be defined as a function of the initial nominal enrichment.

   b. The spent fuel storage rack should be evaluated with spent fuel at the highest reactivity following removal from the reactor (usually after the decay of xenon-135). Operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in storage cells of the designated region(s).

   c. Subsequent decay of longer-life nuclides, such as Pu-241, over the rack storage time may be accounted for to reduce the minimum burnup required to meet the reactivity requirements.

   d. A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

   e. A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution.

B. Additional Considerations

1. The reactivity consequences of incidents and accidents such as (1) a fuel assembly drop and (2) placement of a fuel assembly on the outside and immediately adjacent to a rack must be evaluated. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions

2. If either credit for burnup is assumed or racks of different enrichment capability are in the same fuel pool, fuel assembly misloadings must be considered Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions

3. The analysis must also consider the effect on criticality of natural events (e.g., earthquakes) that may deform, and change in the relative position of, the storage racks and fuel in the spent fuel pool.

4. Abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormally elevated temperature conditions.

5. Normally, credit may only be taken for neutron absorbers that are an integral (nonremovable) part of a fuel assembly or the storage racks. Credit for added absorber (rods, plates, or other configurations) will be considered on a case-by-case basis, provided it can be clearly demonstrated that design features prevent the absorbers from being removed, either inadvertently or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.

6. If credit for soluble boron is taken, the minimum required pool boron concentration (typically, the refueling boron concentration) should be incorporated into the plant technical specifications or operating procedures A boron dilution analysis should be performed to ensure that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum technical specification boron concentration to the boron concentration required to maintain the $0.95k_{eff}$ design basis limit. The analysis should consider all possible dilution initiating events (including operator error), dilution sources, dilution flow rates, boration sources, instrumentation, administrative procedures, and piping. This analysis should justify the surveillance interval for verifying the technical specification minimum pool boron concentration.

7. Consolidated fuel assemblies usually result in low values of reactivity (undermoderated lattice). Nevertheless, criticality calculations, using an explicit geometric description (usually triangular pitch) or as near an explicit description as possible, should be performed to assure a k_{eff} less than 0.95.

6. REFERENCES

1. NRC, "Standard Review Plan " NUREG-0800, Rev 2, Section 9 1 1, "New Fuel Storage," July 1981.

2. NRC, "Standard Review Plan " NUREG-0800, Rev 2, Section 9 1.2, "Spent Fuel Storage," July 1981.

3. Brian K. Grimes, NRC, letter to all power reactor licensees, with enclosure "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978

4. *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

5. Westingnouse Electric Corporation, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, November 1996.

6. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

7. Babcock & Wilcox Company, "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, 1981.

8. National Bureau of Standards, *Experimental Statistics*, Handbook 91, August 1963.

9. J. M. Cano, R. Caro, and J. M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology*, Volume 48, May 1980.

CONTENTION TC-2: EXHIBIT 5

Letter from Donna B. Alexander, CP&L, to U.S. NRC, enclosing response to April 29, 199, RAI (June 14, 1999)

Carolina Power & Light Company
Harris Nuclear Plant
P.O. Box 165
New Hill NC 27562

SERIAL: HNP-99-094

JUN 14 1999

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

P

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE
HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Dear Sir or Madam:

By letter dated April 29, 1999, the NRC issued a request for additional information (RAI) regarding the Harris Nuclear Plant (HNP) license amendment request, submitted by CP&L letter Serial: HNP-98-188, dated December 23, 1998, to place spent fuel pools C and D in service. The HNP response to the NRC RAI is enclosed. The enclosed information is provided as a supplement to our December 23, 1998 license amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

Donna B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosure

9906210117 990614
PDR ADOCK 05000400
P PDR

c:

Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator - Region II

bc:

Mr. K. B. Altman
Mr. G. E. Attarian
Mr. R. H. Bazemore
Mr. C. L. Burton
Mr. S. R. Carr
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
Mr. W. J. Dorman (BNP)
Mr. R. S. Edwards
Mr. R. J. Field
Mr. K. N. Harris
Ms. L. N. Hartz
Mr. W J. Hindman
Mr. C. S. Hinnant
Mr. W. D. Johnson
Mr. G. J. Kline
Ms. W. C. Langston (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. J. H. O'Neill, Jr.
Mr. J. S. Scarola
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File
Files: H-X-0511
H-X-0642

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE
HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Requested Item 1

Although the burnup criteria for storage in Pools C or D will be implemented by administrative procedures to ensure verified burnup prior to fuel transfer into these pools, an administrative failure should be assumed and evaluation of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per Technical Specifications (TS) Figure 5.6.1) should be analyzed.

Response to Requested Item 1

The presence of soluble boron in the spent fuel pool water will assure that the reactivity is maintained substantially less than the design limitation in the event of a misloading event as described above. The Double Contingency Principle provides that neither the utility nor the staff is required to assume two unlikely, independent, concurrent events. Therefore, a failure of the administrative controls related to fuel assembly placement and the inadvertent dilution of the spent fuel pool water need not be considered to occur simultaneously. As a result, credit for the presence of soluble boron in the spent fuel pool water may be taken for an assembly misloading event as described. A minimum spent fuel pool boron concentration of 2000 ppm is maintained in accordance with HNP chemistry procedure CRC-001. This minimum boron concentration is more than adequate to offset the reactivity addition from a postulated fuel assembly misloading event. Based on analysis performed by Holtec International, it has been determined that a soluble boron concentration of 400 ppm would be sufficient to maintain k_{eff} less than 0.95 in the event of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1).

Requested Item 2

How will the burnup requirements needed to meet TS Figure 5.6.1 be ascertained for fuel assemblies shipped from other PWR plants (Robinson)?

Response to Requested Item 2

The burnup curve (proposed TS Figure 5.6.1) applies to the Robinson 15 x 15 fuel assembly types identified in Table 4.3.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98.

The selection of spent fuel for shipment to Harris is made in accordance with procedure NFP-NGGC-0003, entitled "Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask." The purpose of this procedure is to assure that the requirements of the IF-300

Cask Certificate of Compliance No. 9001 are met with regard to the selection of irradiated fuel to be shipped and that the fuel selected for shipment is acceptable for storage at CP&L's Harris plant. This procedure has been in use since 1990 for Robinson spent fuel shipments.

A computer program, which has also been in use since 1990 for Robinson spent fuel shipments, is used in conjunction with the above-referenced fuel selection procedure. For candidate assemblies to be shipped, the program retrieves the fuel type, enrichment, burnup, and decay heat from the special nuclear materials database. The initial enrichment data for each fuel assembly is contained in this database along with the other fuel data, and this data is based on manufacturing records. The burnup data for each fuel assembly is also included in the database along with the other isotopic inventories, and this data is obtained from the core monitoring software used for the Robinson plant. The special nuclear material database and core monitoring software have also been in use since 1990 for Robinson shipments.

The burnup curve proposed as TS Fig. 5.6.1 for pools C and D has already been programmed into the software for use in conjunction with fuel selection procedure NFP-NGGC-0003; however, this version is not yet in production as testing and documentation per CP&L's computer code quality assurance requirements are in progress. This new version will screen candidate PWR (Robinson) fuel against the burnup curve.

Revision to fuel selection procedure NFP-NGGC-0003 to reflect criticality screening requirements for fuel to be stored in Harris pools C or D has begun, but will not be completed until after: (1) the software changes identified above have been tested and the revised software placed in production status, and (2) the NRC has approved CP&L's license amendment application to place spent fuel pools C and D in service.

Requested Item 3

The fuel enrichment tolerance is specified in Section 4.5.2.5 as +0.0/-0.05. Why isn't a positive tolerance of +0.05 assumed (i.e., 5.0+0.05 weight percent U-235)?

Response to Requested Item 3

A maximum U-235 enrichment of 5.0 weight percent was specified, because it is the maximum enrichment allowed by both the Robinson and Harris Technical Specifications. Robinson TS 4.3.1.1.a states that the spent fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Robinson TS 4.3.1.2.a states that the new fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Harris TS 5.3.1 states that the initial core loading shall have a maximum enrichment of 3.5 weight percent U-235 and that reload fuel shall have a maximum enrichment of 5.0 weight percent U-235.

Also, the manufacturing facility of Siemens Power Corporation (SPC), the current fuel supplier for both the Robinson and Harris plants, is limited by license to a maximum U-235 enrichment of 5.0 weight percent. The SPC manufacturing tolerance is 0.05 weight percent U-235. Therefore, for enrichments with a tolerance of +/- 0.05%, the nominal design enrichment may not exceed

4.95 weight percent U-235 to ensure that the nominal plus the tolerance does not exceed 5.0 weight percent. The fuel enrichment and density tolerances specified in Section 4.5.2.5 appropriately supports a maximum allowable enrichment of 5.0 weight percent U-235.

Requested Item 4

Justify that the allowance that was assumed for possible differences between the fuel vendor and the Holtec calculations is sufficient to also encompass burnup calculational uncertainties.

Response to Requested Item 4

The Criticality Safety Calculations for the BWR Fuel Racks are summarized in Table 4.2.2 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98. An uncertainty on depletion was not explicitly included in the uncertainties summarized in Table 4.2.2. Instead, the 0.01 additive allowance for comparisons to vendor calculations discussed in Section 4.4.2.2 also accounts for burnup uncertainty. This practice is acceptable for the following two reasons:

First, the BWR calculations consider the peak reactivity during burnup. The k_{inf} in the rack corresponding to a peak k_{inf} in the Standard Cold Core Geometry (SCCG) of 1.32 was calculated in the analysis. The burnup corresponding to this peak reactivity value is simply a by-product of this calculation and, in contrast to PWR analysis, burnup is not used as a criteria for establishing acceptability for fuel storage. Any uncertainty in the burnup calculation would simply decrease or increase, with burnup, the location of the peak reactivity. However, the k_{inf} in the SCCG and the k_{inf} in the rack would remain the same at the peak in reactivity. As a result, an additional uncertainty on depletion is not necessary.

Second, the fuel vendor performs similar depletion calculations to those discussed in Section 4. Therefore any uncertainty in depletion is an inherent part of the comparison between those calculations in Section 4 and those performed by the vendor to determine the peak k_{inf} in SCCG as a function of burnup. Again, it is noted that the actual burnup at which the peak occurs is not used in the BWR acceptable fuel storage criteria.

Requested Item 5

The summary of criticality safety calculations shown in Tables 4.2.1 and 4.2.2 indicates that the total uncertainty is a statistical combination of the manufacturing tolerances but do not indicate methodology biases and uncertainties. Were these included?

Response to Requested Item 5

Section 4.4.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98, discusses the fact that CASMO-3, because it is a two-dimensional code, can not be directly compared to critical experiments and as a result a calculational/methodology bias is not available for CASMO-3. This section also discusses MCNP, which is a full three-dimensional Monte Carlo code, which has been benchmarked against critical experiments. CASMO-3 was used as the primary method of calculation and the results from CASMO-3 were compared to the regulatory limit of $k_{eff} \leq 0.95$ in Tables 4.2.1 and 4.2.2. As noted, the methodology bias and uncertainty were not included in these tables. However, these factors were implicitly included in a code-to-code comparison between CASMO-3 and MCNP shown in Table 4.5.1.

As discussed above, a methodology bias can not be developed for CASMO-3. Therefore, CASMO-3 results were compared to MCNP results to either verify that it produces conservative results relative to the benchmarked MCNP, or to determine a code-to-code bias. This comparison is discussed in Sections 4.5.1 and 4.6.1 with the results presented in Table 4.5.1. In the comparison between MCNP and CASMO-3, the methodology bias, uncertainty on the bias, calculational statistics, and a correction from 20°C to 4°C were added to the MCNP results. These results indicate that CASMO-3 is conservative relative to the benchmarked code MCNP and therefore the code-to-code bias was 0.0 for CASMO-3. Since the code-to-code bias was 0.0, it was not included in Tables 4.2.1 and 4.2.2. In conclusion, it can be stated that even though a methodology bias and uncertainty were not directly included in the final results shown in Tables 4.2.1 and 4.2.2, they were implicitly included through comparison of CASMO-3 and the benchmarked MCNP, provided in Table 4.5.1.

CONTENTION TC-2: EXHIBIT 6

Transcript of Deposition of Michael J. DeVoe, P.E.
(October 20, 1999)

COPY

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-400-LA
ASLBP NO. 99-762-02-LA

In the Matter of:	)
	)
CAROLINA POWER AND LIGHT COMPANY	)
	)
(Shearon Harris Nuclear Power Plant)	)
	)

- - - - - - - - - - - - - - - - - - -

DEPOSITION

OF

MICHAEL J. DEVOE, P.E.

At the Offices of Carolina Power & Light Company
411 Fayetteville Street Mall
Raleigh, North Carolina

October 20, 1999
9:40 a.m.

B. JORDAN & CO.
CERTIFIED VERBATIM REPORTERS
P.O. BOX 3372 CHAPEL HILL, NORTH CAROLINA 27515 (919) 929-6592

A P P E A R A N C E S

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(For the U.S. Nuclear Regulatory Commission)

Also Present

DR. GORDON THOMPSON

JOHN R. CAVES, P.E.

LAWRENCE KOPP

C O N T E N T S

E X H I B I T S

This being the deposition of MICHAEL J. DEVOE, P.E., taken by the Intervenor, on October 20, 1999, beginning at 9:40 a.m., at the offices of Carolina Power & Light Company, 411 Fayetteville Street Mall, Raleigh, North Carolina, before Betty Jordan, Certified Verbatim Reporter and Notary Public.

(Whereupon,

MICHAEL J. DEVOE, P.E.,

having first been duly sworn, was examined and testified as follows:)

DIRECT EXAMINATION BY MS. CURRAN:

Q Good morning, Mr. DeVoe.

A Good morning.

Q I'm going to be asking you some questions today. If there's any question that I ask that you don't understand, it's not because I'm trying to confuse you. And you can ask me to clarify the question. If you don't ask for clarification, I'll assume that you've understood the question.

If you need to talk with your counsel or take a break, you can ask me and take a break. Okay?

A Yes.

Q All right. I'm going to ask you some questions first about your resume, which has been provided to Orange County, and ask the court reporter to mark the resume of Michael J. DeVoe as Orange County Deposition Exhibit A.

(Whereupon, Deposition Exhibit A was marked for identification.)

Q First, are you aware that you've been identified by CP&L as a potential declarant or affiant in the licensing proceeding for the activation of spent fuel pools C and D?

A Yes.

Q Can you tell me what your current position is now with CP&L?

A Yes. I'm a project engineer in the nuclear fuel section. I work in a unit called Nuclear Fuel Services.

Q Do you have responsibilities in relation to criticality control in the spent fuel pools?

A Yes.

Q And what are they?

A I provide fuel-related data as input to the analyses and I review the analyses that are performed by our contractors.

Q So have you reviewed the criticality analyses that have been prepared by Holtec in this case?

A I've reviewed two reports in particular, yes.

Q And which two are those?

A I don't recall the particular number, but one is what I refer to as the base criticality analysis for pools C and D, and then a supplemental evaluation that looked at the misload of a fresh fuel assembly.

Q Have you performed any criticality analysis yourself?

MR. HOLLAWAY: A clarification in time. What time period are you talking about?

Q In relation to this license amendment application.

A No.

Q Are you familiar with CP&L's operations related to criticality control in the spent fuel pools at Harris?

A Could you clarify?

Q Well, I'll break it down. Are you familiar with CP&L operations related to boron, the addition of boron to the spent fuel pool water?

MR. HOLLAWAY: Object. A clarification. When you say "operations," I'm not sure I know exactly what that means specifically.

Q Are you familiar with the measures that are taken to introduce boron to the water in the spent fuel pool at Harris?

A No.

Q Are you familiar with the measures that are taken with respect to the tracking of spent fuel assemblies at the Harris plant?

A Yes.

Q So I take it that your contribution in this particular proceeding, which is the license amendment application for activation of spent fuel pools C and D, is to provide Holtec with information about the nature, the characteristics of the fuel at the plant that would be subject to a criticality analysis. That's one of your contributions; is that correct?

A Yes.

Q And the other is that you have reviewed the calculations done by Holtec for its criticality analysis.

A I reviewed the report, the documents, the results of those calculations.

Q And what aspect of the report are you competent to evaluate?

A To insure that they use the appropriate fuel data that was provided, that they analyzed all the fuel designs that we intend to store in those pools, and that the results appear to be reasonable and in accordance with applicable requirements.

Q Have you provided Holtec with any information other than information about the characteristics of the spent fuel itself?

MR. HOLLAWAY: Objection. Could you clarify--I think I know what you're talking about--but whether you're speaking about this particular license amendment application and these analyses.

Q In providing Holtec with information necessary for the criticality analysis that it performed with respect to this particular license amendment application, have you provided Holtec with any information other than information about the characteristics of the spent fuel itself?

A No, I don't believe so.

Q Have you provided Holtec with any information about the boron concentrations in the spent fuel pools?

A No.

Q Have you provided Holtec with any information about CP&L's system for tracking spent fuel movements at Harris?

A No.

Q I'm going to ask you some questions about CP&L's measures for identifying and keeping track of the spent fuel assemblies that come into the Harris plant and reside there.

Would I be correct in saying that there are three basic steps involved here? One would be the cataloguing or describing of the characteristics of each spent fuel assembly.

Another would be tracking the specific location of each spent fuel assembly, and--I'm sorry. I have to strike the word "spent"--each fuel assembly. And the last would be to verify that steps 1 and 2 have been taken appropriately.

Is that a correct description of the steps involved?

A I believe so.

Q I'd like to ask you about the first step, which would be describing or cataloguing the nature of the fuel.

Can you tell me what are the characteristics that are catalogued or described when you make a record of fuel assemblies coming into the plant?

A The assembly ID, their serial number, the amount of fissile material contained in that bundle in terms of grams and the uranium enrichment.

Q There's a record of that information that's made when the assembly enters the plant; is that correct?

A Are you talking about fuel coming fresh to the Harris plant or being brought from other plants to Harris?

Q I'm talking about both.

A Well, there's also a record of the burn-up and there's also a record of all of its previous locations.

Q All right. When an assembly comes to the plant--and it may be a fresh assembly or a spent one--this record is made for each assembly, right?

A Correct.

Q Okay. And some of this information has been provided already by someone else; is that correct? The ID is put on by someone else.

A Correct.

Q In the case of fresh fuel, who puts the ID on?

A The fuel manufacturer.

Q Does every assembly have the same ID from birth to death, from when it's created to when it's finally put into the storage or the spent fuel pool? That doesn't change.

A That's correct.

Q The amount of fissile material, where does that information come from? Who generates that information?

A The fuel manufacturer.

MR. HOLLAWAY: Could you clarify at what time you're talking about? I know you talked about fresh fuel and burned fuel and both together.

And it's not clear to me when you ask that question as to what time you're talking about. I think I know, but--

MS. CURRAN: Well, I'm talking about when the fuel arrives at Harris, where have you gotten the information.

MR. HOLLAWAY: Regardless of where it's coming from.

MS. CURRAN: Regardless of where it's coming from.

Q So answer the question with respect to both fresh and spent fuel if it's different. So for the amount of fissile material.

A For the fresh, that comes from the fuel vendors. For the exposed, it comes from our reactor records, our special nuclear material accountability.

Q Now, when Harris gets spent fuel assemblies, at the moment they are coming from other CP&L plants; is that correct?

A (Witness nods affirmatively.)

Q So when you say "our" you mean other CP&L--that's the CP&L organization.

A Correct. I mean the Brunswick plant or the

Robinson plant.

Q Has CP&L ever accepted fuel assemblies from any other facility other than--at Harris from any facility other than a CP&L facility?

A Not to my knowledge.

Q You had mentioned a special nuclear material, accountability something. What did you say it was?

A It's just a program, computer program.

Q All right. I'll come back to that. Let's finish going through this list of the information that's included in the first record that's made.

So the amount of fissile material in a spent fuel assembly that's coming from a CP&L plant or from the Harris plant is recorded--when you get it at the Harris plant, you get information from CP&L's special nuclear material accountability program; is that right?

A (Witness nods affirmatively.)

Q Okay. What about with respect to burn-up? Where does that information come from for fresh and spent fuel assemblies?

A Well, for the fresh fuel assemblies the burn-up is zero. For the exposed fuel assemblies, that's part of the core monitoring program output.

Q Is that another computer program?

A Yes.

Q Okay. And then the last item was previous locations. Who provides that information to you with respect to the fresh fuel?

A With the fresh fuel, that would not have-- it's not a valid record for fresh fuel.

Q And how about for the spent fuel?

A It's a history of the approved and executed fuel movement instructions.

Q And who provides that information? Where does that information come from?

A That comes from the completed fuel movement procedure instructions.

Q Is that also a computer program?

A This information is stored on a database, computer database, yes. But the locations are not necessarily computer-generated. The movement instructions can be written by hand.

Q Could you explain that? Is it a piece of paper?

A Yes.

Q It's a piece of paper.

A Uh-huh (affirmative).

Q And that has a list of the previous locations.

A I'm not sure of that, of how far back it goes.

Q Okay. So when this information is received, what happens to it? How is it recorded?

A It's maintained on its database.

Q Does CP&L have a procedure for verifying all of the information that's provided when an assembly is received?

A I don't know.

Q Okay. Could you describe for me the special nuclear material accountability program?

A I could describe my understanding of it. It's not one of my specialty areas.

Q Okay.

A As a licensee, we're required to keep track of certain information associated with what we call special nuclear material, which in this case is uranium. And we keep track of how much we have and where it's at.

And also, when we, if you will, generate special nuclear material by irradiating a fuel in a reactor, there are other isotopes besides uranium 235 that we are required to account for. And that program is more of a--not necessarily a computer program, but a program that we have in place to keep track of that material.

Q So is this the program that you go to when you want to ask where is a specific fuel assembly?

A Yes.

Q Is this program universal to all of the CP&L plants or is it just at Harris?

A All three sites use the program.

Q Using this special nuclear material accountability program, can you track the history of each spent fuel assembly?

A Yes.

Q So is this database a unified database that covers the complete history for every fuel assembly that's used at any CP&L plant?

A I'm not sure.

Q I'm sorry. Were you about to say something more?

A No.

Q Is it a realtime database?

A I'm not sure what you mean by that question.

Q If I want to know right now what is the complete history of each fuel assembly from the database, can I get that information immediately?

A Yes.

Q That's what I mean by realtime.

A Yes.

Q Yes. And that covers all of the information up to the present.

A Provided the records have been loaded into the system.

Q Okay. Why don't you tell me about that?

A I'm not sure what you mean by that question.

Q Well, is the information not recorded immediately into the database when it's received?

A Obviously, between the physical movement of the fuel and the recording on the paper and the transferring of the paper to the person responsible for the database and entering that information into the database, there is a certain time lapse.

Q Are there any requirements for the maximum amount of time that is allowed to pass?

A I don't know.

Q Do you know what the backup is for this database?

A No, I don't.

Q What is the physical location of the database?

A I'm not sure.

Q How is the database accessible to a Harris plant operator?

A The database is maintained on our computer network. And if a person has been granted access to that database, they would be able to access it from a desktop personal computer.

Q Is there a paper copy of the database that's kept?

A I'm not sure.

Q You also mentioned a core monitoring program. That's correct?

A Correct.

Q Could you describe that for me?

A Basically, that's the on-line reactor core monitoring system that monitors power shapes, power levels, and what we use to demonstrate conformance with technical specification requirements.

Q Does the core monitoring program have any relationship to the SNM accountability program?

A Yes.

Q How are they related?

A The core monitoring program tracks the assembly burn-up and the isotopes, the generation of the isotopes as a result of that burn-up. And then that information is transferred to the special nuclear material program.

Q When new information is received or generated that needs to be input into the special nuclear material accountability program, but it hasn't been input yet--I just want to take that situation--is any notation made in the program at that point?

A I don't know.

Q You don't know. Do you know who does know?

A Yes.

Q Can you tell me that?

A The names of the people?

Q Uh-huh (affirmative).

A Bob Kunita and Linda Young.

Q And why are they people that know? What are their responsibilities?

A One of their responsibilities is the special nuclear material program.

Q This program, the special nuclear material accountability program, deals with fuel that is in a CP&L system. That's correct?

A Yes.

Q If CP&L takes fuel from other plants that are not currently in the system, are there any other measures that would be added to this program?

MR. HOLLAWAY: I object to that question. There's no foundation.

Q It's my understanding that CP&L has just purchased a nuclear plant in Florida. If CP&L were to take fuel from the plant in Florida for which the information about the fuel characteristics is not currently in the SNM accountability program, what measures would be taken?

MR. HOLLAWAY: Well, I object again.

MR. DEVOE

There's no foundation for that scenario occurring.

Q You may answer the question.

A I mean, that's highly speculative. You know, the purchase has not been finalized. And you're asking me to predict what would happen in the future.

My best estimate or expectation of what would happen is that it would be treated the same way, that that information would be added. But I'm not aware of any thoughts of even doing that.

Q I'd like to go to the step which would be the tracking of the fuel. Suppose that you need to know where a specific fuel assembly is in a plant. How do you find out?

A One method would be to go to the special nuclear material database.

Q And assuming the information has been input into the computer, the database would tell you where it was.

A Correct.

Q What would another method be?

A Depending upon where the fuel was, if it was in the core, in the reactor, there's other--you can look at the loading pattern. The core map describes the loading pattern.

Q The core map? Is that what you said?

A Yes.

Q And where is that kept?

A I'm not sure what--

Q What kind of a map is it? Is it a paper map?

A Yes. It's a paper map.

Q And where is the paper map located?

A It would be in the reload modification package, what we call an engineering service request.

Q Are there other methods for tracking the fuel?

A You could--yes. You could go through the completed fuel-handling procedures.

Q And is that a paper document or a computer file?

A It would be a paper document.

Q And where is that kept?

A In the vault, the quality assurance vault.

Q Suppose that a fuel assembly is moving down a canal and I want the tracking system to tell me where it is and where it was and where it's going.

Would the tracking system give me that information?

A I don't know.

Q Do you know if Federal Express would do that for you if it was a package?

MR. HOLLAWAY: Object. That's not relevant to the proceeding. The witness may answer.

A I've seen it on advertisements.

Q If I wanted to get the history of a fuel assembly that was in a particular rack position, could I get that by going to the SNM database?

A I believe so, yes.

Q And I could get the complete history of that assembly?

MR. HOLLAWAY: I'm going to object as ambiguous. When you say "history," I'm not sure what you mean.

Q The question is that for each location can you--first of all, can you identify each location where the fuel has been?

A Yes.

Q And for each of those locations, can you get information about the fuel characteristics, the burn-up, enrichment, amount of fissile material?

A Yes.

Q Can I get this information that we just discussed--when I'm looking for the previous history of locations and the characteristics at each location, can I get this information immediately?

A What do you mean by "immediately"?

Q Within a minute.

A I believe so.

Q If I knew the characteristics of the fuel and the previous location of the fuel, could I use that information to find the current location using this program?

A Yes.

Q So it will search--even if you don't have all the information, you can put in some input and get an ID?

MR. HOLLAWAY: Object. When you say "some input"--

Q Well, if you put in some of the characteristics of the fuel.

A Yes. It's a database and it's searchable, as a database would be.

Q Supposing that I were trying to verify whether the information regarding, say, the burn-up level that's recorded in the database for a particular fuel assembly is correct. How would I go about doing that? What measures does CP&L have for doing that?

A I'm not sure.

Q You don't know anything about measures for validating the data that's been input into the program.

A I'm not involved in those activities or familiar with the activities.

Q Can you tell me who would be?

A Yes. Bob Kunita and Linda Young.

MR. HOLLAWAY: At some point I'd like to take a break just as a regular break to talk to him.

MS. CURRAN: Okay.

MR. HOLLAWAY: I don't know if this would be a good time.

MS. CURRAN: Maybe in just a few minutes.

MR. HOLLAWAY: Okay.

Q If you look at a specific rack position, how can you be sure that the fuel in that position in the rack has the same ID as the database says it does?

A How can you be sure? One way is the surveys of the pool, the video camera.

Q Now, tell me, why would that tell you?

A You could lower the video camera into the rack location of interest and actually read the serial number on the bundle and then compare that to your records.

Q Is that done?

MR. HOLLAWAY: Object as ambiguous. You said, "Is that done?" Is that done when, by who? I don't know what that means.

Q Before a fuel assembly is moved, are any steps taken to verify the identity of the fuel assembly that's being moved?

A I don't know.

MS. CURRAN: This is a good time for a break.

(Whereupon, a brief recess was taken.)

Q Mr. DeVoe, have you searched the database that we've been speaking of?

A No.

Q You haven't?

A No.

Q How do you know about this database?

A I request it to be queried to provide data that I might need.

Q So someone else uses the database, but you provide the request for information?

A Yes.

Q You've provided some information to me about the characteristics of the database today. How did you get that information?

A From being aware of what is in the database by looking at the results of the data queries and by providing the information that gets put into the database and by being familiar with the special nuclear materials plant.

Q You're saying you provide information that gets input to the database; is that correct?

A When you say "I," do you mean myself personally or in my work functions?

Q Well, you were the one that said it, so I guess--

A Okay. Could you repeat the question?

Q You said that you provide input to the database, I believe; is that correct?

A Yes.

Q What kind of information have you provided to the database?

A Information on fresh fuel when it's first delivered to the reactor site.

Q Can you tell me how you got the information that went into the database? Did you read a piece of paper?

A It's provided as part of the QA documentation by the fuel vendor when the fuel is manufactured. And I participate in surveillances of the vendor while he's manufacturing our fuel. And one of the things that I review is the documentation.

Q And what is the documentation of?

A In reference to our discussions here, it's the assembly ID and amount of uranium as manufactured.

Q When you review it, are you reviewing a piece of paper?

A Yes.

Q And then do you hand that piece of paper to a person who's inputting the information to the computer?

A Yes.

Q So you don't transcribe or create a new document. You take the information that's been given to you and you hand it to the person who is inputting it to the computer.

A It's provided that way, yes.

Q It's provided--

A Well, I physically don't give the sheet of paper to the person.

Q But you make sure that person gets it.

A Yes.

Q In the course of providing information to Holtec regarding spent fuel characteristics for purposes of Holtec's criticality analysis in this particular proceeding, did you make any attempt to validate the data that you were providing?

A Yes.

Q And how did you do that?

A First off, the fuel-related data describes the fresher or unexposed fuel and its, primarily, dimensions and enrichments and the physical characteristics of the fuel assembly.

And we worked with the respective fuel

vendors to obtain that data, and they provided it under their quality assurance plans.

And then when it was received by CP&L, we perform a review of it to insure that it's the information that Holtec has requested and it's appropriate for use in this project.

Q What do you mean by "appropriate"?

A That we provided information on all the fuel types present as opposed to maybe just the most current, to make sure that we covered all the fuel types that were planned to be loaded, that it was in fact describing our fuel.

The vendors make fuel for many customers, and we insure that it describes our fuel, the CP&L fuel.

Q It sounds like the information you provided was from the vendors, not from the CP&L database; is that correct?

A Correct. The fuel information describes the fresh fuel.

Q Let me make sure I understand what you've told me. It's my understanding that you have attempted to validate data that you obtained from vendors in order to give to Holtec. Is that correct?

A Yes, we do validate it.

Q Have you ever attempted to validate data that you got from the CP&L SNM accountability database?

A I haven't.

Q In your current position, have you searched the SNM accountability database?

A Yes. I requested a query.

Q For what purpose?

A I requested a listing of the fuel assemblies that were, in this case, presently in the Robinson spent fuel pool.

Q And did you attempt--scratch that. And for what purpose did you request that information?

A To support a criticality evaluation.

Q And did you attempt to validate the information that you had obtained from the database, this listing?

A No.

Q Would you know how to do that?

A Would I know how to validate it?

Q Right.

A Yes.
Q How would you do it?
A I would compare the database records against the reactor records.
Q What reactor records?
A As I mentioned earlier, there's an on-line process computer that tracks fuel assembly burn-up. And I would--the records produced by that are the records that are intended to be transferred into the special nuclear material database.
Q Is that a separate source of information? If it gets transferred into the special nuclear material database, how do you know that's not the source of information you originally queried?
A I'm not sure I understand the question.
Q You had said that the on-line process computer puts information into the SNM accountability database; is that correct?
A The information generated is transferred to the SNM. It's not automatically put there, but it's transferred. In this case, we're talking about a limited set of the information, just the burn-up and the isotopics at this point.

Q Are you competent to physically validate the burn-up level of the spent fuel assembly?

A I'm not sure what you mean by "physically."

Q What I'm talking about is, if you have a specific fuel assembly in a rack and you're required to experimentally verify the characteristics, the burn-up characteristics that are described in the database, would you be able to that?

MR. HOLLAWAY: I object to that for a foundation. I don't know that there's any foundation for a requirement, as you stated, for experimental validation of fuel assembly burn-up.

Q Can you answer the question?

A I am not.

Q You would not know how to do that?

A I know of ways it could be done, but I have never done that myself.

Q What are the ways that it could be done?

A One technique is called gamma scan.

Q A gamma what?

A Scan.

Q How does that work?

A The assembly is--measurements are taken of the assembly for a particular isotope that is representative of the power and burn-up distribution.

Q Can that operation be done reliably for an assembly that is in a rack with other assemblies nearby?

A I don't know.

Q Did you participate in the design of the SNM accountability database?

A No.

Q Have you ever been involved in any changes to the program?

A Yes.

Q Can you describe that for me?

A We are in the process of making and implementing changes to support adding to the database the information required to facilitate implementation of pools C and D. And I'm providing the input. I'm not doing any manipulation of coding.

Q And what kind of input are you providing?

A The maximum planar average enrichment for the fuel assemblies.

Q Who is responsible for actually changing the program?

A Bob Kunita.

Q The same person you mentioned before?

A Yes.

Q Do you know what kinds of changes are being made?

A In general, I don't know the specific coding changes. But the functionality changes I'm aware of.

Q Would you please describe those for me?

A It's to add a data field that records the maximum planar average enrichment for the PWR fuel assemblies and for the boiling water reactor, BWR fuel, the maximum lattice planar average enrichment, and the standard cold core geometry K-infinity.

Q Mr. DeVoe, are you familiar with the physical process for introducing boron into the spent fuel pools at Harris?

A No.

Q You know nothing about it?

A (No response.)

Q Do you know how the boron gets into the pool?

A I believe I do.

MR. DEVOE

Q Can you explain it to me?
A I believe it's added by the operators.
Q How do the operators do it?
A I don't know those details.
Q Do you know whether boron concentrations are monitored in the pools?
A Yes.
Q Do you know how frequently they're monitored?
A No, I don't.
Q Are records maintained of boron measurements?
A Yes.
Q For how long are they maintained?
A I do not know.
Q Has CP&L prepared any boron dilution analyses that you know of?
A Could you clarify that? We're restricting this to the pool.
Q Yes.
A And what do you mean by "boron dilution"?
Q I'm actually using a term that's provided in an NRC guidance document, which isn't defined

further than that.

A I'm not aware of any calculations.

Q Do you know if the physical correspondence between spent fuel pools makes any difference in maintaining the boron concentration in the pools?

A I do not know that.

Q Do you know if CP&L has ever done any studies or analyses of its own experience with maintaining boron concentrations in its spent fuel pools?

A No, I do not.

Q Do you know if there are any industry studies that have been done on industry experience with controlling boron that was in spent fuel pools?

A No, I do not.

Q Do you know of any studies prepared by CP&L or any other entity that would describe the probability or consequences of accidents resulting from errors in boron concentration levels?

A No.

Q Do you know of any studies or analyses by CP&L or any other entity of the probability or consequences of criticality accidents in spent fuel pools?

A I don't recall seeing any.

Q Do you know of any analyses or studies done by CP&L or any other entity regarding errors or accidents caused by the mishandling or misplacement of fuel assemblies?

A No.

Q Can you tell me what regulations or guidance documents are followed by CP&L in attempting to maintain criticality control at the Harris plant?

A Yes.

Q And what are they?

A GDC-62 and Reg Guide 1.13.

Q Any others?

A Not that I'm aware of.

MS. CURRAN: We're going to take a short break.

(Whereupon, a brief recess was taken.)

(Mr. Caves and Mr. O'Neill exit.)

MS. CURRAN: I have no further questions.

MS. UTTAL: I don't have any questions.

MR. DEVOE

MR. HOLLAWAY: I don't have any questions.

AND FURTHER DEPONENT SAITH NOT.

(Deposition concluded at 11:30 a.m.)

S I G N A T U R E   O F   W I T N E S S

I have read the foregoing pages numbered 4 through 39, inclusive, which contain a correct transcription of answers made by me to the questions therein recorded, with the exceptions and/or additions reflected on the errata sheet attached hereto, if any.

Signed this 9th day of December, 1999.

Michael J. DeVoe

MICHAEL J. DEVOE, P.E.

****

STATE OF ______________________________

COUNTY OF ______________________________

Subscribed and sworn to before me this _______ day of ________________________, 1999.

____

NOTARY PUBLIC

(SEAL)

My Commission Expires:

____

ERRATA

I, Michael J. DeVoe, the witness herein, have read my deposition and request that the following changes be made:

PAGE	LINE	CHANGE	REASON FOR CHANGE
8	5	", the documents," to "that documents"	Typographical
8	9	"use" to "used"	Typographical
14	10	"have - -" to "have significance,"	Clarification
15	12	"its database" to "the database"	Typographical
16	5	"irradiating a fuel" to "irradiating fuel"	Typographical
27	17	"materials plant" to "material plan"	Typographical
29	22	"fresher" to "fresh"	Typographical

CONTENTION TC-2: EXHIBIT 8

Internal AEC memorandum from G.A. Arlotto to J.J. DiNunno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966)
(relevant excerpt)

Those Listed Below

October 7, 1966

G. A. Arlotto
Facilities Standards Branch, SS

REVISED DRAFT - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached is a revised draft of the General Design Criteria for Nuclear Power Plant Construction Permits dated October 6, 1966, which I developed for your consideration. In comparison with the previous draft, which was dated July 25, 1966, the attached version reflects the following:

1. Changes suggested by ACRS Subcommittee members at meetings of August 10 and September 21, 1966.

2. Changes suggested in the Backup Document dated August 9, 1966.

3. Changes suggested in memorandum from Robert H. Bryan to J. J. DiNunno dated October 3, 1966.

4. Changes resulting from discussions among the addressees and myself.

5. My suggestions which time did not permit resolution of with the addressees.

Attachment:
As Stated Above

Addressees:
J. J. DiNunno, Assistant Director for Reactor Standards, SS
Robert H. Bryan, Chief, Facilities Standards Branch, SS

OFFICE ▶	SS:FSB					
SURNAME ▶	Arlotto:jjb					
DATE ▶	10-7-66					

Form AEC-318 (Rev. 9-53) U. S. GOVERNMENT PRINTING OFFICE 16—62761-8

Revised Draft
10/6/66

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The purpose of these criteria is to define or describe the basic safety objectives to be met in the design of a nuclear power plant. They are intended: (1) to serve as guidance to the applicant in preparing an application for an AEC construction permit and (2) to aid the AEC staff in reviewing that application.

The application of these criteria to a specific design involves a considerable amount of engineering judgment. There may be instances in which one or more of these criteria are unnecessary or are insufficient. It is not intended that the criteria be used as a check list of design objectives for all proposed plants, and the applicant is free to establish the safety of his design by alternative criteria. The criteria will be modified if, or as, future technological developments and experience warrant.

An applicant for a construction permit is expected to present a design approach together with data and analyses sufficient to give assurance that the design can reasonably be expected to fulfill all applicable criteria. It is recognized that the nature and detail of technical information and analysis required at the construction permit stage to provide such assurance may vary, depending on the particular criterion under consideration. Category A criteria encompass critical safety areas so fundamental in the design, procurement, fabrication, and construction of the plant that modification for reasons of safety at the operating license review stage would be exceedingly difficult and costly; in essence, for practical purposes, decisions made at the construction permit stage in these areas are irrevocable. Where novel features

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are associated with criteria which are site-sensitive or are directly related to limiting the accidental release of radioactivity into the public domain, they must be dealt with in a relatively complete way at the construction permit stage even if the "irrevocable" condition is not met. Category B criteria encompass safety areas where the modifications can be made for reasons of safety at the operating license review stage without placing an undue burden on the parties concerned. These criteria principally concerned with protecting the operational capability of the reactor may be dealt with in relatively less detail at the construction permit stage if more detailed information and analysis are not available at that time.

All applicable safety criteria must, of course, be fulfilled as a condition for issuance of a license to operate the plant.

CRITERION 1 (Category A) QUALITY AND PERFORMANCE STANDARDS

Those features of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to:

(a) Quality standards* that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are applicable, they shall be used. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented as necessary.

* A showing of sufficiency and applicability of standards used shall be required.

(2) Active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

(3) A capability is provided to test periodically the delivery capability at a position as close to the spray nozzles as is practical.

(4) A capability is provided to test under conditions as close to the design as practical the full operational sequence that would bring the systems into action, including the transfer to alternate power sources.

CRITERION 10 (Category B) FUEL AND WASTE STORAGE SYSTEMS

Storage and handling systems for fuel and waste shall be designed on the basis that:

1. Possibilities for inadvertent criticality must be prevented by engineered systems or processes toevery extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.

2. Reliable decay heat removal means must be provided as necessary to prevent fuel or storage volume damage that could result in radioactivity release to plant operating areas or the public environs. Such means must be assured for all anticipated normal and abnormal conditions as well as those accident situations whereby normal cooling could credibly become lost.

CONTENTION TC-2: EXHIBIT 7

AEC Press Release entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits"
(November 22, 1965)

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. H-252
Tel. 973-3335 or
973-3446

FOR IMMEDIATE RELEASE
(Monday, November 22, 1965)

AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The Atomic Energy Commission is seeking comment from the nuclear industry and other interested persons on proposed general design criteria which have been developed to assist in the evaluation of applications for nuclear power plant construction permits.

The proposed criteria have been developed by the AEC regulatory staff and discussed with the Commission's Advisory Committee on Reactor Safeguards (ACRS). They represent an effort to set forth design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such, they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It is recognized that further efforts by the AEC regulatory staff and the ACRS will be necessary to fully develop these criteria. However, the criteria as now proposed are sufficiently advanced to submit for public comment. Also, they are intended to give interim guidance to applicants and reactor equipment manufacturers.

The development and publication of criteria for nuclear power plants was one of the key recommendations of the special Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures.

In the further development of these criteria, the AEC intends to hold discussions with organizations in the nuclear industry and to issue from time to time explanatory information on each criterion. Following such discussions with industry and receipt of other public comment, the AEC expects to develop and publish criteria that will serve as a basis for evaluation of applications for nuclear power plant construction permits.

(more)

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached hereto are general design criteria used by the AEC in judging whether a proposed nuclear power facility can be built and operated without undue risk to the health and safety of the public. They represent design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It should be recognized that additional criteria will be needed for evaluation of a detailed design, particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Moreover, there may be instances in which it can be demonstrated that one or more of the criteria need not be fulfilled. It should also be recognized that the application of these criteria to a specific design involves a considerable amount of engineering judgment.

An applicant for a construction permit should present a design approach together with data and analysis sufficient to give assurance that the design can reasonably be expected to fulfill the criteria.

FACILITY

CRITERION 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.

CRITERION 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

CRITERION 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

CRITERION 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

CRITERION 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

CRITERION 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

CRITERION 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

CRITERION 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can intefere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to

CRITERION 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

CRITERION 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

CRITERION 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

CRITERION 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent

CRITERION 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid, or solid effluents.

CRITERION 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

CONTENTION TC-2: EXHIBIT 9

Letter from J J DiNunno, AEC, to David Okrent, ACRS (October 25, 1966), and attached October 20, 1966 draft of General Design Criteria (relevant excerpt)

October 25, 1966

Dr. David Okrent, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D.C. 20545

Dear Dr. Okrent:

Enclosed for consideration of the ACRS are draft copies of the General Design Criteria for Nuclear Power Plant Construction Permits. This redrafted material includes a comparison of criteria contained in the Press Release dated November 22, 1965, and those contained in our latest draft dated October 20, 1966. In addition, we have included along with a revised draft of the criteria dated October 20, 1966, a comparison of the October 20 draft with the July 25 draft previously submitted and discussed with the ACRS Criteria Subcommittee.

Our October 20, 1966,draft attempts to reflect results of our last discussion with the ACRS Subcommittee, and we would like to have the scheduled November 9th meeting on criteria be based on the October 20th draft.

Sincerely yours,

J. J. DiNunno
Assistant Director for
Reactor Standards
Division of Safety Standards

Enclosures:
1. Rev. Draft dated 10/20/66 of
   General Design Criteria (18)
2. Comparison of Drafts dated 7/25/66 and
   10/20/66 for General Design Criteria (18)
3. Comparison of Criteria in Press Release
   dated 11/22/65 and Those in Rev. Draft
   dated 10/20/66 (18)

bcc: Harold L. Price, Director of Regulation, w/encl.
~~Clifford K. Beck, Deputy Dir. of Reg., w/encl.~~
Peter A. Morris, Director, DRL, w/encl. (6)
SS:ADIR M. M. Mann, Asst. Dir. for Nuclear Safety, REG, w/encl.

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SURNAME ▶	DiNunno:jjb					
DATE ▶	10-25-66					

REVISED DRAFT OF

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

October 20, 1966

The purpose of these criteria is to define or describe the basic safety objectives to be met in the design of a nuclear power plant. They are intended: (1) to serve as guidance to the applicant in preparing an application for an AEC construction permit and (2) to aid the AEC staff in reviewing that application.

The application of these criteria to a specific design involves a considerable amount of engineering judgment. There may be instances in which one or more of these criteria are unnecessary or are insufficient. It is not intended that the criteria be used as a check list of design objectives for all proposed plants, and the applicant is free to establish the safety of his design by alternative criteria. The criteria will be modified if, or as, future technological developments and experience warrant.

An applicant for a construction permit is expected to present a design approach together with data and analyses sufficient to give assurance that the design can reasonably be expected to fulfill all applicable criteria. It is recognized that the nature and detail of technical information and analysis required at the construction permit stage to provide such assurance may vary, depending on the particular criterion under consideration.

To provide guidance as to the relative emphasis expected at the construction permit stage, the criteria have been divided into two broad categories. Category A criteria involve aspects of facility design that are site-sensitive or are directly related to limiting the accidental release of radioactivity into the public domain. These aspects of facility design are also categorized by their marked influence on plans for construction and operation. From a practical viewpoint, aspects of facility design satisfying Category A criteria are relatively fixed at the construction permit stage and not amenable to change without serious disruptions of construction plans and incurrence of considerable costs. For these reasons, those aspects of facility design provided in fulfillment of Category A criteria must be dealt with in a relatively complete way at the construction permit stage.

Category B criteria are intended to reflect primarily those aspects of design that provide for safe operational control of the facility. Such features are generally less unique to a facility than those required for satisfying Category A criteria and are much less determinate of facility construction schedules. Modifications to such features that might prove necessary, for safety reasons, following issuance of a construction permit are much more likely to be accommodated without the pressures for compromise that might well accompany the more time-consuming and costly type changes. Under these circumstances, criteria principally concerned with the safe operational control of the reactor and designated as Category B may be dealt with in relatively less detail at the construction permit stage, if more detailed information is not available at that time.

All applicable safety criteria must, of course, be fulfilled as a condition for issuance of a license to operate the plant.

9.2.4.4 A capability is provided to test under conditions as close to the design as practical the full operational sequence that would bring the systems into action, including the transfer to alternate power sources.

FUEL AND WASTE STORAGE SYSTEMS

CRITERION 10 (Category B) FUEL AND WASTE STORAGE

10.0 Storage and handling systems for fuel and waste shall be designed on the basis that:

10.1 Possibilities for inadvertent criticality must be prevented by engineered systems or processes to every extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.

10.2 Reliable decay heat removal means must be provided as necessary to prevent fuel or storage volume damage that could result in radioactivity release to plant operating areas or the public environs. Such means must be assured for all anticipated normal and abnormal conditions as well as those accident situations whereby normal cooling could credibly become lost.

10.3 Shielding for radiation protection shall be provided as required from considerations of 10 CFR 20.

10.4 Containment of the systems shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

CONTENTION TC-2: EXHIBIT 10

Letter from J. J. DiNunno, AEC, to Nunzio J. Palladino, ACRS (February 8, 1967), and attached draft of General Design Criteria (relevant excerpt)

February 8, 1967

Mr. Nunzio J. Palladino, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Palladino:

Enclosed for consideration by the Committee is a redraft of General Design Criteria. The format of the criteria has been changed. The subparts previously listed in earlier drafts have been made into separate criteria. The wording of these criteria is essentially the same as those in the October 20, 1966, draft, modified to reflect subsequent discussions held with the ACRS Subcommittee in November and recent developments of criteria for emergency core cooling systems.

An additional document showing the changes made from the last draft discussed with the ACRS is under preparation and will be forwarded by separate correspondence.

Sincerely yours,

J. J. DiNunno
Assistant Director for
Reactor Standards
Division of Safety Standards

Enclosure:
General Design Criteria for Nuclear
Power Plant Construction Permits (18)

bcc: Harold L. Price, Director of Regulation, w/encl.
Clifford K. Beck, Deputy Director of Regulation, w/encl.
M. M. Mann, Asst. Dir. for Nuclear Safety, w/encl.
C. L. Henderson, Asst. Dir. for Administration, w/encl.
Peter A. Morris, Director, DRL, w/encl. (6)
Edson G. Case, Deputy Director, DRL, w/encl.
Forrest Western, Director, DRL, w/encl.

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SURNAME ▶	DiNunno:jjb	PAM				
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GENERAL DESIGN CRITERIA

FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS

February 6, 1967

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VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 61 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Possibilities for criticality in new and spent fuel storage shall be prevented by physical systems or processes to every extent practicable. Such means as favorable geometries shall be emphasized over procedural controls.

CRITERION 62 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to ensure damage to the fuel or storage facilities that could result in radioactivity release to plant operating areas or the public environs is prevented. Such means must be assured for all anticipated normal and abnormal conditions as well as those accident situations whereby normal cooling could credibly become lost.

CRITERION 63 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category A)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required from consideration of 10 CFR 20.

CRITERION 64 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

CRITERION 65 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over plant radioactive effluents, whether solid, liquid, or gaseous. Appropriate

CONTENTION TC-2: EXHIBIT 11

Note by the Secretary, W.B. McCool, to AEC Commissioners re: Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits (June 16, 1967)
(relevant excerpts)

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AEC-R 2/57

June 16, 1967

ATOMIC ENERGY COMMISSION

PROPOSED AMENDMENT TO 10 CFR 50: GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Note by the Secretary

1. The Director of Regulation has requested that the attached report be circulated for consideration by the Commission at an early date.

2. The Commission approved the proposed design criteria, as revised, during consideration of AEC-R 2/49 at Regulatory Meeting 223 on November 10, 1965.

W. B. McCool

Secretary

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Commissioner Nabrit	2	Congr. Relations	2
Commissioner Johnson	2	Inspection	1
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... staff has considered all ...

the criteria. In addition, subsequent redrafts were circulated to other divisions within the Commission. Principal comments from these divisions have been reflected in the revised criteria. Other comments from within the Commission will be considered in conjunction with public comments received after publication in the Federal Register.

6. The regulatory staff has worked closely with the Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment. The ACRS has stated that it believes that the revised criteria are appropriate to publish for public comment.

7. It is proposed that the criteria be included as Appendix A to 10 CFR 50. The proposed amendment, which is attached as Appendix "B," provides that the General Design Criteria be used for guidance by an applicant in developing the principal design criteria for the facility. For a specific reactor case, some of the General Design Criteria may be unnecessary or inappropriate and the criteria, as a whole, may be insufficient. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced reactor types. In any case, there must be assurance that the principal design criteria proposed by an applicant encompass all those facility design features required in the interest of public safety.

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

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

STAFF JUDGMENTS

10. 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 concurs in recommendation 11.c.

RECOMMENDATION

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

a. Approve publication of the proposed amendments to 10 CFR Part 50 contained in Appendix "B."

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

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

LIST OF ENCLOSURES

APPENDIX "A"

LIST OF INCOMING CORRESPONDENCE ON "AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS" PRESS RELEASE NO. H-252 DATED NOVEMBER 22, 1965

1. J. B. McCarty, Jr., U.S. Coast Guard, 1/26/66.
2. E. P. Epler, Oak Ridge National Laboratory, 1/26/66.
3. Dr. Emerson Jones, Technical Management, Inc., 2/2/66.
4. H. C. Paxton and D. B. Hall, Los Alamos Scientific Laboratory, 2/2/66.
5. C. Starr, Atomics International, 2/4/66.
6. C. T. Chave, Stone and Webster Engineering Corporation, 2/11/66.
7. R. L. Junkins, Pacific Northwest Laboratory, 2/8/66.
8. Richard Hughes, Governor of New Jersey, 2/10/66.
9. Royce J. Rickert, Combustion Engineering, Inc., 2/11/66.
10. W. B. Cottrell, Oak Ridge National Laboratory, 2/11/66.
11. Peter A. Morris, Director, Division of Operational Safety, 2/11/66.
12. Holmes & Narver, Inc., 2/11/66.
13. CDR J. C. Ledoux, BuY&D, Dept. of Navy, 2/11/66.
14. Richard H. Peterson, Pacific Gas and Electric Company, 2/14/66.
15. Norbert L. Kopchinski, Professional Engineer, California, 2/14/66.
16. D. L. Crook, Dept. of Commerce, Maritime Adm., Wash., D.C., 2/15/66.
17. R. H. Harrison, Babcock & Wilcox, 2/22/66.
18. Theodore Stern, Westinghouse Electric Corporation, 2/25/66.
19. E. A. Wiggin, Atomic Industrial Forum, 2/28/66.
20. James G. Terrill, Jr., Dept. of Health, Education, and Welfare, Washington, D.C., 3/7/66.
21. J. P. Hogan, General Atomic, 4/30/66.
22. H. G. Rickover, Director, Division of Naval Reactors, 7/26/66.

APPENDIX "B"

[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.

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 divisions within the Commission, 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 (1) 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

---

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. /Additions are underscored./

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:

(See Attachment)

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

Dated at ______________________ this ____________ day of ____________ 1967.

For the Atomic Energy Commission.

____

W. B. McCool
Secretary

APPENDIX A

GENERAL DESIGN CRITERIA FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS[3/]

Table of Contents

3/ 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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[illegible]

Every applicant for a construction permit is required by the provisions of §50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

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

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

CONTENTION TC-2: EXHIBIT 12

Notice of proposed rulemaking for General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967)

52.

Published 32 Federal Register 10213, July 11, 1967

FEDERAL REGISTER will be considered before action is taken on the proposed amendment. No hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Administration officials may be made by contacting the Chief, Air Traffic Branch. Any data, views, or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The Birmingham 1,200-foot transition area described in § 71.181 (32 F.R. 2148 and 3765) would be altered as follows:

"* * * thence southwest along the southeast boundary of V-209 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VOR; thence clockwise along this arc to longitude 87°30'00" W.; thence north along longitude 87°30'00" W. to point of beginning, excluding that portion that coincide with R-2101 and the Gadsden, Ala., transition area * * *." would be deleted and "* * * thence southwest along the southeast boundary of V-209 to longitude 88°00'00" W.; thence north along longitude 88°00'00" W., to the north boundary of V-18; thence northeast along the north boundary of V-18 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VORTAC; thence clockwise along this arc to longitude 87°30'00" W.; thence north along longitude 87°30'00" W. to point of beginning, excluding that portion that coincides with R-2101 and the Gadsden, Ala., transition area * * *." would be substituted therefor.

The proposed additional airspace is required for the protection of IFR operations and for radar vectoring of aircraft arriving and departing the Birmingham area.

The official docket will be available for examination by interested persons at the Southern Regional Office, Federal Aviation Administration, Room 724, 3400 Whipple Street, East Point, Ga.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348(a)).

Issued in East Point, Ga., on June 30, 1967.

JAMES G. ROGERS,
Director, Southern Region.

[F.R. Doc. 67-7949; Filed, July 10, 1967; 8:49 a.m.]

[14 CFR Part 71]

[Airspace Docket No. 67-SO-64]

TRANSITION AREA

Proposed Designation

The Federal Aviation Administration is considering an amendment to Part 71 of the Federal Aviation Regulations that would designate the Camden, S.C., transition area.

Interested persons may submit such written data, views, or arguments as they may desire. Communications should be submitted in triplicate to the Area Manager, Atlanta Area Office, Attention: Chief, Air Traffic Branch, Federal Aviation Administration, Post Office Box 20636, Atlanta, Ga. 30320. All communications received within 30 days after publication of this notice in the FEDERAL REGISTER will be considered before action is taken on the proposed amendment. No hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Administration officials may be made by contacting the Chief, Air Traffic Branch. Any data, views, or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The Camden transition area would be designated as:

That airspace extending upward from 700 feet above the surface within a 7-mile radius of Woodward Field (latitude 34°17'03" N., longitude 80°33'53" W.); within 2 miles each side of the 040° bearing from the Camden RBN (latitude 34°17'02" N., longitude 80°33'42.5" W.), extending from the 7-mile radius area to 8 miles northeast of the RBN.

The proposed transition area is required for the protection of IFR operations at Woodward Field. A prescribed instrument approach procedure to this airport utilizing the Camden (private) nondirectional radio beacon is proposed in conjunction with the designation of this transition area.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348(a)).

Issued in East Point, Ga., on June 21, 1967.

GORDON A. WILLIAMS, Jr.
Acting Director, Southern Region.

[F.R. Doc. 67-7950; Filed, July 10, 1967; 8:49 a.m.]

[14 CFR Part 71]

[Airspace Docket No. 67-EA-1]

FEDERAL AIRWAYS

Supplemental Proposed Alteration

On March 1, 1967, a notice of proposed rule making was published in the FEDERAL REGISTER (32 F.R. 3402) stating that the Federal Aviation Agency was considering amendments to Part 71 of the Federal Aviation Regulations that would realign V-1 from Cape Charles, Va., via the INT of Cape Charles 013° and Salisbury, Md., 206° True radials; to Salisbury; that would designate a segment of V-139 from Norfolk, Va., via Cape Charles; to Snow Hill, Md., including a west alternate from Norfolk to Snow Hill via INT of Norfolk 360° and Snow Hill 226° True radials; and that would revoke the segment of V-194 from Norfolk to INT of Norfolk 001° and Cape Charles 313° True radials. Floors of 1,200 feet above the surface were proposed for these airway segments. These actions were proposed to simplify air traffic control procedures and flight planning in the Norfolk area.

Subsequent to publication of the notice, it was determined that the Snow Hill 226° True radial would not support a Federal airway. Accordingly, the proposals published in the notice are hereby cancelled and in lieu thereof, consideration is given to the following airway alignments that would serve the same purpose.

1. Redesignate the segment of V-194 from Norfolk via the intersection of Norfolk 001° T (008° Mag.) and Harcum, Va., 072° T (079° Mag.) radials; to the intersection of Harcum 072° and Snow Hill 211° True radials.

2. Realign V-1 from Cape Charles via the intersection of Cape Charles 009° T (016° Mag.) and Salisbury 206° T (214° Mag.) radials; to Salisbury.

Interested persons may participate in the proposed rule making by submitting such written data, views, or arguments as they may desire. Communications should identify the airspace docket number and be submitted in triplicate to the Director, Eastern Region, Attention: Chief, Air Traffic Division, Federal Aviation Administration, Federal Building, John F. Kennedy International Airport, Jamaica, N.Y. 11430. All communications received within 45 days after publication of this notice in the FEDERAL REGISTER will be considered before action is taken on the proposed amendment. The proposal contained in this notice may be changed in the light of comments received.

An official docket will be available for examination by interested persons at the Federal Aviation Administration, Office of the General Counsel, Attention: Rules Docket, 800 Independence Avenue SW., Washington, D.C. 20590. An informal docket will be available for examination at the office of the Regional Air Traffic Division Chief.

These amendments are proposed under the authority of section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348).

Issued in Washington, D.C., on July 3, 1967.

T. MCCORMACK,
Acting Chief, Airspace and Air Traffic Rules Division.

[F.R. Doc. 67-7951; Filed, July 10, 1967; 8:49 a.m.]

ATOMIC ENERGY COMMISSION

[10 CFR Part 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plant Construction Permits

The Atomic Energy Comission 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

No. 132——9

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Plant Construction Permits."[1] 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).

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, U.S. 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 NW., Washington, D.C.

1. Section 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 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:

[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.

[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.

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS[3]

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[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.

Introduction. Every applicant for a construction permit is required by the provisions of § 50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

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

I. Overall Plant Requirements

Criterion 1—Quality Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Criterion 2—Performance Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) Appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Criterion 3—Fire Protection (Category A). The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Criterion 4—Sharing of Systems (Category A). Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Criterion 5—Records Requirements (Category A). Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. Protection by Multiple Fission Product Barriers

Criterion 6—Reactor Core Design (Category A). The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Criterion 7—Supression of Power Oscillations (Category B). The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Criterion 8—Overall Power Coefficient (Category B). The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Criterion 9—Reactor Coolant Pressure Boundary (Category A). The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 10—Containment (Category A). Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. Nuclear and Radiation Controls

Criterion 11—Control Room (Category B). The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Criterion 12—Instrumentation and Control Systems (Category B). Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Criterion 13—Fission Process Monitors and Controls (Category B). Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Criterion 14—Core Protection Systems (Category A). Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Criterion 15—Engineered Safety Features Protection Systems (Category B). Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Criterion 16—Monitoring Reactor Coolant Pressure Boundary (Category B). Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Criterion 17—Monitoring Radioactivity Releases (Category B). Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Criterion 18—Monitoring Fuel and Waste Storage (Category B). Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. Reliability and Testability of Protection Systems

Criterion 19—Protection Systems Reliability (Catergory B). Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Criterion 20—Protection Systems Redundancy and Independence (Category B). Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Criterion 21—Single Failure Definition (Category B). Multiple failures resulting from a single event shall be treated as a single failure.

Criterion 22—Separation of Protection and Control Instrumentation Systems (Category B). Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Criterion 23—Protection Against Multiple Disability for Protection Systems (Category B). The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Criterion 24—Emergency Power for Protection Systems (Category B). In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Criterion 25—Demonstration of Functional Operability of Protection Systems (Category B). Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Criterion 26—Protection Systems Fail-Safe Design (Category B). The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. Reactivity Control

Criterion 27—Redundancy of Reactivity Control (Category A). At least two independent reactivity control systems, preferably of different principles, shall be provided.

Criterion 28—Reactivity Hot Shutdown Capability (Category A). At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Criterion 29—Reactivity Shutdown Capability (Category A). At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Criterion 30—Reactivity Holddown Capability (Category B). At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Criterion 31—Reactivity Control Systems Malfunction (Category B). The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Criterion 32—Maximum Reactivity Worth of Control Rods (Category A). Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. Reactor Coolant Pressure Boundary

Criterion 33—Reactor Coolant Pressure Boundary Capability (Category A). The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Criterion 34—Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A). The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Criterion 35—Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A). Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120° F. above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60° F. above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Criterion 36—Reactor Coolant Pressure Boundary Surveillance (Category A). Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. Engineered Safety Features

Criterion 37—Engineered Safety Features Basis for Design (Category A). Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Criterion 38—Reliability and Testability of Engineered Safety Features (Category A). All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Criterion 39—Emergency Power for Engineered Safety Features (Category A). Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Criterion 40—Missile Protection (Category A). Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Criterion 41—Engineered Safety Features Performance Capability (Category A). Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Criterion 42—Engineered Safety Features Components Capability (Category A). Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Criterion 43—Accident Aggravation Prevention (Category A). Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Criterion 44—Emergency Core Cooling Systems Capability (Category A). At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Criterion 45—Inspection of Emergency Core Cooling Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Criterion 46—Testing of Emergency Core Cooling Systems Components (Category A). Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 47—Testing of Emergency Core Cooling Systems (Category A). A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Criterion 48—Testing of Operational Sequence of Emergency Core Cooling Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Criterion 49—Containment Design Basis (Category A). The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Criterion 50—NDT Requirement for Containment Material (Category A). Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30° F. above nil ductility transition (NDT) temperature.

Criterion 51—Reactor Coolant Pressure Boundary Outside Containment (Category A). If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Criterion 52—Containment Heat Removal Systems (Category A). Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Criterion 53—Containment Isolation Valves (Category A). Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Criterion 54—Containment Leakage Rate Testing (Category A). Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Criterion 55—Containment Periodic Leakage Rate Testing (Category A). The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Criterion 56—Provisions for Testing of Penetrations (Category A). Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Criterion 57—Provisions for Testing of Isolation Valves (Category A). Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Criterion 58—Inspection of Containment Pressure-Reducing Systems (Category A). Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Criterion 59—Testing of Containment Pressure-Reducing Systems Components (Category A). The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 60—Testing of Containment Spray Systems (Category A). A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Criterion 61—Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A). A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Criterion 62—Inspection of Air Cleanup Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

Criterion 63—Testing of Air Cleanup Systems Components (Category A). Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Criterion 64—Testing of Air Cleanup Systems (Category A). A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Criterion 65—Testing of Operational Sequence of Air Cleanup Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. Fuel and Waste Storage Systems

Criterion 66—Prevention of Fuel Storage Criticality (Category B). Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Criterion 67—Fuel and Waste Storage Decay Heat (Category B). Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Criterion 68—Fuel and Waste Storage Radiation Shielding (Category B). Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

Criterion 69—Protection Against Radioactivity Release From Spent Fuel and Waste Storage (Category B). Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

Criterion 70—Control of Releases of Radioactivity to the Environment (Category B). The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

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

Dated at Washington, D.C., this 28th day of June 1967.

For the Atomic Energy Commission.

W. B. McCOOL,
Secretary.

[F.R. Doc. 67-7901; Filed, July 10, 1967; 8:45 a.m.]

CONTENTION TC-2: EXHIBIT 13

Letter from William B. Cottrell, ORNL, to H. L. Price, AEC (September 6, 1967) and enclosed ORNL comments on proposed GDC.

DOCKET NUMBER PROPOSED RULE 50

OAK RIDGE NATIONAL LABORATORY

OPERATED BY

UNION CARBIDE CORPORATION

NUCLEAR DIVISION

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OAK RIDGE, TENNESSEE 37830

September 6, 1967

DOCKETED USAEC SEP 15 1967 Office of the Secretary Public Proceedings Branch

Mr. H. L. Price
Director of Regulation
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

Subject: Review of USAEC "General Design Criteria for Nuclear Power Plant Construction Permits" Federal Register, July 11, 1967

The subject document has been reviewed by members of the staff of the Nuclear Safety Information Center. We realize and appreciate the great amount of work that your staff has done in bringing these criteria to their present form. We participated in the initial review of the criteria when they were issued in November 1965 and we are pleased to have the opportunity to review this later version. Our comments are enclosed in two parts: (1) general comments which apply to the entire set of criteria and (2) specific comments on the individual criteria and in a few cases on sections such as VII, Engineered Safety Features.

With a few exceptions, the scope of the criteria seems broad enough and generally well organized. We do have rather extensive comments on those criteria which deal with protection systems. A difficult problem is that of assessing reliability. The "single failure criterion" is an attempt to relieve this situation, but its application is subjective and it has different meanings to different individuals. Another problem area is that of the use of the same instruments for both operating the plant and providing protection. We believe that such interdependence can only degrade the reliability and performance of the protection system. Problems such as these make the task of writing criteria and standards quite difficult.

Further, the absence of clear definitions of terms, which to many are rather loosely understood, could limit the effectiveness of the criteria. We feel that there is a critical need for these definitions.

RECEIVED SEP 11 1967 U.S. ATOMIC ENERGY COMMISSION

We again wish to commend you for the significant contribution represented by these criteria. If you have questions concerning our comments, we will be glad to discuss them with you.

Sincerely yours,

Wm. B. Cottrell

Wm. B. Cottrell, Director
Nuclear Safety Information Center

WBC:JRB:jt

Enclosure

cc A. J. Pressesky

General Comments

1. The ramifications of civil disobedience, riots, strikes, sabotage, and the like have not even been mentioned. With this vast potential risk in mind, should not the physical security of the plant be considered?

2. Since these criteria will be used by many groups whose terminology is not always (or even usually) in agreement, a set of definitions is badly needed. For example - what is a system, component, engineered safety feature, failure, redundancy, channel, surveillance, monitoring, malfunction, protection system, loss of coolant accident, etc.?

3. Since "single failure criteria" are to be applied to systems other than those for control (for which criterion 21 is the definition), it is extremely important that they be clearly defined for all systems.

4. Since the introduction uses the phrase "nuclear reactor plant" why is the phrase "reactor facility" used in the text of several of the criteria to mean the same thing?

Specific Comments

Title - General Design Criteria for Nuclear Power Plant Construction Permits

The title is really not grammatically correct, since it infers that we are designing a "construction permit".

Criterion 2 - Performance Standards

1. Line 7: Delete "performance" since this could be construed as applying to operating performance only.

2. In regard to earthquakes the "appropriate margin for withstanding forces greater than those recorded . . ." has not been defined here and furthermore it would be extremely difficult to do so at least with our present understanding of earthquake phenomena. Therefore, the criterion should state what constitutes an adequate margin.

Criterion 4 - Sharing of Systems

We agree with criterion 4 as it applies to the nuclear reactor plant but it should be extended to apply to systems, sub-systems, and especially engineered safety features.

Criterion 5 - Records Requirements

1. Line 2: Should read, "Records of the design, fabrication, inspection, testing and construction of . . ." to be sufficiently inclusive. The performance of engineered safety features must be determined as a datum for evaluation of subsequent tests required of the system. For example, criterion 46 states that active components be periodically tested for required performance.

2. Line 5: Change "its" to "his" to refer to the operator's control.

Criterion 8 - Overall Power Coefficient

For this entire criterion it might be better to say that "the reactor shall be designed so that either the overall power coefficient in the power operating range shall not be positive or reliable controls which will eliminate or minimize the undesirable effects of a positive power coefficient shall be provided, tested and proved effective."

Criterion 10 - Containment

We infer from subsequent criteria that the protection system is not considered an engineered safety feature even though there are reactors that depend upon the protection systems to work in order not to overstress the containment. Thus, either "engineered safety features" should be defined to include the reactor protective system, i.e., scram functions, or this and other functions should be specifically mentioned. We prefer the former alternative.

Criterion 11 - Control Room

The aims of this criterion are certainly desirable but it is difficult if not impossible to prove the criterion has been met. However, some clarification is needed, for example, if a fire in a panel renders the controls of some emergency system inoperable, the criterion can be interpreted to mean that two separate control rooms are required. Is this the intent?

Criterion 13 - Fission Process Monitors and Controls

1. Line 4: Delete "throughout core life and" since it is redundant.

2. The examples cited should either be deleted or augmented by a more comprehensive set including flux, hot spots, etc.

Criteria 14 and 15 - Core Protection Systems and Engineered Safety Features

These criteria exemplify the fact that a more detailed definition of containment and engineered safety features needs to be included. One could define the engineered safety features as including scram system, core protection system, etc., and then eliminate Criterion 14.

Suggested Criterion - Monitoring Engineered Safety Features

We suggest that this criterion be inserted at this point: Instrumentation shall be provided to monitor the performance of engineered safety features during the course of the accident and to monitor the condition of the reactor itself under these conditions.

Criterion 16 - Monitoring Reactor Coolant Pressure Boundary

This criterion defines the monitoring that is necessary to prove compliance with Criterion 9. (Similar proof is required by Criterion 36) In cases of this nature cross referencing of criteria should be made for the sake of clarity.

Criterion 17 - Monitoring Radioactivity Releases

This criterion was written to specify monitoring to meet the specifications of Criterion 70, which should be cross referenced here.

Criterion 18 - Monitoring Fuel and Waste Storage

Specification of criticality monitoring should be included in this criterion; for example, as by reference to 10 CFR, Part 70.34.

Criterion 19 - Protection Systems Reliability

There is no guide for determining whether or not the functional reliability and in-service testability is commensurate with the safety functions to be performed. Every designer could claim that his system met this criterion, and challenge a reviewer to show otherwise. Arguments about this criterion most likely will include comparisons to somewhat similar protection systems for somewhat similar nuclear power plants that have been reviewed and approved.

This criterion is of questionable value and we recommend its omission. A set of rules for designing protection systems would be more useful than a general statement of desirable results.

Criterion 20 - Protection Systems Redundancy and Independence

The criterion is not clear as to the extent of the effects of a single failure that need consideration. Apparently, considerations of effect are to be limited to a component or channel - resulting in a severe limitation in the value of this criterion. This is another example of a criterion where definitions are needed; for example, component, channel, and system need to be defined.

Criterion 21 - Single Failure Definition

A judgment of the extent of failures caused by a single event hinges on credibility. First, there is the probability of the initiating event, then the probability of progressive failures. A single event of sufficient magnitude will certainly prevent the functioning of the protection system. Detailed guidelines for describing the required independence of redundant equipment are needed. Examples are spacing between cables carrying redundant signals, methods of separating electronic equipment handling redundant signals, methods of isolating redundant logic devices which combine redundant signals, etc. Unless more detailed information is given as to what is to be considered credible, this criterion serves little purpose.

Criterion 22 - Separation of Protection and Control Instrumentation Systems

This criterion apparently recognizes the need for separating protective and control instrumentation but compromises this objective with the qualifications permitted. The net effect is to permit the intimate intermingling of the system that normally operates the plant and the system that is intended to afford protection. We strongly recommend that no exceptions be permitted to the separation of these two systems as the only effective means to insure the vital integrity of the protection system.

Both of these systems in the new and larger reactors are complex. Despite the use of buffer amplifiers in attempting to isolate the effects of failures in the two systems, the systems are not independent when the same signals are coupled into each. Additionally, the objectives of operation are not those of protection. When the two systems are intermingled, signal processing equipment is invariably designed for operating the plant rather than for protection. Inadequate control demands that corrections must be made in the equipment to allow operation, but inadequate protection equipment may be discovered only after their need during an accident. Mixing of the two systems as allowed by this criterion diverts design attention from the requirements of protection to those of operation. Such mixing also increases the probability that protection will be lost as the result of a failure in the control system that initiates the accident requiring protection.

The basic justification for independence of protection and operation systems, in our opinion, is the relative ease with which the protection function can be assured with independence, and the great difficulty of realizing such assurance with interdependence. We believe it is easier to separate the systems than to assure that their interactions are harmless. We believe it is easier to maintain independence than to insure, for the lifetime of the plant, that deliberate changes or inadvertent alteration of the operation system will not adversely affect the protection function.

The dismal list of accidents caused by design errors, and the much larger list of design errors caught before they caused accidents, lead us to believe that design errors will continue to occur. We believe further that independence of operation and protection is one of the best defenses against the possibility that a design error may cause an unprotected accident.

It may be possible that for some combinations of protection and operation instruments no conceivable failure of the operation function involved can result in a situation requiring action of the protection function involved. To the extent that this can be proved, both initially and throughout reactor lifetime, the particular interdependence could be acceptable. A hypothetical example is the instrumentation used to measure and control the pressure of a sealed containment enclosure. The operation function is used principally to provide a pressure differential between the inside of the containment and the outside, and thus to provide a means for surveillance of the leakage rate.

The protection function might be to initiate reactor shutdown, emergency cooling, and isolation of process piping if a rise in containment pressure should indicate the presence of a serious leak of potentially radioactive fluids. It might be demonstrable that no failure whatever of this instrumentation could induce a substantial leak of radioactive fluid, in which case no real interdependence of operation system and protection system would in fact exist.

The basis of the above example is the impossibility that failure of the operational function or equipment could ever, under any circumstances, lead to a situation where the protection function would be needed. Therefore, sharing of equipment (common elements) between the protection system and the operation system could not lead to interaction between the two systems. It is difficult to prove conclusively this lack of functional interaction. More difficult is the problem of ensuring that this lack of interaction can and will be maintained throughout the life of the plant. Operators are not designers; operators in charge of the plant at the end of its 40-year life are not the ones who may have discussed protection problems with the designers at the beginning. Subtle considerations are apt to be forgotten or ignored. It is easy to forget that plant protection was originally based on the impossibility that failure of certain operation instruments could result in a need for protection-system function.

Criterion 24 - Emergency Power for Protection Systems

Design requirements related to power supply include consideration of both Criteria 24 and 26. There is an anomaly here in that Criterion 24 permits the protection system to require power to provide protection, whereas Criterion 26 requires the system to fall into a safe or tolerable state on loss of power. To the extent that Criterion 26 can be met, alternate power sources become an economic or operational consideration rather than being needed for safety.

Criterion 25 - Demonstration of Functional Operability of Protection Systems

We agree with the intent of this criterion but suggest that the wording be changed to state ". . . demonstrate that no failure causing a reduction of redundancy . . ." rather than ". . . demonstrate that no failure or loss of redundancy . . .". Some systems may have extra elements whose failures do not reduce the redundancy claimed for the system.

Criterion 26 - Protection Systems Fail-Safe Design

This criterion places a requirement not only on the protection system but on the plant as well. For example, a plant design could be such that operation of the protection mechanism when not needed would be highly undesirable. (An illustration is the closure of the steam stop valves in a

BWR.) Criterion 26 requires the plant to be able to accept operation of the protection system when not needed. We believe this is a good objective and we support this criterion.

Section V - Reactivity Control

1. The title of this section should be "Reactivity Control for Reactor Shutdown".

2. This group of criteria should distinquish more clearly between functions of reactivity control; namely, the dynamic reactivity reduction process and the static holddown functions. The first function must be performed at such times as in power transients and loss-of-coolant accidents with the objective of preventing exceeding "acceptable fuel damage limits" referred to in Criteria 28 and 29. Margins expressed in terms of shutdown parameters are inappropriate and inadequate for the dynamic function.

   The reliability with which each function must be carried out depends upon the seriousness of the consequences of failure of that function.

Criterion 27 - Redundancy of Reactivity Control

This criterion is not clear. It does not state whether the two reactivity control systems (1) should both be capable of both increasing and decreasing reactivity for operation, or (2) should both be capable of fast shutdown, or (3) should one be for fast shutdown and one for holddown. We recommend that the word "shutdown" be substituted for "control" in this criterion. These systems should also meet the requirements of Criteria 28, 29, 30, 31, and 32.

Criteria 28, 29, and 30 taken together indicate that one of the shutdown systems is not required to cope with positive transients and is essentially a method of obtaining reactivity holddown capability. However, reactors that must be shut down rapidly to allow the containment system to function need two separate and fast shutdown systems. A single fast or "primary" shutdown system together with a "holddown", or slow, "secondary" shutdown system is not satisfactory in this case.

Criterion 29 - Reactivity Shutdown Capability

As stated in our comments on Criterion 27, some reactors require a shutdown to allow the containment to function. In such cases, this criterion should require that two shutdown systems be applied. Each such system should be capable of preventing an unacceptable situation.

This criterion carries a reference to shutdown margin that could well be made a separate criterion as the shutdown requirements are a function of the number of rods, reactor operating conditions and function desired (e.g., reduction of nuclear power level or holddown of the subcritical reactor). Although we have not addressed ourselves to these conditions in detail, we believe that a margin much greater than the worth of the most effective control rod is needed for reactors having many rods.

Criterion 30 - Reactivity Holddown Capability

In cases requiring the reactor to be shut down in order to achieve containment, two of these systems should be required. See comments on Criteria 27 and 29.

Criterion 31 - Reactivity Control Systems Malfunction

This criterion should be expanded to include all failures of the plant operating system that are capable of increasing reactivity. In particular this criterion should not be limited to the unplanned withdrawal of only one control rod since a failure of the control rod operating system may not be restricted to the withdrawal of only one rod. All failures that may affect the performance of the control rod operating system must be considered. Of a more general nature, all failures that can introduce reactivity increases must be considered. In addition to control rods, there are coolant temperature changes, and perhaps even void effects that need analysis.

Criterion 33 - Reactor Coolant Pressure Boundary Capability

We agree with the intent of the criterion but it is not clear what is meant by "positive mechanical means" for preventing a rod ejection. A definition is needed.

Section VII - Engineered Safety Features

With the exception of reactor shutdown systems, all other engineered safety features are discussed in this section. These are: emergency power system, emergency core cooling system, containment enclosure system, containment pressure-reducing system (including containment heat removal), and air cleaning systems.

For each of these systems, there should be criteria for design of the system and their components as well as criteria for testing and inspection.

The objective of these criteria would be clearer if each system were treated in separate subsections and the criteria for each were set up in parallel form. Thus, there would be criteria for the inspection and testing of emergency power system (now covered in only Criterion 39) as well as the inspection and testing criteria for the other engineered safety features. Criterion 52, "Containment Heat Removal Systems," would be grouped with Criteria 58-61 with which it is generally associated. Such a rearrangement raises questions on other points of apparent inconsistancy, e.g., Criterion 60 is seen to be but a special case of Criterion 61, etc.

Criterion 37 - Engineered Safety Features Basis for Design

Again a definition of engineered safety features is necessary. For example, if the scram must work in order that the containment not be overstressed, then the scram system must be considered part of an engineered safety feature.

Criterion 38 - **Reliability and Testability of Engineered Safety Features**

We agree with this criterion. However, its title and inclusion in Section VII, both of which pertain only to engineered safety features, does not reflect its more general applications which include "inherent" as well as "engineered safety features". It would more appropriately be included in Section I.

Criterion 39 - Emergency Power for Engineered Safety Features

A difficult point in the application of this criterion is that of redundancy in the offsite power system. For example, a plant failure that results in shutting off the electric generator driven by the reactor could produce the loss of all offsite power. The probability of this consequential loss of offsite power varies widely as a result of changes in the power system and of variations in power system load. As a result of this wide variation in the reliability of offsite power, we recommend that this criterion require that redundant and independent onsite power system be required such that onsite power alone be capable of supplying the needs of the engineered safety features after a failure of a single active component in the onsite power system. We do not believe that the offsite power is really independent of the power from a main generator operated from the reactor to be safeguarded.

Criterion 40 - Missile Protection

Analysis shall be made to show that fragments and components that could be ejected from highly pressurized system's rotating equipment would not impair the function of an engineered safety feature. Typical missiles requiring analyses are such items as primary system valves, flanges, instrumentation, etc. When rotating equipment is not completely contained, such as in a concrete vault, a missile map should be provided for rotating equipment (e.g., main turbines, pumps, etc.)

Criterion 41 - Engineered Safety Features Performance Capability

We agree with this criterion as far as it goes. In particular the detailed requirements for the emergency core cooling system as contained in Criterion 44 illustrate the desired amplification (but for that system only). Thus, it could be generalized and added to Criterion 41 as follows: "The performance of each engineered safety feature shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident."

Criterion 42 - Engineered Safety Features Components Capability

We see no need to limit this criterion to the loss-of-coolant accident and suggest that . . . "by the effects of a loss-of-coolant accident" be changed to read "the effects of the accident for which the function is required."

Criterion 43 - Accident Aggravation Prevention

It is not obvious what purpose this criterion is intended to serve. If something specific is in mind here it should be stated, i.e., are we worried about the core becoming critical again, or inducing a thermal shock, etc. Perhaps this should not even appear here but be in the general discussion.

Criterion 44 - Emergency Core Cooling Systems Capability

As noted in the discussion on Criterion 41, we would restrict this criterion to the first two sentences (having already included the remainder of this criterion as a general requirement in Criterion 41). However, as we interpret the intent of these sentences, each of the two emergency cooling systems should cover the whole range of pipe break conditions up to the maximum. To make this point clearer, it might be better to rephrase the second sentence defining the cooling system requirements as follows: "For each size break in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe, at least two emergency core cooling systems, preferably of different design principles and each with a capability for accomplishing abundant emergency core cooling, shall be provided."

Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling Systems

We agree with the intent of this criterion and suggest that in addition to "the transfer to alternate power sources" the operation of the reactivity control system (which must shutdown the reactor and then provide holddown in the cold condition after the loss-of-coolant accident) should be mentioned.

Criterion 49 - Containment Design Basis

We agree with the intent of this criterion but feel that the following need some elaboration:

Line 10: "Considerable Margin" should be defined in some manner.

Line 13: What degree of failure of the emergency core cooling system is assumed?

Criterion 50 - NDT Requirement for Containment Material

This criteria needs further clarification. The temperature of the steel members in question under normal operating and testing conditions should be defined, i.e., the temperature of the component when the ambient temperature is at its lowest recorded (or perhaps expected) value. Furthermore, the requirement of NDT + 30° F has no meaning in the eyes of the stress analyst although it has found some usage. This temperature is half way between NDT and FTE and unless there is adequate justification of which we are unaware, we recommend using NDT + 60° F which defines the transition, e.g., temperature at which cracks won't propagate at stresses less than yield.

Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment

The intent of this criterion is not clear. It would appear that Criterion 53 which requires redundant valving would also cover reactor containment coolant boundaries outside containment. If, however, it is intended to require extensions of the containment, it should be specifically stated. In any event . . . delete "appropriate" and "as necessary" in lines 4 and 5 and the entire last sentence which begins, "Determination of . . .". These words do not materially contribute to the sense of the statement of the criterion and therefore should be omitted.

Criteria 54, 55, and 56 - Containment Leakage Rate Testing, Containment Periodic Leakage Rate Testing, and Provisions for Testing of Penetrations

Following the words "design pressure" it is suggested that "defined by Criterion 49" be inserted.

Criterion 56

This criterion is not sufficiently inclusive. The types of penetrations which should be tested should NOT be limited to the two that are mentioned, but for instance should also include electrical penetrations and piping penetrations that do not require expansion joints. The penetration testing is usually done at greater than design pressure.

Criterion 66 - Prevention of Fuel Storage Criticality

We do not understand the implication of "or processes" at the end of the first sentence, nor do we believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. Hence, the last sentence of this criterion should be changed to read as follows: "Such means as geometrically safe configuations shall be used to insure that criticality cannot occur."

Criterion 67 - Fuel and Waste Storage Decay Heat

To the extent that removal of decay heat is a function necessary to prevent escape of fission products, decay heat removal systems should be designed to the same requirements for redundancy, inspectability, and testability as engineered safety features on reactors. This should include facilities for supplying additional coolant fluid in the event of accidental loss.

CONTENTION TC-2: EXHIBIT 14

Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), enclosing General Design Criteria for Nuclear Power Units (July 15, 1969) (relevant excerpts)

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 23, 1969

Dr. Stephen H. Hanauer, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Hanauer:

Enclosed are 18 copies of:

1. "General Design Criteria for Nuclear Power Units" revision dated July 15, 1969, which reflects the comments made by the ACRS Subcommittee at our meeting July 9, 1969, and

2. A "Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969)."

Regarding the differences between the published and revised criteria, please note that the revised criteria:

a. Reflect comments received from industry on the published criteria and developments that have occurred since their release. In addition, they reflect comments received from the ACRS and the regulatory staff on interim drafts.

b. Establish "minimum requirements" for water-cooled reactors, whereas the published criteria were "guidance" for all reactors.

c. Are arranged in six sections, include definitions, and are not categorized (Category A or Category B).

d. Do not include the term "engineered safety features." The requirements in the published criteria for "engineered safety features" have been incorporated in the revised criteria by including the requirements in the criteria for individual systems.

e. Include criteria which do not have direct counterparts in the published criteria; these are located in the back of Enclosure 2.

ACRS review is requested as soon as possible.

Sincerely,

Edson G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
As stated

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER UNITS

July 15, 1969

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER UNITS

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INTRODUCTION

Pursuant to the provisions of § 50.34, applications for construction permits 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 previously approved for construction 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 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 or accidents which could cause undue risk to the health and safety of the public. There will be some nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be established 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 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 reactor and associated equipment necessary for electrical power generation and 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.

REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary means all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, within the following systems or portions of systems of pressurized and boiling water-cooled nuclear power units:

(a) The reactor coolant system. For a nuclear power unit of the boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valves capable of external actuation in the main steam and feed-water lines, and the reactor safety and relief valves.

(b) Portions of associated auxiliary systems connected to the reactor coolant system. For piping of these systems which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation. For piping of these systems which contains two valves both of which are normally closed during normal reactor operation, the boundary extends to and includes the second of these two valves (the second of which must be capable of external actuation), whether or not the system piping penetrates primary reactor containment.

(c) Portions of the emergency core cooling system connected to the reactor coolant system. For piping of this system which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside containment capable of external actuation. For piping of this system which does not penetrate primary reactor containment, the boundary extends to and includes the second of two valves normally closed during normal reactor operation.

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 within 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 a loss of capability of a structure, system, or component to perform its intended functions. Multiple failures resulting from a single occurrence are considered to be a single failure.

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 geometrically safe configurations.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Instrumentation shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of decay 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 unit environs for radioactivity that may be released from normal operations, from anticipated operational occurrences, and from postulated accidents.

CONTENTION TC-2: EXHIBIT 15

Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970), and enclosed letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970)

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 12 1970

Harold L. Price, Director of Regulation
Clifford K. Beck, Deputy Director of Regulation
Marvin M. Mann, Assistant Director of Regulation for Reactors
C. L. Henderson, Assistant Director of Regulation for Administration
S. H. Hanauer, Technical Advisor to the Director of Regulation
L. D. Low, Director, Division of Compliance
P. A. Morris, Director, Division of Reactor Licensing

REVISED GENERAL DESIGN CRITERIA

My memorandum of September 24, 1970, to Harold L. Price forwarded the latest revision of the General Design Criteria for your comments. Additions and changes to the June 4 version of the criteria were annotated.

Enclosed is a letter and enclosures which provide the AIF comments of the June 4 version of the criteria. Please note that the major Forum comments are discussed in the third enclosure to its October 6 letter. The revised criteria forwarded by my memorandum of September 24 appear to satisfy all of these major comments.

Please provide your comments on the revised criteria by Monday, October 19, so that review by the ACRS and final issuance of the criteria can be expedited.

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
AIF Letter dated October 6, 1970,
to Edson G. Case w/encls
(except second enclosure)

cc: G. A. Arlotto, DRS

ATOMIC INDUSTRIAL FORUM INC.

475 PARK AVENUE SOUTH · NEW YORK, N. Y. 10016 · (212) 725-8300

October 6, 1970

Mr. Edson G. Case, Director
Division of Reactor Standards
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Ed:

The purpose of this letter and the enclosed material is to provide you with a commentary, developed by an ad hoc group convened under the aegis of the Forum's Committee on Reactor Safety, on the AEC-proposed "General Design Criteria for Nuclear Power Plants," as set forth in the AEC draft of June 4, 1970.

This commentary has been developed by, and represents the consensus view of, the following industry representatives, who have had an opportunity to participate either in redrafting and modifying the criteria or reviewing the same:

Robert D. Allen (Chairman) - Bechtel Corp.
Edwin A. Wiggin (Secretary) - Atomic Industrial Forum

Rennie Anderson - Combustion Engineering, Inc.
William Bley - Stone & Webster Engineering Corp.
Henry E. Bliss - Commonwealth Edison Co.
A. Philip Bray - General Electric Co.
Allan R. Collier - Westinghouse Electric Corp.
Walter D. Gilbert - General Electric Co.
Gilbert S. Keeley - Consumers Power Co.
Douglas V. Kelly - Pacific Gas & Electric Co.
William J. L. Kennedy - Stone & Webster Engineering Corp.
William Little - Babcock & Wilcox Co.
Lawrence E. Minnick - Yankee Atomic Electric Co.
James S. Moore - Westinghouse Electric Corp.
John N. Noble - Stone & Webster Engineering Corp.
Harold Oslick - Ebasco Services, Inc.
Warren H. Owen - Duke Power Co.

Rec'd Off. Dir. of Reg.
Date 10/12/70
Time 1...

Richard F. Ranellone - General Electric Co.
William Smith - Babcock & Wilcox Co.
James E. Tribble - Yankee Atomic Electric Co.
Michael F. Valerino - Combustion Engineering, Inc.
Robert E. Wascher - Babcock & Wilcox Co.
John M. West - Combustion Engineering, Inc.
Robert A. Wiesemann - Westinghouse Electric Corp.

The enclosed material, which in its entirety comprises our commentary, includes the following five items:

1. A marked up version of the AEC draft of June 4 indicating the changes we believe should be incorporated prior to publication of the criteria.

2. A retyped version of the AEC draft of June 4 incorporating the changes referred to above.

3. A discussion of the major changes recommended. Our consensus agreement with the criteria as modified is dependent upon their acceptance.

4. An explanation of certain detailed changes which we believe to be both necessary and desirable if the criteria are to prove of maximum usefulness to the AEC and the industry. Omitted from this listing are minor changes, for the most part self-explanatory, which have been suggested in the interest of enhancing the clarity of certain criteria but which do not alter either their scope or intent.

5. An excerpt which we believe should be incorporated in the Statement of Considerations at the time the criteria are published.

We wish to emphasize the importance attached to the concerns underlying the major changes recommended. We very much hope that these concerns can be accommodated by adoption of the recommended changes or in some other equally satisfactory manner.

Submission of this consensus commentary is not intended to preclude the subsequent submission of individual comments by those named above or by other industry representatives, once the criteria have been published. Conversely, it is not expected that the group named above or the Forum Committee on Reactor Safety would wish to offer further comments if the recommendations set forth in this commentary are adopted.

Please let us know if you desire further clarification of these comments. Also, should you wish further elaboration of the comments, we would be pleased to convene a representative group of those named above to meet with you and your associates.

We appreciate the opportunity to comment on this important document.

Sincerely,

Ed

Edwin A. Wiggin

EAW:erk
Enc.

DRAF

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

June 4, 1970

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. 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 provide guidance in establishing the principal design criteria for such other types of nuclear power 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.

Accordingly, these General Design Criteria are intended to reflect current licensing review practice.

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 identified and satisfied by the design in the interest of public safety. It is expected that additional or different criteria ~~will~~ may 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.

A Insert (A)-see next page

The requirements of these General Design Criteria shall be supplemented or modified as necessary to cope with the existence or consequences of a previously unidentified physical condition important to safety. The effective date for the application of industry code and standards shall be as specified in Title 10 of the Code of Federal Regulations.

insert (i)

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, certain of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined so that they can be generalized as criteria. For these reasons it is expected that the criter a will be augmented and revised from time to time as important new or charged requirements such as these are identified and developed.

DEFINITIONS AND EXPLANATIONS

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 system used for normal reactor coolant makeup, ~~system~~ from ~~any size~~ breaks 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.[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. 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~~ selected passive components (assuming active components function properly), results in a

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 as a general design criterion.

loss of the capability of the system to perform its safety functions.[2] ~~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.

2 Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a mechanical system should be considered in designing the system against a single failure are under development as a general design criterion.

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, if required by unusual design or site characteristics, 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, as defined in 10 CFR Part 50, Appendix B, 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 _ - 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 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 and testing, and postulated accidents. These structures, systems, and components shall be

to the extent necessary
appropriately protected/~~against~~ dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from/ the effects of events and conditions ~~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 damage ~~design~~ limits are not exceeded during 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.

CONTROL

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.~~ controlled.

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 onsite and offsite power systems shall each ~~The safety function for each system alone shall be~~ ~~to~~ provide sufficient capacity and capability to assure that (1) specified acceptable fuel damage ~~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.

Element of system important to safety

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.

Electrical power from the transmission network to the switchyard shall be supplied ~~Two physically independent transmission lines, each with the capability~~ by two transmission lines designed and located so as to suitably minimize the likelihood of ~~of supplying electrical power from the transmission network to the switch-~~ their simultaneous failure under operating, accident and expected environmental conditions. ~~yard, and two physically independent circuits from the switchyard to the~~ Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. ~~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~~ all onsite electrical power sources, in the absence of a loss-of-coolant accident, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. Assuming a loss-of-coolant accident, One of these circuits from the switchyard to the onsite electrical power distribution system shall ~~shall be designed to~~ be available in sufficient time ~~immediately following a loss-of-coolant accident~~ to assure that core cooling, containment integrity, and other ~~vital safety~~ functions important to safety are maintained.

Provisions shall be included to minimize the probability of losing electrical power from the offsite electrical power system sources ~~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 important to ~~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 emergency 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, although not necessarily while the plant is at power, the full operational sequence that brings the systems into operation, including operation of the initiation logic ~~protection system~~, and the required transfer of power among the nuclear power unit, the offsite power system, and the onsite emergency 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 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/damage ~~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. 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.

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 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, 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. ~~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 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 or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems, ~~preferably~~ of different design principles shall be provided. ~~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 failure of the highest worth rod to insert, ~~appropriate margin for malfunctions such as stuck rods~~, specified acceptable fuel damage ~~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, including consideration of any rods failing to insert as a consequence of the accident.

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. 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 in accordance with applicable industry codes. ~~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 stressed 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 in accordance with applicable industry codes (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, suitable interconnections, and leak detection and isolation capabilities shall be provide to assure that for either onsite or ~~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 a ~~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, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for either onsite or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 36 - INSPECTION AND PRESSURE TESTING 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 components ~~areas and features, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping,~~ to assure their structural and leaktight integrity as a measure of 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 which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves,~~ and (2) to the extent practical, 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 initiation logic ~~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 a ~~any~~ loss-of-coolant accident and maintain them at acceptable ~~low~~ levels.

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

CRITERION 39 - INSPECTION AND PRESSURE TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM ~~COMPONENTS~~

~~Components of~~ The containment heat removal system shall be designed, insofar as practical, to permit periodic inspection and appropriate pressure testing of important components ~~areas and features, such as the torus, sumps, spray nozzles, and piping~~ to assure their structural and leaktight integrity ~~and~~ as a measure of 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 which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves~~ and (2) to the extent practical 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 initiation logic ~~protection system~~, the transfer between normal and emergency power sources, and operation of the associated cooling water system.

CRITERION 41 - CONTROL OF 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 limit ~~reduce~~, consistent with the functioning of other associated systems, the release ~~concentration and quantity~~ of fission products ~~released~~ to the environment following postulated accidents, such that acceptable limits are not exceeded, 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, suitable interconnections, and leak detection and isolation capabilities shall be provided to assure that for either onsite ~~and for~~ or offsite electrical power system operation its safety function can be accomplished assuming a single failure.

CRITERION 42 - INSPECTION AND PRESSURE TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS ~~COMPONENTS~~

~~Components of~~ The containment atmosphere cleanup systems shall be designed, insofar as practical, to permit periodic inspection and appropriate pressure testing of important components ~~areas and features such as filter frames, ducts, and piping~~ to assure their structural and leaktight integrity as a measure of ~~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 which will provide a measure of (1) the operability and performance of the active components of the systems ~~such as fans, filters, dampers, pumps and valves~~ and (2) to the extent practical, 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 initiation logic, ~~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 or ~~and~~ accident conditions.

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

CRITERION 45 - INSPECTION AND PRESSURE TESTING OF COOLING WATER SYSTEM ~~COMPONENTS~~

~~Components of t~~The cooling water system shall be designed insofar as practical to permit periodic inspection and appropriate pressure testing of important components ~~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 which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves,~~ and (2) to the extent practical 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 initiation logic ~~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 allowable ~~design~~ leakage rate, ~~and, with sufficient margin,~~ the calculated pressure and temperature conditions resulting from a ~~any~~ loss-of-coolant accident. The design ~~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 pressures up to and, if necessary, including the ~~containment~~ design pressure.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

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

INSERT (2) - see next page

~~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.~~

INSERT (2)

CRITERION 54 PROVISIONS FOR CONTAINMENT ISOLATION

Piping which penetrates the containment must be provided with two isolation barriers; one barrier must be located outside the containment and one must be inside the containment, unless it can be demonstrated that the design is acceptable on some other defined basis.

The definition of an isolation barrier is either a suitably designed closed system trip valve, check valve or a manually closed valve under administrative control.

Using this definition four general classifications are derived:

1. Two closed systems - one inside, one outside, no isolation valves required.
2. No closed systems - one valve inside and one valve outside required.
3. Closed system inside - no valve inside, valve required outside.
4. Closed system outside - no valve outside, valve required inside.

NOTE 1: The same criteria apply to lines which are used after an accident except that manual isolation is acceptable and in the case of instrument lines, a check valve or manual valve inside or outside containment is acceptable.

NOTE 2: An isolation valve outside containment shall be located as close to to the 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 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 o 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~~ the handling and release of
~~control over~~ radioactive materials in gaseous and liquid effluents and in solid wastes produced during normal reactor operation, including anticipated operational occurrences, within acceptable limits. 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 RADIOACTIVE WASTE SYSTEMS ~~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 provided with ~~designed~~ (1) with a capability to permit inspection and testing of ~~important areas and features of the~~ components important to safety ~~of these systems~~, (2) with suitable shielding for radiation protection, (3) ~~with~~ appropriate ~~containment~~, confinement, and filtering systems, and (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 designed ~~(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

Instrumentation ~~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 alarm any. ~~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

____

W. B. McCool
Secretary

A Discussion of Major Changes Recommended

There are a number of criteria which as drafted cannot be accepted by the industry for one or more of the following reasons: (1) it represents an unnecessary and unjustified escalation of licensing requirements, (2) there is no clear or common understanding on the part of the AEC and the licensee as to what it would take to meet the requirement, and (3) it is premature to attempt to incorporate the requirement into general design criteria inasmuch as the technical rationale for the requirement has not been fully developed.

Loss-of-Coolant Accident

The definition of the loss-of-coolant accident as set forth in the AEC draft of June 4 clearly represents an escalation of licensing requirements inasmuch as it refers to "any size break" in the "pressure vessels, pumps, and valves connected to the reactor pressure vessel" as well as to a break in the piping. These additional breaks should not be postulated by license reviewers and certainly should not be incorporated into general design criteria in the absence of a realistic technical rationale, the basis for which can be developed only through further study. That study is now being undertaken by an ACRS subcommittee and by an ad hoc Forum group.

Single Failure

As the definition of "single failure" appears in the AEC draft of June 4, it postulates the failure of passive components in both mechanical and electrical systems. Although current licensing review practice assumes the failure of passive components in electrical systems, the extension of the general concept to mechanical systems represents an escalation of licensing requirements for which no technical rationale has been developed. Further, the definition leaves open ended the number and type of mechanical systems to which it could be applied. Indeed, an undisciplined application of the definition would presumably lead to postulating such failures as to make it impossible to design operable systems. Clearly, a single failure concept which would permit the indiscriminate application of postulated failures of passive components in mechanical systems should not be incorporated into general design criteria.

Industrial Sabotage

The AEC draft of June 4 includes as Criterion 5 "Protection Against Industrial Sabotage" which reads" "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.'

Policy considerations involved in the proposed requirement are of such significance that a direct discussion of top utility management personnel with members of the Commission would appear to be prerequisite to resolution of the issues that would be raised in implementing the proposed criterion.

Transmission of Offsite Electrical Power

Criterion 17, "Electrical Power Systems," as it appears in the June 4 draft, includes the requirement: "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."

A literal interpretation of this requirement would call for two transmission lines mounted on different sets of towers located on different rights-of-way. Not only is this an unwarranted escalation of licensing requirements, but for many sites the requirement would neither be desirable nor possible to meet. Further, such a requirement would be contradictory in many instances with requirements being imposed on licensees by environmental considerations.

License applicants should be permitted the option of satisfying the integrity of emergency offsite electrical power service by means other than would be permitted by the criterion as now drafted.

Systematic, Nonrandom, Concurrent Failures of Redundant Elements

Criteria 22, 24 and 29, as set forth in the AEC draft of June 4, all deal with protection and reactivity control systems and all postulate "systematic, nonrandom, concurrent failures of redundant elements." This postulated failure mode is not acceptable to the industry for the following reasons: (1) there is no indication of what requirements are involved, (2) it would provide a "hunting license" for an undisciplined imposition of requirements, (3) there is no logical basis for limiting the concept to protection and reactivity control systems, and (4) the reactor systems suppliers are only now in the early stages of studies which the AEC regulatory staff has asked them to undertake in this area.

Until such time as the requirements which would be imposed by this postulated failure mode can be clearly defined and supported by sound technical rationale, they should not be incorporated into general design criteria.

Containment Isolation

Criterion 54 through Criterion 57, as set forth in the AEC draft of June 4, provide a number of requirements dealing with containment isolation. As drafted, some of these requirements are difficult to interpret and appear to represent an escalation of current licensing practice. Informal discussions with the AEC regulatory staff have not proved successful in developing a mutually satisfactory format for these criteria.

CONTENTION TC-2: EXHIBIT 16

Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971)

Act of February 2, 1903, as amended the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, the reference to the State of Ohio in the introductory portion of paragraph (e) and paragraph (e)(9) relating to the State of Ohio are deleted.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended.)

Effective date. The foregoing amendment shall become effective upon issuance.

The amendment excludes a portion of Clinton County, Ohio, from the areas quarantined because of hog cholera. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not apply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the excluded area. No areas in Ohio remain under the quarantine.

The amendment relieves certain restrictions presently imposed but no longer deemed necessary to prevent the spread of hog cholera and must be made effective immediately to be of maximum benefit to affected persons. It does not appear that public participation in this rule making proceeding would make additional information available to this Department. Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and unnecessary, and good cause is found for making it effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 16th day of February 1971.

F. J. MULHERN,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2380 Filed 2-19-71;8:49 am]

[Docket No. 71-520]

PART 76—HOG CHOLERA AND OTHER COMMUNICABLE SWINE DISEASES

Areas Quarantined

Pursuant to provisions of the Act of May 29, 1884, as amended, the Act of February 2, 1903, as amended, the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, in paragraph (e)(13) relating to the State of Texas, subdivision (xvi) relating to Smith County is deleted, and new subdivisions (xxii) and (xxiii) relating to Bexar County are added to read:

(13) *Texas.* * * *

(xxii) That portion of Bexar County bounded by a line beginning at the junction of Interstate Highway 410 and Farm-to-Market Road 78; thence, following Farm-to-Market Road 78 in a northeasterly direction to Farm-to-Market Road 1518; thence, following Farm-to-Market Road 1518 in a southeasterly and then southwesterly direction to U.S. Highway 87; thence, following U.S. Highway 87 in a northwesterly direction to Interstate Highway 410; thence, following Interstate Highway 410 in a northwesterly direction to its junction with Farm-to-Market Road 78.

(xxiii) That portion of Bexar County bounded by a line beginning at the junction of the Bexar-Medina County line and State Highway 16; thence, following State Highway 16 in a southeasterly direction to Farm-to-Market Road 471; thence, following Farm-to-Market Road 471 in a southwesterly and then northwesterly direction to Farm-to-Market Road 1957; thence, following Farm-to-Market Road 1957 in a southeasterly and then southwesterly direction to the Bexar-Medina County line; thence, following the Bexar-Medina County line in a northerly direction to its junction with State Highway 16.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended)

Effective date. The foregoing amendments shall become effective upon issuance.

The amendments quarantine portions of Bexar County, Tex., because of the existence of hog cholera. This action is deemed necessary to prevent further spread of the disease. The restrictions pertaining to the interstate movement of swine and swine products from or through-quarantined areas as contained in 9 CFR Part 76, as amended, will apply to the quarantined portions of such county.

The amendments also exclude a portion of Smith County, Tex., from the areas quarantined because of hog cholera. No areas in Smith County, Tex., remain under the quarantine. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not comply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the area excluded from quarantine.

Insofar as the amendments impose certain further restrictions necessary to prevent the interstate spread of hog cholera, they must be made effective immediately to accomplish their purpose in the public interest. Insofar as they relieve restrictions, they should be made effective promptly in order to be of maximum benefit to affected persons.

Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendments are impracticable, unnecessary, and contrary to the public interest, and good cause is found for making them effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 16th day of February 1971.

F. J. MULHERN,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2339 Filed 2-19-71;8:46 am]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

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

Section 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 bases and the relation of the design bases 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 to the design bases with adequate margin for safety.

The "General Design Criteria for Nuclear Power Plants" added as Appendix A to Part 50 establish the 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. They also provide guidance in establishing the principal design criteria for other types of nuclear power plants. Principal design criteria established by an applicant and accepted by the Commission will be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission will require assurance that these criteria have been satisfied in the detailed design and construction of the facility and that any changes in such criteria are justified.

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

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants 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 criteria have been reduced to 55 in number, include definitions of important terms, and have been rearranged to increase their usefulness in the licensing process. Additional criteria describing specific requirements on matters covered in more general terms in the previously proposed criteria have been added to the criteria. The Categories A and B used to characterize the amount of information needed in Safety Analysis Reports concerning each criterion have been deleted since additional guidance on the amount and detail of information required to be submitted by applicants for facility licenses at the construction permit stage is now included in § 50.34 of Part 50. The term "engineered safety features" has been eliminated from the revised criteria and the requirements for "engineered safety features" incorporated in the criteria for individual systems.

Further revisions of these General Design Criteria are to be expected. In the course of the development of the revised criteria, important safety considerations were identified, but specific requirements related to some of these considerations have not as yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion in the criteria at this time. 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:

(i) Consideration of the need to design against single failures of passive components in fluid systems important to safety.

(ii) 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.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, DC.

1. Section 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.[1] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units:

* * * * *

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

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

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

INTRODUCTION

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

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

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

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

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

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

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

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

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

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical 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 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.[2]

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

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[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 electrical 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.

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, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

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

Criterion 4—Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Criterion 5—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 during any condition of normal operation, including the effects of anticipated operational occurrences.

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

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

Criterion 13—Instrumentation and control. Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

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

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

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

Criterion 17—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 (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 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.

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. 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 all onsite alternating current power sources and the other offsite electrical 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 electrical power from any of the remaining sources 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 important to 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 components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

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

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

III. Protection and Reactivity Control Systems

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

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

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

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

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

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

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems 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 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.

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 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 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 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.

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.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment 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.

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

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

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

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite 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.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit 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 electrical power system operation (assuming offsite power is not available) and for offsite electrical 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 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 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.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit 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 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 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 surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

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

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit 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.

(Secs. 161, 182, 68 Stat. 948, 953; 42 U.S.C. 2201, 2232)

Dated at Washington, D.C., this 10th day of February 1971.

For the Atomic Energy Commission.

W. B. McCOOL,
Secretary of the Commission.

[FR Doc.71-2370 Filed 2-19-71;8:48 am]

Title 14—AERONAUTICS AND SPACE

Chapter I—Federal Aviation Administration, Department of Transportation

[Docket No. 71-EA-13; Amdt. 39-1155]

PART 39—AIRWORTHINESS DIRECTIVES

American Aviation Corp.

The Federal Aviation Administration is amending § 39.13 of Part 39 of the Federal Aviation Regulations so as to issue an airworthiness directive applicable to

CONTENTION TC-2: EXHIBIT 17

Letter from Donna B. Alexander, CP&L, to U.S. NRC (October 15, 199), enclosing letter from Scott H. Pellet, Holtec International, to Steven Edwards, CP&L (October 11, 1999)

CP&L

OCT 15 1999

Carolina Power & Light Company
Harris Nuclear Plant
PO Box 165
New Hill NC 27562

SERIAL: HNP-99-156

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
SUPPLEMENTAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE HNP SPENT FUEL POOLS 'C' AND 'D' IN SERVICE

Dear Sir or Madam:

Enclosure 8 of the HNP license amendment request (ref. SERIAL: HNP-98-188, dated December 23, 1998) provided a detailed Alternative Plan for demonstrating compliance with ASME Boiler & Pressure Vessel Code requirements for spent fuel pool cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i). By letter dated March 24, 1999, the NRC issued a request for additional information (RAI) related to the Harris Nuclear Plant (HNP) license amendment request to place spent fuel pools C and D in service. The March 24, 1999 RAI included a request to identify each of the embedded field welds within the scope of the Alternative Plan. The HNP response (ref. SERIAL: HNP-99-069, dated April 30, 1999) provided a field weld matrix which identified the field welds to be inspected by using a high resolution remote video camera. The sample size was selected based on a feasibility walkdown with the camera vendor. CP&L has continued, however, to investigate alternative inspection methods with other vendors. Through these efforts with another vendor, CP&L has successfully performed a remote camera inspection of all 15 embedded field welds included within the scope of the Alternative Plan. In the course of the inspection, two field welds (2-SF-1-FW-3 and 2-SF-1-FW-6) which were not embedded in concrete, but within the scope of the Alternative Plan, were cut out to facilitate removal of piping to provide access for the camera inspections. An updated field weld matrix will be provided to reflect the removal of these two welds and the inspection of all 15 embedded field welds.

In addition, by letter dated April 29, 1999, the NRC issued an RAI related to the criticality control provisions in the HNP license amendment request. Item 1 of this RAI requested information regarding a postulated fresh fuel assembly misloading event. As a supplement to our June 14, 1999 response (ref. SERIAL: HNP-99-094) to requested item 1 of the RAI, we had our vendor, Holtec International, perform additional fuel assembly misloading analyses. The results of these analyses are included as an Enclosure to this letter. These analyses demonstrate that criticality will not occur as a result of the postulated misloading of a fresh fuel assembly in the spent fuel storage racks for HNP pools C and D.

This information is provided as a supplement to our December 23, 1998 license amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

J.H. Eads for DBA

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosure:

c: (all w/ Enclosure)

Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator - Region II

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bc: (all w/ Enclosure)

Mr. K. B. Altman
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October 11, 1999

Mr. Steven Edwards
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References: Holtec Project 70324
CP&L Contract XTA7000024

Subject: Additional Criticality Analysis Results

Dear Mr. Edwards,

Per your request, and in support of the recent NRC RAIs pertaining to the criticality evaluations performed for fuel storage in pools C and D, we have performed additional analyses.

RAI #1 from the NRC stated that an evaluation of a fuel assembly misloading event should be analyzed. Holtec's previous response drew upon earlier spent fuel rack evaluations and stated that the k_{eff} would remain below 0.95 with a minimum of 400 ppm soluble boron in the pool.

As a supplement to this response, Holtec International has performed additional analyses for the Harris Spent Fuel Pools C and D to determine the amount of soluble boron required to maintain k_{inf} below 0.95 with a misloaded fresh PWR fuel assembly. The results of this analysis are summarized here.

The inadvertent misloading of a fresh PWR fuel assembly into Harris Pools C and D was analyzed using MCNP-4A and CASMO-3. A delta-k_{inf} for the misloading event was calculated using MCNP and this delta-k_{inf} was applied to the maximum k_{inf} in the licensing amendment report (LAR) to determine the maximum k_{inf} under the misloading scenario. This accident scenario consisted of a single 5 wt.% ^{235}U PWR fresh fuel assembly misloaded into the PWR racks surrounded by fuel of maximum reactivity as determined by the burnup and enrichment curve in the LAR. The k_{inf} for the PWR racks with the misloaded fresh assembly, without taking credit for soluble boron, was determined to be 0.9916 with a 95%/95% confidence level.

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A second scenario was also analyzed in which the fresh 5 wt.% ^{235}U PWR fuel assembly was placed in a PWR storage cell adjacent to the BWR storage racks. The PWR and BWR racks were filled with fuel of maximum permissible reactivity. The k_{inf} for this scenario with the misloaded fresh 5 wt.% ^{235}U PWR fuel assembly, without taking credit for soluble boron, was 0.9932 with a 95%/95% confidence level.

These results clearly demonstrate that the spent fuel pool will remain subcritical even with a fresh 5 wt.% ^{235}U PWR fuel assembly misloaded in the PWR racks.

The April 1978 NRC letter to All Power Reactor Licensees states that "The double contingency principle of ANSI N-16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident." Consistent with this approach, credit for soluble boron, which is normally in the spent fuel pool, was taken when the misloaded fresh 5 wt.% ^{235}U PWR fuel was analyzed. It was determined that the maximum k_{inf} for the misloading accident is 0.9352 with 400 ppm soluble boron in the spent fuel pool water. Therefore, the minimum amount of soluble boron required to maintain k_{inf} less than the regulatory limit of 0.95 under all postulated abnormal and accident conditions is 400 ppm.

Additional calculations were also performed to determine the k_{inf} for the misloading accident with 1000 and 2000 ppm soluble boron in the spent fuel pool water. The maximum k_{inf} was calculated to be 0.8671 and 0.7783 for the 1000 and 2000 ppm respectively. These results demonstrate that there is considerable un-credited margin in the criticality analysis of Harris Spent Fuel Pools C and D.

If you have any questions please feel free to contact me.

Sincerely,

Scott H. Pellet

Scott H. Pellet
Project Manager

cc: Holtec Engineering File 80964
Holtec Contracts file

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