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50-269 270/287

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SUBJECT:

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FURNISHING IN RE SMALL BREAK ECCS ANALYSIS OF THE 0.04 SQ FT BREAK FOR
SUBJECT FACILITY'S AND ADVISING EMERGENCY CORE COOLING REQUIREMENTS FOR 0.04
FT 2 BREAK AT PUMP DISCHARGE ARE SATISFIED, AND CONCLUDED THAT FACILITY UNITS
CAN CONTINUE TO BE SAFELY OPERATED AT THE RATED POWER

PLANT NAME: OCONEE - UNIT 1
OCONEE - UNIT 2
OCONEE - UNIT 3

REVIEWER INITIAL: XJM
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DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

April 14, 1978

TELEPHONE: AREA 704
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DISTRIBUTION
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Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RE: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

REGULATORY DOCKET FILE COPY

Dear Mr. Case:

As you are aware, the small break ECCS analysis of the 0.04 sq. ft. break for Oconee class plants was performed with the assumption that this break occurs at the pump suction of the reactor coolant piping, as reported in BAW-10103, Revision 3, and in BAW-10052. Recently, it has been identified that a 0.04 sq. ft. break occurring at the pump discharge would be more limiting if it is assumed that only one high pressure injection (HPI) pump is available for core cooling. B&W has now performed an evaluation of this break at the pump discharge for Oconee 1, 2, and 3 by considering that two HPI pumps are available at 25 minutes. The results indicate that the core remains covered by fluid at all times without any cladding temperature excursion.

Each of the Oconee units has three HPI pumps normally available, and all three pumps are automatically started upon Engineered Safeguards signal actuation. The flow discharged from these pumps is injected into the reactor coolant system through two independent injection lines, each branching into two smaller lines, and terminating into the reactor coolant cold leg piping between the pump discharge and the reactor vessel nozzle, as shown in Figures 6-2 and 9-2 of the Oconee FSAR. The two injection paths and two HPI pumps are adequate to provide the necessary flow into the core to maintain the core covered with fluid at all times.

The HPI pumps are powered by three independent 4160V switchgears. If a single failure is assumed to occur, there would still be two HPI pumps available to provide flow to the core. If a single failure were to occur on HPI Pump C, then the normally closed manual valves, HP-116 and HP-117, would have to be opened to ensure that sufficient flow is available through both injection lines. B&W's evaluation shows that at least 25 minutes is available to the operators to accomplish this task, and it is considered an adequate time interval (5 minutes to confirm no flow indication in HPI Line B, 5 minutes transit time, and 10 minutes to open the valves) to accomplish this operation. If, however, the single failure

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Mr. Edson G. Case, Acting Director
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were to occur with Valve HP-26, then it would be the A injection line without the flow. In this case, the operator would have to manually open Valve HP-26, and 25 minutes is more than adequate time to accomplish this task.

The emergency operating procedures for loss of coolant accident conditions will be revised by April 14, 1978, to include sufficient guidance and instructions for the operators to take the above described actions. It should be pointed out that the above manual actions are needed only under one of two independent single failure conditions (failure of HPI Pump C or failure of Valve HP-26), and this situation can be easily recognized from flow, pump status, and/or valve indications.

Also, the manual valves HP-99 and HP-100, which are currently closed during normal operation will now be kept open during normal operation, and the ES valve HP-25, currently kept open, will be kept closed. This change in the valve lineup is being made to assure that adequate suction flow is available for the HPI pumps if a single failure were to occur in Valve HP-24. This valve lineup will be applied at the time of implementing the change to the emergency operating procedure. It is pointed out that the HPI pump suction line will be full of water with the revised valve configuration also.

By April 21, 1978, we will submit a proposed revision to Technical Specification Section 3.3 to require the operability of all three HPI pumps, except as permitted by Specification 3.3.5. All three pumps are currently operable, and we will notify the NRC-ONRR if an HPI pump becomes inoperable for a period exceeding 24 hours.

In summary, the emergency core cooling requirements for the 0.04 ft² break at the pump discharge are satisfied, and it is concluded that Oconee units can continue to be safely operated at the rated power.

Very truly yours,

William O. Parker, Jr.
William O. Parker, Jr. *By [Signature]*

PMA:ge



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

April 14, 1978

✓ Dockets Nos.: 50-269/270/287

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,

A handwritten signature in cursive script, reading "Brian K. Grimes", is written over the typed name.

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

Enclosures:

1. NRC Guidance
2. Notice

OT POSITION FOR REVIEW AND ACCEPTANCE OF
SPENT FUEL STORAGE AND HANDLING APPLICATIONSI. 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 uncertainty 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 uncertainty 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/cm² 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, including all uncertainties, 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 APCS 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 lesser 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 + E¹

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 amended 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, boron, B₄C, 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

SPENT FUEL STORAGE

Intent To Prepare Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel

From the early days of the nuclear power industry in this country, electric utilities planning to construct and operate light water nuclear power reactors contemplated that the used or spent fuel discharged from the reactors would be chemically reprocessed to recover the remaining quantities of fissile and fertile materials (uranium and plutonium), and that the materials so recovered would be recycled back into fresh reactor fuel. It was contemplated by the nuclear industry that spent fuel would be discharged periodically from operating reactors, stored in onsite fuel storage pools for a period of time to permit decay of radioactive materials contained within the fuel and to cool, and periodically shipped offsite for reprocessing. Typically, space was provided in onsite storage pools for about one and one-third nuclear reactor cores. Assuming a four-year reactor fuel reload cycle, such onsite storage pools were planned to hold an average of one year's discharge with sufficient remaining capacity to hold a complete core should unloading of all of the fuel from the reactor be necessary or desirable because of operational difficulties. Under normal operating conditions, an average of five years' discharge could be accommodated before the pools were filled.

Persons planning to conduct commercial reprocessing of spent reactor fuels provided sufficient storage capacity for the spent fuels at their facilities to allow some operational flexibility. Typically, space has been provided or planned for several spent fuel core reloads. Three commercial reprocessing plants have been planned for operation in the United States. The only such plant that has actually operated, Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for extensive alterations and expansion. There is a pending proceeding before the Nuclear Regulatory Commission (Commission) on NFS's application for a permit to construct these alterations and expansion (docket no. 50-201). The second plant, General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois, has never operated and is in a decommissioned condition. The third

plant, Allied General Nuclear Services (AGNS), proposed plant in Barnwell, South Carolina, is under construction and is the subject of pending proceedings before the Commission regarding the continuation, modification or suspension of the construction permit from an environmental protection standpoint, and the possible issuance of an operating license (docket no. 50-332), as well as a related matter (docket no. 70-1729).

On May 8, 1975, the Nuclear Regulatory Commission published a notice in the FEDERAL REGISTER setting forth its provisional views that, subject to consideration of comments, (1) a cost-benefit analysis of alternative safeguards programs should be prepared and set forth in draft and final environmental impact statements before a Commission decision is reached on wide-scale use of mixed oxide (recycle plutonium) fuels in light water nuclear power reactors, (2) there should be no additional licenses granted for use of mixed oxide fuel in light water nuclear power reactors except for experimental purposes, (3) with respect to light water nuclear power reactor fuel cycle activities which depend for their justification on wide-scale use of mixed oxide fuel in light water nuclear power reactors, there should be no additional licenses granted which would foreclose future safeguards options or result in unnecessary "grandfathering" and (4) the granting of licenses would not be precluded for fuel cycle activities for experimental and/or technical feasibility purposes.

In light of the status of the three planned commercial reprocessing plants in the United States, as outlined above, the earliest that spent fuel reprocessing could begin on a commercial basis, if authorized, would be late 1976. This assumes that the pending licensing proceedings are completed and licenses issued by this date. However, the spent fuel pools at a number of reactors may soon be filled, and still other reactors will have their pools filled before the end of 1978. Accordingly, even if limited reprocessing should begin in late 1976, there would still be a shortage in spent fuel storage capacity.

The existing pools at the GE and NFS reprocessing plants have some remaining marginal licensed storage capacity which may be able to accommodate the fuel discharges from some reactors; any increases planned at these plants may not be sufficient for industry in the future. Consequently, there is the possibility of a future shortage in licensed spent fuel capacity regardless of the outcome of the proceedings on the May 8th notice.

The Commission has not promulgated any regulation which specifies a given size for on-site reactor spent fuel pools; however, proposals by reactor licensees to significantly change the manner of spent fuel storage or spent fuel pool size would be subject to licensing review by the Commission. In the event that a particular on-site spent fuel pool should become filled, and no alternative form of spent fuel storage could be found,

the reactor would be eventually forced to shut down and "store" the last spent reactor fuel in the reactor pressure vessel. While no serious adverse consequences to the public health and safety, the common defense and security, or the environment would likely result, the reactor shutdown would, of course, remove the plant from service, and this in turn could adversely affect the electric utility's ability to meet electrical energy needs, or force the utility to operate other plants that are less economical to operate or which have greater environmental impact, and thereby adversely affect the public interest.

There appear to be a number of possible alternatives for increasing spent fuel storage capacity including, among other things, increasing the storage capacity at present reactor sites, and construction of independent spent fuel storage facilities. The shortage of spent fuel storage capacity will occur at individual reactors, and the Commission could adequately address the issues involved on a case-by-case basis within the context of individual licensing reviews. Indeed, the Commission has not, to date, found it necessary, in the discharge of its licensing and related regulatory functions, to develop any overall program of action to deal with the problem. The Commission does, however, have the discretion to deal with issues of this type on a generic basis through the exercise of its rulemaking authority and/or the issuance of a "generic" environmental impact statement. Rulemaking proceedings and/or the issuance of a generic environmental impact statement might, as appropriate, serve as the context for the promulgation of more definitive criteria regarding size and design of spent fuel pools and/or the licensing of independent spent fuel storage facilities, and for consideration of possible revision of the fuel cycle environmental impacts set forth in 10 CFR § 51.20(e) in light of additional spent fuel storage and attendant transportation. Also, the possible implications of increased spent fuel storage on the options available for intermediate and long-term storage of nuclear waste materials could profitably be examined within this context.

One group of interested organizations (Natural Resources Defense Council, Sierra Club, and Businessmen for the Public Interest) has requested the Commission to prepare a generic environmental impact statement on the handling and storage of spent reactor fuel and related matters (letter to L. V. Gossick from Anthony Z. Roisman, dated May 20, 1975, copy on file at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C.).

While the Commission believes, as earlier indicated, that the matter of spent fuel storage capacity can adequately be addressed on a case-by-case basis within the context of individual licensing reviews, it also believes that, from the standpoint of longer range policy, this matter can profitably be examined in a broader context. It views the preparation of a generic environmental impact state-

ment as a suitable vehicle for such an examination. Notice is hereby given that a generic environmental impact statement on the handling and storage of spent light water power reactor fuels will be prepared by the Commission. The statement will focus on the time period between now and the mid 1980's and will address:

(1) The magnitude of the possible shortage of spent fuel storage capacity;
(2) The alternatives for dealing with the problem, including, but not necessarily limited to:

(a) Permitting the expansion of spent fuel storage capacity at power reactors;
(b) Permitting the expansion of spent fuel storage capacity at reprocessing plants;

(c) Licensing of independent spent fuel storage facilities;

(d) Storage of spent fuel from one or more reactors at the storage pools of other reactors;

(e) Ordering that generation of spent fuel (reactor operation) be stopped or restricted;

(3) A cost-benefit analysis of the alternatives listed in (2), along with any other reasonably feasible alternatives, including:

(a) Impacts on public health and safety and the common defense and security;

(b) Environmental, social, and economic costs and benefits;

(c) Commitments of resources;

(d) Implications regarding options available for the intermediate and long-term storage of nuclear waste materials;

(e) Relationship between local short-term uses of the environment and long-term productivity;

(4) The impacts of possible additional transportation of spent fuel that may be required should one or more of the alternatives be adopted;

(5) More definitive standards and criteria to govern the licensing of one or more of the alternatives for dealing with the problem; and

(6) Possible amendments to 10 CFR § 51.20(e).

If appropriate, rulemaking proceedings on items (5) and (6) listed above, or on other issues related to the handling and storage of spent reactor fuel, will be initiated on or about that time of issuance of the draft generic environmental impact statement.

The Commission has also given careful consideration to the question whether licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity, including such actions as the issuance of operating license amendments to permit increases in the storage capacity of reactor spent fuel pools or reprocessing plant spent-fuel storage pools, or the licensing of independent spent fuel storage facilities, should be deferred pending completion of the generic environmental impact statement. Such a deferral was requested in the letter on behalf of Natural Resources Defense Council, Sierra Club, and Businessmen for the Public Interest noted above. In considering this matter, the

Commission had two basic objectives in mind: on the one hand, the generic impact statement should not serve as a justification for a fait accompli; on the other hand, the public interest considerations associated with such a deferral should be carefully weighed. The Commission has concluded that there should be no such general deferral, and that these related licensing actions may continue during the period required for preparation of the generic statement, subject to certain conditions. In reaching this conclusion, the Commission has considered the following specific factors:

(1) It is likely that each individual licensing action of this type would have a utility that is independent of the utility of other licensing actions of this type;

(2) It is not likely that the taking of any particular licensing action of this type during the time frame under consideration would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing action of this type;

(3) It is likely that any environmental impacts associated with any individual licensing action of this type would be such that they could adequately be addressed within the context of the individual license application without overlooking any cumulative environmental impacts;

(4) It is likely that any technical issues that may arise in the course of a review of an individual license application can be resolved within that context; and

(5) A deferral or severe restriction on licensing actions of this type would result in substantial harm to the public interest. As indicated, such a restriction or deferral could result in reactor shutdowns as existing spent fuel pools become filled. It now appears that the spent fuel pools of as many as ten reactors could be filled by mid-1978. These ten reactors represent a total of about 6 million kilowatts of electrical energy generating capacity. The removal of these reactors from service could reduce the utilities' service margins to a point where reliable service would be in jeopardy, or force the utilities to rely more heavily on less economical or more polluting forms of generation that would impose economic penalties on consumers and increase environmental impacts.

The Commission expects that any licensing action intended to ameliorate a possible shortage of spent fuel storage capacity during this interim period would be accompanied by an environmental impact statement (10 CFR § 51.5(a)) or impact appraisal (10 CFR § 51.5(c)) tailored to the facts of the case. Since the Commission's general conclusions with respect to the five factors, as set forth above, may not fit the factual circumstances of particular licensing actions, the five factors will be applied, weighed and balanced within the context of these statements or appraisals in reaching licensing determinations.

Dated at Washington, D.C. this 10th day of September 1975.

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