January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | Docket No. 50-400-LA |
| CAROLINA POWER & LIGHT | ) | |
| COMPANY | ) | ASLBP No. 99-762-02-LA |
| | ) | |
| (Shearon Harris Nuclear Power Plant) | ) | |
| | ) | |

AFFIDAVIT OF LAURENCE I. KOPP IN SUPPORT OF THE NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON TECHNICAL CONTENTION 2

Laurence I. Kopp, being duly sworn, does hereby state as follows:

1. I have been employed by the U.S. Nuclear Regulatory Commission (NRC), and its predecessor, the Atomic Energy Commission (AEC), since 1965. My current position is Senior Reactor Engineer in the Reactor Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation (NRR). My responsibilities include review and evaluation of the criticality aspects of on-site fuel storage at commercial nuclear power reactors. I have a Ph.D. degree in Nuclear Engineering from the University of Maryland, a Master of Science degree in Physics from Stevens Institute of Technology, and a Bachelor of Science degree in Physics from Fairleigh Dickinson University. I have 42 years experience in the nuclear power industry, including 5 years at the Martin-Marrietta Nuclear Division and 2 years at the Westinghouse Astronuclear Division. A statement of my professional qualifications is attached hereto (Exhibit 1).

2. The purpose of this affidavit is to address the Board of Commissioners of Orange County's (BCOC) Contention 2 as set forth in Orange County's Supplemental Petition to Intervene and in the Atomic Safety and Licensing Board (Board) Memorandum and Order of July 12, 1999 (LBP-99-25).

3. In a letter from J. Scaraola to the NRC, dated December 23, 1998 ("Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment Spent Fuel Storage") (Exhibit 2), Carolina Power and Light Company (CP&L) submitted a request to place spent fuel pools C and D at the Shearon Harris Nuclear Power Plant (Harris) in service. Specifically, CP&L proposed to increase the spent fuel storage capacity by adding storage racks to pools C and D.

4. In preparation for this affidavit, I reviewed the criticality aspects of the CP&L application for the proposed license amendment as well as the correspondence and technical documents identified below.

5. BCOC's Contention 2 states:

> Storage of pressurized water reactor ("PWR") spent fuel in pools C and D at the Harris plant, in the manner proposed in CP&L's license amendment application, would violate Criterion 62 of the General Design Criteria ("GDC") set forth in Part 50, Appendix A. GDC 62 requires that: "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." In violation of GDC 62, CP&L proposes to prevent criticality of PWR fuel in pools C and D by employing administrative measures which limit the combination of burnup and enrichment for PWR fuel assemblies that are placed in those pools. This proposed reliance on administrative measures rather than physical systems or processes is inconsistent with GDC 62.

The Board admitted the contention with the following bases:

> Basis 1 - - CP&L's proposed use of credit for burnup to prevent criticality in pools C and D is unlawful because GDC 62 prohibits the use of administrative measures, and the use of credit for burnup is an administrative measure.
>
> Basis 2 - - The use of credit for burnup is proscribed because Regulatory Guide 1.13 requires that criticality not occur without two independent failures, and one failure, misplacement of a fuel assembly, could cause criticality if credit for burnup is used.

BASIS 1

6. My response to Basis 1 of Contention 2 is contained in the following paragraphs.

7. Criticality is the achievement of a self-sustaining nuclear chain reaction. The chain reaction proceeds as atoms of a fissile material absorb slow (thermal) neutrons and split (fission) into new lighter atoms (*i.e.*, fission products) and additional neutrons that, in turn, interact with additional fissile atoms. Neutrons resulting from fission have high energy and are called "fast" neutrons. Fast neutrons are not readily captured in U-235, the fissile material originally present in fresh fuel. Rather, a neutron must lose energy and "slow down," or become "thermalized" (a thermal neutron), in order to be readily captured in U-235 and cause fission.

8. In order for fast neutrons to slow down, they must collide with, and transfer energy to, atoms. This process is called "moderation." A light element (such as hydrogen) is an effective moderator because the mass of its nucleus is on the same order as that of a neutron. Therefore, upon initial collision, the neutron imparts most of its energy to the

hydrogen nucleus and becomes thermalized. Water, with its high hydrogen content, is the moderator in a light water reactor (LWR) such as Harris.

9. After being created through fission, during the process of moderation, and after reaching thermal energy levels, a neutron may undergo several events. It may be absorbed by nonproductive capture in the fuel, the moderator, or the structural materials. It may leak from the reactor system and either be reflected back into the system or be lost. Finally, it may be absorbed by the U-235, cause fission, and produce more fast neutrons.

10. When the process continues on its own, the system of atoms of fissile material is said to be critical. The measure of criticality is the effective neutron multiplication factor, k-effective, or $k_{eff}$. The multiplication factor is the ratio of the rate of neutron production to neutron loss due to fission, nonproductive capture and leakage. Well-developed mathematical models (equations) exist in present-day computer codes and are used to compute $k_{eff}$. Criticality is achieved when $k_{eff}$ is equal to 1.0. When $k_{eff}$ is less than 1.0, the system is subcritical. When $k_{eff}$ is greater than 1.0, the system is supercritical. Criticality can only occur in an array of LWR fuel if sufficient fissile material is available in a near-optimum geometry and a moderator (water) is present. As previously mentioned, no array of LWR fuel can achieve criticality without water moderation present in the array.

11. "Reactivity" is defined as $(k_{eff} - 1)/k_{eff}$. When fuel is irradiated in a reactor as a result of operation and power generation, the reactivity of the fuel decreases. This reduction of reactivity with irradiation is called "burnup." Burnup is caused by the change in fissile content of the fuel (*i.e.*, depletion of U-235 and production of Pu-239 and other fissile actinides), the production of actinide neutron absorbers, and the production of fission

product neutron absorbers. Before each reactor operating cycle, licensees perform reload analyses that predict burnup of each fuel assembly during the cycle. These calculations are confirmed during the cycle by measurements of various operating characteristics, such as boron concentration and power distribution. After every operating cycle (typically 1 to 2 years), approximately 1/3 of the fuel in a reactor is removed because its reactivity is too low to effectively contribute to power generation in the reactor environment. This irradiated (or spent) fuel is generally placed in a spent fuel pool at the reactor site and is replaced in the reactor by fresh (unirradiated) fuel.

12. The NRC regulations (10 CFR Part 50, Appendix A, GDC 62) require that licensees prevent criticality in their spent fuel pools. GDC 62 states that "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by the use of geometrically safe configurations." A proposed version of the GDC was sent to the Commission in a paper dated June 16, 1967 ("Proposed Amendment to 10 CFR 50; General Design Criteria for Nuclear Power Plant Construction Permits," AEC-R2/57) (Exhibit 2A). The AEC first formally published the general design criteria for comment on July 11, 1967 (32 FR 10213, "General Design Criteria for Nuclear Power Plant Construction Permits")(Exhibit 3). At that time, the proposed criterion for prevention of fuel storage criticality was labeled GDC 66, which stated "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." The AEC received only one comment regarding Criterion 66. This comment was received from the Oak Ridge National Laboratory (ORNL) on September 6, 1967 (Letter from W.B. Cottrell to H.L. Price,

"Review of USAEC 'General Design Criteria for Nuclear Power Plant Construction Permits' Federal Register, July 11, 1967," September 6, 1967)(Exhibit 4). Specifically, the ORNL comment on proposed GDC 66 stated that they did not understand the implication of "or processes" at the end of the first sentence, nor did they believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. They suggested that the last sentence of the criterion should read as follows: "Such means as geometrically safe configurations shall be used to insure that criticality cannot occur." The AEC staff considered these comments and decided that it was not necessary to change the phrase "or processes" and, therefore, it was retained. The AEC staff agreed that geometrically safe configurations was the preferable means for preventing criticality. However, procedural controls were not specifically ruled out, as suggested by ORNL. Rather, GDC 66 (renumbered as GDC 62) was revised to state that geometrically safe configurations are the preferable means for preventing criticality in fuel handling and storage ("Status Report on General Design Criteria," memorandum from Harold L. Price to the Chairman and Commissioners, July 6, 1970 (Exhibit 4A); "Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969)(Exhibit 4B)). However, it did not specifically rule out other means.

13. Burnup credit is the practice of accounting for the reduced reactivity of spent fuel due to fissile material decay and fission product buildup described above in evaluating criticality safety. The regulations do not elaborate on how or how much subcriticality should be assured, nor do they prohibit the use of burnup credit for criticality safety. As explained above, burnup of fuel occurs as a natural consequence of the fuel being used in a

reactor. Therefore, fuel burnup is a physical process and credit for burnup to prevent criticality in spent fuel storage pools is permitted under GDC 62.

14. The NRC has established a 5% subcriticality margin for wet storage of spent fuel assemblies to assure that licensees meet the requirements of GDC 62. The NRC staff stated this acceptance criterion for criticality in a generic communication from Brian K. Grimes sent to all power reactor licensees (B.K. Grimes, Letter to All Power Reactor Licensees, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978)(Exhibit 5). This letter states that "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." (Page III-3). This requirement is also stated in Section 4.3.1 of the Standard Technical Specifications for all PWR's ("Standard Technical Specifications Babcock and Wilcox Plants," NUREG-1430, "Standard Technical Specifications for Westinghouse Plants," NUREG-1431, "Standard Technical Specifications for Combustion Engineering Plants," NUREG-1432)(Exhibits 6, 7, and 8, respectively), which state "The spent fuel storage racks are designed and shall be maintained with $k_{eff}$ less than or equal to 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]." The brackets indicate that the reference for a description of the uncertainties is plant-dependent. In the case of Harris, the proper reference is Section 4.3.2.6, pages 4.3.2-19 through 4.3.2-22 of the Final Safety Analysis Report (FSAR) ("Shearon Harris Nuclear Power Plant Final Safety Analysis Report")(Exhibit 9).

15. CP&L proposes to use administrative procedures at Harris to verify that a fuel assembly has achieved the required amount of burnup to be placed in the pool C or D

proposed storage racks. CP&L is currently licensed to store fuel from two other CP&L plants, H. B. Robinson Steam Electric Plant (Robinson), Unit 2, and Brunswick Steam Electric Plant Units 1 and 2 (Brunswick), as well as fuel from the Harris reactor core, in the existing spent fuel pools A and B at Harris. By letter dated June 14, 1999 (Letter from D.B. Alexander (CP&L) to U.S. Nuclear Regulatory Commission, "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," June 14, 1999)(Exhibit 10), CP&L stated that it selects spent fuel assemblies for shipment to Harris from Robinson and Brunswick in accordance with plant procedure NFP-NGGC-0003, (Carolina Power & Light Company, Nuclear Generation Group, Standard Procedure, Volume 99, Book/Part 99, NFP-NGGC-0003, "Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask")(Exhibit 11). The purpose of this procedure is to assure that the selection of spent fuel to be shipped to Harris is acceptable for transportation and storage in the Harris A and B spent fuel pools.

16. CP&L uses a computer program in conjunction with this fuel selection procedure. For candidate assemblies to be shipped, the program retrieves the fuel type, enrichment, and burnup from a database. The fuel type and initial enrichment data for each fuel assembly contained in this database is based on manufacturing records. The burnup data for each fuel assembly included in this database is obtained from the reload core design calculations and confirmed by periodic core monitoring of boron concentration and power distribution. The letter (Exhibit 10) further states that revision to NFP-NGGC-0003 to incorporate the burnup curve proposed as technical specification Figure 5.6.1 (Shearon

Harris Unit 1 Technical Specifications, Section 5.6.1.2. "Fuel Storage - Criticality") (Exhibit 12) to reflect criticality screening requirements for fuel from all three CP&L plants (Robinson, Brunswick, and Harris) to be stored in Harris pools C or D has begun. However, the revision is not yet complete and will be put into production if CP&L's license amendment application to place spent fuel pools C and D in service is approved.

17. Licensees have used administrative procedures to determine the acceptability for essentially all burnup-dependent storage pools since the early 1980's. These procedures generally consist of verification that the licensee has selected a fuel assembly that has achieved the required amount of burnup, based on plant operating records, and the licensee has stored it in the intended position in the spent fuel pool. Section 4.2.1 of American National Standards Institute (ANSI) standard ANSI/ANS-8.1-1983 ("American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 1983)(Exhibit 13) states that nuclear criticality safety may be achieved by controlling one or more parameters of the system within subcritical limits and that control may be exercised administratively through procedures. The NRC endorsed ANSI/ANS-8.1.1983 in revision 2 to Regulatory Guide 3.4 (Regulatory Guide 3.4, Rev. 2, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities," U.S. Nuclear Regulatory Commission, March 1986) (Exhibit 14).

18. In addition, the Code of Federal Regulations, 10 C.F.R. § 50.68, allows the use of administrative controls to prevent inadvertent criticality in fuel handling and storage. The Commission developed 10 C.F.R. § 50.68 to allow holders of a construction permit or

operating license for a nuclear power reactor issued under 10 C.F.R. Part 50 to opt out of the 10 C.F.R. § 70.24 requirement to maintain a criticality accident monitoring system in each area where nuclear fuel is handled, used, or stored, if criticality is precluded in these areas. Specifically, 10 C.F.R. § 50.68(b)(1) allows a licensee to rely upon plant procedures to "prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse conditions feasible by unborated water." (In addition, GDC 62 applies to fuel handling systems, as well as fuel storage systems. While the fuel handling systems may move only one fuel assembly at a time, administrative controls must be used, for example, to prevent temporary storage of multiple assemblies in close proximity.) Section (b)(2) and (b)(3) of 10 C.F.R.§ 50.68 allow licensees to use administrative controls or design features or both to prevent accidental flooding of new fuel racks to preclude criticality. Therefore, the industry uses administrative measures to prevent criticality in fuel storage and the NRC has accepted this practice since the early 1980's. As set forth above, NRC regulations allow the use of administrative controls to prevent criticality of fuel in storage. Further, since human action is necessary to move fuel between the reactor and fuel storage facilities, it is inescapable that administrative controls on fuel movement must be used to ensure that the physical measures for preventing criticality are properly employed. To date, there have been no reported incidents of inadvertent criticality in U.S. spent fuel pools for any reason, including violation of administrative procedures. In fact, there have been no known instances where even the 5% subcriticality margin has not been maintained due to violations of administrative procedures.

19. To date, more than 50 plants have obtained NRC approval for the use of burnup credit for spent fuel storage. I have been the NRC principal criticality reviewer for most of these plants. The NRC first approved burnup credit in spent fuel pool storage analyses in the early 1980's.[1] Licensees have established their ability to predict core burnup behavior over hundreds of reactor years of operation. They have also established the ability to predict isotopic inventories of reprocessed fuel by comparison of calculations of data available from several cores of the Yankee reactor (R.J. Nodvik, "Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel," WCAP-6068, Westinghouse Electric Corporation, March 1966)(Exhibit 15). In view of the above, the NRC has allowed licensees to take credit for burnup in criticality analyses of spent fuel storage pools.

20. In summary, GDC 62 does not prohibit the use of burnup credit nor does it prohibit the use of administrative measures to determine if adequate burnup has been achieved to allow storage in pools C and D.

## BASIS 2

21. My response to Basis 2 is contained in the following paragraphs.

22. Draft Regulatory Guide 1.13 (RG 1.13) (Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," U.S. Nuclear Regulatory Commission, December 1981)(Exhibit 16) recommends that the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely,

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[1]Several plants which were initially approved for burnup credit include Fort Calhoun (1983), St. Lucie 2 (1984), Ginna (1984), Turkey Point 3&4 (1984), and Summer (1984).

independent, and concurrent failures. This additional safety assurance is based on application of the "double contingency principle" as defined in ANSI/ANS-8.1-1983 (Exhibit 13), which was endorsed by the NRC staff in a generic communication from Brian K. Grimes sent to all power reactor licensees on April 14, 1978 (Exhibit 5). More recently, the Commission included similar criteria in 10 C.F.R. § 72.124(a), which requires at least two unlikely, independent, concurrent or sequential events to have occurred before a nuclear criticality accident is possible. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered one unlikely accident condition and a second concurrent independent accident need not be assumed. Alternatively, credit for the presence of soluble boron in PWR pools may be assumed in evaluating other accident conditions such as the misloading of fresh fuel or fuel that has not attained the required minimum burnup into the proposed pool C or D storage racks.

23. The staff considers fuel misplacement in the Harris pool C and D storage racks to be an unlikely event for several reasons. First, proposed technical specification 5.6.1.2 (Exhibit 12) will control fuel storage limitations and selection procedure NFP-NGGC-0003 (Exhibit 11), described above, will control fuel assembly selection. Therefore, both technical specifications and plant procedures would have to be violated for a fuel assembly misplacement to occur. In addition, fresh fuel assemblies have a bright, metallic color and are visually distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding. Finally, the burnup limit curve (Figure 5.6.1) proposed for the Harris technical specifications for safe storage in pools C and D (Exhibit 12) is based on a minimum required burnup. This is a bounding value that results

in just meeting the 5% subcriticality margin in pools C and D. In practice, unless an assembly is prematurely removed from the reactor, permanently discharged fuel assemblies would be expected to exceed these burnup requirements (have a lower reactivity). Such fuel assemblies, therefore, should fall in the acceptable burnup domain of Figure 5.6.1, thereby minimizing the number of available fuel assemblies that could cause an increase in reactivity if misloaded. Although there have been several reported fuel assembly misplacements in spent fuel pools at other plants in the past, the fact that these misplacements were reported and corrected indicates that administrative controls are effective in precluding permanent fuel misloadings.

24. Dr. Gordon Thompson suggested that a single failure in the administrative or the management process may lead to misplacement of multiple out-of-compliance assemblies and this multiple misplacement, with or without boron dilution, may lead to a criticality (Transcript of Deposition of Gordon Thompson, Ph.D., at 162)(Exhibit 16A). However, the placement of a fuel assembly in pools C or D that does not meet the technical specification burnup requirements and the continued failure to detect this misplacement is a highly unlikely event. Multiple misplacements would be even more unlikely. Therefore, Dr. Thompson's suggested scenario is highly improbable, and well beyond the application of the double contingency principle discussed previously.

25. It is possible that loss of borated water might occur either by leakage or by overfill of the pool by unborated water. However, attachment 1.2, sheet 10, of the Shearon Harris Chemistry and Radiochemistry Procedure CRC-001 (Shearon Harris Nuclear Power Plant, Plant Operating Manual, Volume 5, Part 3, Chemistry and Radiochemistry, CRC-001,

SHNPP Environmental and Chemistry Sampling and Analysis Program)(Exhibit 17) specifies that the spent fuel pool boron concentration be maintained between 2000 and 2600 parts per million (ppm) and that the minimum concentration be confirmed by monthly surveillance measurements. In addition, Harris technical specification 3.9.11 (Shearon Harris Unit 1 Technical Specifications, Section 3.9.11, "Refueling Operations, Water Level - New and Spent Fuel Pools," Amendment 88)(Exhibit 18) requires at least 23 feet of water above the top of the fuel rods. Also, FSAR Section 9.1.3 (Shearon Harris Nuclear Power Plant Final Safety Analysis Report, Amendment 49, Section 9.1.3, "Fuel Pool Cooling and Cleanup System," pages 9.1.3-1 through 9.1.3-6c)(Exhibit 19) states that high and low level alarms are provided that would indicate water level changes and, therefore, potential dilution due to leakage or overfill by unborated water. Visual indication of water level is also observed during each shift. Therefore, the staff considers significant boron dilution to be highly improbable.

26. In Dr. Gordon Thompson's deposition of October 21, 1999, he asserts that the NRC staff should have required a boron dilution analysis. Thompson Dep. Tr. at 157 (Exhibit 16A). The NRC staff does, in fact, request a boron dilution analysis. Standard Review Plan (SRP) 9.1.2 (U.S. Nuclear Regulatory Commission, Standard Review Plan, Rev. 3, NUREG-0800, Section 9.1.2, Spent Fuel Storage, July 1981)(Exhibit 20) specifies that the reactivity of each spent fuel pool be at least 5% subcritical if moderated by unborated water. This subcriticality margin is demonstrated in the criticality analysis for pools C and D of the proposed Harris amendment assuming no boron in the pool ("Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D' (proprietary version),

Section 4.0, Criticality Safety Evaluation")(Exhibit 21).[2] I reviewed this criticality safety analysis submitted with the CP&L amendment request. The analysis showed that $k_{eff}$ in the proposed spent fuel pool C and D storage racks would be no greater than 0.95 if accidentally flooded with unborated water. This is an extremely conservative accident condition since the pool is about 25% or 30% subcritical under normal conditions with a minimum of 2000 ppm of boron and a complete boron dilution with loss of all soluble boron would be highly improbable for the reasons stated above.

27. The primary analysis of the reactivity effects of fuel storage in the Harris spent fuel storage racks proposed for pools C and D was performed by Holtec International with the CASMO-3 two-dimensional transport theory code ("CASMO-3, A Fuel Assembly Burnup Program, Methodology," STUDSVIK/NFA-89/2)(Exhibit 22). CASMO-3 was also used for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. The MCNP-4A Monte Carlo code ("RSICC Computer Code Collection, MCNP4B2, Monte Carlo N-Particle Transport Code System," CCC-660, Oak Ridge National Laboratory)(Exhibit 23) was used to determine reactivity effects, to calculate the reactivity for fuel misloading outside the racks and to determine the effect of having PWR and BWR racks adjacent to each other. MCNP-4A was also used for independent verification calculations against CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. (Benchmarking is the comparison of code predictions to known values for the

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[2] Exhibit 21 is proprietary and has been provided under separate cover to the Board and parties.

purpose of validating the code.) These experiments simulate the Harris spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, these two independent methods of analysis (MCNP-4A and CASMO-3) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. These methods have been used and approved by the NRC staff in numerous other criticality analyses of spent fuel pools. Based on the foregoing, I have concluded that the analysis methods used are acceptable and capable of predicting the reactivity of the Harris storage racks proposed for pools C and D with a high degree of confidence.

28. In addition to the extremely conservative assumption of unborated water mentioned above, the Harris criticality analysis was performed with several other conservative assumptions that maximize the storage pool reactivity. These include:

(a) Racks were fully loaded with the most reactive fuel authorized to be stored in the facility.

(b) Unborated water at the temperature yielding the highest reactivity over the expected range of water temperatures.

(c) Assumption of infinite array (no neutron leakage) of storage cells except for the assessment of peripheral effects and certain accidents.

(d) Neutron absorption in minor structural material is neglected.

(e) Uncertainties due to manufacturing tolerances were included to maximize the calculated $k_{eff}$.

(f) Calculational uncertainties and biases were incorporated to maximize the calculated $k_{eff}$..

29. As part of my review of the CP&L amendment request to place fuel storage racks in pools C and D, I reviewed Holtec Report HI-992283 ("Evaluation of Fresh Fuel Assembly Misload in Harris Pools C and D," HI-992283, Holtec International, September 1999)(Exhibit 24),[3] which presented the criticality evaluation of a fresh fuel misload in the Harris C and D pools. Based on analysis performed by Holtec and described in this report, it has been determined that a soluble boron concentration of only 400 ppm would be sufficient to maintain a 5% subcriticality margin in the event of a fuel assembly misloading event (i.e., a fresh PWR assembly enriched to 5 weight-percent U-235 inadvertently placed in a location restricted to a burned assembly as per proposed Shearon Harris Technical Specification Figure 5.6.1 (Exhibit 12). Based on my experience in evaluating the criticality safety of spent fuel pools, I find the calculational methods and the assumptions made in these analyses to be acceptable. The results indicate that the minimum boron concentration of 2000 ppm required in the Harris spent fuel pools is more than adequate to offset the reactivity addition from a postulated fuel assembly misloading event. Although beyond the staff's request, Holtec also presented the results of an additional calculation in HI-992283, using the same NRC-acceptable methods, which showed that criticality would not be achieved for this misloading event even for a concurrent accident condition of loss of all soluble boron.[4]

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[3] Exhibit 24 is proprietary and has been provided under separate cover to the Board and parties.

[4] In addition, the NRC staff performed a calculation that assumed the misloading of an entire burnup-dependent rack with fresh fuel assemblies enriched to 5 weight-percent U-235 and the pool borated to the minimum required 2000 ppm (See Affidavit of Anthony

30. Therefore, the staff has concluded that a fuel loading error in the proposed burnup-dependent Shearon Harris spent fuel storage racks in pools C and D, although highly unlikely, will not cause an inadvertent criticality.

31. In conclusion, GDC 62 does not prohibit the use of administrative controls to prevent criticality in spent fuel storage. In particular, licensees may take credit for burnup to prevent criticality in spent fuel pools. At Shearon Harris, a misloaded fresh fuel assembly will not cause a criticality in pools C or D, even if there is no boron in the pool water. With only 400 ppm of boron in the pool water (a minimum of 2000 ppm is required at Harris), such a fuel misloading event would not cause $k_{eff}$ to be greater than the Staff's acceptance criterion set forth in draft RG 1.13 of 0.95. CP&L's proposed amendment satisfies GDC 62.

32. The exhibits attached hereto are true and correct copies of the documents relied upon in this affidavit.

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P. Ulses). This scenario bounds the broader spectrum of misplacements of more than one assembly suggested by Dr. Thompson on page 157 of his deposition on October 21, 1999, and would require multiple administrative errors, including selection of a large number of improper fuel assemblies as well as failure of independent verification of proper storage in the pool C and D racks. Although the staff considers this scenario to be highly improbable, the results showed that subcriticality is maintained even for an entire misloaded rack.

33. I hereby certify that the foregoing is true and correct to the best of my knowledge, information and belief.

Laurence I. Kopp

Laurence I. Kopp

Subscribed and sworn to before me
this 4 day of January 2000

Circe E. Martin

Notary Public

CIRCE E. MARTIN
NOTARY PUBLIC
MONTGOMERY CO., MD.

My commission expires: March 1, 2003

Exhibits

1. Statement of Professional Qualifications of Laurence I. Kopp.

2. Letter from J. Scaraola (CP&L) to U.S. Nuclear Regulatory Commission, "Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment Spent Fuel Storage," December 23, 1998.

2A. "Proposed Amendment to 10 CFR 50; General Design Criteria for Nuclear Power Plant Construction Permits," AEC-R2/57, June 16, 1967.

3. "General Design Criteria for Nuclear Power Plant Construction Permits," 32 Fed. Reg. 10,213 (July 11, 1967).

4. Letter from W.B. Cottrell (ORNL) to H.L. Price (AEC), "Review of USAEC 'General Design Criteria for Nuclear Power Plant Construction Permits' Federal Register, July 11, 1967," September 6, 1967.

4A. "Status Report on General Design Criteria," memorandum from H.L. Price, Director of Regulation, NRC, to the Chairman and Commissioners, July 6, 1970.

4B. Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969).

5. B.K. Grimes, Letter to All Power Reactor Licensees, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

6. Standard Technical Specifications, Babcock and Wilcox Plants, NUREG-1430, Vol. 1, Rev.1, April 1995.

7. Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Vol. 1, Rev. 1, April 1995.

8. Standard Technical Specifications, Combustion Engineering Plants, Vol. 1, Rev. 1, April 1995.

9. Shearon Harris Nuclear Power Plant Final Safety Analysis Report, Amendment 45, Section 4.3.2.6, "Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies," pages 4.3.2-19 through 4.3.2-22.

10. Letter from D.B. Alexander (CP&L) to U.S. Nuclear Regulatory Commission, "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," June 14, 1999.

11. Carolina Power & Light Company, Nuclear Generation Group, Standard Procedure, Volume 99, Book/Part 99, "NFP-NGGC-0003, Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask," Revision 4.

12. Shearon Harris Unit 1 Technical Specifications, Section 5.6.1.2, "Fuel Storage-Criticality."

13. American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 1983.

14. Regulatory Guide 3.4, Rev. 2, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities," U.S. Nuclear Regulatory Commission, March 1986.

15. R.J. Nodvik, "Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel," WCAP-6068, Westinghouse Electric Corporation, March 1966.

16. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," U.S. Nuclear Regulatory Commission, December 1981.

16A. Deposition of Gordon Thompson, Ph.D., October 21, 1999 (Excerpts).

17. Shearon Harris Nuclear Power Plant, Plant Operating Manual, Volume 5, Part 3, Chemistry and Radiochemistry, CRC-001, SHNPP Environmental and Chemistry Sampling and Analysis Program.

18. Shearon Harris Unit 1 Technical Specifications, Section 3.9.11, "Refueling Operations, Water Level - New and Spent Fuel Pools," Amendment 88.

19. Shearon Harris Nuclear Power Plant Final Safety Analysis Report, Amendment 49, Section 9.1.3, "Fuel Pool Cooling and Cleanup System," pages 9.1.3-1 through 9.1.3-6c.

20. U.S. Nuclear Regulatory Commission, Standard Review Plan, Rev. 3, NUREG-0800, Section 9.1.2, Spent Fuel Storage, July 1981.

21. Letter from J. Scaraola (CP&L) to U.S. Nuclear Regulatory Commission, "Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment Spent Fuel Storage," Enclosure 6, Section 4.0, Criticality Safety

Evaluation, December 23, 1999. [PROPRIETARY-provided under separate cover to the Board and parties.]

22. "CASMO-3, A Fuel Assembly Burnup Program, Methodology," STUDSVIK/NFA-89/2.

23. "RSICC Computer Code Collection, MCNP4B2, Monte Carlo N-Particle Transport Code System," CCC-660, Oak Ridge National Laboratory.

24. "Evaluation of Fresh Fuel Assembly Misload in Harris Pools C and D," HI-992283, Holtec International, September 1999. [PROPRIETARY-provided under separate cover to the Board and parties.]

Laurence I. Kopp
Senior Reactor Engineer

Education

Ph.D., Nuclear Engineering, University of Maryland, 1968
M.S., Physics, Stevens Institute of Technology, 1959
B.S., Physics, Fairleigh Dickinson College, 1956.

Employment

U.S. Nuclear Regulatory Commission, Senior Reactor Engineer, 1965 - present
Performs safety evaluations of reactor license applications, technical specifications, core reloads, spent fuel storage facilities, and topical reports. Developed regulatory guides, information notices, generic letters, rulemaking related to reactor physics, safety analysis, and fuel storage. Assisted in development of improved technical specifications in areas of reactivity control, power distribution limits, and fuel storage.

Westinghouse Astronuclear Laboratory, Senior Scientist, 1963-1965
Evaluated nuclear analytical methods to be used in the design of NERVA rocket reactors. Analyzed experiments performed in the Los Alamos zero power reactor.

Martin-Marietta Nuclear Division, Senior Engineer, 1959-1963
Performed core physics calculations on fluidized bed and PM-1 reactors. Performed parametric studies of reactors applicable to nuclear rocket applications. Programmed several FORTRAN computer codes.

Federal Electric Corporation, Senior Programmer, 1957-1959

Curtiss-Wright Research Division, Programmer/Physicist, 1956-1957

Professional Societies

American Nuclear Society
ANS-10 Mathematics and Computations Standards Committee (Chairman 1986-1988)
ANSI N-17 Standards Committee on Research Reactors, Reactor Physics & Radiation Shielding

Publications

"The NRC Activities Concerning Boraflex Use in Spent-Fuel Storage Racks," invited paper, American Nuclear Society Annual Meeting, June 1996.

"Potential Loss of Required Shutdown Margin During Refueling Operations," invited paper, American Nuclear Society Annual Meeting, June 1990.

"Recommended Programming Practices to Facilitate the Portability of Scientific Computer Programs," ANS Proceedings of the Topical Meeting on Computational Methods in Nuclear Engineering, April 1979.

"The Neutron Resonance Integral of Natural Dysprosium," Ph.D. thesis, University of Maryland, 1968.

"Pool Reactor Experiments with Control Rods," Transactions of the American Nuclear Society, Vol. 10, Pg. 16, 1967 (co-author).

"Procedures for Obtaining Few-Group Constants for Systems Having Rapid Spectral Variation With Position," Transactions of the American Nuclear Society, Vol. 8, pg. 303, 1965 (co-author).

"Improved Nuclear Design Method for NERVA Calculations - NSDM II," WANL-TME-1091, Westinghouse Astronuclear Laboratory, 1965 (co-author).

"Analysis of Experiments Performed in Los Alamos ZEPO Reactor," WANL-TME-273, Westinghouse Astronuclear Laboratory, 1963.

Kopp Exhibit 1

Laurence I. Kopp
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Kopp Exhibit 2

2

**CP&L**

**Carolina Power & Light Company**
PO Box 165
New Hill NC 27562

**James Scarola**
Vice President
Harris Nuclear Plant

DEC 23 1998

SERIAL: HNP-98-188
10CFR50.90
10CFR50.59(c)
10CFR50.55(a)

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
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

Dear Sir or Madam:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) requests a license amendment to place spent fuel pools 'C' and 'D' in service. Specifically, Harris Nuclear Plant (HNP) proposes to revise TS 5.6 "Fuel Storage" to increase the spent fuel storage capacity by adding rack modules to pools 'C' and 'D'. The enclosures to this letter support the proposed license amendment.

Enclosure 1 provides background information, a description of the proposed changes, and the basis for the changes.

Enclosure 2 details, in accordance with 10 CFR 50.91(a), the basis for the CP&L's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment is required for approval of this amendment request.

Enclosure 4 provides page change instructions for incorporating the proposed revisions.

Enclosure 5 provides the proposed Technical Specification pages.

Enclosure 6 provides a report entitled "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D'" which contains supporting technical documentation. Please note that Enclosure 6 contains information which is considered proprietary pursuant to 10 CFR 2.790. In this regard, CP&L requests Enclosure 6 be withheld from public viewing.

Enclosure 7 is identical to Enclosure 6, except that the proprietary information has been removed and replaced by highlighting and/or a note of explanation at each location where the information has been omitted. CP&L provides this additional version for the purposes of public review.

Enclosure 8 provides a detailed description of the proposed alternatives to demonstrate compliance with ASME B&PV Code requirements for the cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i).

Enclosure 9 provides results of the thermal hydraulic analysis of the cooling water systems that support placing pools 'C' and 'D' in service. The analysis resulted in changes to previously reviewed and approved cooling water flow requirements. These changes have been identified as an unreviewed safety question and are being submitted for NRC review and approval pursuant to the requirements of 10 CFR 50.59(c) and 10 CFR 50.90.

CP&L requests the issuance date for this amendment be no later than December 31, 1999. This issuance date is necessary to support loading of spent fuel in pool 'C' starting in early 2000. CP&L also requests the proposed amendment be issued such that implementation will occur within 60 days of issuance to allow time for procedure revision and orderly incorporation into copies of the Technical Specifications.

Please refer any questions regarding this submittal to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

James Scarola

RSE/KWS/kws

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Technical Specification Pages
6. Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D' (proprietary version)
7. Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D' (non-proprietary version)
8. 10 CFR 50.55a(a)(3) Alternative Plan
9. Unreviewed Safety Question Analysis

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Lisa M. Randall
Notary (Seal)

LISA M. RANDALL NOTARY PUBLIC WAKE COUNTY, N.C.

My commission expires: 6-7-2003

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. S. C. Flanders, NRC Project Manager
Mr. Mel Fry, Director, N.C. DRP
Mr. L. A. Reyes, NRC Regional Administrator

bc:
Ms. D. B. Alexander
Mr. K. B. Altman
Mr. G. E. Attarian
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
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. G. J. Kline
Ms. W. C. Langston (PE&RAS File)
Mr. R. D. Martin
Mr. J. W. McKay
Mr. P. M. Odom (RNP)
Mr. W. S. Orser
Mr. P. M. Sawyer (BNP)
Mr. J. M. Taylor
Nuclear Records
Licensing File
File: H-X-0512
File: H-X-0642

Enclosure 1 to Serial: HNP-98-188

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

**BASIS FOR CHANGE REQUEST**

## BASIS FOR CHANGE REQUEST

Background:

The Harris Plant was originally planned as a four nuclear unit site (Harris 1, 2, 3 and 4). In order to accommodate four units at Harris, the Fuel Handling Building (FHB) was designed and constructed with four separate pools capable of storing spent fuel. The two pools at the south end of the FHB, now known as Spent Fuel Pools (SFPs) 'A' and 'B', were to support Harris Units 1 and 4. The two pools at the north end of the FHB, now known as Spent Fuel Pools 'C' and 'D', were to support Harris Units 2 and 3. The multi-unit design included a spent fuel pool cooling and cleanup system to service SFPs 'A' and 'B' and a separate cooling and cleanup system to support SFPs 'C' and 'D'.

Harris Units 3 and 4 were canceled in late 1981. Harris Unit 2 was canceled in late 1983. The FHB, all four pools (including liners), and the cooling and cleanup system to support SFPs 'A' and 'B' were completed and turned over. However, construction on the spent fuel pool cooling and cleanup system for SFPs 'C' and 'D' was discontinued after Unit 2 was canceled and the system was not completed. Harris Unit 1 began operation in 1987 with SFPs 'A' and 'B' in service. The need to eventually activate SFPs 'C' and 'D' (depending on the availability of a permanent DOE spent fuel storage facility) was anticipated at the time the operating license for Harris Unit 1 was issued. The spent fuel storage capacity currently identified in Section 5.6.3 of the Harris Plant Technical Specifications (1832 PWR assemblies and 48 interchangeable (7 x 7 cell) PWR or (11 x 11 cell) BWR racks) assumes installation of racks in all four of the spent fuel pools.

Since the time that construction of the spent fuel pool cooling and cleanup system for SFPs 'C' and 'D' was halted, CP&L has implemented a spent fuel shipping program because DOE spent fuel storage facilities are not available and are not expected to be available for the foreseeable future. Spent fuel from Brunswick (2 BWR units) and Robinson (1 PWR unit) is shipped to Harris for storage in the Harris SFPs. Shipment of spent fuel to Harris is necessary in order to maintain full core offload capability at Brunswick and Robinson. As a result of the operation of the Harris Plant, shipping program requirements, and the unavailability of DOE storage, it will be necessary to activate SFPs 'C' and 'D' and the associated cooling and cleanup system by early in the year 2000. Activation of these two pools will provide storage capacity for all four CP&L nuclear units (Harris, Brunswick 1 and 2, and Robinson) through the end of their current licenses.

SFP 'A' now contains six Region 1 flux trap style (6 x 10 cell) PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. SFP 'A' has been, and will continue to be, used to store fresh (unburned) and recently discharged Harris fuel.

SFP 'B' now contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) PWR Region 1 style racks. SFP 'B' also currently contains seventeen (11 x 11 cell) BWR racks. SFP 'B' is licensed to store one more (11 x 11 cell) BWR rack, which would increase the total pool storage capacity to 2946 assemblies. Harris is postponing installation of the last BWR rack and prefers to reserve the pool open area for fuel examination and repair. Therefore, the total installed capacity in SFP 'B' will temporarily remain as 768 PWR cells and 2,057 BWR cells for a total of 2,825 storage cell locations.

Proposed Changes:

The proposed changes will allow CP&L to increase the spent fuel storage capacity at the Harris plant by placing SFPs 'C' and 'D' in service. In order to activate the pools, CP&L requests that the NRC review and approve the following changes:

1. Revised Technical Specification 5.6 to identify PWR burnup restrictions, BWR enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D'.

   The use of the high density region 2 racks has been shown to be acceptable based on the analysis performed by Holtec International.

2. 10CFR50.55a Alternative Plan to demonstrate acceptable level of quality and safety in the completion of the component cooling water (CCW) and SFP 'C' and 'D' cooling and cleanup system piping.

   The cooling system for SFPs 'C' and 'D' cannot be N stamped in accordance with ASME Section III since some installation records are not available, a partial turnover was not performed when construction was halted following the cancellation of Unit 2 and CP&L's N certificate program was discontinued following completion of Unit 1. The Alternative Plan demonstrates that the originally installed equipment is acceptable for use and that the design and construction on the remaining portion of the cooling system piping (estimated at about 20%) maintains the same level of quality and safety through the use of the CP&L Appendix B QA program supplemented by additional QA requirements integrated into the plant modification package which completes the system

3. Unreviewed safety question for additional heat load on the component cooling water (CCW) system.

   The acceptability of the 1.0 MBtu/hr heat load from SFPs 'C' and 'D' was demonstrated by the use of thermal-hydraulic analyses of the CCW system under various operating scenarios. The dynamic modeling used in the thermal-hydraulic analyses identified a decrease in the minimum required CCW system flow rate to the RHR heat exchangers. This change has not been previously reviewed by the NRC and is deemed to constitute an unreviewed safety question.

Basis for Change

Installation of spent fuel storage racks in SFPs 'C' and 'D':

The FHB and SFPs 'C' and 'D' (including pool liners) were fully constructed and turned over as part of the construction and licensing of Harris Unit 1. However, the decision was made to not place SFPs 'C' and 'D' in service until needed (depending on the availability of DOE spent fuel storage). SFPs 'C' and 'D' are flooded but have not been previously used for spent fuel storage. CP&L proposes to expand the storage capacity at Harris by installing Region 2 (non-flux trap style) rack modules in Pools 'C' and 'D' in incremental phases (campaigns), on an as needed basis. SFP 'C' will provide the initial storage expansion for both PWR and BWR fuel. In its fully implemented storage configuration, SFP 'C' can accommodate 927 PWR and 2763 BWR assemblies. Expansion of storage capacity by installing racks in SFP 'D' will occur once SFP 'C' is substantially filled. SFP 'D' will contain only PWR fuel and can accommodate 1025 maximum density storage cells.

Following this proposed change, Spent Fuel Pool capacities will be as follows:

| Pool | PWR spaces | BWR spaces | Total |
|---|---|---|---|
| 'A' | 360 | 363 | 723 |
| 'B' | 768 | 2178 | 2946 |
| 'C' | 927 | 2763 | 3690 |
| 'D' | 1025 | 0 | 1025 |
| Total | 3080 | 5304 | 8384 |

Racks in SFP 'C' and 'D' will be installed in the following phases:

SFP 'C' - 1st Campaign - install by early 2000
4 PWR racks → 360 PWR spaces
10 BWR racks → 1320 BWR spaces

SFP 'C' - 2nd Campaign - install approximately 2005
4 PWR racks → 324 PWR spaces
6 BWR racks → 936 BWR spaces

SFP 'C' - 3rd Campaign - install approximately 2014
3 PWR racks → 243 PWR spaces
3 BWR racks → 507 BWR spaces

SFP 'D' - 1st Campaign - install approximately 2016
6 PWR racks → 500 PWR spaces

SFP 'D' - 2nd Campaign - installation date to be determined
6 PWR racks → 525 PWR spaces

(Note: The projected rack installation dates listed above are based on the current spent fuel shipping schedule. These dates may change as the shipping schedule is revised).

This configuration represents the mixture of PWR and BWR storage which will accommodate future storage requirements based on currently identified needs. Within SFP 'C', eighteen (18) of the racks are sized to allow interchangeability between BWR and PWR storage if required in the future. The dimensions of the (9 x 9 cell) PWR rack and the (13 x 13 cell) BWR rack are virtually identical. Therefore, rack configurations other than those identified above are possible.

Enclosure 6 of this license amendment request provides a report developed in conjunction with Holtec International which describes the evaluations performed to show the acceptability of the proposed change to install the racks in pools 'C' and 'D'. (Enclosure 7 is a non-proprietary version of enclosure 6). The report includes listings of the applicable regulations, codes and standards, descriptions of the evaluation methodology, acceptance criteria, and evaluation results. The licensing report also includes discussions on the need for the proposed change and considerations of other alternatives. Technical Specification Section 5.6, Fuel Storage, will be revised to identify PWR burnup restrictions, BWR enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D' (See Enclosure 5).

Completion of Cooling and Cleanup System for SFPs 'C' and 'D':

In order to activate Spent Fuel Pools 'C' and 'D', it is necessary to complete construction of the cooling and cleanup system for these pools and to install tie-ins to the existing Harris Unit 1 component cooling water system to provide heat removal capabilities. Approximately 80% of the SFP cooling and cleanup system piping and the majority of the CCW piping was installed during the original plant construction. In addition, other major system components such as the SFP cooling heat exchangers and pumps were also installed before original construction was discontinued. The cooling and cleanup system for pools 'C' and 'D' will be completed such that system design and operation is consistent with the design and operation of the cooling and cleanup system for pools 'A' and 'B'. The spent fuel pool cooling system for pools 'C' and 'D' is nuclear safety related with two fully redundant 100% capacity trains.

At the time that construction on the SFP cooling system was discontinued following cancellation of Harris Unit 2, a formal turnover of the partial system was not performed and CP&L has since discontinued its N certificate program. Also, some of the field installation records for the completed piping are no longer available. As a result, the system when completed will not satisfy ASME Section III code requirements (i.e. will not be N stamped). Therefore, an Alternative Plan in accordance with 10CFR50.55a(a)(3) is provided as Enclosure 8 to demonstrate that the completed system will provide an acceptable level of quality and safety. The majority of the ASME Section III piping was already installed when original construction was discontinued. As identified in the Alternative Plan, that piping to the extent that it was completed, was designed, constructed and inspected to Section III requirements. The remainder of the system will also be designed, constructed, inspected and tested to Section III requirements to the extent practical considering CP&L no longer has an N certificate program. Work will be performed in accordance with CP&L's 10CFR50 Appendix B QA program with any differences between Section III requirements and Appendix B requirements conservatively dispositioned. Supplemental QA requirements will be integrated into the modification package(s) as appropriate.

Calculations have been performed to verify that the existing CCW system is adequate to provide heat removal for near-term pool operation. The Spent Fuel Pool 'C' and 'D' heat loads will be limited to 1.0 MBtu/hr for near-term operation. Technical Specification section 5.6.3 will be revised to identify this heat load limit (Enclosure 5). This heat load limit is being established since additional CCW heat loads resulting from the power uprate project (potential to increase post-accident containment temperature resulting in an increased containment sump temperatures and increased load on RHR during long term recirculation phase) are not quantified at this time. Therefore, it has been determined that the most prudent action is to establish limiting heat loads based on current system loads. Additional heat load analysis will be performed concurrent with the power uprate project to establish the maximum heat loads on the CCW system that will exist at the end of plant licensed life when all spent fuel pools are expected to be full. Any CCW modifications necessary to increase system heat removal capability will be identified and implemented at that time. As part of the licensing required to support the power uprate project (currently planned for implementation concurrent with the steam generator replacement in late 2001), the technical specification heat load limit will either be revised or removed completely.

The plant design change package and supporting analyses for the CCW tie-in demonstrated that adequate capacity exists on the CCW system to add the 1.0 MBtu/hr for the near-term operation of SFPs 'C' and 'D'. The thermal-hydraulic analysis performed in support of this plant design change package modeled the dynamic RHR heat exchanger performance based on fluid property changes. Previous analyses evaluated RHR heat exchanger performance at a fixed data sheet value. This results in a reduction in the required CCW flow to the RHR heat exchanger. While technically valid, the lower required flow rate has not been previously reviewed by the NRC and, therefore, is deemed to constitute an unreviewed safety question. Included in Enclosure 9 are the results of the 10CFR50.59 evaluation for the unreviewed safety question identified by the tie-in to Unit 1 CCW.

Conclusion:

CP&L has concluded that placing SFPs 'C' and 'D' in service at this time to provide spent fuel storage is the safe and prudent alternative for increasing spent fuel storage capacity in the nuclear generating system. This option has been shown to be safe and in conformance with the appropriate regulations, codes and standards. Expansion of storage capacity by using Pools 'C' and 'D' will support continued operation of the Harris, Brunswick and Robinson facilities until the end of their current operating licenses.

Kopp Exhibit 2A

...ICIAL U... ONLY

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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| [illegible] | 3 | Plan... & P... | [illegible] |

ATOMIC ENERGY COMMISSION

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

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

## THE PROBLEM

1. To consider the publication for public comment of a proposed amendment to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits." The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria for nuclear power plants to be included in applications for construction permits. Under the proposed amendments to this Part, specifically to §50.34, which were published for public comment in the Federal Register on August 16, 1966, applicants for an AEC construction permit would be required to specify these principal design criteria for a proposed facility. The proposed new guide would be substituted for the present Appendix A to Part 50.

## BACKGROUND AND SUMMARY

2. The development and publication of criteria for nuclear power plants was one of the key recommendations of the Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures. The Panel particularly stressed the need for design criteria to be used at the construction permit stage of a licensing proceeding. Work on the development of general criteria had been in progress at the time of the Review Panel's study. This effort was accelerated and led to the issuance in a Commission press release dated November 22, 1965, of draft criteria f[illegible]

Construction Permits." The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria for nuclear power plants to be included in applications for construction permits. Under the proposed amendments to this Part, specifically to §50.34, which were published for public comment in the Federal Register on August 16, 1966, applicants for an AEC construction permit would be required to specify these principal design criteria for a proposed facility. The proposed new guide would be substituted for the present Appendix A to Part 50.

## BACKGROUND AND SUMMARY

2. The development and publication of criteria for nuclear power plants was one of the key recommendations of the Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures. The Panel particularly stressed the need for design criteria to be used at the construction permit stage of a licensing proceeding. Work on the development of general criteria had been in progress at the time of the Review Panel's study. This effort was accelerated and led to the issuance in a Commission press release dated November 22, 1965, of draft criteria for use in the evaluation of applications for nuclear power plant construction permits.* The criteria were largely statements of design principles and objectives previously used by the staff in evaluating applications for reactor construction permits. Although they reflected the predominating experience with water reactors, they were considered to be generally applicable to other reactors as well.

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

3. As invited in the press release, twenty-two groups of individuals submitted comments, as listed in Appendix "A." Because of the volume, the correspondence is not attached. Copies of all comments received except those originated within the Commission have been placed in the Public Document Room.

4. The general reaction was that the criteria fulfilled a need and the AEC should continue their development. None of the correspondents objected to the issuance of general criteria and their comments were constructive. The Atomic Industrial Forum, for example, submitted a complete proposed revision reflecting considerable interest and effort on the part of that organization. The comments received fell into the following broad categories:

a. Title each criterion. This was suggested as an aid in indexing and referencing.

b. Improve the organization of the criteria. Comments included suggestions for arranging criteria according to type of systems and for grouping the criteria according to the degree of public protection.

c. Simplify the format. A number of suggestions were made for eliminating repetition for combining criteria and for clarification.

d. Eliminate details. Some comments suggested that the criteria should state only objectives, and that specific details and manner of implementation should not be stated. A number of comments expressed a desire for less general and for more comprehensive and detailed criteria.

e. Relate the criteria only to the protection of the public. Views were expressed that some criteria as written related to operational problems and should be eliminated.

f. Retitle the document. A belief was expressed that as written these were not truly criteria, but principles or fundamentals.

g. Apply the criteria more broadly than construction permits alone. This comment essentially urged that the restriction of the criteria to construction permits should be deleted and that they should be made applicable to all stages of licensing, including the operating license

5. The staff has considered all comments received in further developing 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.

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## 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).

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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.

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 (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

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

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

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

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

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

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

* * * * *

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

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

INTRODUCTION



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.

## 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 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 off-site power.

CRITERION 7 - SUPPRESSION 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 B)

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 (Category 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$^{\circ}$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$^{\circ}$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 leaktightness 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.

APPENDIX "C"

DRAFT LETTER TO JOINT COMMITTEE ON ATOMIC ENERGY

1. Enclosed for the information of the Joint Committee on Atomic Energy is a Notice of Proposed Rule Making which would add to the proposed amendments to the Commission's regulations 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which were published in the Federal Register for comment on August 16, 1966. This amendment would add a new Appendix A to Part 50 "General Design Criteria for Nuclear Power Plant Construction Permits" to assist in the preparation of applications for construction permits for nuclear power plants.

2. The proposed change implements one of the key recommendations of the Regulatory Review Panel in which the Panel expressed the need for criteria to be used at the construction permit stage. As you know, work had been in progress on criteria development at the time of the Panel's recommendation. This effort was accelerated and led to the issuance of preliminary proposed criteria for public comment in Press Release H-252 dated November 22, 1965. The General Design Criteria included in the enclosed proposed amendment reflect comments and suggestions on the preliminary criteria received from industry, divisions within the Commission, the Advisory Committee on Reactor Safeguards, and the public.

3. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant as contemplated by the previously published revisions to Part 50. The framework within which the criteria are presented provides sufficient flexibility for applicants to establish design requirements using alternate and/or additional criteria so long as safety can be assured. In particular,

additional criteria will be needed for unusual sites and environmental conditions and for new or advanced types of reactors. In every case, however, the applicant will be required to identify its principal design criteria and provide assurance that they encompass all those facility design features required in the interest of public health and safety.

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

5. The notice of proposed rule making has been transmitted to the Office of the Federal Register for publication. Sixty days for public comment are provided. Enclosed also is a copy of an announcement we plan to issue in the next few days on this matter.

## APPENDIX "D"

## DRAFT PUBLIC ANNOUNCEMENT

### AEC PUBLISHES GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

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

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

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

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

The General Design Criteria reflect the predominating experience to date with water reactors, but they are considered to be generally applicable to all power reactors. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant. The framework within which the criteria are presented provides suffi-

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

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

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

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

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

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

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

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

Kopp Exhibit 3

(3)

32 FR 10213
Published 7/11/67
Comment period
expires 9/9/67

## General Design Criteria for Nuclear Power Plant Construction Permits

The Atomic Energy Comission [is] 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."[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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[a] 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 B).* 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 (Category 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.]

Kopp Exhibit 4

Shapar

DOCKET NUMBER PR 50
PROPOSED RULE

(4)

# OAK RIDGE NATIONAL LABORATORY

OPERATED BY

UNION CARBIDE CORPORATION

NUCLEAR DIVISION

POST OFFICE BOX Y

OAK RIDGE, TENNESSEE 37830

September 6, 1967

DOCKETED
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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.

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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.

<u>Criteria 54, 55, and 56</u> - 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.

<u>Criterion 56</u>

This criterion is not sufficiently inclusive. The types of penetrations which should be tested should <u>NOT</u> 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.

<u>Criterion 66</u> - 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."

<u>Criterion 67</u> - 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.

Kopp Exhibit 4A

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

OGC

File:
Part 50 –
Design Criteria
& Part 52

JUL 6 1970

Chairman Seaborg
Commissioner Ramey
Commissioner Johnson
Commissioner Thompson
Commissioner Larson

STATUS REPORT ON GENERAL DESIGN CRITERIA

Enclosed for the information of the Commissioners is the latest draft of the revised General Design Criteria for Nuclear Power Plants. This is the revision which is now being reviewed by an ad hoc committee of the Atomic Industrial Forum. Some Forum members believe that revised Criterion 5 (the need to consider the probability and effects of industrial sabotage) and revised Criteria 22, 24, and 29 (because of the reference in each of these criteria to systematic, nonrandom, concurrent failures) are not acceptable. Some Forum members also believe that there should be changes in wording (but not in intent) of about 25 additional criteria. The wording of the remaining revised criteria is considered to be acceptable by Forum members.

As you may know, this revision of the criteria has been concurred in by all interested regulatory divisions and also reflects agreements with the ACRS. This version also takes into account the oral and written comments of those AIF members who reviewed a previous draft of the criteria and participated in a day long discussion with the staff in February of this year. The criteria were extensively revised as a result of the February discussions (at least 27 of the criteria were substantially changed) and at least four of the six Forum representatives at the meeting appeared to be satisfied with the changes made. Any further substantial changes in the criteria would probably require another period of review by the regulatory staff and the ACRS.

We have been informed that the additional comments being developed by the Forum will be completed in about a month. Another meeting with Forum members will be held at that time to discuss the changes they suggest.

(Signed) HLP

Harold L. Price
Director of Regulation

Enclosure:
Revised General Design Criteria
Draft dated June 4, 1970

cc w/encl:
Secretary (2)
General Manager (2)
General Counsel (2)

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| OFFICE ▷ | DRS:DIR | DR | | | | |
| SURNAME ▷ | Case:jjb | Harold L. Price | | | | |
| DATE ▷ | 6/29/70 | / /70 | | | | |

# GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

June 4, 1970

# APPENDIX A

## GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

### Table of Contents

# INTRODUCTION

Pursuant to the provisions of §50.34, an application for a construction permit must include the principal design criteria for a proposed facility. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The principal design criteria for a nuclear power unit establish necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that prevent or mitigate the consequences of accidents which could cause undue risk to the health and safety of the public. There will be some water-cooled nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For units such as these, departures from the General Design Criteria must be identified and justified.

## DEFINITIONS 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 reactor coolant makeup system from any size break in the piping, pressure vessels, pumps, and valves connected to the reactor pressure vessel and which are part of the reactor coolant pressure boundary, up to and including a break in these components equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

### SINGLE FAILURE

A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Mechanical and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of any passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions. The failure of a passive component need not be considered in the design of mechanical systems if it can be demonstrated that the design is acceptable on some other defined basis, such as an appropriate combination of unusually high quality, high strength or low stress, inspectability, repairability, or short-term use.

ANTICIPATED OPERATIONAL OCCURRENCES

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

## CRITERIA

## I. OVERALL REQUIREMENTS

### CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

### CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, (2) sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (3) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (4) the importance of the safety functions to be performed.

## CRITERION 3 - FIRE PROTECTION

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

## CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from sources outside the nuclear power unit.

CRITERION 5 - PROTECTION AGAINST INDUSTRIAL SABOTAGE

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

CRITERION 6 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

## II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded 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.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

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

CRITERION 13 - REACTOR INSTRUMENTATION AND CONTROL

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

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

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

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

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

CRITERION 16 - CONTAINMENT DESIGN

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

CRITERION 17 - ELECTRICAL POWER SYSTEMS

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

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

Two physically independent transmission lines, each with the capability of supplying electrical power from the transmission network to the switchyard, and two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of electrical power from all other alternating current sources, including onsite electrical sources, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available immediately following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

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

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 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 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 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

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

## CRITERION 19 - CONTROL ROOM

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

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 acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

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

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

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

CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These 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 to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

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

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

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

CRITERION 33 - REACTOR COOLANT MAKEUP

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

CRITERION 34 - RESIDUAL HEAT REMOVAL

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

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

CRITERION 35 - EMERGENCY CORE COOLING

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

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

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM COMPONENTS

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

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

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

CRITERION 38 - CONTAINMENT HEAT REMOVAL

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

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

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM COMPONENTS

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

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

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

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

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

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

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS COMPONENTS

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

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

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

CRITERION 44 - COOLING WATER

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

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

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM COMPONENTS

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

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

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

## V. REACTOR CONTAINMENT

### CRITERION 50 - CONTAINMENT DESIGN BASIS

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

### CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

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

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

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

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

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

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

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

## CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

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

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

## CRITERION 56 - CONTAINMENT PRESSURE BOUNDARY ISOLATION VALVES

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

CRITERION 57 - CLOSED SYSTEMS ISOLATION VALVES

Each line which penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one isolation valve, other than a simple check valve. This valve shall be outside of containment and shall be located as close to containment as practical.

## VI. FUEL AND RADIOACTIVITY CONTROL

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

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

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

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

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

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

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

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

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

Dated at ______________________ this ______ ______

day of ______________________ 1970.

For the Atomic Energy Commission

______________________

W. B. McCool
Secretary

Kopp Exhibit 4B

COMPARISON OF PUBLISHED CRITERIA (JULY 11, 1967) AND REVISED CRITERIA (JULY 15, 1969)

## PUBLISHED VERSION (JULY 11, 1967)

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.

## REVISED VERSION (JULY 15, 1969)

INTRODUCTION

Pursuant to the provisions of § 50.34, applica permits must include the principal design criteria : These General Design Criteria establish minimum req principal design criteria for water-cooled nuclear design and location to units previously approved fo Commission. The General Design Criteria are also c applicable to other types of nuclear power units an used for guidance in establishing the principal des units.

The principal design criteria for a nuclear po necessary design, fabrication, construction, testin requirements for structures, systems, and component that is, structures, systems, and components that p consequences or accidents which could cause undue r safety of the public. There will be some nuclear p these General Design Criteria are not sufficient fo tional criteria must be established in the interest is expected that additional or different criteria w into account unusual sites and environmental condit cooled nuclear power units of advanced design. Als power units for which fulfillment of some of the Ge may not be necessary or appropriate. For units suc from the General Design Criteria must be identified

PUBLISHED VERSION (JULY 11, 1967)

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.

ALSO SEE PUBLISHED VERSION OF CRITERION 5 (PAGE 6)

REVISED VERSION (JULY 15, 1!

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components importa designed, fabricated, erected, and tested to qua with the importance of the safety function to be generally recognized codes and standards are use Codes and standards shall be supplemented and mo assure a quality product in keeping with the req quality assurance program shall be established a provide adequate assurance that these structures, will satisfactorily perform their safety functio fabrication, erection, and testing of structures, important to safety shall be maintained by or un nuclear power plant licensee throughout the life

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15, 196**

**CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NAT**

**Structures, systems, and components important to the effects of natural phenomena such as earthquakes, floods, tsunami, and seiches without loss of capabili safety functions. The design bases for these structu components shall reflect: (1) appropriate considerat of the natural phenomena that have been historically and surrounding area, (2) an appropriate margin for t quantity, and period of time in which the historical lated, (3) appropriate combinations of the effects of conditions with the effects of the natural phenomena of the safety function to be performed.**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15,**

**CRITERION 3 - FIRE PROTECTION**

**Structures, systems, and components importa designed and located to minimize the probability explosions. Noncombustible and heat resistant m wherever practicable throughout the unit particu the containment and control room. Fire detectio appropriate capacity and capability shall be pro minimize the adverse effects of fires on structu important to safety. Fire fighting systems shall their rupture or inadvertent operation does not s capability of these structures, systems, and comp**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

**CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AI**

**Structures, systems, and components importa shared between nuclear power units unless it is to perform their safety functions is not signifi sharing.**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

ALSO SEE PUBLISHED VERSION OF CRITERION 1 (PAGE 2)

**REVISED VERSION (JULY 15,**

**CRITERION 1 - QUALITY STANDARDS AND RECORDS**

**Structures, systems, and components importa designed, fabricated, erected, and tested to qua with the importance of the safety function to be generally recognized codes and standards are use Codes and standards shall be supplemented and mo assure a quality product in keeping with the rec quality assurance program shall be established a provide adequate assurance that these structures will satisfactorily perform their safety functio fabrication, erection, and testing of structures important to safety shall be maintained by or un nuclear power plant licensee throughout the life**

PUBLISHED VERSION (JULY 11, 1967)

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 off-site power.

**REVISED VERSION (JULY :**

**CRITERION 10 - REACTOR DESIGN**

**The reactor core and associated coolant systems shall be designed with appropriate m acceptable fuel damage limits are not exceed system designs shall assure this fuel integr normal operation, including the effects of a rences such as loss of power to recirculatio the capability of the reactor coolant makeup generator set, isolation of the main condens power.**

**PUBLISHED VERSION (JULY 11, 1967)**

CRITERION 7 - SUPPRESSION 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.

**REVISED VERSION (JUL**

CRITERION 12 - SUPPRESSION OF REACTOR POWE

The reactor core and associated coola systems shall be designed to assure that p cause damage in excess of specified accept. not possible or can be reliably and readil

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant so that in the power operating range the effe feedback characteristics tends to compensate reactivity.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15, 1**

**CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY**

**The reactor coolant pressure boundary shall be erected, and tested so as to have an extremely low leakage, rapidly propagating failure, or gross rupt**

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY 15

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment shall be provided. Th
systems shall be designed to provide an essenti
against the uncontrolled release of radioactivi
to assure that the containment design condition
long as any postulated accident condition requi

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY

CRITERION 19 - CONTROL ROOM

A control room shall be provided from wh to operate the nuclear power unit safely unde to maintain it in a safe condition under acci loss-of-coolant accidents. Adequate radiatio room shall be provided to permit access and o conditions without personnel receiving radiat 5 rem whole body, or its equivalent to any pa: duration of the accidents.

Equipment at appropriate locations outsi( provided (1) having a design capability for p reactor, including necessary instrumentation a unit in a safe condition during hot shutdown a capability for subsequent cold shutdown of the suitable emergency procedures.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JUI**

CRITERION 13 - REACTOR INSTRUMENTATI

Instrumentation and control sha variables and systems which can affe integrity of the reactor core are mo prescribed operating ranges.

**PUBLISHED VERSION (JULY 11, 1967)**

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

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.

**REVISED VERSION (JULY**

CRITERION 20 - PROTECTION SYSTEM FUNCTION

The protection systemshall be design to assure that specified acceptable fuel as a result of anticipated operational oc accident conditions and to initiate the o components important to safety.

**PUBLISHED VERSION (JULY 11, 1967)**

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

**REVISED VERSION (JULY 15, 19**

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE B

Components within the reactor coolant pressure designed, fabricated, erected, and tested to the hig standards practicable. Means shall be provided for to the extent practicable, identifying the location reactor coolant leakage.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 1**

**CRITERION 64 - MONITORING RADIOACTIVITY RELEASE**

**Means shall be provided for monitoring the atmosphere, spaces containing components for re coolant accident fluids, effluent discharge pat for radioactivity that may be released from nor anticipated operational occurrences, and from p**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

CRITERION 63 - MONITORING FUEL AND WASTE STORA

Instrumentation shall be provided in fuel active waste systems and associated handling a conditions that may result in loss of decay he excessive radiation levels and (2) to initiate actions.

PUBLISHED VERSION (JULY 11, 1967)

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category 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.

ALSO SEE PUBLISHED VERSION OF CRITERION 25 (PAGE 23)

REVISED VERSION (JULY 15, 196

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TES

The protection system shall be designed for hig and inservice testability commensurate with the safe performed. Redundancy and independence designed int shall be sufficient to assure that (1) no single fai of the protection function and (2) removal from serv channel does not result in loss of redundancy. Mear for testing the protection system when the reactor i determine failures and losses of redundancy and ind occurred.

"DIFFERENT PRINCIPLES..."COVERED B

OF CRITERION 22 (PAGE 21)

**PUBLISHED VERSION (JULY 11, 1967)**

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

**REVISED VERSION (JULY 15,**

**SINGLE FAILURE**

**A single failure means an occurrence which**
**bility of a structure, system, or component to p**
**Multiple failures resulting from a single occurr**
**a single failure.**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

CRITERION 24 - SEPARATION OF PROTECTION AND C

The protection system shall be separated the extent that failure or removal from servi component or channel, or any one of those com tection systems, leaves intact a system satis redundancy, testability, and independence req protection system.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**ALSO SEE PUBLISHED VERSION OF CRITERION 20 (PAGE 18)**

**REVISED VERSION (JULY 15, 19**

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The effects of adverse conditions to which re

protection system may be exposed in common, either

or those of an accident, shall not result in loss

tion, or shall be demonstrated to be acceptable on

basis. Design techniques, such as diversity in co

principles of operation, shall be used to the exte

prevent loss of the protection function in the eve

nonrandom, concurrent failures of redundant elemen

PUBLISHED VERSION (JULY 11, 1967)

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.

ALSO SEE PUBLISHED VERSION OF CRITERION 39 (PAGE 33)

REVISED VERSION (JULY 15

CRITERION 17 - ELECTRICAL POWER SYSTEMS

Onsite and offsite electrical power systems
sufficient capacity and capability to assure that
fuel damage limits and design conditions of the r
boundary are not exceeded during anticipated oper
(2) the core is cooled and containment integrity
are maintained following postulated accidents. T
for the onsite and offsite electrical power syste
provide sufficient capacity to permit functioning
and components important to safety. Offsite elec
provided to the site preferably by two physically
lines. The onsite system and the onsite portions
shall be designed with sufficient independency, r
to perform their safety function assuming failure
nent. Provisions shall be included to minimize t
offsite electrical power as a result of or coinci
electrical power generated by the nuclear power u

**PUBLISHED VERSION (JULY 11, 1967)**

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.

ALSO SEE PUBLISHED VERSION OF CRITERIA 19 AND 20 (PAGE 18)

**REVISED VERSION (JULY 15,**

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND

The protection system shall be designed for and inservice testability commensurate with the s performed. Redundancy and independence designed shall be sufficient to assure that (1) no single of the protection function and (2) removal from s channel does not result in loss of redundancy. M for testing the protection system when the reacto determine failures and losses of redundancy and i occurred.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 1**

**CRITERION 23 - PROTECTION SYSTEM FAILURE MO**

**The protection system shall be designe**
**or into a state demonstrated to be acceptab**
**basis if conditions such as disconnection o**
**(e.g., electric power, instrument air), or**
**ments (e.g., extreme heat or cold, fire, pr**
**radiation) are experienced.**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15,**

**CRITERION 26 - REACTIVITY CONTROL SYSTEM REDI**

**Two independent reactivity control syste**
**design principles, shall be provided. Each s**
**bility to control reactivity changes (includi**
**from planned, normal power changes without ex**
**damage limits. One of the systems shall be c**
**reactivity changes to assure that under condi**
**including anticipated operational occurrences**
**for malfunctions such as stuck rods, specifie**
**are not exceeded. One of the systems shall b**
**reactor core subcritical under cold condition**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15**

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR MALFUNCTIONS

The protection system shall be capable of pr single malfunction of the reactivity control syst withdrawal (not ejection or dropout) of control r soluble poison, without exceeding acceptable fuel

**PUBLISHED VERSION (JULY 11, 1967)**

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.

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.

**REVISED VERSION (JULY 15**

CRITERION 28 - REACTIVITY LIMITS FOR ACCIDE

The reactivity control systems shall b

limits on the potential amount and rate of

assure that the effects of postulated react

(1) result in damage to the reactor coolant

than limited local yielding nor (2) suffici

its support structures, or other reactor pr

to impair significantly the capability to c

reactivity accidents shall include consider

(unless prevented by positive means), rod d

coolant temperature and pressure, and cold

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15, 1**

**CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDA**

**The reactor coolant pressure boundary sha erected, and tested so as to have an extremely leakage, rapidly propagating failure, or gross**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15, 19**

CRITERION 31 - FRACTURE PREVENTION OF REACTOR

The fracture toughness properties and th
the reactor coolant pressure boundary shall a
under operating, testing, and postulated acci

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15**

CRITERION 32 - DESIGN OF COMPONENTS WITHIN REAC BOUNDARY

Components within the reactor coolant pres designed to permit periodic inspection and test and features, including an appropriate material the reactor pressure vessel, to assess their st integrity.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15,**

**LOSS-OF-COOLANT ACCIDENTS**

**Loss-of-coolant accidents mean those post from the loss of reactor coolant at a rate in the reactor coolant makeup system from any siz vessels, pumps, and valves connected to the re within the reactor coolant pressure boundary, in these components equivalent in size to the largest pipe of the reactor coolant system.**

ALSO COVERED BY REQUIRING INDIVIDU

VERSION OF CRITERIA 16, 17, 35, 38

33, 37, 43, 62 AND 63)

**PUBLISHED VERSION (JULY 11, 1967)**

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.

COVERED BY REQUIREMENTS ON INDIVIDU

VERSION OF CRITERIA 16, 17, 18, 35

(PAGES 11, 33, 58, 37, 38, 39, 43, 48

64, 40, 41, 45, 46, 44, 42, 65)

**PUBLISHED VERSION (JULY 11, 1967)**

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.

ALSO SEE PUBLISHED VERSION OF CRITERION 24 (PAGE 22)

**REVISED VERSION (JULY 15,**

**CRITERION 17 - ELECTRICAL POWER SYSTEMS**

**Onsite and offsite electrical power systen**
**sufficient capacity and capability to assure th**
**fuel damage limits and design conditions of the**
**boundary are not exceeded during anticipated op**
**(2) the core is cooled and containment integrit**
**are maintained following postulated accidents.**
**for the onsite and offsite electrical power sys**
**provide sufficient capacity to permit functioni**
**and components important to safety. Offsite el**
**provided to the site preferably by two physical**
**lines. The onsite system and the onsite portic**
**shall be designed with sufficient independency,**
**to perform their safety function assuming failı**
**nent. Provisions shall be included to minimize**
**offsite electrical power as a result of or coir**
**electrical power generated by the nuclear powe**

PUBLISHED VERSION (JULY 11, 1967)

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.

ALSO SEE PUBLISHED VERSION OF CRITERIA 42 AND 43 (PAGE 36)

REVISED VERSION (JULY 15,

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN

Structures, systems, and components import
designed to accommodate the effects of and to b
ronmental conditions associated with normal ope
postulated accidents. These structures, system
appropriately protected against dynamic effects
result from equipment failures and sources outs

**PUBLISHED VERSION (JULY 11, 1967)**

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.

COVERED BY REQUIREMENTS ON INDIVIDUAL SY

VERSION OF CRITERIA 17, 35, 38, 41, 44,

33, 37, 43, 62, 63, 44, 42, 65)

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**ALSO SEE PUBLISHED VERSION OF CRITERION 40 (PAGE 34)**

**REVISED VERSION (JULY 15,**

**CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGI**

**Structures, systems, and components impo:**
**designed to accommodate the effects of and to**
**ronmental conditions associated with normal o**
**postulated accidents. These structures, syst**
**appropriately protected against dynamic effec**
**result from equipment failures and sources ou**

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY 15, 1

CRITERION 35 - EMERGENCY CORE COOLING SYSTEM

A system to provide abundant emergency core through two system flow paths and by different de be provided. The system safety function shall be the reactor core following any loss of coolant ac that (1) fuel and clad damage that could interfer effective core cooling are prevented and (2) clad is limited to negligible amounts. The performanc be evaluated conservatively in each area of uncer

Redundancy in components and features, suita and leak detection, isolation, and containment ca provided to assure that for onsite and for offsit system operation the system safety function can b (1) failure of any single active component and (2 passive component unless it can be demonstrated t acceptable on some other defined basis.

PUBLISHED VERSION (JULY 11, 1967)

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all ·itical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

**REVISED VERSION (JULY 15,**

CRITERION 36 - DESIGN OF EMERGENCY CORE COOLING S'

**Components of the emergency core cooling sys to permit periodic inspection and testing of impo features, such as spray rings in the reactor pres injection nozzles, and piping to assure their str integrity and the full design capability of the s**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

CRITERION 37 - TESTING OF EMERGENCY CORE COO

The emergency core cooling system shall capability to test periodically (1) the oper performance of the active components of the s valves and (2) the operability of the system conditions as close to design as practicable, sequence that brings the system into operatio between normal and emergency power sources, a associated cooling water system.

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY 15, 19

CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, incl
penetrations, and any necessary containment
be designed so that the containment structur
ments can accommodate, without exceeding the
with an appropriate margin, the calculated p
conditions resulting from any loss-of-coolan
margin shall reflect consideration of (1) the
energy sources which have not been included i
peak conditions, such as energy in steam gene
metal water and other chemical reactions that
emergency core cooling functioning, (2) the p
of phenomena may be more severe than predicte
experience and experimental data available fo
ena and containment response.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15**

CRITERION 51 - FRACTURE PREVENTION OF CONTAINM

The fracture toughness properties and the of the reactor containment ferritic materials s behavior under operating, testing, and postulat

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 1**

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDA

Each line which is part of the reactor co and which penetrates primary reactor containme one isolation valve inside and one isolation v The valve outside of containment shall be loca as practicable. The primary mode for actuatio automatic and upon loss of actuating power the to fail safe.

Other appropriate requirements to minimiz consequences of an accidental rupture of these connected to them shall be provided as necessa safety. Determination of the appropriateness c such as higher quality in design, fabrication a provisions for inservice inspection, protectior natural phenomena, and additional isolation val shall include consideration of the population d acteristics, and physical characteristics of th

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 15**

CRITERION 38 - CONTAINMENT HEAT REMOVAL SYST

A system to remove heat from the reacto: through two system flow paths and by differe be provided. The system safety function sha consistent with the functioning of other ass containment pressure and temperature followi accident and maintain them at low levels.

Redundancy in components and features, and leak detection, isolation, and containme provided to assure that for onsite and for o system operation the system safety function assuming (1) failure of any single active co of any single passive component unless it ca system is acceptable on some other defined b

PUBLISHED VERSION (JULY 11, 1967)

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

ALSO SEE PUBLISHED VERSION OF CRITERION 57 (PAGE 47)

REVISED VERSION (JULY 15,

CRITERION 54 - SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor c
provided with leak detection, isolation, and cont
having redundancy, reliability, testability, and
ities which reflect the importance to safety of
systems. Such piping systems shall be designed w
test periodically the operability of the isolatio
apparatus and to determine if valve leakage is wi

ALSO COVERED BY REQUIREMENTS FOR IS

REVISED VERSION OF CRITERIA 55 AND

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY 15

CRITERION 52 - CAPABILITY FOR CONTAINMENT

The reactor containment and other equ to containment text conditions shall be de: integrated leakage rate testing can be conc pressure.

**PUBLISHED VERSION (JULY 11, 1967)**

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 leaktightness to be demonstrated at design pressure at any time.

**REVISED VERSION (JULY 15,**

**CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTIN(**

**The reactor containment shall have provisioı of all important areas including penetrations, (: materials surveillance program, and (3) for peri( leaktightness of penetrations which have resilieı bellows at containment design pressure.**

PUBLISHED VERSION (JULY 11, 1967)

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.

ALSO SEE PUBLISHED VERSION OF CRITERION 53 (PAGE 44)

**REVISED VERSION (JULY 15**

**CRITERION 54 - SYSTEMS PENETRATING CONTAINME**

**Piping systems penetrating primary reac provided with leak detection, isolation, and having redundancy, reliability, testability, ities hich reflect the importance to safet systems. Such piping systems shall be desig test periodically the operability of the iso apparatus and to determine if valve leakage**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

**CRITERION 39 - DESIGN OF CONTAINMENT HEAT RE**

**Components of the containment heat remo**
**to permit periodic inspection and testing of**
**tures, such as the torus, sumps, spray nozzl**
**their structural and leaktight integrity and**
**of the system.**

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JU**

**CRITERION 40 - TESTING OF CONTAINMENT HE**

**The containment heat removal system**
**capability to test periodically (1) the**
**performance of the active components of**
**valves and (2) the operability of the sy**
**conditions as close to the design as pra**
**sequence that brings the system into ope**
**between normal and emergency power sourc**
**ciated cooling water system.**

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY

CRITERION 42 - DESIGN OF CONTAINMENT ATMOSPHE COMPONENTS

Components of the containment atmosphere designed to permit periodic inspection of imp such as filter frames, ducts, and piping to a leaktight integrity and the full design capab

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY 1**

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE

The containment atmosphere cleanup systems

a capability to test periodically (1) the operab

performance of the active components of the syst

dampers, pumps, and valves and (2) the operabili

whole and, under conditions as close to design a

operational sequence that brings the systems int

the transfer between normal and emergency power

of associated systems.

**PUBLISHED VERSION (JULY 11, 1967)**

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.

**REVISED VERSION (JULY**

**CRITERION 62 - PREVENTION OF CRITICALITY IN F**

**Criticality in the fuel storage and hand**
**prevented by physical systems or processes, p**
**safe configurations.**

PUBLISHED VERSION (JULY 11, 1967)

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.

REVISED VERSION (JULY 1

CRITERION 60 - FUEL STORAGE AND HANDLING AND RAI

The fuel storage and handling and radioacti be designed to assure adequate safety under norn accident conditions. These systems shall be des significant reduction in fuel storage coolant ir conditions (2) with a decay heat removal capabil testability, and performance that reflect the in decay heat removal, (3) with suitable shielding (4) with a capability to permit inspection and t areas and features of the components of these sy appropriate containment, confinement, and filter

## PUBLISHED VERSION (JULY 11, 1967)

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 exceeding1y 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.

## REVISED VERSION (JULY :

CRITERION 61 - CONTROL OF RELEASES OF RADIO

The nuclear power unit design shall in

able control over gaseous, liquid, and soli

may be released from the unit during normal

operational occurrences, and postulated acc:

capacity shall be provided for retention of

effluents, particularly where unfavorable s:

can be expected to impose operational limit

radioactive effluents.

REVISED VERSION (JULY 15,

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear rea
necessary for electrical power generation and
and components required to prevent or mitigat
which could cause undue risk to the health an

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 syst reactor coolant system. For piping c penetrates primary reactor containmen to and includes the first containment side the containment capable of exter piping of these systems which contain which are normally closed during norm the boundary extends to and includes two valves (the second of which must actuation), whether or not the system primary reactor containment.

(c) Portions of the emergency core cooling the reactor coolant system. For pipir penetrates primary reactor containment and includes the first containment iso containment capable of external actuat system which does not penetrate primar the boundary extends to and includes t normally closed during normal reactor

CRITERION 15 - REACTOR COOLANT SYSTEM DESIG

The reactor coolant system and associa and protection systems shall be designed wi that the design conditions of the reactor c exceeded. The reactor coolant system and a assure these design conditions under all co including the effects of anticipated operat of power to the recirculation pumps, trippi isolation of the main condenser, and loss o

CRITERION 18 - INSPECTION AND TESTING OF ELEC

Electrical power systems shall be design
tion and testing of important areas and featu
connections, and switchboards to assess the c
the condition of their components. The syste
capability to test periodically (1) the opera
mance of the active components of the system,
relays, switches, and buses, and (2) the oper
whole and, under conditions as close to desig
operational sequence that brings the system i
transfer of power among the nuclear power uni
and the onsite power system.

CRITERION 27 - COMBINED REACTIVITY CONTROl

The reactivity control systems shall
reliably controlling reactivity changes tc
accident conditions the capability to cool

CRITERION 33 - REACTOR COOLANT MAKEUP SYST

A system to supply reactor coolant ma operation, preferably through two system f The system safety function shall be to ass fuel damage limits are not exceeded as a r leakage from the reactor coolant pressure piping within the boundary.

Redundancy in components and features and leak detection and isolation capabilit assure that for onsite and for offsite ele the system safety function can be accompli any single active component and (2) failur nent unless if can be demonstrated that th some other defined basis.

CRITERION 34 - DECAY HEAT REMOVAL SYSTEM

A system to remove decay heat, pref paths, shall be provided. The system sa transfer fission product decay heat and core when the reactor is shutdown at a r able fuel damage limits and the design coolant pressure boundary are not exceed

Redundancy in components and featur and leak detection and isolation capabil assure that for onsite and for offsite el the system safety function can be accompl any single active component and (2) failu nent unless it can be demonstrated that t some other defined basis.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLE/

Systems to control fission products, substances which may be released into the provided. The systems safety functions sh tent with the functioning of other associa and quantity of fission products released any postulated accident and (2) to control hydrogen, oxygen, and other substances in following any postulated accident to assur is maintained.

Each system shall have redundancy in suitable interconnections, and leak detect to assure that for onsite and for offsite tion its safety function can be accomplish single active component and (2) failure of unless it can be demonstrated that the sys defined basis.

CRITERION 44 - COOLING WATER SYSTEM

A system to transfer heat from struc important to safety, preferably through t ultimate heat sink shall be provided. Th be to transfer the combined heat load of components under normal operating and acc

Redundancy in components and feature and leak detection and isolation capabili required to assure that for onsite and fo system operation the system safety functi (1) failure of any single active componen passive component unless it can be demons acceptable on some other basis.

CRITERION 45 - DESIGN OF COOLING WATER SY!

Components of the cooling water syst( periodic inspection of important areas an( exchangers and piping, to assure their st: grity and the full design capability of tl

CRITERION 46 - TESTING OF COOLING WATER S)

The cooling water system shall be de: test periodically (1) the operability and active components of the system, such as p operability of the system as a whole and, design as practicable, and full operationa system into operation for reactor shutdowr accidents, including the transfer between sources.

CRITERION 56 - CONTAINMENT ATMOSPHERE ISO

Each line which connects directly to and penetrates primary reactor containmen isolation valves. One of these valves sh and shall be located as close to containm primary mode for actuation of the valves loss of actuating power these valves shal unless it can be demonstrated that the sy some other defined basis.

CRITERION 57 - CLOSED SYSTEMS ISOLATION VA

Each line which penetrates primary re neither part of the reactor coolant pressu directly to the containment atmosphere sha valve. This valve shall be outside of con as close to containment as practicable.

Kopp Exhibit 5

(5)

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 $k_\infty$ of the nominal fuel storage lattice cell and the changed $k_\infty$) 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 $k_\infty$ 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 | Acceptance Limit |
|---|---|
| **Elastic Analysis** | |
| 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** | Limits of XVII-4000 of Appendix XVII of ASME Code Section III |
| 1.7 (D + L) | |
| 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

Kopp Exhibit 6

NUREG–1430
Vol. 1, Rev. 1

# Standard Technical Specifications
# Babcock and Wilcox Plants

Specifications

**Issued by the**
**U.S. Nuclear Regulatory Commission**

**Office of Nuclear Reactor Regulation**

**April 1995**

# 4.0 DESIGN FEATURES

## 4.1 Site Location [Text description of site location.]

## 4.2 Reactor Core

### 4.2.1 Fuel Assemblies

The reactor shall contain [177] fuel assemblies. Each assembly shall consist of a matrix of [Zircalloy or ZIRLO] fuel rods with an initial composition of natural or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

### 4.2.2 CONTROL RODS

The reactor core shall contain [60] safety and regulating and [8] axial power shaping CONTROL RODS. The control material shall be [silver indium cadmium, boron carbide, or hafnium metal] as approved by the NRC.

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of [4.5] weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

(continued)

Kopp Exhibit 7

NUREG–1431
Vol. 1, Rev. 1

# Standard Technical Specifications
# Westinghouse Plants

Specifications

Issued by the
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# 4.0 DESIGN FEATURES

## 4.1 Site Location [Text description of site location.]

## 4.2 Reactor Core

### 4.2.1 Fuel Assemblies

The reactor shall contain [157] fuel assemblies. Each assembly shall consist of a matrix of [Zircalloy or ZIRLO] fuel rods with an initial composition of natural or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

### 4.2.2 [Control Rod] Assemblies

The reactor core shall contain [48] [control rod] assemblies. The control material shall be [silver indium cadmium, boron carbide, or hafnium metal] as approved by the NRC.

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of [4.5] weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

(continued)

Kopp Exhibit 8

NUREG-1432
Vol. 1, Rev. 1

# Standard Technical Specifications
# Combustion Engineering Plants

Specifications

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1995

# 4.0 DESIGN FEATURES

## 4.1 Site Location [Text description of the site location.]

## 4.2 Reactor Core

### 4.2.1 Fuel Assemblies

The reactor shall contain [217] fuel assemblies. Each assembly shall consist of a matrix of [Zircalloy or ZIRLO] fuel rods with an initial composition of natural or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

### 4.2.2 [Control Rod] Assemblies

The reactor core shall contain [91] control element assemblies (CEAs). The control material shall be [silver indium cadmium, boron carbide, or hafnium metal] as approved by the NRC.

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of [4.5] weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

(continued)

Kopp Exhibit 9

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped, but assuming that the highest worth assembly remains fully withdrawn and no changes in xenon or boron take place. The loss of control rod worth due to the material irradiation is negligible since only bank D may be in the core under normal operating conditions (near full power). The values given in Table 4.3.2-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies. Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, considering possible variations, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor ($K_{eff}$) of the fuel assembly array will be less than or equal to 0.95 as recommended in ANSI 57.2-1983. The following are the conditions that are assumed in meeting this design basis:

1. PWR fuel: The fuel assembly contains the highest enrichment authorized at its most reactive point in core life. No credit is taken for control rods. Refer to Section 9.1.2 for fuel storage rack parameters used in criticality calculations. Fuel parameters used in the criticality analysis of fuel irradiated at SHNPP are for the Westinghouse optimized 17 x 17 fuel design which is more reactive than the Westinghouse standard 17 x 17 fuel design of the same enrichment.

Since the spent fuel storage racks at SHNPP are identical to those at H. B. Robinson Unit 2 (HBR2), criticality analyses performed by Westinghouse for the HBR2 high density spent fuel racks (Reference 4.3.2-29) are applicable at SHNPP.

Fuel manufactured by Siemens Power Corporation is of the HTP design with parameters identified in Reference 4.3.2.31. The enrichment is less than 5.0% maximum. The design shall include natural blankets at 2.5" long at each end and it shall include gadolinia in the central ≥100" of the pellet stack. A "cutback" zone of ≤19.5" long may be between the ≥100" zone with gadolinia rods and the ≥2.5" natural zone at each end of the pellet stack. No gadolinia is required in the cutback zone. The central zone shall include at least four gadolinia rods with at least 1.8 wt% $Gd_2O_3$. The above fuel design limits are adequate to assure criticality safety for SPC HTP fuel design at SHNPP.

BWR fuel: The fuel assembly contains the highest enrichment authorized at its most reactive point in core life. No credit is taken for control rods or burnable poison. Refer to Section 9.1.2 for fuel storage rack parameters used in criticality calculations. Fuel parameters used in the criticality analysis are for General Electric (GE) 8 x 8R fuel design at 3.2 w/o U235. A study has been performed to confirm that other GE bundle designs currently stored at Brunswick (BSEP) are bounded by the analyzed 8 x 8R assembly at 3.2 w/o (Reference 4.3.2-28). From a reactivity standpoint, as measured by K-infinity, the existing criticality analysis conservatively bounds all fuel assemblies loaded in BSEP Unit 1 through reload 5 and all fuel assemblies loaded in BSEP Unit 2 through reload 6.

2. For flooded conditions, the moderator is pure water at the temperature within the design limits which yields the largest reactivity.

3. The array is infinite in lateral and axial extent and precludes any neutron leakage from the array for Westinghouse fuel. The SPC fuel is restricted by design parameters identified in Reference 4.3.2-31.

4. Mechanical uncertainties are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

5. Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption.

6. Where borated water is present, credit for the dissolved boron is not taken except under postulated accident conditions where the double contingency principle of ANSI N16.1-1975 is applied. This principle states that it shall require at least two unlikely, independent, and concurrent events to produce a criticality accident.

For fuel storage application, water is usually present. However, the design methodology also prevents accidental criticality when fuel assemblies are stored in the dry condition. For this case, possible sources of moderation such as those that could arise during fire fighting operations are included in the analysis.

The design method for the Westinghouse criticality analysis which insures the criticality safety of fuel assemblies outside the reactor uses the AMPX system of codes (Reference 4.3.2-12) for cross-section generation and KENO IV (Reference 4.3.2-13) for reactivity determination.

The cross-section library (Reference 4.3.2-11) that is the common starting point for all cross-sections has been generated from ENDF/B-V data. The NITAWL program (Reference 4.3.2-12) includes in this library the self-shielded resonance cross-sections that are appropriate for a particular geometry. The Nordheim Integral Treatment is used.

Energy and spatial weighting of cross-sections is performed by the XSDRNPM program (Reference 4.3.2-12) which is a one dimensional SN transport theory code. These multi-group cross-section sets are then used as input to KENO IV (Reference 4.3.2-13) which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. This benchmarking demonstrates that the calculational method is capable of determining the multiplication factor with an uncertainty less than 0.5 percent at a 95/95 percent probability/confidence level.

The criticality design criteria are met when the calculated effective multiplication factor ($k_{eff}$) described below for the PWR analysis is less than or equal to 0.95 for Westinghouse fuel designs:

$$K_{eff} = K_{worst} + B_{method} + B_{part} + \sqrt{[\ (ks)^2_{worst} + (ks)^2_{method} + (ks)^2rr]}$$

where:

$k_{worst}$ = worst case KENO $K_{eff}$ that includes material tolerances and mechanical tolerances which can result in spacings between assemblies less than nominal

$B_{method}$ = method bias determined from benchmark critical comparisons

$B_{part}$ = bias to account for poison particle self-shielding

$ks_{worst}$ = 95/95 uncertainty in the worst case KENO $K_{eff}$

$ks_{method}$ = 95/95 uncertainty in the method bias

$ks_{rr}$ = uncertainty in reactivity to account for enrichment and $K_{\infty}$ calculational uncertainties

It has been determined in Reference 4.3.2-26 that $K_{eff}$ will remain less than or equal to 0.95 as long as the maximum infinite core geometry lattice multiplication factor for a PWR assembly is less than or equal to 1.470 at 68°F. Specific credit for burnable absorbers integral with the fuel (i.e., gadolinia and boron-coated pellets) may be utilized when verifying that the multiplication factor remains below 1.470 at all times.

A criticality analysis performed at SPC for the HTP fuel design described above shows that the criticality design criteria of $K_{eff}$ being $\leq$ 0.95 is met when the K-infinity of the SPC fuel is less than 1.466. The codes, cross sections, and other data from SCALE 4.1 (Reference 4.3.2-32) were used for this analysis. The "CSAS25" option was used with 16-group cross sections. The codes executed, in sequence, are: DRIVE, CSAS25, BONAMI, NITAWL, AND KENO-VA.

The SCALE 4.1 system was developed for use by the USNRC and its licensees methodology validation was performed by modeling critical experiments of 4.31%-enriched assemblies (Reference 4.3.2-33) using the same methodology used in the SPC calculations.

Additional cases from Reference 4.3.2-34 were also modeled. All of the cases selected from Reference 4.3.2-34 employed Boral absorber plates between the fuel assemblies in a 3x3 array. These cases were selected to be as close as possible to the conditions modeled in this analysis and therefore provide the best estimate of the calculation bias. Calculation results are reported in Reference 4.3.2-31.

For the BWR analysis, the criticality design criteria are met when the calculated effective multiplication factor ($K_{eff}$) described is less than or equal to 0.95:

$$K_{eff} = K_{nominal} + B_{method} + B_{part} + [(ks_{nominal})^2 + (ks_{mech})^2 + (ks_{method})^2 + (ks_{mat})^2]^{1/2}$$

where:

| | | |
|---|---|---|
| $K_{nominal}$ | = | nominal case KENO $K_{eff}$ |
| $B_{method}$ | = | method bias determined from benchmark critical comparisons |
| $B_{part}$ | = | bias to account for poison particle self-shielding |
| $ks_{nominal}$ | = | 95/95 uncertainty in the nominal KENO $K_{eff}$ |
| $ks_{mech}$ | = | 95/95 uncertainty in the calculation of the bias due to construction tolerances |
| $ks_{method}$ | = | 95/95 uncertainty in the method bias |
| $ks_{mat}$ | = | 95/95 uncertainty associated with material thickness tolerances |

It has been determined in Reference 4.3.2-27 that $K_{eff}$ will remain less than or equal to 0.95 for BWR 8 x 8R fuel with a maximum lattice average enrichment of 3.2 w/o U235.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI 57.2-1983, "Design Objectives for LWR Fuel Storage Facilities at Nuclear Power Stations," Section 6.4.2; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety;" NRC Standard Review Plan, Section 9.1.2, "Fuel Storage;" and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

4.3.2.7 Stability.

4.3.2.7.1 Introduction. The stability of the pressurized water reactor cores against xenon-induced spatial oscillations and the control of such transients are discussed extensively in References 4.3.2-17, 4.3.2-18 and 4.3.2-19. A summary of these reports is given in the following discussion and the design bases are given in Section 4.3.1.6.

Kopp Exhibit 10

10

**CP&L**

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

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

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).

Surveillance interval? monthly

### 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.

Kopp Exhibit 11

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CP&L

R
Reference
Use

CAROLINA POWER & LIGHT COMPANY

NUCLEAR GENERATION GROUP

STANDARD PROCEDURE

VOLUME 99

BOOK/PART 99

# NFP - NGGC - 0003

# *PROCEDURE FOR SELECTION OF IRRADIATED FUEL FOR SHIPMENT IN THE IF-300 SPENT FUEL CASK*

REVISION 4

CPL 00470001

TABLE OF CONTENTS



CPL 00470002

## 1.0 PURPOSE

1.1 The purpose of this procedure is to assure that the requirements of the IF-300 Cask Certificate of Compliance are met with regards 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.

1.2 This procedure shall be used to document that the spent fuel selected for shipment meets the requirements of the IF-300 Cask Certificate of Compliance No.9001 and the Harris fuel pool license requirements prior to loading in the IF-300 series Irradiated Fuel Shipping Cask for offsite shipment. Both PWR and BWR shipments are addressed.

## 2.0 REFERENCES

2.1 Developmental

1. 10CFR71, "Packaging and Transportation of Radioactive Material"
2. MAGIC - Code Description and Users manual - NA10, File: NF-2189.002
3. ORIDATA - Code and Users Manual - NA11, File:NF-2190.004
4. BWR 7x7, 8x8S and 8x8R K-infinity Calculations: NFS Design Activity - 89.0018 ; File : NF-1489.0018
5. Criticality Analysis of Shearon Harris Spent Fuel Racks , January 1987 File: NF-1084.02

2.2 Implementing

R1 1. Certificate of Compliance for Radioactive Materials Packages, Model No. IF-300, Certificate No.9001 (NGGM-PM-0009)

R2 2. IF-300 Shipping Cask Consolidated Safety Analysis Report - NEDO 10084-4 (NGGM-PM-0009)

R3 3. Harris Nuclear Plant - Final Safety Analysis Report

R4 4. Harris ESR 97-00152, "Cask Closure Head Analysis Owner's Review"

R5 5. Robinson ESR 97-00191, "Perform Offsite Dose Calculation for Spent Fuel Shipping Cask"

CPL 00470003

## 3.0 DEFINITIONS

R1 **3.1 Certificate of Compliance**

Documentation issued by the NRC certifying that the IF-300 cask, with approved contents, meet the applicable safety standards as stated in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

R2 **3.2 Consolidated Safety Analysis Report**

The IF-300 Consolidated Safety Analysis Report ( CSAR ) NEDO 10084-4 represents the technical basis for the IF-300 Certificate of Compliance Number 9001.

## 4.0 RESPONSIBILITIES

4.1 Supervisor - Spent Fuel Management Sub-Unit (HNP) is responsible for:

- Ensuring that the pertinent requirements in the current revision of the IF-300 Certificate of Compliance (i.e., any changes affecting fuel selection) are incorporated into this procedure.

- Ensuring that pertinent changes to the IF-300 CSAR and the Harris FSAR are incorporated into this procedure.

- Analyzing new fuel types for shipment in the IF-300 and for updating this procedure and the HNP FSAR to reflect the new fuel types available for shipment.

4.2 Principle Engineer - Spent Fuel Management Sub-Unit (HNP) is responsible for:

- The two year review of this procedure.

4.3 BNP - Reactor Systems is responsible for:

- Selecting fuel for shipment per this procedure.

- Preparing and verifying Attachment 1, the appropriate Irradiated Fuel Data Sheets and Cask Loading Diagrams.

- Providing the documentation of the fuel selection to the plant Shipment Director.

CPL 00470004

4.4 RNP - Reactor Systems or SFM Sub-Unit (HNP) is responsible for:

- Selecting fuel for shipment per this procedure.
- Preparing and verifying Attachment 1, the appropriate Irradiated Fuel Data Sheets and Cask Loading Diagrams.
- Providing the documentation of the fuel selection to the plant Shipment Director.

4.5 Plant Shipment Director is responsible for:

- Receiving the fuel selection documentation and is responsible for transmitting the data to E&RC for purposes of advance notification and to the site Nuclear Materials Custodian.

## 5.0 PREREQUISITES

5.1 The plant Shipment Director has notified BNP - Reactor Systems, RNP - Reactor Systems, or SFM Sub-Unit (HNP) ( as appropriate ) that a fuel shipment is planned and has provided an approximate shipping date, allowing sufficient advance notice for the fuel selection preparation and verification.

5.2 The ORIDATA code has been installed on a PC with access to the MAGIC code database.

## 6.0 PRECAUTIONS AND LIMITATIONS

R1 6.1 All fuel assemblies selected shall meet the requirements of the Certificate of Compliance No. 9001.

R1 6.2 Fuel assemblies with known leaking fuel rods shall not be shipped.

R1&2 6.3 Per Section A-3.1.1 of the IF-300 CSAR (Ref. 2.2 ) any BWR fuel shipped in the Channeled BWR Fuel Basket (17 bundle basket) must be cooled a minimum of 3 years from discharge prior to shipment as well as meeting the cask 40,000 Btu/hour limit in the Certificate of Compliance.

R2 6.4 Per Section 5(b)(1)(i) of the IF-300 CSAR (Ref. 2.2) any BWR fuel shipped in the BWR Fuel Basket (18 bundle basket) for unchanneled BWR fuel has a 120 day minimum time based cooling requirement and must meet the decay heat limit.

R1 6.5 Per Section 10 of the IF-300 Certificate of Compliance - No.9001 (Ref. 2.1) any Robinson PWR fuel to be shipped with a burnup greater than 35 GWD/MTU (but less than 45 GWD/MTU) must be cooled a minimum of

5 years from discharge prior to shipment and have an initial uranium loading of less than or equal to 439 kilograms.

R3 6.6 Per Sections 4.3.2.6, 9.1.1.3 and 9.1.2.3 of the Harris FSAR, only Brunswick Unit 1 fuel from the Initial Core through Reload 5 (i.e, Batch 9) and Brunswick Unit 2 fuel from the Initial Core through Reload 6 (i.e, Batch 11) may be safely stored in the Harris BWR fuel racks based on criticality concerns.

R3 6.7 Per Section 4.3.2.6 of the Harris FSAR, any Robinson fuel with up to a 4.2 w/o enrichment may be safely stored in the Harris PWR racks.

R2&3 6.8 Attachment 2 identifies fuel which has been analyzed (Ref. 2.6 and 2.7) as meeting the Harris FSAR criticality concern and therefore acceptable for storage at Harris.

R1 6.9 In addition to meeting any cooling time requirements that may be specified in the Certificate of Compliance (Ref. 2.1), the decay heat limits specified in the Certificate must also be met.

R2 6.10 Only BWR fuel <u>with channels installed</u> shall be shipped in the 17 bundle capacity basket.

R2 6.11 Only BWR fuel <u>without</u> channels installed shall be shipped in the 18 bundle capacity basket.

6.12 Robinson fuel located at Brunswick may only be shipped after review and written approval by the SFM Sub-Unit (HNP).

R4&5 6.13 The 2.5 year minimum cooling time for Robinson fuel has been added to Attachment 1, Fuel Acceptance Form, to comply with cooling times used in the Harris ESR 97-00152 and Robinson ESR 97-00191.

## 7.0 SPECIAL TOOLS AND EQUIPMENT

7.1 PC with the ORIDATA code installed and access to the appropriate MAGIC code database.

## 8.0 ACCEPTANCE CRITERIA

R1,2 &3 8.1 All fuel assemblies selected shall meet the requirements of the Certificate of Compliance No. 9001, the IF-300 CSAR and the Harris FSAR as specified in this procedure. This acceptability for shipment and storage at Harris shall be documented on Attachment 1.

CPL 00470006

## 9.0 INSTRUCTIONS

### 9.1 Selection of Fuel Assemblies

9.1.1 Obtain a listing of the available fuel assemblies currently in storage in the spent fuel pool from the MAGIC code database.

| R1 NOTE: | BWR or PWR assemblies with known leaking fuel rods, missing fuel rods ( ie: empty rod locations ) or questionable structural damage are not to be shipped and are listed on Attachment 3 . |
|---|---|

9.1.2 Select the appropriate number of fuel assemblies to be shipped using the MAGIC listing and the listing of Acceptable Fuel For Shipment - Attachment 2.

9.1.3 From the ORIDATA Main Menu select the Shipping Option.

9.1.4 From the Shipping Menu select the Enter/edit components option.

9.1.5 Enter the desired decay date and the selected fuel assemblies and alternates into ORIDATA using their respective MAGIC fuel assembly ID.

9.1.6 After entering the data return to the Shipping Menu and select the Calculate/report activity option.

| NOTE: | The IFDS may be filled out by hand instead of using ORIDATA. However, the listed Values and Certificate of Compliance Requirements shall be verified as being correct for the fuel type selected. |
|---|---|

9.1.7 From the Calculation/Report Menu select the Print irradiated fuel data sheet option.

| NOTE: | IFDS fuel type template files are listed in Attachment 2 along with the acceptable fuel for shipment. The ORIDATA fuel type template files may be modified as needed. Any changes shall be verified against the requirements of the Certificate of Compliance. Attachment 4 is a sample template used for 7x7 BWR assemblies and Attachment 5 is a sample template for 8x8 BWR assemblies. Attachment 6 is an example used for 15x15 PWR assemblies. |
|---|---|

CPL 00470007

## 9.1 Selection of Fuel Assemblies

9.1.8 Enter the appropriate fuel type template file name when prompted by ORIDATA to generate the Irradiated Fuel Data Sheet or alternately generate one using Attachment 4, 5, 6, or 7 as appropriate. Return to Calculation/Report Menu.

9.1.9 From the Calculation/Report Menu select the Print summary shipping report option. Return to Shipping Menu.

9.1.10 On the IFDS record the Reactor name from which the fuel is being shipped and the Cask ID. No. for the cask the fuel is to be loaded into.

9.1.11 Assign an IFDS No. to each IFDS and also place the IFDS No. in the upper right hand corner of each page of the Summary report. The IFDS No. is to be assigned using the following format:

Year, Plant, Cask ID, # (shipment no.)
Where:
Year = 96 for 1996
Plant = B1 for Brunswick 1
B2 for Brunswick 2
R2 for Robinson 2
Cask ID = 3 for IF-303
4 for IF-304
# = sequential shipment no. for each cask in the current year

**EXAMPLE:** IFDS No. 96R24#05 would mean:

1996, Robinson 2, IF-304, Shipment No. 5

9.1.12 A Cask Loading Diagram ( CLD ) will be completed for each cask to be loaded. The placement of the fuel into each cask cell should correspond to the same order in which the assemblies were input into ORIDATA.

**EXAMPLE:** CLD Cell No. 1 should be the first assembly entered into ORIDATA, CLD Cell No. 2 should be the second assembly entered into ORIDATA, etc.

CPL 00470008

### 9.1 Selection of Fuel Assemblies

9.1.13 The IFDS Shipment No. will be put on the CLD to identify which IFDS the CLD is to be used with. ( Attachments 8 - 10 show the CLD for the 18 bundle basket for unchanneled BWR fuel, the 17 bundle basket for channeled BWR fuel and the 7 assembly PWR basket, respectively. )

**EXAMPLE:**

| | |
|---|---|
| Fuel Acceptable: | Value "Acceptable" or "Unacceptable per review of Attachment 4, IFDS values vs. C of C requirements. |
| 17 BWR . . . | Value of Decay Time (days) from Attach. 4, IFDS, Channeled ? |
| 18 BWR . . . | Value of Decay Time (days) from Attach. 4, IFDS, Channeled ? |
| RNP PWR . . . | a) Value of Decay Time (days) from Attach. 4, IFDS |
| Harris FSAR . . . | a) Bundle Reload per Attach. 2<br>b) Bundle Reload per Attach. 3 |
| Harris FSAR PWR . . . | Value of max. enrichment listed on Isotopic Summary Report, page 2 |

R1,2 &3 9.1.14 The Fuel Acceptance Form -Attachment 1, shall be completed for each cask and verified indicating that the selected fuel complies with the cask Certificate of Compliance, the IF-300 CSAR, the Robinson UFSAR, and the Harris FSAR. The appropriate IFDS No. corresponding to the IFDS being verified is to be placed on the Fuel Acceptance Form.

9.1.15 Steps 9.1.2 through 9.1.14 will be completed for each spent fuel shipping cask to be loaded.

### 9.2 Use of Alternate Fuel Assemblies

9.2.1 Upon notification of the use of an alternate assembly, generate a Revised Fuel Acceptance Form, IFDS and Cask Loading Diagram by repeating the Steps 9.1.2 through 9.1.14.

## 10. RECORDS

10.1 A completed Fuel Acceptance Form (Attachment 1) along with the corresponding Irradiated Fuel Data Sheet (IFDS) and the appropriate summary tables from ORIDATA and a Cask Loading Diagram shall be prepared for each PWR and BWR fuel shipment. The above shall be retained per the appropriate site procedures for the retention of QA documents.

CPL 00470009

## 10. RECORDS

10.2 The completed Fuel Acceptance Form ,IFDS ( including the attached summary tables) and Cask Loading Diagram( CLD) shall be transmitted to the Shipment Director at both the shipping and receiving plants and the Principal Engineer - SFM Sub-Unit (HNP).

CPL 00470010

**FUEL ACCEPTANCE FORM**

IFDS SHIPMENT No.____________

| ATTRIBUTE | REQUIRED VALUE | VALUE | REFERENCE |
|---|---|---|---|
| Fuel Acceptable per IF-300 Cert. of Compl. | IFDS Values Acceptable | | IF-300 Cert. of Compliance - Cert. No.9001 |
| 17 BWR Assembly Basket<br>a) Cool Time<br>b) Fuel Channeled | Each Assembly:<br>a) ≥ 3 Years Cool Time<br>b) Yes | a)<br>b) | IF-300 CSAR -<br>a) Sect. A-3.1.1<br>b) Sect. A-1.2.3 |
| 18 BWR Assembly Basket<br>a) Cool Time<br>b) Fuel Channeled | Each Assembly:<br>a) 120 day min. Cool Time<br>b) No | a)<br>b) | IF-300 CSAR -<br>a)Sect. 5(b)(1)(i)<br>b)Sect. 3.4 |
| RNP PWR Fuel<br>a) Cool Time | Each Assembly<br>a) ≥ 2.5 Years Cool Time | a) | HNP ESR 97-00152<br>RNP ESR 97-00191<br>Robinson UFSAR and Harris FSAR-<br>Sections 15.7.5 |
| Harris FSAR BWR Racks-Allowable Offsite Fuel | a) B1 IC thru Reload 5<br>b) B2 IC thru Reload 6 | a)<br>b) | Harris FSAR -<br>Section 4.3.2.6<br>Section 9.1.1.3<br>Section 9.1.2.3 |
| Harris FSAR PWR Racks - Allowable Offsite Fuel | All 15x15 ≤ 4.2 w/o<br>( A thru Y-series ID ) | | Harris FSAR -<br>Section 4.3.2.6 |

FUEL ACCEPTANCE FORM PREPARED BY:

_______________________________________ DATE: ________

FUEL ACCEPTANCE FORM VERIFIED BY:

_______________________________________ DATE: ________

(Form NFP-NGGC-0003-1-4)

CPL 00470011

**ACCEPTABLE FUEL FOR SHIPMENT**

BRUNSWICK UNIT 1 FUEL

Note: Refer to Attachment 3 for a list of Unacceptable Fuel prior to final selection

| Fuel Region ID | Fuel Serial No. ID | IFDS Fuel Type File |
|---|---|---|
| B1 - Initial Core, Batches 1-4 | LJ0196 - LJ0755 | GE8x8 |
| B1 - Reload 1, Batch 5 | LJB642 - LJB649<br>LJD584 - LJD751 | GEP8x8R |
| B1 - Reload 2, Batch 6 | LJM295 - LJM450 | GEP8x8R |
| B1 - Reload 3, Batch 7 | LJZ667 - LJZ810<br>LY3965 - LY4000 | GEP8x8R |
| B1 - Reload 4, Batch 8 | LY9020 - LY9203 | GEP8x8R |
| B1 - Reload 5, Batch 9 | LYG461 - LYG636 | GEP8x8R |

BRUNSWICK UNIT 2 FUEL

Note: Refer to Attachment 3 for a list of Unacceptable Fuel prior to final selection

| Fuel Region ID | Fuel Serial No. ID | IFDS Fuel Type File |
|---|---|---|
| B2 - Initial Core Batches 1-5 | BR001 - BR560<br>GED007, GED012<br>GED014, GED042 | GE7x7 |
| B2 - Reload 1, Batch 6 | LJ6326 - LJ6465 | GE8x8 |
| B2 - Reload 2, Batch 7 | LJB146 - LJB277 | GEP8x8R |
| B2 - Reload 3, Batch 8 | LJL874 - LJL999<br>LJM001 - LJM006 | GEP8x8R |
| B2 - Reload 4, Batch 9 | LJX476 - LJX611<br>LY1853 - LY1876 | GEP8x8R |
| B2 - Reload 5, Batch 10 | LY7029 - LY7212 | GEP8x8R |
| B2 - Reload 6, Batch 11 | LYE325 - LYE472 | GEP8x8R |

CPL 00470012

## ACCEPTABLE FUEL FOR SHIPMENT

ROBINSON UNIT 2

Note: Refer to Attachment 3 for a list of Unacceptable Fuel prior to final selection

| Fuel Region ID | Fuel Serial No. ID | IFDS Fuel Type File |
|---|---|---|
| R2 Initial Core Batches 1-4 | A01 - A53; B01 - B52<br>C01 - C52 | W15x15 |
| R2 Reload 1, Batch 5 | D01 - D53 | W15x15 |
| R2 Reload 2, Batch 6 | E01 - E52; F01 - F52 | W15x15 |
| R2 Reload 3, Batch 7 | UD10-G01 - UD10-G52 | EX15x15 |
| R2 Reload 4, Batch 8 | UD10-H01 - UD10-H52 | EX15x15 |
| R2 Reload 5, Batch 9 | UD10-J01 - UD10-J53 | EX15x15 |
| R2 Reload 6, Batch 10 | UD10-K01 - UD10-K52 | EX15x15 |
| R2 Reload 7, Batch 11 | UD10-L01 - UD10-L52 | EX15x15 |
| R2 Reload 8, Batch 12 | UD10-M01 - UD10-M17<br>UD10-M26 - UD10-M52 | EX15x15 |
| R2 Reload 9, Batch 13 | UD10-M18 - UD10-M25<br>UD10-M53<br>UD10-N01 - UD10-N56 | EX15x15 |
| R2 Reload 10, Batch 14 | UD10-P01 - UD10-P48 | EX15x15 |
| R2 Reload 11, Batch 15 | UD10-S01 - UD10-S48 | EX15x15 |
| R2 Reload 12, Batch 16 | UD10-T01 - UD10-T48 | ANF15x15 |

CPL 00470013

## FUEL UNACCEPTABLE FOR SHIPMENT

BRUNSWICK UNIT 1 FUEL

| Fuel Region ID | Fuel Serial No. ID |
|---|---|
| B1 - Initial Core, Batches 1-4 | LJ0292, LJ0323, LJ0351, LJ0500 LJ0630 |
| B1 - Reload 1, Batch 5 | LJD610, LJD659, LJD718 |
| B1 - Reload 2, Batch 6 | LJM317, LJM330, LJM334, LJM351, LJM358, LJM403, LJM426, LJM431 |
| B1 - Reload 3, Batch 7 | LY3971, LY3977, LY3980, LY3995 |
| B1 - Reload 4, Batch 8 | LY9181, LY9194 |
| B1 - Reload 5, Batch 9 | LYG475,LYG491, LYG563, LYG577, LYG612 |
| B1 - Reload 6, Batch 10 | LYL717 thru LYL900 (See Sect. 6.6) |
| B1 - Reload 7, Batch 11 | LYV333 thru LYV456 (See Sect. 6.6)<br>LYV962 thru LYV997 (See Sect. 6.6) |
| B1 - Reload 8, Batch 12 | YJ1888 thru YJ1995 (See Sect. 6.6)<br>YJ2004 thru YJ2019 (See Sect. 6.6)<br>YJ2013 |
| B1 - Reload 9, Batch 13 | YJB757 thru YJB912 (See Sect. 6.6)<br>YJB787, YJB806, YJB856 |
| B1 - Reload 10, Batch 14 | YJG573 thru YJG772 (See Sect. 6.6) |

CPL 00470014

**FUEL UNACCEPTABLE FOR SHIPMENT**

BRUNSWICK UNIT 2 FUEL

| Fuel Region ID | Fuel Serial No. ID |
|---|---|
| B2 - Initial Core, Batches 1-5 | BR 081, BR 081R, BR 125, BR 128, BR 131, BR 132, BR 135, BR 138, BR 139, BR 144, BR 148, BR 164, BR 165, BR 166, BR 173, BR 176, BR 179, BR 184, BR 187, BR 190, BR 191, BR 193, BR 204, BR 205, BR 217, BR 219, BR 222, BR 250, BR 251, BR 261, BR 263, BR 265, BR 267, BR 270, BR 273, BR 277, BR 285, BR 286, BR 298, BR 301, BR 307, BR 326, BR 330, BR 358, BR 394, BR 404, BR 406, BR 433, BR 444, BR 463, BR 480, BR 484, BR 485, BR 486, BR 491, BR 492, BR 498, BR 540, BR 551,GED007 |
| B2 - Reload 1, Batch 6 | LJ6352, LJ6413, LJ6421, LJ6451 |
| B2 - Reload 2, Batch 7 | LJB197, LJB250 |
| B2 - Reload 3, Batch 8 | LJL894, LJL904,LJL981 |
| B2 - Reload 4, Batch 9 | LJX491, LJX514, LJX515 |
| B2 - Reload 5, Batch 10 | LY7060, LY7063, LY7070, LY7073, LY7081, LY7101, LY7136, LY7150, LY7168, LY7171, LY7174, LY7178, LY7181, LY7204 |
| B2 - Reload 6, Batch 11 | LYE325 thru LYE472 (See Sect. 6.6) |
| B2 - Reload 7, Batch 12 | LYJ748 thru LYJ931 (See Sect. 6.6)<br>LYJ855 |
| B2 - Reload 8, Batch 13 | LYS778 thru LYS945 (See Sect. 6.6) |
| B2 - Reload 9, Batch 14 | YJ0001 thru YJ0148 (See Sect. 6..6) |
| B2 - Reload 10, Batch 15 | YJ1996 thru YJ2003 (See Sect. 6.6)<br>YJ6939 thru YJ7050 (See Sect. 6.6)<br>YJ8587 thru YJ8618 (See Sect. 6.6) |

CPL 00470015

**FUEL UNACCEPTABLE FOR SHIPMENT**

BRUNSWICK UNIT 2 FUEL (CONT.)

| Fuel Region ID | Fuel Serial No. ID |
|---|---|
| B2 - Reload 11, Batch 16 | YJE377 thru YJE576 (See Sect. 6.6) |

ROBINSON UNIT 2 FUEL

**NOTE:** See Section 6.9 regarding Robinson spent fuel located at Brunswick.

| Fuel Region ID | Fuel Serial No. ID |
|---|---|
| R2 Initial Core, Batches 1-4 | A01 - A53<br>B01 - B52<br>C01 - C52 |
| R2 Reload 1, Batch 5 | |
| R2 Reload 2, Batch 6 | |
| R2 Reload 3, Batch 7 | UD10-G19, UD10-G20, UD10-G38 |
| R2 Reload 4, Batch 8 | UD10-H24 |
| R2 Reload 5, Batch 9 | UD10-J17 |
| R2 Reload 6, Batch 10 | UD10-K16, UD10-K29 |
| R2 Reload 7, Batch 11 | |
| R2 Reload 8, Batch 12 | UD10-M01 |
| R2 Reload 9, Batch 13 | UD10-N09, UD10-N23 |
| R2 Reload 10, Batch 14 | UD10-P12 thru UD10-P17, UD10-P26, UD10-P27 (These have burnups >45 GWD/MTU) |
| R2 Reload 11, Batch 15 | UD10-S15, UD10-S15H; UD10-S25 thru UD10-S32 (S25 thru S32 have burnup > 45 GWD/MTU) |

CPL 00470016

**FUEL UNACCEPTABLE FOR SHIPMENT**

ROBINSON UNIT 2 FUEL (CONT.)

| Fuel Region ID | Fuel Serial No. ID |
|---|---|
| R2 Reload 12, Batch 16 | UD10-T13, UD10-T15, UD10-T18, UD10-T20, UD10-T26 [All except T26 have burnup > 45 GwD/MTU] |

CPL 00470017

Attachment 4
Page 1 of 1

**IRRADIATED FUEL DATA SHEET**
**IF-300 - 7X7 BWR**

IFDS No.________________ Bundle I.D. Numbers
Reactor ______________________________
**Date of Discharge______________________________
Date of Decay Calculation______________________________
Cask ID No.______________________________

| FUEL DATA | VALUE | C of C REQUIREMENTS BWR |
|---|---|---|
| Form | | Clad U02 Pellets |
| Cladding | | Zr or SS |
| * Initial U (kg/Bundle) | | 198 max |
| * Initial Enrichment (w/o) | | 4.0 max |
| * Bundle Cross Section (in.) | | 5.75 max |
| Fuel Pin Array | | 7 x 7 |
| Fuel Pin Diameter (in.) | | 0.500-0.600 |
| Fuel Pin Pitch (in.) | | 0.647-0.809 |
| Active Fuel Length (in.) | | 146 max |
| ** Decay Time (Days) | | 120 minimum |
| * Weight Per Bundle (lbs) | | 840 max |
| * Decay Heat Per Bundle (BTU/hr) | | 2225 max |
| Decay Heat Per Shipment (BTU/hr) | | 40000 max |
| * Specific Power (kw/kgU) | | 40 max |
| * Burnup (MWD/MTU) | | 35000 max |
| Number of Known Failed Fuel Rods | | 0 |
| End of Life gas content - Meets C of C requirement (Y/N) | | 0.50 lb moles max per shipment |
| Comments: | | |

IFDS Completed By: ____________________________/________________
Date

IFDS Verified By:____________________________/________________
Date

* Enter Largest Assembly Value Per Shipment
** Enter Smallest Assembly Value Per Shipment

CPL 00470018

# IRRADIATED FUEL DATA SHEET
# IF-300 - 8X8 BWR

IFDS No.______________ Bundle I.D. Numbers
Reactor ______________
**Date of Discharge______________
Date of Decay Calculation______________
Cask ID No. ______________

| FUEL DATA | VALUE | C of C REQUIREMENTS BWR |
|---|---|---|
| Form | | Clad U02 Pellets |
| Cladding | | Zr or SS |
| * Initial U (kg/Bundle) | | 198 max |
| * Initial Enrichment (w/o) | | 4.0 max |
| * Bundle Cross Section (in.) | | 5.75 max |
| Fuel Pin Array | | 8 x 8 |
| Fuel Pin Diameter (in.) | | 0.475-0.505 |
| Fuel Pin Pitch (in.) | | 0.630-0.645 |
| Active Fuel Length (in.) | | 150 max |
| ** Decay Time (Days) | | 120 minimum |
| * Weight Per Bundle (lbs) | | 840 max |
| * Decay Heat Per Bundle (BTU/hr) | | 2225 max |
| * Decay Heat Per Shipment (BTU/hr) | | 40000 max |
| * Specific Power (kw/kgU) | | 40 max |
| * Burnup (MWD/MTU) | | 35000 max |
| Number of Known Failed Fuel Rods | | 0 |
| End of Life gas content - Meets C of C requirement (Y/N) | | 0.50 lb moles max per shipment |

Comments:

IFDS Completed By: ______________ / ______________
Date

IFDS Verified By: ______________ / ______________
Date

* Enter Largest Assembly Value Per Shipment
** Enter Smallest Assembly Value Per Shipment

CPL 00470019

# IRRADIATED FUEL DATA SHEET
# IF-300 - 15X15 PWR [BURNUP (MWD/MTU) <= 35,000]

IFDS No._______________ Bundle I.D. Numbers
Reactor _______________
**Date of Discharge_______________
Date of Decay Calculation_______________
Cask ID No. _______________

| FUEL DATA | VALUE | C of C REQUIREMENTS PWR |
|---|---|---|
| Form | | Clad U02 Pellets |
| Cladding | | Zr or SS |
| * Initial U (kg/Bundle) | | 465 max |
| * Initial Enrichment (w/o) | | 4.0 max |
| * Bundle Cross Section (in.) | | 8.75 max |
| Fuel Pin Array | | 15 x 15 |
| Fuel Pin Diameter (in.) | | 0.380-0.460 |
| Fuel Pin Pitch (in.) | | 0.502-0.582 |
| Active Fuel Length (in.) | | 145 max |
| ** Decay Time (Days) | | 120 minimum |
| * Weight Per Bundle (lbs) | | 2300 max |
| * Decay Heat Per Bundle (BTU/hr) | | 5725 max |
| Decay Heat Per Shipment (BTU/hr) | | 40000 max |
| * Specific Power (kw/kgU) | | 40 max |
| * Burnup (MWD/MTU) | | 35000 max |
| Number of Known Failed Fuel Rods | | 0 |
| End of Life gas content - Meets C of C requirement (Y/N) | | 0.50 lb moles max per shipment |
| Comments: | | |

IFDS Completed By: _______________/__________
Date

IFDS Verified By: _______________/__________
Date

* Enter Largest Assembly Value Per Shipment
** Enter Smallest Assembly Value Per Shipment

CPL 00470020

# IRRADIATED FUEL DATA SHEET
## IF-300 - 15X15 PWR [35,000 > BURNUP (MWD/MTU) <= 45,000]

IFDS No.________________ Bundle I.D. Numbers
Reactor ________________
**Date of Discharge________________
Date of Decay Calculation________________
Cask ID No. ________________

| FUEL DATA | VALUE | C of C REQUIREMENTS PWR |
|---|---|---|
| Form | | Clad U02 Pellets |
| Cladding | | Zr or SS |
| * Initial U (kg/Bundle) | | 439 max |
| * Initial Enrichment (w/o) | | 4.0 max |
| * Bundle Cross Section (in.) | | 8.75 max |
| Fuel Pin Array | | 15 x 15 |
| Fuel Pin Diameter (in.) | | 0.380-0.460 |
| Fuel Pin Pitch (in.) | | 0.502-0.582 |
| Active Fuel Length (in.) | | 145 max |
| ** Decay Time (Days) | | 1826 minimum |
| * Weight Per Bundle (lbs) | | 2300 max |
| * Decay Heat Per Bundle (BTU/hr) | | 5725 max |
| Decay Heat Per Shipment (BTU/hr) | | 40000 max |
| * Specific Power (kw/kgU) | | 40 max |
| * Burnup (MWD/MTU) | | 45000 max |
| Number of Known Failed Fuel Rods | | 0 |
| End of Life gas content - Meets C of C requirement (Y/N) | | 0.50 lb moles max per shipment |
| Comments: | | |

IFDS Completed By: ________________/________________
Date

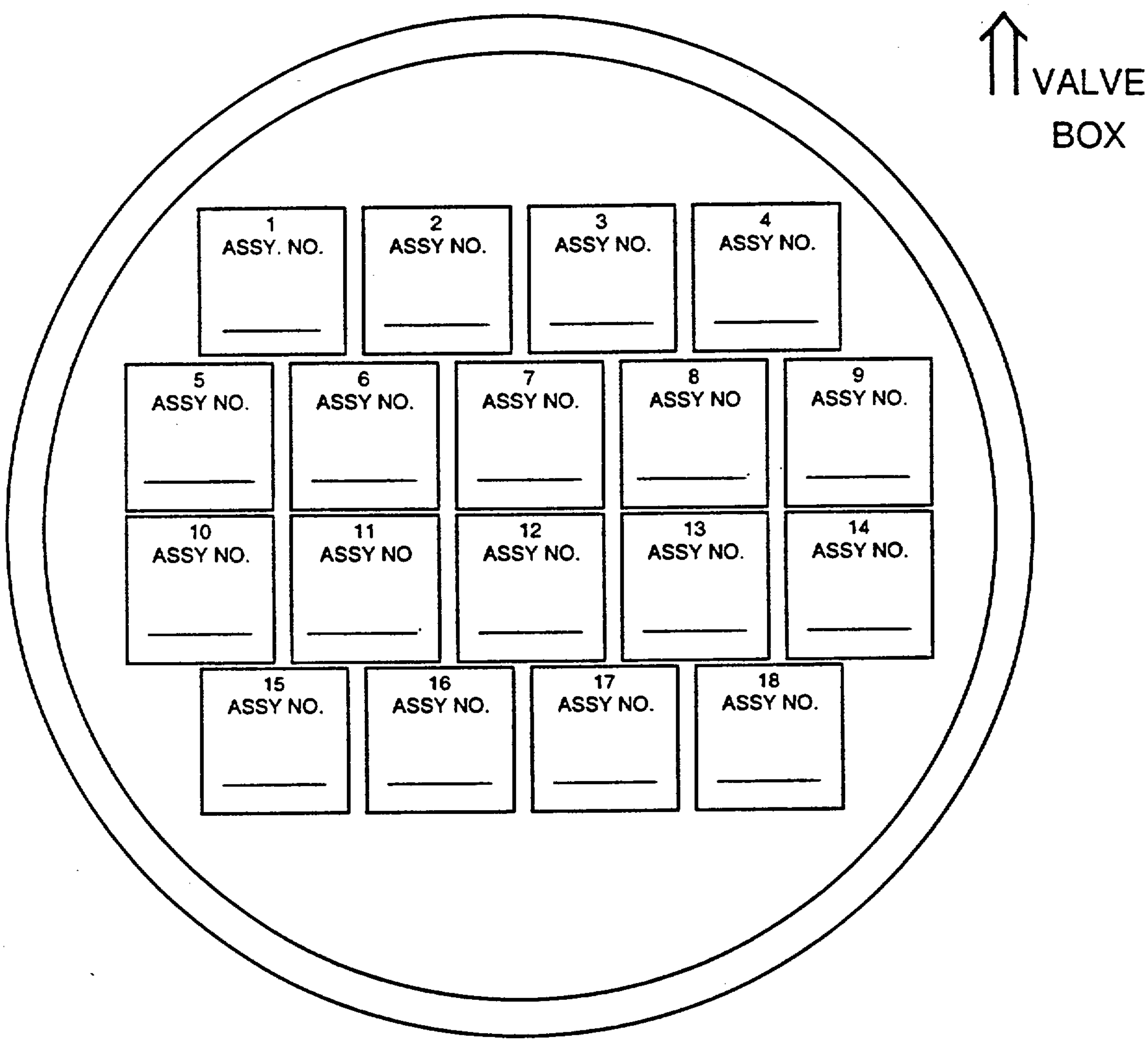
IFDS Verified By: ________________/________________
Date

* Enter Largest Assembly Value Per Shipment
** Enter Smallest Assembly Value Per Shipment

CPL 00470021

# IF-300 CASK LOADING DIAGRAM
## 18 BUNDLE BASKET FOR UNCHANNELLED BWR FUEL

**IFDS No:** ____________




**ALTERNATE FUEL ASSEMBLIES:**

**1)** __________

**2)** __________

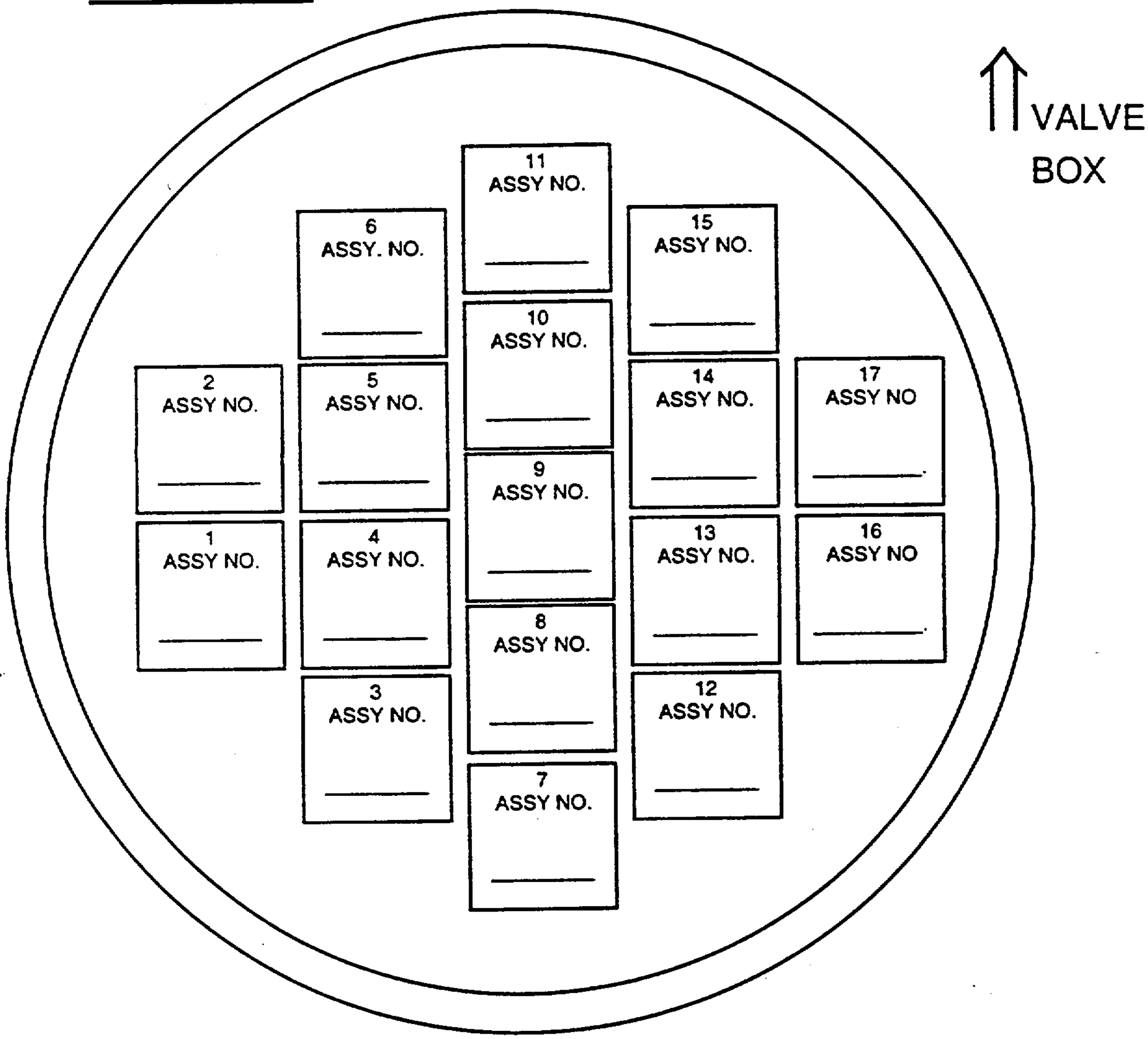
**3)** __________

(Form NFP-NGGC-0003-8-4)

CPL 00470022

# IF-300 CASK LOADING DIAGRAM
## 17 BUNDLE BASKET FOR CHANNELLED BWR FUEL

**IFDS No:** ____________




**ALTERNATE FUEL ASSEMBLIES:**

1) ____________

2) ____________

3) ____________

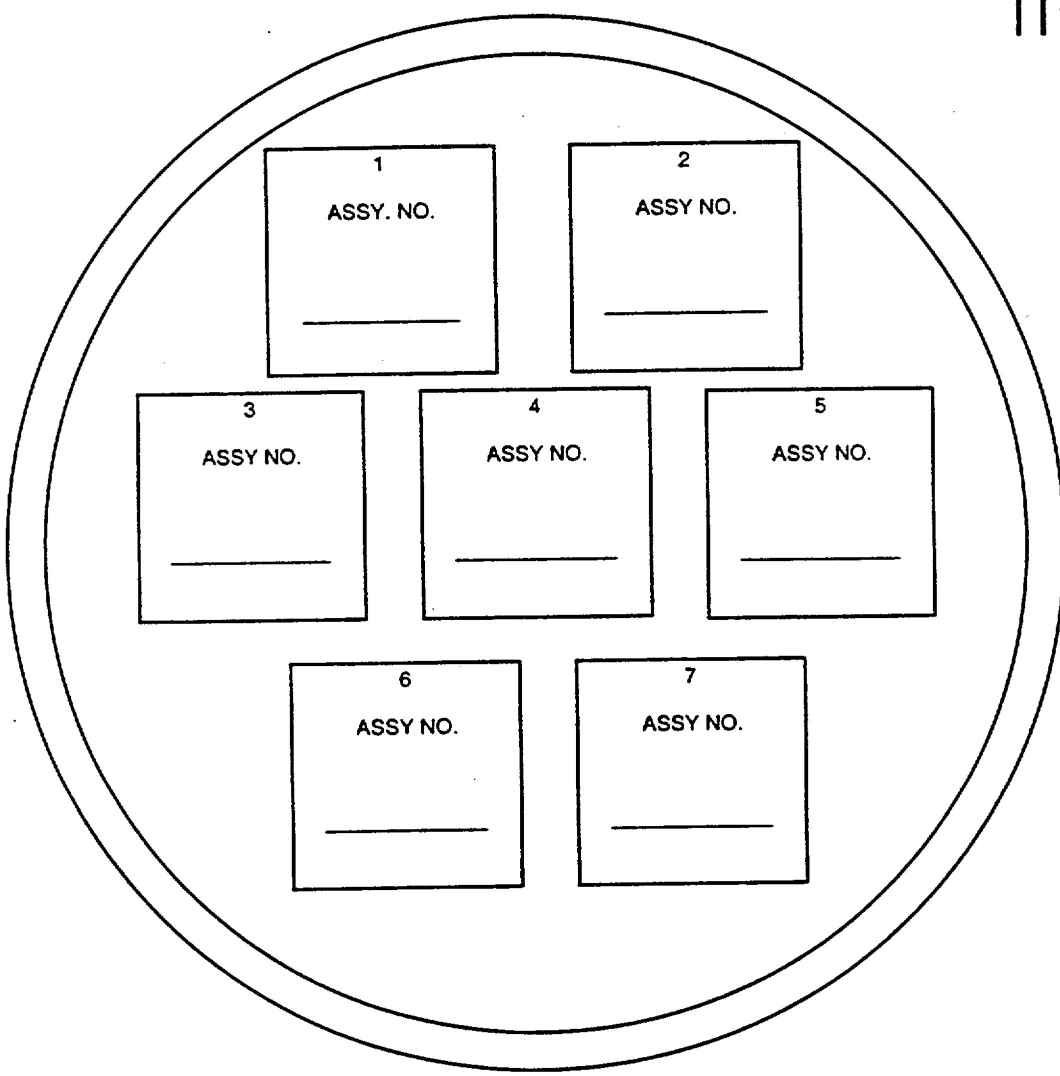
(Form NFP-NGGC-0003-9-4)

CPL 00470023

# IF-300 CASK LOADING DIAGRAM
## 7 ASSEMBLY - PWR BASKET

**IFDS No:** ____________

⇑ VALVE BOX




**ALTERNATE FUEL ASSEMBLIES:**

**1)** ____________

**2)** ____________

**3)** ____________

(Form NFP-NGGC-0003-10-4)

CPL 00470024

## REVISION SUMMARY

The following changes were made in revision 4:

- Administrative Correction (DCF # 19980037) to bring into compliance with PRO-NGGC-0201, "NGG Standard Procedure Writer's Guide", Rev. 4. This closes Corrective Action HNP 98-0165-44.

CPL 00470025

Kopp Exhibit 12

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with a $k_{eff}$ less than or equal to 0.95 when flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3.2.6 of the FSAR.

1. The reactivity margin is assured for pools 'A' and 'B' by maintaining a nominal 10.5 inch center-to-center distance between fuel assemblies placed in the flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks.

2. The reactivity margin is assured for pools 'C' and 'D' by maintaining a nominal 9.017 inch center-to-center distance between fuel assemblies placed in the non-flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks. The following restrictions are also imposed through administrative controls:

   a. PWR assemblies must be within the "acceptable range" of the burnup restrictions shown in Figure 5.6.1 prior to storage in Pools 'C' or 'D'

   b. BWR assemblies are acceptable for storage in Pool 'C' provided that the maximum planar average enrichments is less than 4.6 wt% U235 and $K_{inf}$ is less than or equal to 1.32 for the standard cold core geometry (SCCG).

DRAINAGE

5.6.2 The pools 'A', 'B', 'C' and 'D' are designed and shall be maintained to prevent inadvertent draining of the pools below elevation 277.

CAPACITY

5.6.3.a Pool 'A' contains six (6 x 10 cell) flux trap type PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. Pool 'B' contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) flux trap style PWR racks and seventeen (11 x 11 cell) BWR racks and is licensed for one additional (11 x 11 cell) BWR rack that will be installed as needed. The combined pool 'A' and 'B' licensed storage capacity is 3669 assemblies.

5.6.3.b Pool 'C' is designed to contain a combination of PWR and BWR assemblies. Pool 'C' can contain two (11 x 9 cell) and nine (9 x 9 cell) PWR racks for storage of 927 PWR assemblies. Pool 'C' can contain two (8 x 13 cell), two (8 x 11cell), six (13 x 11 cell), and nine (13 x 13 cell) BWR racks for storage of 2763 BWR assemblies. The (9 x 9 cell) PWR racks and the (13 x 13 cell) BWR racks are dimensioned to allow interchangeability between PWR or BWR storage rack styles as required. The racks in pool 'C' will be installed as needed.

5.6.3.c Pool 'D' contains a variable number of PWR storage spaces. These racks will be installed as needed. Pool 'D' is designed for a maximum storage capacity of 1025 PWR assemblies.

5.6.3.d The heat load from fuel stored in Pools 'C' and 'D' shall not exceed 1.0 MBtu/hr.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

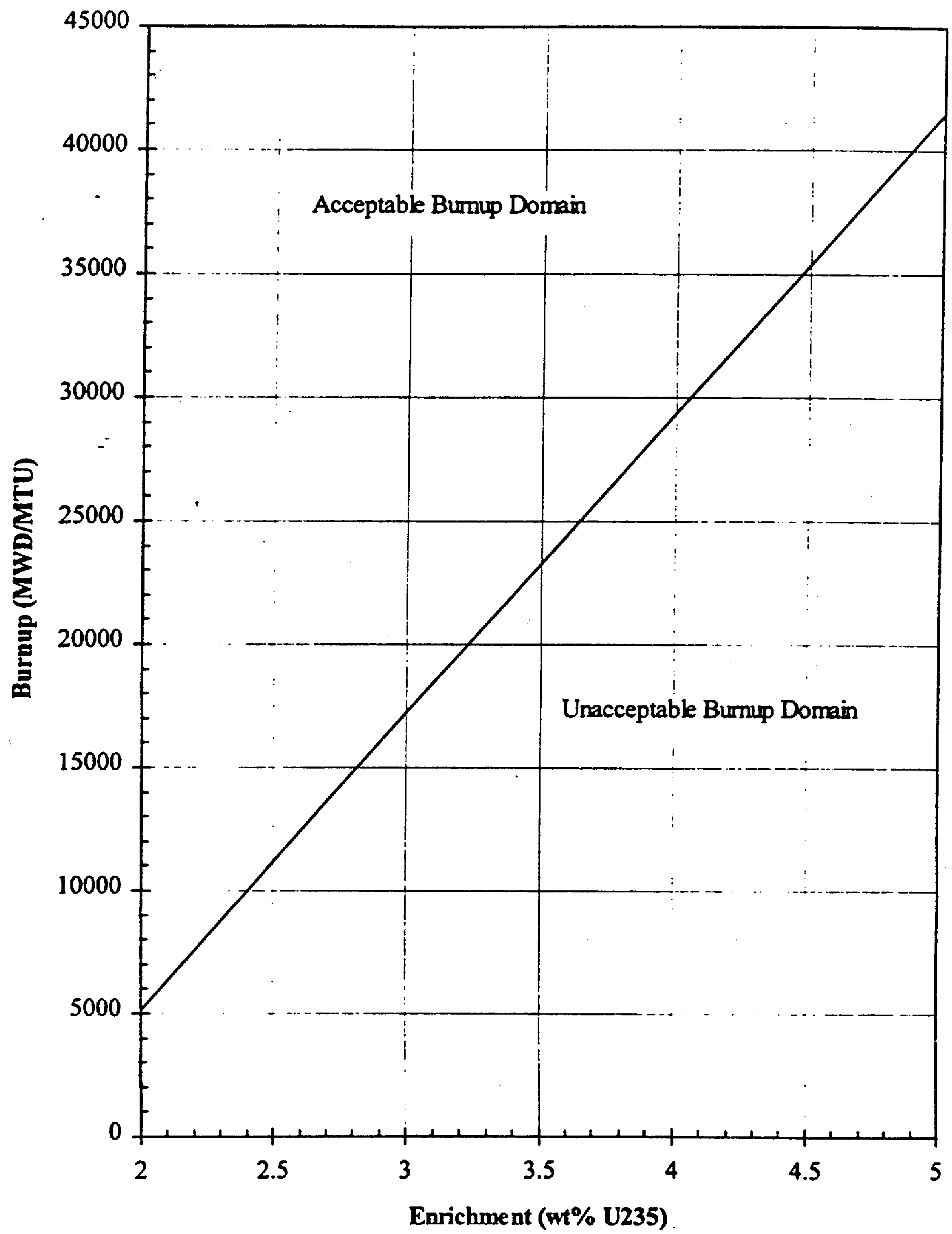



**Figure 5.6.1: Burnup Versus Enrichment for PWR Fuel**

Kopp Exhibit 13

ANSI/ANS-8.1-1983
Revision of
ANSI N16.1-1975

# American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

Secretariat
American Nuclear Society

Prepared by the
American Nuclear Society
Standards Committee
Subcommittee ANS-8

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by the
American National Standards Institute, Inc.

# Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

## 1. Introduction

Operations with some fissionable materials introduce risks of a criticality accident resulting in a release of radiation that may be lethal to nearby personnel. However, experience has shown that extensive operations can be performed safely and economically when proper precautions are exercised. The few criticality accidents that have occurred show frequency and severity rates far below those typical of nonnuclear accidents. This favorable record can be maintained only by continued adherence to good operating practices such as are embodied in this standard; however, the standard, by itself, cannot establish safe processes in an absolute sense. Good safety practices must recognize economic considerations, but the protection of operating personnel[1] and the public must be the dominant consideration.

## 2. Scope

This standard is applicable to operations with fissionable materials outside nuclear reactors, except the assembly of these materials under controlled conditions, such as in critical experiments. Generalized basic criteria are presented and limits are specified for some single fissionable units of simple shape containing $^{233}U$, $^{235}U$, or $^{239}Pu$, but not for multiunit arrays.[2] Requirements are stated for establishing the validity and areas of applicability of any calculational method used in assessing nuclear criticality safety. This standard does not include the details of administrative controls, the design of processes or equipment, the description of instrumentation for process control, or detailed criteria to be met in transporting fissionable materials.

[1]Guidance for establishing an alarm system is contained in American National Standard Criticality Accident Alarm System, ANSI/ANS-8.3-1979.

[2]Limits for certain multiunit arrays are contained in American National Standard Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI/ANS-8.7-1982.

## 3. Definitions

**3.1 Limitations.** The definitions given below are of a restricted nature for the purposes of this standard. Other specialized terms are defined in American National Standard Glossary of Terms in Nuclear Science and Technology, ANSI N1.1-1976/ANS-9 [1].[3]

**3.2 Shall, Should, and May.** The word "shall" is used to denote a requirement, the word "should" to denote a recommendation, and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform with this standard, all operations shall be performed in accordance with its requirements, but not necessarily with its recommendations.

**3.3 Glossary of Terms**

**area(s) of applicability.** The ranges of material compositions and geometric arrangements within which the bias of a calculational method is established.

**areal density.** The total mass of fissionable material per unit area projected perpendicularly onto a plane. (For an infinite, uniform slab, it is the product of the slab thickness and the concentration of fissionable material within the slab.)

**bias.** A measure of the systematic disagreement between the results calculated by a method and experimental data. The uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data.

**calculational method (method).** The mathematical equations, approximations, assumptions, associated numerical parameters (e.g., cross sections), and calculational procedures which yield the calculated results.

**controlled parameter.** A parameter that is kept within specified limits.

**criticality accident.** The release of energy as a result of accidentally producing a self-sustaining or divergent neutron chain reaction.

[3]Numbers in brackets refer to corresponding numbers in Section 7, References.

**effective multiplication factor ($k_{eff}$).** The ratio of the total number of neutrons produced during a time interval (excluding neutrons produced by sources whose strengths are not a function of fission rate) to the total number of neutrons lost by absorption and leakage during the same interval.

**nuclear criticality safety.** Protection against the consequences of an inadvertent nuclear chain reaction, preferably by prevention of the reaction.

**subcritical limit (limit).** The limiting value assigned to a controlled parameter that results in a subcritical system under specified conditions. The subcritical limit allows for uncertainties in the calculations and experimental data used in its derivation but not for contingencies; e.g., double batching or failure of analytical techniques to yield accurate values.

## 4. Nuclear Criticality Safety Practices

### 4.1 Administrative Practices

**4.1.1 Responsibilities.** Management shall clearly establish responsibility for nuclear criticality safety. Supervision should be made as responsible for nuclear criticality safety as for production, development, research, or other functions. Each individual, regardless of position, shall be made aware that nuclear criticality safety in his work area is ultimately his responsibility. This may be accomplished through training and periodic retraining of all operating and maintenance personnel. Nuclear criticality safety differs in no intrinsic way from industrial safety, and good managerial practices apply to both.

Management shall provide personnel skilled in the interpretation of data pertinent to nuclear criticality safety and familiar with operations to serve as advisors to supervision. These specialists should be, to the extent practicable, administratively independent of process supervision.

Management shall establish the criteria to be satisfied by nuclear criticality safety controls. Distinction may be made between shielded and unshielded facilities, and the criteria may be less stringent when adequate shielding and confinement assure the protection of personnel.[4]

[4]Guidance is provided in American National Standard Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement, ANSI/ANS-8.10-1983.

**4.1.2 Process Analysis.** Before a new operation with fissionable materials is begun or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions.[5] Care shall be exercised to determine those conditions which result in the maximum effective multiplication factor ($k_{eff}$).

**4.1.3 Written Procedures.** Operations to which nuclear criticality safety is pertinent shall be governed by written procedures. All persons participating in these operations shall understand and be familiar with the procedures. The procedures shall specify all parameters they are intended to control. They should be such that no single, inadvertent departure from a procedure can cause a criticality accident.

**4.1.4 Materials Control.** The movement of fissionable materials shall be controlled. Appropriate materials labeling and area posting shall be maintained specifying material identification and all limits on parameters that are subjected to procedural control.

**4.1.5 Operational Control.** Deviations from procedures and unforeseen alterations in process conditions that affect nuclear criticality safety shall be reported to management and shall be investigated promptly. Action shall be taken to prevent a recurrence.

**4.1.6 Operational Reviews.** Operations shall be reviewed frequently (at least annually) to ascertain that procedures are being followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluation. These reviews shall be conducted, in consultation with operating personnel, by individuals who are knowledgeable in nuclear criticality safety and who, to the extent practicable, are not immediately responsible for the operation.

**4.1.7 Emergency Procedures.** Emergency procedures shall be prepared and approved by management. Organizations, local and offsite, that are expected to respond to emergencies shall be made aware of conditions that might be encountered, and they should be assisted in preparing suitable procedures governing their responses.

[5]In some cases it may be necessary to resort to *in situ* neutron multiplication measurements to confirm the subcriticality of proposed configurations. Guidance for safety in performing such measurements is contained in American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ, ANSI/ANS-8.6-1983.

## 4.2 Technical Practices

**4.2.1 Controlling Factors.** The effective multiplication factor ($k_{eff}$) of a system containing fissionable material depends on:

(1) The mass and distribution of all fissionable materials and

(2) The mass, distribution, and nuclear properties of all other materials with which the fissionable materials are associated.

Nuclear criticality safety is achieved by controlling one or more parameters of the system within subcritical limits. Control may be exercised administratively through procedures (e.g., by requiring that a mass not exceed a posted limit), by physical restraints (e.g., by confining a solution to a cylindrical vessel with diameter no greater than the subcritical limit), through the use of instrumentation (e.g., by keeping a fissile concentration below a specific limit by devices that measure concentration and prevent its buildup through reflux in a chemical system), by chemical means (e.g., by prevention of conditions that allow precipitation, thereby maintaining concentration characteristic of an aqueous solution), by relying on the natural or credible course of events (e.g., by relying on the nature of a process to keep the density of uranium oxide less than a specified fraction of theoretical), or by other means. All controlled parameters and their limits shall be specified.

**4.2.2 Double Contingency Principle.** Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

**4.2.3 Geometry Control.** Where practicable, reliance should be placed on equipment design in which dimensions are limited[6] rather than on administrative controls. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. All dimensions and nuclear properties on which reliance is placed shall be verified prior to beginning operations, and control shall be exercised to maintain them.

**4.2.4 Neutron Absorbers.** Reliance may be placed on neutron-absorbing materials, such as cadmium and boron, that are incorporated in process materials or equipment, or both.[7] Control shall be exercised to maintain their continued presence with the intended distributions and concentrations. Extraordinary care should be taken with solutions of absorbers because of the difficulty of exercising such control.

**4.2.5 Subcritical Limits.** Where applicable data are available, subcritical limits shall be established on bases derived from experiments, with adequate allowance for uncertainties in the data. In the absence of directly applicable experimental measurements, the limits may be derived from calculations made by a method shown by comparison with experimental data to be valid in accordance with 4.3.

**4.3 Validation of a Calculational Method.** There are many calculational methods suitable for determining the effective multiplication factor ($k_{eff}$) of a system or for deriving subcritical limits. The methods vary widely in basis and form, and each has its place in the broad spectrum of problems encountered in the nuclear criticality safety field. However, the general procedure to be followed in establishing validity is common to all.

4.3.1 Bias shall be established by correlating the results of criticality experiments with results obtained for these same systems by the method being validated. Commonly the correlation is expressed in terms of the values of $k_{eff}$ calculated for the experimental systems, in which case the bias is the deviation of the calculated values of $k_{eff}$ from unity. However, other parameters may be used. The bias serves to normalize a method over its area(s) of applicability so that it will predict critical conditions within the limits of the uncertainty in the bias. Generally neither the bias nor its uncertainty is constant; both should be expected to be functions of composition and other variables.

4.3.2 The area(s) of applicability of a calculational method may be extended beyond the range of experimental conditions over which the bias is established by making use of the trends in the bias. Where the extension is large, the method should be supplemented by other calculational methods to provide a better estimate of the bias in the extended area(s).

[6]Guidance for assessing the safety of piping systems for uranyl nitrate solutions is contained in American National Standard Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate, ANSI/ANS-8.9-1978.

[7]Guidance for the use of a particular absorber is contained in American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, ANSI/ANS-8.5-1979.

4.3.3 A margin in the correlating parameter, which margin may be a function of composition and other variables, shall be prescribed that is sufficient to ensure subcriticality. This margin of subcriticality shall include allowances for the uncertainty in the bias and for uncertainties due to any extensions of the area(s) of applicability.

4.3.4 If the method involves a computer program, checks shall be performed to confirm that the mathematical operations are performed as intended. Any changes in the computer program shall be followed by reconfirmation that the mathematical operations are performed as intended.

4.3.5 Nuclear properties such as cross sections should be consistent with experimental measurements of these properties.

4.3.6 A written report of the validation shall be prepared.[8] This report shall:

(1) Describe the method with sufficient detail, clarity, and lack of ambiguity to allow independent duplication of results.

(2) State computer programs used, the options, recipes for choosing mesh points where applicable, the cross section sets, and any numerical parameters necessary to describe the input.

(3) Identify experimental data and list parameters derived therefrom for use in the validation of the method.

(4) State the area(s) of applicability.

(5) State the bias and the prescribed margin of subcriticality over the area(s) of applicability. State the basis for the margin.

## 5. Single-Parameter Limits for Fissile Nuclides

Operations with fissile materials may be performed safely by complying with any one of the limits given in 5.1, 5.2, 5.3, and 5.4 for single units provided the conditions under which the limit applies are maintained; these limits were calculated by methods satisfying the requirements of 4.3. A limit shall be applied only when surrounding materials, including other nearby fissionable materials, can be shown to increase the effective multiplication factor ($k_{eff}$) no more than does enclosing the unit by a contiguous layer of water of unlimited thickness. A limit may be applied to a mixture of fissile nuclides by considering all components of the mixture to be the one with the most restrictive limit.

***Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.***

**5.1 Uniform Aqueous Solutions.** Any one of the limits of Table 1 is applicable provided a uniform aqueous solution is maintained. It is therefore implied that the concentrations of the saturated solutions are not exceeded. The $^{239}Pu$ limits apply to mixtures of plutonium isotopes provided the concentration of $^{240}Pu$ exceeds that of $^{241}Pu$ and provided $^{241}Pu$ is considered to be $^{239}Pu$ in computing mass or concentration. (Less restrictive limits are provided in 6.3 for plutonium isotopic compositions containing appreciable concentrations of $^{240}Pu$.) The limit on atomic ratio is equivalent to the limit on solution concentration, but the ratio limit may also be applied to non-aqueous solutions regardless of the chemical form of the fissile nuclide.

**5.2 Aqueous Mixtures.** The areal densities of Table 1 are independent of chemical compound and are valid for mixtures which may have density gradients provided the areal densities are uniform. The subcritical mass limits for $^{233}U$, $^{235}U$, and $^{239}Pu$ in mixtures that may not be uniform are 0.50, 0.70, and 0.45 kg, respectively, and are likewise independent of compound [2-4].

**5.2.1 Enrichment Limits.** Table 2 contains $^{235}U$ enrichment limits for uranium compounds mixed homogeneously[9] with water with no limitations on mass or concentration.

[8]Management may limit the distribution of the report to protect proprietary information.

[9]In the "homogeneous" mixtures to which calculations of these limits were normalized the average particle size of dry $UO_3$ was 60 microns [V. I. NEELEY and H. E. HANDLER, "Measurement of Multiplication Constant for Slightly Enriched Homogeneous $UO_3$-Water Mixtures and Minimum Enrichment for Criticality," HW-70310, Hanford Atomic Products Operations (August 1961)]. It seems likely that the average particle size of the dihydrate of $UO_2(NO_3)_2$ was approximately 100 microns [V. I. NEELEY, J. A. BERBERET and R. H. MASTERSON, "$k_\infty$ of Three Weight Per Cent $^{235}U$ Enriched $UO_3$ and $UO_2(NO_3)_2$ Hydrogeneous Systems," HW-66882, Hanford Atomic Products Operations (September 1961)]. Various H/U ratios in the nitrate mixtures were achieved with 1/8-inch spheres of polyethylene [S. R. BIERMAN and G. M. HESS, "Minimum Critical $^{235}U$ Enrichment of Homogeneous Uranyl Nitrate," ORNL-CDC-5, Oak Ridge Criticality Data Center (June 1968)].

**5.3 Metallic Units.** The enrichment limit for uranium and the mass limits given in Table 3 apply to a single piece having no concave surfaces. They may be extended to an assembly of pieces provided there is no interspersed moderation.

The $^{233}U$ and $^{235}U$ limits apply to mixtures of either isotope with $^{234}U$, $^{236}U$, or $^{238}U$ provided $^{234}U$ is considered to be $^{233}U$ or $^{235}U$, respectively, in computing mass [3]. The $^{239}Pu$ limits apply to isotopic mixtures of plutonium provided the concentration of $^{240}Pu$ exceeds that of $^{241}Pu$ and all isotopes are considered to be $^{239}Pu$ in computing mass [4]. Density limits may be adjusted for isotopic composition.

**5.4 Oxides.** The limits in Tables 4 and 5 apply only if the oxide contains no more than 1.5% water by weight. The mass limits apply to a single piece having no concave surfaces. They may be extended to an assembly of pieces provided there is no additional interspersed moderation.

The mass limit is given equivalently as mass of nuclide and as mass of oxide (including moisture). It is emphasized that the limits in Tables 4 and 5 are valid only under the specified bulk density restrictions.[10] With water content limited to 1.5% the enrichment limit of Table 2 for uranium oxides is increased to 3.2% $^{235}U$ [3].

## 6. Multiparameter Control

Although the single-parameter limits are adequate for many purposes, they are inconveniently and uneconomically small for many others. Simultaneous limitation of two or more parameters results in a less restrictive limit for the one of interest. A few particularly useful examples are given in 6.1 through 6.4. All were calculated by methods satisfying 4.3. These limits shall be applied only when surrounding materials can be shown to increase the effective multiplication factor ($k_{eff}$) no more than does enclosing the system by a contiguous layer of water of unlimited thickness. General guidance for multiparameter control may be found in the technical literature.[11-14]

***Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.***

**6.1 Uranium Metal- and Uranium Oxide-Water Mixtures at Low $^{235}U$ Enrichment.** An application of multiparameter control is control of both the $^{235}U$ enrichment of uranium and one of the parameters of Section 5. Subcritical limits [5] applicable to aqueous systems containing uranium metal or uranium oxide ($UO_2$), regardless of the size and shape of metal or oxide pieces, are specified as functions of enrichment in Figs. 1 through 5 which give, respectively, the mass of $^{235}U$, the cylinder diameter, the slab thickness, the volume, and the areal density.[15]

[10]The user is cautioned that, particularly for $UO_3$, material densities in excess of the full densities of Table 4 may be possible and hence that the limits of Table 4 may not be valid for highly compacted oxides. However, it is expected that oxides will generally be in the form of loose powders or, in the case of $UO_2$, of accumulations of pellets and that the limits of Table 4 and perhaps Table 5 will be valid. Where other density limits are desired, where it is inconvenient to maintain the water content below 1.5% ($H/U \cong 0.47$), or where oxides are non-stoichiometric, the limits may be useful as points of departure in deriving more appropriate values.

The maximum bulk densities were derived from CRC Handbook values of 10.96, 8.3, 7.29, and 11.46 $g/cm^3$ for $UO_2$, $U_3O_8$, $UO_3$, and $PuO_2$ together with the assumption of additive volumes of oxide and water. However, x-ray densities of $UO_3$ as high as 8.46 $g/cm^3$ have been reported. Moreover, the assumption of additive volumes may be incorrect; with $H_2O$ assigned a density of unity, an effective $UO_3$ density of 10.47 $g/cm^3$ is required to produce a reported x-ray density of 6.71 $g/cm^3$ for $\alpha$-$UO_2(OH)_2$.

[11]H. C. PAXTON, J. T. THOMAS, D. CALLIHAN, and E. B. JOHNSON, "Critical Dimensions of Systems Containing $^{235}U$, $^{239}Pu$, and $^{233}U$," TID-7028, U.S. Atomic Energy Commission (1964).

[12]J. T. THOMAS, "Nuclear Safety Guide, TID-7016, Rev. 2," NUREG/CR-0095 (ORNL/NUREG/CSD-6), Oak Ridge National Laboratory (1978).

[13]H. K. CLARK, "Handbook of Nuclear Safety," DP-532, Savannah River Laboratory (1961).

[14]R. D. CARTER, G. R. KEIL, K. R. RIDGWAY, "Criticality Handbook," ARH-600, Atlantic Richfield Hanford Company (1973).

[15]The data points through which the curves in Figs. 1-5 were drawn are the subcritical values listed in Tables VI-VIII of Ref. [5].

**6.2 Aqueous Uranium Solutions at Low $^{235}U$ Enrichment.** A similar application of multiparameter control is control of both $^{235}U$ enrichment and one of the parameters of Table 1, together with the maintenance of a uniform aqueous solution. Table 6 lists subcritical limits for uniform aqueous solutions of uranium where the enrichment is controlled within the stated limit. Concentrations of saturated solutions, which are here taken to be 5 molar for $UO_2F_2$ solutions and 2.5 molar for $UO_2(NO_3)_2$ solutions, shall not be exceeded.

**6.3 Uniform Aqueous Solutions of $Pu(NO_3)_4$ Containing $^{240}Pu$.** Reliance on, and hence control of, the isotopic concentration of $^{240}Pu$ in plutonium permits greater limits for $Pu(NO_3)_4$ solutions than are listed in Table 1.[16] However, the amount of the increase is dependent on $^{241}Pu$ concentration. Table 7 contains limits for uniform aqueous solutions of $Pu(NO_3)_4$ as a function of isotopic composition. Any $^{238}Pu$ or $^{242}Pu$ present shall be omitted in computing the isotopic composition.

**6.4 Aqueous Mixtures of Plutonium Containing $^{240}Pu$.** Subcritical mass limits for plutonium as $PuO_2$ in aqueous mixtures, which may be nonuniform, where $^{240}Pu$ and $^{241}Pu$ are subject to the three pairs of restrictions on isotopic composition of Table 7, are, in increasing order of $^{240}Pu$ concentration, 0.53, 0.74, and 0.99 kg, respectively [4].

[16] Where plutonium, in addition, is intimately mixed with natural uranium, limits are even greater. Limits for this case are included in American National Standard for Nuclear Criticality Control and Safety of Homogeneous Plutonium-Uranium Fuel Mixtures Outside Reactors, ANSI/ANS-8.12-1978.

## 7. References


[1] American National Standard Glossary of Terms in Nuclear Science and Technology, ANSI/N1.1-1976/ANS-9. American Nuclear Society, La Grange Park, Ill.

[2] H. K. CLARK, "Subcritical Limits for $^{233}U$ Systems," *Nucl. Sci. Eng.* 81, 379-395 (1982). American Nuclear Society, La Grange Park, Ill.

[3] H. K. CLARK, "Subcritical Limits for $^{235}U$ Systems," *Nucl. Sci. Eng.* 81, 351-378 (1982). American Nuclear Society, La Grange Park, Ill.

[4] H. K. CLARK, "Subcritical Limits for Pu Systems," *Nucl. Sci. Eng.* 79, 65-84 (1981). American Nuclear Society, La Grange Park, Ill.

[5] H. K. CLARK, "Critical and Safe Masses and Dimensions of Lattices of U and $UO_2$ Rods in Water," DP-1014, Savannah River Laboratory, Aiken, S. C., (1966).


When the preceding American National Standard referred to in this document is superseded by a revision approved by the American National Standards Institute, Inc., the revision shall apply.

Kopp Exhibit 14

(14)

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

March 20, 1986

REGULATORY GUIDE DISTRIBUTION LIST (DIVISION 3)

SUBJECT: ISSUANCE OF REVISION 2 TO REGULATORY GUIDE 3.4 AND WITHDRAWAL OF REGULATORY GUIDE 3.41

With the issuance of Revision 2 to Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities," the NRC staff is withdrawing Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety."

Revision 2 to Regulatory Guide 3.4 endorses ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," which is a consolidation of ANSI N16.1-1975/ANS-8.1 (endorsed by Revision 1 of Regulatory Guide 3.4) and ANSI N16.9-1975/ANS-8.11 (endorsed by Regulatory Guide 3.41). Regulatory Guide 3.41 is therefore obsolete. However, withdrawal of Regulatory Guide 3.41 is in no way intended to alter any prior or existing licensing commitments based on its use.

Regulatory guides may be withdrawn when they are superseded by the Commission's regulations, when equivalent recommendations have been incorporated in applicable approved codes and standards, or when changes in methods and techniques or in the need for specific guidance have made them obsolete.

Robert B. Minogue

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

U.S. NUCLEAR REGULATORY COMMISSION March 1986

(14)

# REGULATORY GUIDE

## OFFICE OF NUCLEAR REGULATORY RESEARCH

**REGULATORY GUIDE 3.4**
**(Task CE 404-4)**

# NUCLEAR CRITICALITY SAFETY IN OPERATIONS WITH FISSIONABLE MATERIALS AT FUELS AND MATERIALS FACILITIES

## A. INTRODUCTION

Section 70.22, "Contents of Applications," of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," requires that applications for a specific license to own, acquire, deliver, receive, possess, use, or initially transfer special nuclear material contain proposed procedures to avoid accidental criticality. This guide describes procedures acceptable to the NRC staff for preventing accidental criticality in operations with fissionable materials at fuels and materials facilities (i.e., fuel cycle facilities other than nuclear reactors) and for validating calculational methods used in assessing nuclear criticality safety.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 70, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 70 have been cleared under OMB Clearance No. 3150-0009.

## B. DISCUSSION

ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,"** was prepared by Subcommittee 8, Fissionable Materials Outside Reactors, of the Standards Committee of the American Nuclear Society as a consolidation of revisions to ANSI N16.1-1975/ANS-8.1 and ANSI N16.9-1975/ANS-8.11. ANSI/ANS-8.1-1983 was approved by the American National Standards Committee N16, Nuclear Criticality Safety, in 1982 and by the American National Standards Institute (ANSI) on October 7, 1983.

ANSI/ANS-8.1-1983 applies to handling, storing, processing, and transporting fissionable material outside nuclear reactors. The standard presents generalized basic criteria and specific limits (maximum subcritical) for some single units of simple shape containing $^{233}U$, $^{235}U$, $^{239}Pu$, but not for multi-unit arrays. Further, the subcritical limits specified in the standard allow for uncertainties in the calculations and experimental data used in their derivation but not for contingencies such as double batching or failure of analytical techniques to yield accurate values.

This standard also delineates requirements for establishing the validity and area of applicability of a calculational method used in assessing nuclear criticality safety. However, it is concerned only with validating calculational methods and does not address important related questions such as the margin of safety to be used with the method or the qualifications of the personnel responsible for the data input.

This standard does not apply to the assembly of fissionable materials under controlled conditions, e.g., in critical experiments. Nor does the standard include the details of administrative controls, the design of processes or equipment, the description of instrumentation for process control, or detailed criteria to be met in transporting multi-unit arrays of fissionable materials.

## C. REGULATORY POSITION

The nuclear criticality safety practices, the single-parameter limits for fissionable nuclides, and the guidance for multiparameter control contained in ANSI/ANS-8.1-1983 provide procedures acceptable to the NRC staff for preventing accidental conditions of criticality in handling, storing, processing, and transporting special nuclear materials at fuels and materials facilities. However, use of ANSI/ANS-8.1-1983 is not a substitute for detailed nuclear criticality safety analyses for specific operations.

The guidelines for validating calculational methods for nuclear criticality safety contained in ANSI/ANS-8.1-1983

*Lines indicate substantive changes from Revision 1.

**Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

provide a procedure acceptable to the NRC staff for establishing the validity and area of applicability of calculational methods used in assessing nuclear criticality safety. However, it will not be sufficient merely to refer to this guide in describing the validation of a method. The details of validation indicated in Section 4.3.6 of the standard should be provided to demonstrate the adequacy of the safety margins relative to the bias and criticality parameters and to demonstrate that the calculations embrace the range of variables to which the method will be applied.

Section 7 of ANSI/ANS-8.1-1983 lists additional documents referred to in the standard. Endorsement of ANSI/ANS-8.1-1983 by this regulatory guide does not constitute an endorsement of these documents.

## D. IMPLEMENTATION

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

The methods described in this guide were applied to a number of specific cases during reviews and selected licensing actions. These methods reflect the latest general NRC approach to criticality safety in operations with fissionable materials at fuels and materials facilities. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide will be used in the evaluation of submittals in connection with license applications submitted under 10 CFR Part 70.

## VALUE/IMPACT STATEMENT

The NRC staff performed a value/impact assessment to determine the proper procedural approach for updating Revision 1 of Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," issued in February 1978, which endorsed ANSI N16.1-1975/ANS-8.1. The NRC staff has been involved in the development, review, and approval of a revision to ANSI N16.1-1975/ANS-8.1 (designated ANSI/ ANS-8.1-1983), which was approved by the American National Standards Institute on October 7, 1983. The assessment resulted in a decision to develop a revision to Regulatory Guide 3.4 that would endorse, with possible supplemental provisions, ANSI/ ANS-8.1-1983. The results of this assessment were included in a proposed Revision 2 to Regulatory Guide 3.4 that was issued for public comment in April 1985. No comments have been received from the public, and additional NRC staff review has shown that, except for minor clarifications, there was no need to change the regulatory position of the proposed Revision 2 to Regulatory Guide 3.4. Therefore, the value/impact statement published with the proposed revision is applicable. A copy of the draft regulatory guide and the associated value/impact statement (identified by its task number, CE 404-4) is available for inspection or copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, DC.

Kopp Exhibit 15

WCAP-6068



WCAP-6068
UC-80, Reactor Technology

# EVALUATION OF MASS SPECTROMETRIC AND RADIOCHEMICAL ANALYSES OF YANKEE CORE I SPENT FUEL

by

R. J. Nodvik
Nuclear Operations Analysis

March 1966

Prepared for the New York Operations Office
U. S. Atomic Energy Commission
Under AEC Contract Number AT(30-1)-3017

TECHNICAL APPROVAL:

H. W. Graves Jr

H. W. Graves, Jr., Manager
Nuclear Engineering Section
Reactor Development

PROJECT MANAGER APPROVAL:

M. O'Boyle

for L. Chajson, Manager
Yankee Core Evaluation Program

WESTINGHOUSE ELECTRIC CORPORATION
Atomic Power Division
Box 355
Pittsburgh, Pennsylvania 15230

ABSTRACT

*This report presents the results of mass spectrometric, X-ray spectrographic, and radiochemical analyses of spent fuel samples from 219 locations in the Yankee core. These sample locations were preselected to provide: 1) the U and Pu isotopic composition of the fuel as a function of burnup in the asymptotic and perturbed reactor neutron spectra; 2) the spatial distribution of burnup and fuel isotopes in the rods, in the assemblies, and in the core; 3) the total isotopic inventory of the core; and 4) fuel characteristics, including the specific Pu production, the effective capture-to-fission ratio in U-235, and the net mass of fissile materials destroyed per unit of energy release in the fuel.*

*Values of burnup are inferred over a broad range (1,200 to 31,000 MWD/MTU) from relationships between U and Pu concentrations measured before and after irradiation, and from the activities of the Cs-137 and Sr-90 fission products. The calculations used to infer burnup and the various fuel characteristics from the spent fuel data are described in detail.*

*The consistency and reliability of the data are established through the evaluation of the experimental results obtained from a number of interlaboratroy cross-check and monitor sample solutions.*

Kopp Exhibit 16

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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

FOR COMMENT

## 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.

ENCLOSURE 1

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

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

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).

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

1.88 "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,

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

$\Delta k_u$ = the uncertainty in the benchmark experiments, and

$\Delta k_{sc}$ = the combined uncertainties in the parameters listed in paragraph 3.2 below.

3.2 The combined uncertainties, $\Delta k_{sc}$, 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. <u>STORAGE RACK ANALYSIS ASSUMPTIONS</u>

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.

Kopp Exhibit 16A

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

| | |
|---|---|
| In the Matter of<br><br>CAROLINA POWER & LIGHT COMPANY<br>(Shearon Harris Nuclear Power Plant) | Docket<br>No. 50-400-LA<br><br>ASLBP<br>No. 99-762-02-LA |

# DEPOSITION OF GORDON THOMPSON, PH.D.

At Raleigh, North Carolina
October 21, 1999
9:40 AM to 4:14 PM
Reported by: Melody L. Rife, RPR

COPY

## A P P E A R A N C E S

For the Applicant, Carolina Power & Light Company:

**JOHN H.O'NEILL, JR., ESQUIRE, and**
**WILLIAM R. HOLLAWAY, ESQUIRE**
Shaw Pittman
2300 N Street, N.W.
Washington, D.C. 20037-1128
(202) 663-8148 or
(202) 663-8294

For the U. S. Nuclear Regulatory Commission:

**SUSAN L. UTTAL, ATTORNEY AT LAW**
Office of the General Counsel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555
(301) 415-1582

For the Intervenor, Board of County, Commissioners, Orange County, North Carolina:

**DIANE CURRAN, ATTORNEY AT LAW**
Harmon, Curran, Spielberg & Eisenberg, LLP
1726 M Street, N.W., Suite 600
Washington, D.C. 20036
(202) 328-3500

Also Present:

Messieurs Michael Devoe, John Caves, Richard Laufer, and Laurence Copp

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## EXAMINATIONS



- - - - - - -

A. On Contention 2?
Q. Yes.
A. Yes. I think the request for additional information does not go far enough. I think the NRC staff should have required a broader spectrum of accident analyses, misplacements of more than one assembly. They should have required a boron dilution analysis, and they should have required an assessment of the probability and consequences of an -- a correct accident.
And as mentioned earlier, I'd like to see the reg guide, the Draft Reg. Guide, brought up to date and issued as a final reg guide within an explicit prohibition of burn-up credit.
Q. Do you believe the staff's lack of putting in the things you desired in the RAI, is that demonstrating their complacency in this proceeding?
A. Yes.
Q. Ask you to turn to Exhibit -- a new exhibit.

administrative control.

And we'll be elaborating on the various possible scenarios in our brief; but one possibility is that a single failure in the administrative or the management process leads to misplacement of multiple out-of-compliance assemblies, and this multiple misplacement, with -- with or without boron dilution, might lead to a criticality.

I suppose hypothetically that one could identify a single administrative failure that lead to multiple misplacements, such that criticality occurred with boron dilution with relatively common frequency, within the ordinary variation of boron concentration. Then that would be criticality with a single failure.

Suppose that it required boron dilution of an even higher frequency, and you could argue that this is a double failure, but perhaps not of -- as unlikely

RECEIVED
DEC 03 1999

**WITNESS CERTIFICATION**

I, GORDON THOMPSON, PH.D., do hereby certify:

That I have read and examined the contents of the foregoing two hundred and seven (207) pages of record of testimony as given by me at the time and place herein aforementioned;

And that to the best of my knowledge and belief, the foregoing two hundred and seven (207) pages are a complete and accurate record of all of the testimony given by me at said time, except as to where noted on the attached errata addendum.

G.R. Thompson 11/30/99

* * * * *

Sworn to and subscribed before me on the ______ day of ___________ 1999

____________________
Notary Public

My Commission Expires: _________________

Kopp Exhibit 17

VERIFY FOR OUTSTANDING CHANGES BEFORE USE

19

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

PLANT OPERATING MANUAL

VOLUME 5

PART 3

PROCEDURE TYPE: Chemistry and Radiochemistry (CRC)

NUMBER: CRC-001

TITLE: SHNPP Environmental and Chemistry Sampling and Analysis Program

CPL 00470056

PRIMARY SYSTEMS

| SAMPLE POINT/ PARAMETER | UNITS | FREQUENCY | ADMIN. LIMITS | CONTROL LIMITS | REFERENCES | METHODS |
|---|---|---|---|---|---|---|
| M. SPENT FUEL POOL SAMPLES* (SFPA/SFPB/SFPC/SFPD/1-4TCANAL/2-3TCANAL/CASKPOOL/MAINCANAL)**- I (AT ALL TIMES) | | | | | | CRC-215 |
| ALPPBL | PPB | MN | ≤80 | ----- | 2.3 | CRC-500,CRC-508,CRC-509 |
| BORONPPM | PPM | MN | ----- | 2000-2600 | 2.3, 2.11 | CRC-524,CRC-528 |
| CAPPBL | PPB | MN | ≤40 | ----- | 2.3 | CRC-500,CRC-508,CRC-509 |
| CLPPBL | PPB | MN | ≤150 | ----- | 2.3, 2.11 | CRC-503 |
| FPPBL | PPB | MN | ≤150 | ----- | 2.3, 2.11 | CRC-503 |
| SO4PPBL | PPB | MN | ----- | ----- | 2.30, 2.11 | CRC-503 |
| GSCANS | μCI/ML | MN | ----- | ----- | 2.30, 2.11 | RCP-704,RCP-660 |
| MGPPBL | PPB | MN | ≤40 | ----- | 2.3 | CRC-500,CRC-508,CRC-509 |
| SIO2PPBL | PPB | MN | *** | ----- | 2.3,2.11 | CRC-519 |
| TRITIUM | μCI/ML | MN | ----- | ----- | | RCP-710,RCP-660,RCP-742 |
| TSSPPBL | PPB | MN | ≤250 | ----- | 2.30 | CRC-336 |

* Only one sample/month from alternating sample points is required as long as gates are removed.

**Only quarterly Cl, F, and S04 analyses are required for SFPC and SFPD. Only monthly boron analysis is required for CASKPOOL, MAINCANAL, and 2-3TCANAL.
No boron analysis is required for SFPC and SFPD.

*** Silica cleanup of the Spent Fuel Pools could aggravate the Boraflex degradation. Therefore, silica cleanup should not be attempted without concurrence from Reactor Engineering. The use of SFP demin is not considered silica cleanup since the demin does not remove silica from borated water. (CR 96-03318) R

VERIFY FOR OUTSTANDING CHANGES BEFORE USE

Kopp Exhibit 18

## 3/4.9.11 WATER LEVEL - NEW AND SPENT FUEL POOLS

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of fuel rods within irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the affected pool area and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.11 At least once per 7 days, when irradiated fuel assemblies are in a pool, the water level in that pool shall be determined to be at least its minimum required depth.

Kopp Exhibit 19

## 9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Basis. The Fuel Handling Building (FHB) is split into two storage facilities. The storage facility on the south end of the FHB consists of a new fuel pool, also referred to as Pool A or New Fuel Pool Unit 1 and a spent fuel pool, also referred to as Pool B or Spent Fuel Pool Unit 1. Both new fuel and spent fuel may be stored in either of the pools in this facility, as described in Sections 9.1.1 and 9.1.2. The storage facility on the north end of the FHB consists of a spent fuel pool, also referred to as Pool C or Spent Fuel Pool Unit 2 and a New Fuel Pool, also referred to as Pool D or New Fuel Pool Unit 2. By design, both of the pools in this facility may accommodate both new and spent fuel. Spent fuel may not be loaded into Pools C or D until they are completed and made operational. The design bases for the Fuel Pool Cooling and Cleanup System (FPCCS) for the operational pools, Pools A and B, are as follows:

a) The fuel storage facility consists of two 100 percent cooling systems in addition to cleanup equipment for removing the particulate and dissolved fission and corrosion products resulting from the spent fuel.

b) Fuel can be transferred within the operational storage facility as shown on Figure 1.2.2-55. Fuel handling is described in detail in Section 9.1.4.

c) The FPCCS is designed to maintain water quality in the fuel storage pools and remove residual heat from the spent fuel.

d) The current and typical refueling practice at SHNPP of transferring the entire core to the storage facility is referred to herein as the Full Core Offload Shuffle. The refueling practice of transferring only that portion of the core to be discharged to the storage facility is referred to herein as the Incore Shuffle. Both of these practices are reported as Normal Cases when meeting the requirements of the Standard Review Plan. The Abnormal Case is reported as the transfer of the entire core to the storage facility following startup of the next operating cycle. This case is referred to herein as the Post Outage Full Core Offload.

e) The cooling system serving the operational fuel storage facility has been designed to remove the heat loads generated by the quantities of fuel to be stored in the pools through operation to the end-of-Cycle 9.

f) The Standard Review Plan pool temperature requirement for the Normal Case, assuming a single active failure, is 140°F. The minimum decay time prior to movement of irradiated fuel in the reactor vessel will address both radiological and decay heat considerations. Administrative controls are placed on the minimum cooling time before transfer of spent fuel to the pools, to limit the fuel pool temperature to less than or equal to 137°F. The pool temperature requirement for the Abnormal Case is to be below boiling. The pool concrete design temperature is 150°F.

g) Calculations of the maximum amount of thermal energy to be removed by the spent fuel cooling system are made in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." An uncertainty factor K equal to 0.20 for cooling times ($t_s$) less than $10^3$ seconds and 0.10 for $t_s$ greater than $10^3$ seconds was used.

h) The fuel pool heatup rates were calculated using the following assumptions:

1) No credit for operation of the FPCCS.

2) No evaporative heat losses.

3) No heat absorption by concrete or liner.

4) No heat absorption by spent fuel racks or fuel in pool.

i) The cleanup loop pumps have the capacity to provide makeup water at a rate greater than the loss of water due to normal system leakage and evaporation.

j) Safe water level (and thus sufficient radiation shielding) is maintained in the new and spent fuel pools since the cooling connections are at the tops of the pools.

k) Components and structures of the system are designed to the safety class and seismic requirements indicated in Table 3.2.1-1.

l) The FPCCS will perform its safety related function assuming a single active failure (Reference 9.1.3-1).

9.1.3.2 System Description. The Fuel Pool Cooling and Cleanup System is provided as shown on Figures 9.1.3-1, 9.1.3-2, 9.1.3-3 and 9.1.3-4. The FPCCS is comprised of the two operational fuel pools, Pools A and B; the Cask Loading/Unloading Pool; the Main Fuel Transfer Canal; the south Fuel Transfer Canal; the north Fuel Transfer Canal; two fuel pool heat exchangers; two fuel pool cooling pumps; two fuel pool strainers; a fuel pool demineralizer; a fuel pool demineralizer filter; a fuel pool and a refueling water purification filter; two fuel pool and refueling water purification pumps; provisions for skimmer connections as follows: three fuel Pool A skimmers; five Pool B skimmers; two south transfer canal skimmers; two north transfer canal skimmers, one main transfer canal skimmer, one cask loading/unloading pool skimmer; a fuel pool skimmer pump, a fuel pool skimmer strainer, and a fuel pool skimmer filter.

The new fuel pool, Pool A, and the spent fuel pool, Pool B, are interconnected by the south Fuel Transfer Canal. The Cask Loading/Unloading Pool, the non-operational Pool C, and the non-operational Pool D are interconnected by the north Fuel Transfer Canal. The Main Fuel Transfer Canal connects the south and north Fuel Transfer Canals. Gates are provided to isolate the pools, as needed. Spent fuel is placed in the operational pools during refueling or from shipments of off-site fuel and stored until it is shipped to a reprocessing facility or otherwise disposed. Fuel handling is discussed in detail in Section 9.1.4. The overall arrangement of the pools is shown on Figure 1.2.2-55. Cooling of spent fuel can be accomplished in the operational fuel pools since they are serviced by the fuel pool cooling system. The location of the inlet and outlet connections to the pools precludes the possibility of coolant flow "short circuiting" the pool.

The Fuel Handling Building is designed to Seismic Category I requirements and to the tornado criteria as stated in Section 3.3.

The fuel pools in the Fuel Handling Building will not be affected by any loss of coolant accident in the Containment Building. The water in the pools is isolated from that in the refueling cavity during most of the refueling operation. Only a very small amount of interchange of water will occur as fuel assemblies are transferred during refueling.

The FPCCS is designed for the removal of sensible heat from the fuel pools. Current analyses have evaluated this function for a decay heatload equivalent to that generated by fuel discharged at HNP through operation to the end-of-Cycle 9 and from additional fuel assemblies planned to be shipped from H. B. Robinson Unit 2 and Brunswick Units 1 and 2 through end-of-Cycle 9 (Reference 9.1.3-3). For this mode of operation, the equilibrium temperatures are as shown in Table 9.1.3-2.

The clarity and purity of the fuel pool water is maintained when desired or necessary by passing approximately five percent of the cooling system flow through a cleanup loop consisting of two filters and a demineralizer. The fuel pool cooling pump suction line, which can be used to lower the pool water level, penetrates the fuel pool wall approximately 18 ft. above the fuel assemblies. The penetration location precludes uncovering the fuel assemblies as a result of a postulated suction line rupture.

Piping in contact with fuel pool water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pumps, heat exchangers and control valves to facilitate maintenance.

Control Room and local alarms are provided to alert the operator of high and low pool water level, and high temperature in the fuel pool. A low flow alarm, based on measured flow to the fuel pool, is provided to warn of interruption of cooling flow.

The Fuel Pool Cooling and Cleanup System is comprised of the following components. The component parameters are presented in Table 9.1.3-2.

a) Fuel Pool Heat Exchanger - Two fuel pool heat exchangers are provided. The fuel pool heat exchangers are of the shell and straight tube type. Component cooling water supplied from the Component Cooling Water System (Section 9.2.2) circulates through the shell, while fuel pool water circulates through the tubes. The installation of two heat exchangers assures that the heat removal capacity of the cooling system is only partially lost if one heat exchanger fails or becomes inoperative.

b) Fuel Pool Cooling Pump - Two horizontal centrifugal pumps are installed. The use of two pumps installed in separate lines assures that pumping capacity is only partially lost should one pump become inoperative. This also allows maintenance on one pump while the other is in operation.

c) Fuel Pool Demineralizer - One demineralizer is installed. The demineralizer is sized to pass approximately five percent of the loop circulation flow to provide adequate purification of the fuel pool water and to maintain optical clarity in the pool.

d) Fuel Pool Demineralizer Filter and Fuel Pool and Refueling Water Purification Filter - Two filters are installed - one fuel pool demineralizer filter and one fuel pool and refueling water purification filter. The filters remove particulate matter from the fuel pool water.

e) Fuel Pool Cooling and Cleanup System Skimmers - Provisions for fourteen skimmers are installed: three for Pool A, five for Pool B, two for each fuel transfer canal, one for the main fuel transfer canal, and one for the cask loading/unloading pool. A fuel pool skimmer pump, fuel pool skimmer pump suction strainer, and filter are provided for surface skimming of the fuel pool water. Flow from the pump is routed through the skimmer filter and returned to the fuel pools.

f) Fuel Pool and Refueling Water Purification Pumps - Two fuel pool and refueling water purification pumps are provided. Each pump can take suction from and return fluid to the refueling water storage tank via the Safety Injection System, the transfer canal, the new and spent fuel pools, or the refueling cavity. Fluids from these systems are purified by the fuel pool demineralizer and filter. Each pump can also take suction from the demineralized water storage tank for make-up to the fuel pools and line flushing.

g) Fuel Pool Cooling and Cleanup System Valves - Manual stop valves are used to isolate equipment and lines and manual throttle valves provide flow control. Valves in contact with fuel pool water are of austenitic stainless steel or of equivalent corrosion resistant material.

h) Fuel Pool Cooling and Cleanup System Piping - All piping in contact with fuel pool water is of austenitic stainless steel construction. The piping is welded except where flanged connections are used at the pumps, heat exchanger, and control valve to facilitate maintenance. Also, flanged joints with line blanks are installed at locations to provide isolation capabilities for non-operational portions of Unit 2 (Pools C and D) system flow paths.

i) Fuel Pool Gates - The vertical steel gates on the new fuel pool, spent fuel pools, fuel transfer canals, main fuel transfer canal and cask loading pools allow the spent fuel to be immersed at all times while being moved to its destination. They also allow each area to be isolated for drainage, if necessary, and enable new fuel to be stored dry in the new fuel pool.

Fuel Pool water chemistry limits and guidelines are specified in plant chemistry procedures. These procedures insure the fuel pool water chemistry is consistent with current specifications and guidelines established by the NSSS vendor, fuel manufacturer and EPRI standards. The plant Chemistry subunit routinely monitors the fuel pools water by chemical and radiochemical analysis of grab samples. When chemistry exceeds plant procedure limits, appropriate corrective actions are implemented to restore the parameter within its limit. The performance of the Fuel Pool Demineralizer is routinely monitored and when the ion exchange media is depleted, the resin is replaced.

The Spent Fuel Pool fission and corrosion product activities are discussed in FSAR Section 11.1.7. Design and normal operating specific activities are given in FSAR Table 11.1.7-1.

Radiological monitoring of the various samples for the subject system is described in detail in FSAR Sections 11.5.2.5 and 11.5.2.6.

The differential pressure across the flushable filter is measured with on line instrumentation. Before the differential pressure approaches 60 psig, the filter being deposited with maximum amount of crud requires a back-flushing treatment.

9.1.3.3 Safety Evaluation. All fuel pools are cooled by two independent cooling loops, either of which can remove the decay heat loads generated by the quantities of fuel through operation to the end-of-Cycle 9.

Table 9.1.3-2 provides the fuel pool heat load, equilibrium temperature, and water heat inertia for the Incore Shuffle, Full Core Offload Shuffle and Post Outage Full Core Offload cases. These three cases were evaluated based on operation through end-of-Cycle 9. For cases assuming a single active failure, a single CCW train supplies both essential and non-essential loads, resulting in reduced CCW flow to the fuel pool cooling system heat exchanger. Heat loads were calculated for the three cases above. Each of these cases modeled the spent fuel received from previous plant operation and from spent fuel from H. B. Robinson Unit 2 and Brunswick Units 1 and 2 received through end-of-Cycle 8. A bounding heat load from the additional spent fuel to be received during Cycle 9 was also addressed.

Administrative controls are placed on the minimum cooling time prior to transfer of irradiated fuel from the core to the storage facility in order to maintain the pools at less than or equal to 137°F (Reference 9.1.3-2). The minimum cooling time prior to movement of irradiated fuel in the reactor vessel addresses both radiological and decay heat considerations. The most conservative of these two are used in determining the actual required cooling time.

In the event of a single failure in one of these Spent Fuel Cooling Loops, the other loop will provide adequate cooling. The pool temperature with one Fuel Pool Cooling Loop in operation will be equal to or less than 137°F.

The maximum normal heat load which would exist in the spent fuel pools concurrent with a LOCA would be 16.84 MBTU/hr. The maximum heat load values given in FSAR Table 9.1.3-2 for the Full Core Offload Shuffle and the Post Outage Full Core Offload are not used because a LOCA is not required to be considered concurrent with these conditions (complete core unload).

When the Emergency Core Cooling System is aligned to recirculate from the containment sump to the Reactor Coolant System, the CCW trains are separated from each other and from the nonessential header to maintain protection against single passive failure and to provide sufficient flow to their respective RHR trains. Once separated, each train provides flow to its respective essential header composed of heat loads from the RHR pump and RHR Heat Exchanger. In this alignment, each CCW train is balanced to provide greater than 5 gpm to the RHR pump and 6050 gpm to the RHR Heat Exchanger.

When the CCW trains are isolated from the nonessential header, CCW flow to the Spent Fuel Pool Heat Exchanger is also isolated. At 5.56 hours from the time of LOCA initiation, the heat load in the containment sump will be low enough to permit the realignment of CCW to the spent fuel pool heat exchanger. The pools will heat up to 137°F in 5.56 hours assuming an initial temperature of 112.7°F and a normal maximum heat load subsequent to a LOCA of 16.84 Mbtu/hr. With this heat load, 2.97 hours is available for manual actions to restore CCW to the spent fuel pool heat exchanger prior to reaching 150°F in the pools. The CCW flow required to maintain the pool temperature at 150°F assuming this same heat load is 1789 gpm.

The minimum CCW flow which must be maintained through the RHR Heat Exchanger and the RHR pump subsequent to alignment to recirculation is 5600 gpm and 5 gpm, respectively. Subsequent to alignment to recirculation, operators are directed by Operating Procedures to restore sufficient CCW cooling from one CCW train to the spent fuel pools to maintain temperature less than 150°F. Based on the CCW flows established through the RHR pump and RHR Heat Exchanger when the nonessential header is isolated, each train is capable of individually providing the required 5600 gpm and 5 gpm through the RHR Heat Exchanger and RHR pump and 1789 gpm through the spent fuel pool heat exchanger assuming that all other nonessential loads are isolated. The spent fuel pool heat up time of 2.97 hours from 137°F to 150°F is sufficient to allow operators to isolate any non-essential loads and to throttle the CCW flow through the spent fuel pool heat exchanger as required. All local manual manipulations are performed in areas which are accessible subsequent to a LOCA.

To assure reliability, each of the fuel pool cooling pumps is powered from separate buses so that each pump receives power from a different source. If a total loss of offsite power should occur, the operator has the option of transferring the pumps to the emergency power source.

In addition, emergency cooling connections are provided in the loops to permit the installation of portable pumps to bypass the fuel pool cooling pumps should they become inoperable when cooling is required in either pool.

As shown on Figure 9.1.3-2, valving and blind flange connections are provided at the suction and discharge side of the fuel pool cooling pumps for emergency connection of a spare cooling pump.

Compliance of the Fuel Pool Cooling and Cleanup System to the guidance of NRC Regulatory Guide No. 1.13, "Fuel Storage Facility Design Basis," is addressed in Section 1.8.

The cooling loop piping and components are designed to Seismic Category I criteria. The cleanup loop is not designed to Seismic Category I criteria; however, suitable valving is provided between the cooling loop and the cleanup loop to permit isolation of the cleanup loop. The cooling loop portion of the FPCCS is protected against externally generated missiles. The fuel pool cooling pumps and associated piping are located in an area of the plant where there are no postulated internally generated missiles. The fuel pool cooling pumps have not been considered credible sources of internally generated missiles. The no-load speed of the pumps is equal to the synchronous speed of the electric motors; consequently, there are no pipe-break plus single failure combinations which could result in a significant increase in pump suction or discharge header. In addition, the FPCCS is protected against the effects of high energy and moderate energy fluid system piping failures (Section 3.6).

The FPCCS is manually controlled and may be shut down safely for reasonable time periods for maintenance or replacement of malfunctioning components.

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to a fuel pool, a small quantity of fission products may enter the fuel pool cooling water. The cleanup loop is provided to remove fission products and other contaminants from the water.

The cleanup loop will normally be run on an intermittent basis as required by fuel pool water conditions. It will be possible to operate the purification system with either the ion exchanger or filter bypassed. Local sample points are provided to permit analysis of ion exchanger and filter efficiencies.

In the event of a high radiation alarm in the Fuel Handling Building, the purification system will be manually started. The cleanup loop is not started automatically since the short delay to manually initiate purification would not significantly speed the reduction of contamination in the pool.

The skimmer system for the new and spent fuel pools consists of surface skimmers, a fuel pool skimmer pump, a fuel pool skimmer pump suction strainer and a fuel pool skimmer filter. The surface skimmers float on the water surface and are connected via flexible hose to the pump suction piping at various locations on the perimeter of the pools. Flow from the pump is routed through the skimmer filter and returned to the fuel pools below the water level.

Siphoning of the pools is prevented by limiting the skimmer hose length to approximately five (5) feet. In addition the skimmer system return piping enters the pool at a point five (5) feet below the normal pool water level and terminates flush with the pool liner. Therefore, water loss due to failures in the skimmer system piping would be limited to five (5) feet.

A failure of the skimmer system piping would not uncover spent fuel nor interrupt fuel pool cooling since the fuel pool cooling water suction connections are located more than five (5) feet below the normal water level.

Draining or siphoning of the spent and new fuel pools via piping or hose connections to these pools or transfer canals is precluded by the location of the penetrations, limitations on hose length, and termination of piping penetrations flush with the liner. Hoses connected to temporary equipment used in the new and spent fuel pools are administratively controlled to prevent siphoning. The fuel pool cooling water return piping terminate at elevation 279 ft., 6 in. The spent fuel pool suction piping exists at 278 ft., 6 in. and the new fuel pool exits at 277 ft., 6 in.. Normal pool water level is 284 ft., 6 in. with the top of the spent fuel at approximately 260 ft. Skimmer suction piping exits the pools at elevation 285 ft., 3 in.

The reduction of the normal pool water level by approximately 5 ft. due to any postulated pipe failure will have no adverse impact on the capability of the cooling system to maintain the required temperature and it does not effect the required shield water depth for limiting exposures from the spent fuel. The slow heatup rate of the fuel pool would allow sufficient time to take any necessary action to provide adequate cooling using the backup provided while the cooling capability for the fuel pool is being restored.

Technical Specification 3.9.11 requires a minimum amount of water coverage in the fuel pools to reduce the potential doses resulting from a fuel handling accident. This minimum water depth provides sufficient iodine removal capability to maintain both the whole body and thyroid doses well within the acceptable limits of 10CFR100 which forms the basis for this Technical Specification and the fuel handling accident doses described in Chapter 15. Technical Specification 3.9.11 requires all movement of fuel assemblies and crane operations with loads in the affected pool area be suspended and the water level restored to within its limit within four hours if the water level falls below the minimum required.

The fuel handling accident described in Section 15.7.4 was evaluated with a dropped PWR fuel assembly impacting a stored PWR fuel assembly and ultimately coming to rest in a horizontal position on top of BWR fuel assemblies seated in the BWR fuel storage racks. This scenario results in the minimum water depth above the dropped fuel assembly, which is utilized to determine conservative decontamination factors used for the removal of iodines assumed in the accident evaluation. Assumptions and inputs supporting the fuel handling accident evaluation are located in Section 15.7.4. Maintaining water level in accordance with Technical Specification 3.9.11 assures that water coverages and decontamination factors used in the Chapter 15 fuel handling accident analysis remain bounding.

Alarms are provided for the indication of fuel pool water levels. Alarms for both high and low water levels indicate changing conditions in the pools. The fuel pool low level alarm indicates the minimum required water depth. An additional alarm set at a lower fuel pool water level indicates degraded pool water capacity conditions. The high level alarm provides equipment protection as well as inventory control during pool makeup and water transfer activities.

Normal makeup for evaporative losses and small amounts of system leakage from the fuel pools is accomplished using the Demineralized Water System (DWS), although other sources, such as from the reactor makeup water storage tank or the recycle holdup tank, may also be used. The DWS connects to the fuel pools and refueling water purification pumps, spent fuel pools cooling pumps, and fuel pools skimmer pumps to permit makeup to the fuel pools, or may be directly added to the pools via hoses. The seismic Category I Refueling Water Storage Tank (RWST) may also be aligned to provide borated makeup water to the fuel pools, and a seismic Category I source of emergency makeup water is available from the Emergency Service Water (ESW) system, by connecting flexible hoses to connections on the ESW and fuel pool cooling and cleanup system piping.

Floor and equipment drain sumps and pumping systems are provided to collect and transfer FPCCS leakage to the Waste Management System. High level alarms are annunciated in the Control Room when high sump level is reached.

Fuel handling equipment is designed such that the equipment cannot fall into the pool under SSE conditions (Section 9.1.4). In addition, the Fuel Handling Building is tornado missile resistant (Section 3.5).

The new fuel pool and spent fuel pools are furnished with stainless steel liners. Although they are classified as non-Nuclear Safety, the fuel pool liners are designed and constructed to the applicable portions of the ASME Code, Section III and they are subject to the Quality Assurance Criteria of 10 CFR 50, Appendix B. Other portions of the fuel transfer system in the Fuel Handling Building which are in communication with the new and spent fuel pools; namely, the fuel transfer canal, the main fuel transfer canal and the fuel cask loading pit, are also furnished with stainless steel liners.

Although these liners are qualified to the same requirements as the fuel pool liners, it is impossible for leakage in these portions of the fuel transfer system to jeopardize the inventory of cooling water in the fuel pools due to a difference in floor elevation. These areas may also be isolated from the fuel pools by gates.

A Permanent Cavity Seal Ring (PCSR) has been installed in the annulus of the reactor cavity adjacent to the refueling cavity. The PCSR is furnished with eight hatch covers which are closed and tested prior to flood-up for refueling. The PCSR is classified as nuclear safety related, subject to the quality assurance provisions of 10CFR50 Appendix B. It is designed and constructed to the applicable portions of the ASME Code Section III, Subsection ND, but is not code stamped by an ANI.

Piping and components of the Fuel Pool Cooling and Cleanup System are designed to the applicable codes and standards listed in Section 3.9. Those portions of the FPCCS required to ensure cooling of the fuel pool are Safety Class 3, since their prolonged failure could result in the release to the environment of normally retained gaseous radioactivity. Piping in contact with fuel pool water is austenitic stainless steel.

Fuel pool nozzles shall be stainless steel Seismic Category I designed and fabricated to ASME Section III, Subsection No. ND. However, they are classified as NNS.

9.1.3.4 Inspection and Testing Requirements. Provisions are incorporated in the layout of the system to allow for periodic inspection, using visual and monitoring instrumentation. Equipment is arranged and shielded to permit inspection with limited personnel exposure.

Preoperational and startup tests as described in Section 14.2.12 were conducted in the FPCCS. Periodic tests are required as described in the Technical Specifications. Inservice inspection requirements are described in Section 6.6 and pump and valve testing will be performed as described in Section 3.9.6.

Prior to initial fill, vacuum box testing was performed on the major liner field joints normally exposed to water.

Components of the system were cleaned and inspected prior to installation. Demineralized water was used to flush the entire system. Instruments were calibrated and alarm functions checked for operability and setpoints during testing. The system was operated and tested initially with regard to flow points, flow capacity and mechanical operability.

Data will be taken periodically during normal system operation to confirm heat transfer capabilities, purification efficiency, and differential pressures across components.

Kopp Exhibit 20

U.S. NUCLEAR REGULATORY COMMISSION

# STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

## 9.1.2 SPENT FUEL STORAGE

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Chemical Engineering Branch (CMEB)

### I. AREAS OF REVIEW

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

The ASB reviews the spent fuel storage facility design including the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, the spent fuel pool liner plate, and the associated equipment storage pits to assure conformance with the requirements of General Design Criteria 2, 4, 5, 61, 62, and 63.

1. The facility and components are reviewed with respect to the following:

   a. The quantity of fuel to be stored.

   b. The design and arrangement of the storage racks for maintaining a subcritical array during all conditions.

   c. The degree of subcriticality provided along with the analysis and associated assumptions.

   d. The effects of external loads and forces on the spent fuel storage racks, pool, and liner plate (e.g., safe shutdown earthquake, crane uplift forces, missiles, and dropped objects).

   e. Design codes, materials compatibility, and shielding requirements.

f. The use of applicable codes and standards consistent with the assigned seismic classification.

2. The ASB review of the pool's water level control system, cleanup system and cooling system is performed with the spent fuel cooling system review in SRP Section 9.1.3.

3. The ASB review of provisions to preclude dropping the spent fuel shipping cask into the pool are evaluated during the review of the cask loading pit area in SRP Section 9.1.5.

4. ASB also performs the following reviews under the SRP sections indicated:

   a. Review of flood protection is performed under SRP Section 3.4.1.

   b. Review of the protection against internally generated missiles as well as missiles generated by natural phenomena is performed under SRP Sections 3.5.1.1, 3.5.1.2, 3.5.2, and 3.5.1.4.

   c. Review of structures, systems, and components to be protected against externally generated missiles is performed under SRP Section 3.5.2.

A secondary review is performed by the Chemical Engineering Branch (CMEB) and the results of its evaluation are used by ASB to complete the overall evaluation of the system. The CMEB reviews the compatibility and chemical stability of the materials wetted by the pool water. In addition, CMEB will verify that there are no potential mechanisms that will: (1) alter the despersion of the strong fixed neutron absorbers incorporated in the design of the storage racks, and/or (2) cause physical distortion of the tubes retaining the stored fuel assemblies. The results of CMEB's evaluation are transmitted to ASB for inclusion in the spent fuel storage SER writeup.

In addition, ASB will coordinate reviews performed by other branches, and the results are used by ASB in the overall spent fuel storage evaluation. The coordinated reviews are as follows: The Structural Engineering Branch (SEB) determines the acceptability of the design analyses, procedures, and criteria used to established the ability of seismic Category I structures to withstand the effects of natural phenomena such as safe shutdown earthquakes (SSE), the probable maximum flood (PMF), and missiles as part of its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5. The Core Performance Branch (CPB) determines that the criticality limits are acceptable and in accordance with ANS 57.2 paragraphs 5.1.1.2.1 and 5.1.1.2.2 as part of its primary responsibility for SRP Section 4.3. The Mechanical Engineering Branch (MEB) determines that the components and structures are designed in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 through 3.9.3. The MEB also determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2. The Materials Engineering Branch (MTEB) verifies that inservice inspection requirements are met for system components as part of its primary review responsibility for SRP Section 6.6. The review for Fire Protection, Technical Specifications, and Quality Assurance is coordinated and performed by the Chemical Engineering Branch, Quality Assurance Branch, and Licensing Guidance Branch as part of their primary review responsibilities for SRP Sections 9.5.1, 16.0, and 17.0, respectively. The Equipment Qualification Branch reviews the seismic qualification of Category I instrumentation and the environmental qualification of mechanical and electrical equipment as part of its primary review responsibility for SRP Sections 3.10 and 3.11, respectively.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

## II. ACCEPTANCE CRITERIA

Acceptability of the spent fuel storage facility design as described in the applicant's safety analysis report (SAR) is based on certain General Design Criteria and Regulatory Guides, and on independent calculations and staff judgments with respect to system functions and component selection. The design of the spent fuel storage facility is acceptable if the integrated design is in accordance with the following criteria:

1. General Design Criterion 2, as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes. Acceptance for meeting this criterion is based on conformance to position C.3 of Regulatory Guide 1.13, the applicable portions of Regulatory Guide 1.29, Regulatory Guide 1.117, and ANS 57.2 paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2, and 5.3.4.

2. General Design Criterion 4, as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of environmental conditions and external missiles, and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks, such that safety functions will not be precluded. Acceptance for meeting this criterion is based on meeting position C.3 of Regulatory Guide 1.13, Regulatory Guides 1.115 and 1.117, as well as appropriate paragraphs of ANS 57.2.

3. General Design Criterion 5, as it relates to shared structures, systems, and components important to safety being capable of performing required safety functions.

4. General Design Criterion 61, as it relates to the facility design for fuel storage and handling of radioactive materials. Acceptance for meeting this criterion is based on conformance to position C.1 and C.4 of Regulatory Guide 1.13 and the appropriate paragraphs of ANS 57.2. Acceptance is also based on meeting the fuel storage capacity requirements noted in subsection III.1 of this SRP section.

5. General Design Criterion 62, as it relates to the prevention of criticality by physical systems or processes utilizing geometrically safe configurations. Acceptance for meeting this criterion is based on conformance to position C.1 and C.4 of Regulatory Guide 1.13 and the appropriate paragraphs of ANS 57.2.

6. General Design Criterion 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions. Acceptance for meeting this criterion is based on conformance with paragraph <u>5.4 of ANS 57.2.</u>

## III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design meet the acceptance criteria given in subsection II. For the review of the operating license (OL) application, the review procedures and acceptance criteria will be utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design. The OL review includes verification that the content and intent of the technical specifications prepared by the applicant are in agreement with requirements for system testing, minimum performance, and surveillance developed as a result of the staff's review.

Upon request from the primary reviewer, the coordinating review branches will provide input for the areas of review stated in subsection I of this SRP section. The secondary review branch, CMEB, will provide an input on a routine basis for those areas of review indicated in this SRP section. The primary reviewer (ASB) obtains and uses such input as required to assure that this review procedure is complete.

The review procedures given below are for a typical storage system. Any variance of the review, to take account of a proposed unique design, will be such as to assure that the facility design conforms to the criteria in subsection II of this SRP section. The reviewer selects and emphasizes material from this SRP section as may be appropriate for a particular case.

1. The SAR is reviewed to determine that the design bases and facility description section indicates the storage capacity provided in the design. The minimum storage capacity in the spent fuel storage pool shall be in accordance with ANS 57.2 paragraph 5.1.15, i.e., for a single unit facility the storage capacity shall equal or exceed one full core discharge plus the maximum normal fuel discharge cycle; for a dual shared storage pool facility the storage capacity shall equal or exceed one full core discharge plus two normal fuel discharge cycles. Due to a lack of sufficient away-from-reactor (AFR) storage capacity, the industry trend has been to use high density storage racks. ASB reviews high density storage on a case-by-case basis.

2. The information provided in the SAR relating to the facility design criteria, safety evaluation, system description, and the layout drawings for the spent fuel pool and storage racks is reviewed to verify that:

   a. Criticality information (including the associated assumptions and input parameters) in the SAR must show that the center-to-center spacing between fuel assemblies and any strong fixed neutron absorbers in the storage racks is sufficient to maintain the array, when fully loaded and flooded with nonborated water, in a subcritical condition. A $K_{eff}$ not greater than 0.95 for this condition is acceptable.

   b. The design of the storage racks is such that a fuel assembly cannot be inserted anywhere other than in a design location.

c. Failures of nonsafety-related systems or structures not designed to seismic Category I that are located in the vicinity of the spent fuel storage facility are reviewed to assure that their failure will not cause an increase in $K_{eff}$ to exceed the maximum allowable. The SAR description section, the general arrangement and layout drawings, and the tabulation of seismic design classifications for structures and systems are reviewed and evaluated to assure that this condition is met. A statement in the SAR establishing the above condition as a design criterion is acceptable at the CP review stage.

d. Design calculations should show that the storage racks and any anchorages can withstand the maximum fuel handling equipment uplift forces without an increase in $K_{eff}$ or a decrease in pool water inventory. A statement in the SAR that excessive forces cannot be applied due to the design of the fuel handling equipment is acceptable if justification is presented. The evaluation procedures identified in SRP Sections 9.1.4 and 9.1.5 are used to validate this statement.

e. Conventionally the plant's Technical Specification states that the weight of all loads being handled above stored spent fuel shall not exceed that of one fuel assembly and its associated handling tool. This weight and its normal carrying height above the storage racks establishes what was considered the upper bound on the potential energy available to damage the stored spent fuel if a load drop occurs. It has been subsequently noted that lighter loads handled at greater drop heights may have greater amounts of potential energy. Therefore, the following additional requirement is being made. The licensee is required to demonstrate and the reviewer to verify that the available potential energy of all lighter loads, being handled above stored spent fuel, shall not exceed that of one fuel assembly and its associated handling tool when dropped from its normal operating height above stored spent fuel.

f. Sharing of storage facilities in multi-unit plants will not increase the potential for the loss of pool water or decrease the degree of subcriticality provided.

3. The reviewer verifies that the safety function of the facility will be maintained, as required, if the facility is subjected to adverse natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. In making this determination, the reviewer considers the following points:

   a. The facility design basis and criteria and the component classification tables are reviewed to verify that the spent fuel storage facility including the storage pool, pool liner, and racks have been classified and designed to seismic Category I requirements. The ASB will accept a statement that the facility will be designed and constructed as a seismic Category I system. (CP)

   b. If the spent fuel pool liner plate will not be designed and constructed to seismic Category I requirements, the spent fuel pool liner plate is reviewed to verify that a failure of the liner plate as a result of an SSE will not cause any of the following:[1]

1. Significant releases of radioactivity due to mechanical damage to the fuel;

2. Significant loss of water from the pool which could uncover the fuel and lead to release of radioactivity due to heatup;

3. Loss of ability to cool the fuel due to flow blockage caused by a portion or one complete section of the liner plate falling on top of the fuel racks;

4. Damage to safety-related equipment as a result of the pool leakage; and

5. Uncontrolled release of significant quantities or radioactive fluids to the environs.

c. The essential portions of the spent fuel storage system are reviewed to verify that protection from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles is provided. Flood protection and missile protection criteria are discussed in sections of the SRP contained in Chapter 3. The reviewer utilizes the information in those SRP sections, as appropriate, to assure that the analyses presented are valid. ASB will accept a statement to the effect that the storage facility is located in a seismic Category I structure that is missile and flood protected.

4. The safe handling of spent fuel assemblies necessitates the underwater transfer of spent fuel between the respective areas of the plant including spent fuel cask loading area. The SAR is reviewed to verify that the design basis and facility description section has stated that a separate spent fuel shipping cask loading area (pit) has been provided adjacent to the spent fuel pool. The reviewer verifies that the loading pit has been designed so that the safety function of the integrated system will be maintained during adverse environmental conditions. In addition, the reviewer verifies that the following are included in the design:

   a. An interconnecting fuel transfer canal should be capable of being isolated from the fuel pool and cask loading area. A statement in the SAR that these features are included in the design is acceptable. The reviewer uses engineering judgment to assure himself that the means provided meet the stated intent.

[1]The implementation of this item reflects current regulatory practice. The methods of review described herein will be used in the evaluation of submittals for operating license or construction permit applications docketed after November 17, 1977, which is based on the first application to which this method was specifically applied. Implementation for applications docketed prior to November 17, 1977 is not considered necessary since stresses induced in the fuel pool liner plate welds due to an SSE will usually be well below the maximum allowable stress levels and therefore liner failure is not considered a likely event. Even in the event that a liner plate failed, it would not likely block the coolant outlet of spent fuel assemblies completely and sufficient cooling of stored spent fuel would be maintained. Therefore, the spent fuel pool liner plate seismic design is not considered a significant safety issue and backfit is not required.

b. In regard to the handling of heavy loads, e.g., the spent fuel shipping cask in the vicinity of the spent fuel storage pool, the reviewer is required to establish and verify in SRP Section 9.1.5 that one of the alternative approaches described in Section 5 of NUREG-0612 has been satisfied. If Sections 5.1.1 and 5.1.6 of NUREG-0612 have not been met, the SAR safety evaluations, results of design calculations, and the general arrangement and layout drawings should show that the spent fuel loading put has been designed to withstand the loads from dropped heavy objects including the shipping cask, and that the loading area is not an integral part of the storage pool floor so that if a dropped object should breach the pit area, loss of fuel pool water would not result in an unacceptable level.

## IV. EVALUATION FINDINGS

The reviewer verifies that the information provided and his review support conclusions of the following type, to be included in the staff's safety evaluation report:

The spent fuel storage facility includes the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, and the associated equipment storage pits. Based on the review of the applicant's proposed design criteria, design bases, and safety classification for the spent fuel storage facility and the provisions necessary to maintain a subcritical array, the staff concludes that the design of the spent fuel storage facility and supporting systems is in conformance with the Commission's regulations as set forth in General Design Criteria 2, 4, 5, 61, 62, and 63.

This conclusion is based on the following:

1. The applicant has met the requirements of General Design Criterion 2 by conforming with position C.3 of Regulatory Guide 1.13 and the applicable portions of Regulatory Guides 1.29 and 1.117, as well as paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2, and 5.3.4 of ANS 57.2.

2. The applicant has met the requirements of General Design Criterion 4 pertaining to the environmental and missile protection design basis by conforming to position C.3 of Regulatory Guide 1.13 and the applicable portions of Regulatory Guides 1.115 and 1.117, as well as appropriate paragraphs of ANS 57.2.

3. The applicant has met the requirements of General Design Criterion 5 since the failure of any portion of the shared spent fuel storage facility will not impair the ability of plants systems to perform their safety function.

4. The applicant has met the requirements of General Design Criteria 61 and 62 pertaining to fuel storage, handling, criticality, and radioactivity control by conforming to positions C.1 and C.4 of Regulatory Guide 1.13 and the appropriate paragraphs of ANS 57.2.

5. The applicant has met the requirements of General Design Criterion 63 pertaining to monitoring the status of the stored spent fuel by conforming to paragraph 5.4 of ANS 57.2.

## V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff on its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced NUREG and Regulatory Guides.

## VI REFERENCES


1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

5. 10 CFR Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

6. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel and Waste Storage."

7. Regulatory Guide 1.13, "Design Objectives for Light-Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

8. Regulatory Guide 1.29, "Seismic Design Classification."

9. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

10. Regulatory Guide 1.117, "Tornado Design Classification."

11. ANS 57.2/ANSI N210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

12. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Kopp Exhibit 22

Malte Edenius
Hans Häggblom
Bengt-Herman Forssén

# CASMO-3

## A FUEL ASSEMBLY BURNUP PROGRAM

## Methodology

ABSTRACT

This report describes the methodology used in CASMO-3. The reader should also consult the CASMO-3 User's Manual, which describes other aspects of the program and its usage.

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CCC-660
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