Reference 2

Northeast
Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(860) 447-1791
Fax (860) 444-4277

The Northeast Utilities System

APR 17 2000

Docket No. 50-423
B18077

Re:10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 3
Modification of Proposed Revision to Technical Specification - Spent
Fuel Pool Rerack (TSCR 3-22-98)

In a letter dated March 19, 1999,[1] Northeast Nuclear Energy Company (NNECO) submitted proposed changes to selected Technical Specifications (TS) in support of a planned modification of the Millstone Unit No. 3 spent fuel pool. The proposed changes modify the TS to allow for the installation and use of additional storage racks in the Millstone Unit No. 3 spent fuel pool. Included in this submittal were proposed changes to TS 3.9.1.2 and 4.9.1.2 which would have revised the specification as follows;

1. The APPLICABILITY of this specification would be changed to require surveillance of spent fuel pool Boron concentration only during times of fuel movement within the spent fuel pool,
2. The required Boron concentration would be changed to 800 parts per million (ppm), and
3. The measurement of Boron concentration would be performed every seven days only during fuel movement.

On February 9, 2000, the Atomic Safety and Licensing Board, in a prehearing conference order,[2] granted a request for hearing on this proposed change based on the admissibility of certain (proposed) contentions filed by the petitioners for intervention – the Connecticut Coalition Against Millstone and the Long Island Coalition Against Millstone. After careful evaluation of the contentions, it is NNECO's opinion

[1] R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Proposed Revision to Technical Specification, Spent Fuel Pool Rerack (TSCR 3-22-98)," dated March 19, 1999.

[2] LBP-00-02, Atomic Safety and Licensing Board, Prehearing Conference Order (Granting Request for Hearing), dated February 9, 2000.

that the proposed change in the APPLICABILITY of TS 3.9.1.2 is central to each of the three contentions admitted. Consequently, NNECO believes that a modification of the proposed revision to TS 3.9.1.2 is in the best interest of all Millstone Unit No. 3 stakeholders. The modified proposal would retain the existing Nuclear Regulatory Commission (NRC) approved APPLICABILITY requirement of TS 3.9.1.2, thereby requiring that the proposed Boron concentration of 800 ppm be maintained whenever fuel is stored in the spent fuel pool. Conforming changes are also required for the proposed wording of Surveillance Requirement 4.9.1.2 to reflect this change in applicability. All other changes proposed under TSCR 3-22-98 are unaffected.

Description of Proposed Modification to TSCR 3-22-98

TSCR 3-22-98 is modified to retain the existing NRC approved APPLICABILITY statement for TS 3.9.1.2. Conforming changes are also made to the wording of the proposed revision to Surveillance Requirement 4.9.1.2. The balance of TSCR 3-22-98 is unaffected by this change.

Markup of Proposed Modifications to TSCR 3-22-98

A copy of the page affected by the proposed modification to TSCR 3-22-98 is provided as Attachment 1. The modified markup is meant to be a complete replacement for the markup of TS 3.9.1.2 and the associated INSERT C as presented in the March 1999 submittal. The balance of TSCR 3-22-98 is unaffected by this change. To present the difference as clearly as possible, the modified proposal has been identified by the pen and ink changes noted on INSERT C. Other pending TS Amendments are not reflected in the enclosed markup.

Retype of Proposed Modifications to TSCR 3-22-98

A copy of the retyped TS page affected by the proposed modification to TSCR 3-22-98 is provided as Attachment 2. The retyped page reflects the incorporation of the modified change to the TS 3.9.1.2, and is meant to be a complete replacement for the original retyped page presented in the March 1999 submittal. The balance of TSCR 3-22-98 is unaffected by this change. Other pending TS Amendments are not reflected in the enclosed retype.

Safety Summary, Significant Hazards Consideration and Environmental Considerations

The proposed modification to TSCR 3-22-98 reduces the scope of the change previously justified by retaining the APPLICABILITY requirement currently incorporated into the NRC approved version of TS 3.9.1.2. On this basis, NNECO concludes that the modified proposal does not affect the conclusions of the Safety Summary, Significant Hazards Consideration, or Environmental Considerations assessments as presented in the March 1999 submittal. The Background, Safety Summary, Significant Hazards Consideration, and Environmental Considerations for the balance of changes proposed in TSCR 3-22-98 are bounded by those presented in the March 1999 submittal.

Plant Operations Review Committee and Nuclear Safety Assessment Board Review

The proposed modification to TSCR 3-22-98 has been reviewed by the Plant Operations Review Committee and the Nuclear Safety Assessment Board. These committees concur with the conclusions relative to safety presented in the rationale for the proposed change.

State Notification

In accordance with 10 CFR 50.91(b), we are providing the State of Connecticut with a copy of the modified proposal.

Schedule

NNECO requests review and approval of the initial March 1999 submittal and this proposed revision by June 2000.

There are no regulatory commitments contained within this letter.

If the NRC Staff should have any questions or comments regarding this submittal, please contact Mr. D. W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

R. Necci

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 17 day of April, 2000

Donna Lynne Williams
Notary Public

Date Commission Expires: Nov. 30, 2001

cc: See next page

Attachments (2)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A.C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Ms. Nancy Burton, Esq.
147 Cross Highway
Redding Ridge, CT 06876

Docket No. 50-423
B18077

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Modification to TSCR 3-22-98
Marked Up Pages

4/9/98

LIMITING CONDITION FOR OPERATION

3.9.1.2 The boron concentration of the Spent Fuel Pool shall be maintained uniform and sufficient to ensure that the boron concentration is greater than or equal to 1750 ppm.

Applicability

Whenever fuel assemblies are in the spent fuel pool.

Action

a. With the boron concentration less than 1750 ppm, initiate action to bring the boron concentration in the fuel pool to at least 1750 ppm within 72 hours, and

b. With the boron concentration less than 1750 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 1750 ppm every 72 hours.

Replace w/ INSERT C

7

INSERT C

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be maintained uniform, and greater than or equal to 800 ppm.

Applicability

~~During all fuel assembly movements within~~ Whenever fuel assemblies are in the spent fuel pool.

Action

With the spent fuel poo' soluble boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool.

Surveillance Requirements

4.9.1.2 Verify that the soluble boron concentration is greater than or equal to 800 ppm ~~prior to any movement of a fuel assembly into or within the spent fuel pool, and~~ every 7 days. ~~thereafter during fuel movement.~~

Docket No. 50-423
B18077

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Proposed Modification to TSCR 3-22-98
Retyped Pages

REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be maintained uniform, and greater than or equal to 800 ppm.

Applicability

Whenever fuel assemblies are in the spent fuel pool.

Action

With the spent fuel pool soluble boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the soluble boron concentration is greater than or equal to 800 ppm every 7 days.

Reference 3

Northeast Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(860) 447-1791
Fax (860) 444-4277

The Northeast Utilities System

MAY - 5 2000

Docket No. 50-423
B18025

Re:10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 3
Response to Requests for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)

In a letter dated March 19, 1999,[1] Northeast Nuclear Energy Company (NNECO) submitted a proposed revision to the Millstone Unit No. 3 Technical Specifications for Spent Fuel Pool Rerack. The proposed changes modify the Technical Specifications to allow for additional racks to be installed in the Millstone Unit No. 3 spent fuel pool (SFP) in order to maintain full core reserve capability.

In response to this submittal, the Nuclear Regulatory Commission (NRC) requested additional information in the form of two sets of questions. An NRC memorandum dated March 14, 2000,[2] proposed a set of five questions related to SFP procedures in a revised draft request for additional information. The answers to those questions are presented in Attachment 1 to this letter. An NRC memorandum dated February 25, 2000,[3] proposed a separate set of four questions related to SFP design and structure. The answers to those questions are presented in Attachment 2.

(1) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Proposed Revision to Technical Specification, Spent Fuel Pool Rerack (TSCR 3-22-98)," dated March 19, 1999.

(2) Memorandum from Victor Nerses to James Clifford, "Millstone, Unit 3, Draft Request for Additional Information, Spent Fuel Pool Rerack (TAC No. MA5137)," dated March 14, 2000.

(3) Memorandum from Victor Nerses to James Clifford, "Millstone, Unit 3, Draft Request for Additional Information, Spent Fuel Pool Rerack (TAC No. MA5137)," dated February 25, 2000.

Subsequently, in a letter dated April 17, 2000,[4] NNECO submitted a modification of the proposed revision to the Millstone Unit No. 3 Technical Specifications for Spent Fuel Pool Rerack. The modified proposal would retain the existing applicability requirement for boron concentration, thereby requiring that the proposed boron concentration of 800 ppm be maintained whenever fuel is stored in the SFP.

There are no regulatory commitments contained within this letter.

If the NRC Staff should have any questions or comments regarding this submittal, please contact Mr. David Dodson at (860) 447-1791, extension 2346.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 5 day of May, 2000

Donna Lynne Williams
Notary Public

Date Commission Expires: Nov 30, 2001

Attachments (2)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

---

(4) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98)," dated April 17, 2000.

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Response to Requests for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)

Responses to Revised Draft RAI dated March 14, 2000

Responses to Revised Draft RAI dated March 14, 2000

1. What will be the minimum and maximum boron concentrations in the spent fuel pool specified by chemical procedures if your submitted amendment is approved?

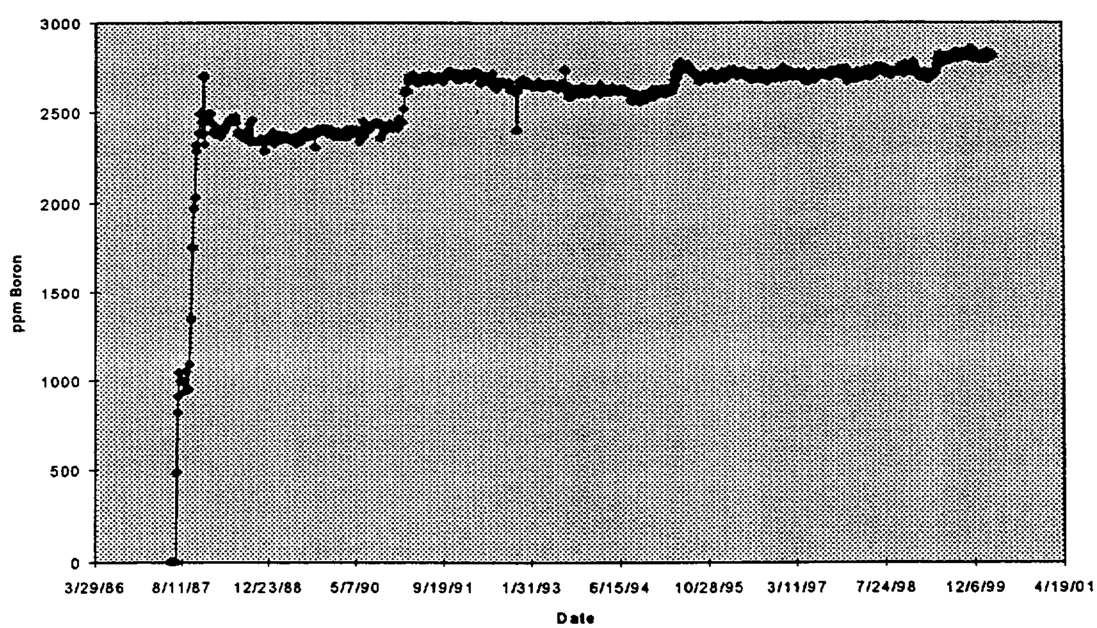
Response

NNECO will maintain the spent fuel pool (SFP) soluble boron concentration ≥ 2600 ppm at all times in accordance with chemistry procedures. This is done as a matter of operational convenience since the SFP boron concentration must be ≥ 2600 ppm during refuelings (per Technical Specification 3.9.1.1) when the SFP and refueling cavity are connected. A value of ≥ 2600 ppm is bounding on all Technical Specification (TS) requirements, including the proposed TS 3.9.1.2 as modified on April 17, 2000[1].

There is no specified maximum SFP boron concentration.

NNECO has historically maintained the SFP boron concentration at high values. The administrative limit of ≥ 2600 ppm was instituted in 1997. Shown below is a plot of Millstone Unit No. 3 SFP boron concentration measurements since the SFP water was initially borated in 1987:

MP3 SFP Boron Concentration



[1] R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98)," dated April 17, 2000.

2. **What is the frequency for surveillance and what are the procedures for surveillance of these boron concentrations?**

Response

The present TS 3.9.1.2 requires a minimum SFP soluble boron concentration of 1750 ppm whenever fuel is stored in the SFP. Surveillance of the boron concentration is performed at least once per 72 hours as required by surveillance procedure SP 3866, "Spent Fuel Pool Boron Concentration." The proposed TS 3.9.1.2, as modified on April 17, 2000, requires a minimum SFP soluble boron concentration of 800 ppm whenever fuel is stored in the SFP. Upon implementation of the proposed TS, the surveillance frequency in SP 3866 will be revised to every 7 days. The 800 ppm concentration is based on the licensing basis criticality analysis, with substantial margin applied.

SP 3866 also requires that the Shift Manager, Reactor Engineering, and Chemistry be notified if boron concentration is less than 2600 ppm. This requirement will be retained.

The weekly surveillance frequency is appropriate because no major replenishment of SFP water or significant change in boron concentration is expected to take place over such a short period of time, a basis that is consistent with Standard Technical Specifications. During the period between weekly SFP boron surveillances, it would take approximately 500,000 to 1,000,000 gallons, depending on the method of dilution, of unborated water to dilute the SFP boron concentration from 2600 ppm to 800 ppm. The volume of the SFP is about 450,000 gallons. An unintentional dilution of this magnitude would be quickly detected either at the source of the unborated water, or by its effect on SFP water level.

The proposed modifications do not affect existing TS 3.9.1.1, which effectively requires that the SFP soluble boron concentration be ≥ 2600 ppm when the SFP and refueling cavity are connected during Mode 6 operation. Surveillance procedure SP 3863, "Reactor Coolant and Reactor Vessel Refueling Cavity Analysis for Boron," implements the boron monitoring requirements of TS 3.9.1.1, and this procedure is unaffected by the proposed changes.

3. **Please describe the administrative procedure used to determine that fuel assemblies have attained proper burn-up for storage in the burn-up dependent racks.**

Response

Surveillance procedure SP 31022, "Spent Fuel Pool Criticality Requirements," controls the process of ensuring that fuel assemblies have attained proper burnup for storage in the burnup-dependent fuel storage region. Currently, Region 2 is the only region of the SFP that has a fuel burnup restriction.

The proposed TS changes will result in a total of three burnup-dependent fuel storage regions in the SFP. SP 31022 will be revised for use with the proposed SFP modifications by expanding the process used to evaluate fuel assemblies for any of the three burnup-dependent fuel storage regions. Provisions to incorporate fuel decay time in the evaluation will also be covered in this procedure so that fuel assemblies may be subsequently relocated based on their actual fuel decay time.

NNECO will perform 10 CFR 50 Appendix B (QA) calculations to determine measured fuel burnups as follows. This aspect of spent fuel management is unaffected by the proposed TS changes.

- The Westinghouse INCORE (or future equivalent) QA computer code will be used to generate measured core power distribution maps. The accuracy of plant power distribution measurements is discussed in WCAP-7308-L-P-A.
- The Westinghouse TOTE (or future equivalent) QA computer code will be used to generate measured individual fuel assembly burnups, using the INCORE measured core power distribution maps. Analytical inputs to TOTE will be determined using QA calculations. An independent review of the INCORE maps will also be documented in these QA calculations. The resulting measured fuel assembly burnups will be documented in QA calculations.

Each fuel assembly to be placed in a burnup-dependent fuel storage region is evaluated per SP 31022, which includes a requirement for independent review. Fuel assemblies may be qualified either individually, or as a group provided the combination of highest initial enrichment and lowest burnup is used in the batch qualification process. Fuel enrichments used in this process can be either the design enrichment value, which is documented by the fuel vendor under their QA program, or the as-built enrichments which are also reported by the vendor per their QA program. It should be noted that the as-built enrichment is bounded by the design enrichment which is limited to the licensed enrichment value for Millstone Unit No. 3. The measured fuel burnup value is documented and then reduced by an appropriate uncertainty value. The result is then checked against the regional TS limits. If the fuel burnup is greater than that required by a regional TS limit, the fuel is qualified for storage in that SFP region. When a fuel assembly or group of assemblies is determined to be qualified for storage in a particular burnup-dependent region, the fuel assembly ID or fuel group ID is entered on a controlled Qualified Fuel Assemblies form which lists all fuel assemblies qualified for storage in each burnup-dependent region.

As a future alternative to qualifying each fuel assembly per SP 31022, QA calculations may be performed to qualify fuel assemblies for each storage region. In either case, whether SP 31022 or a QA calculation is used, an independent reviewer will be used to ensure that each fuel assembly is correctly qualified for regional storage.

4. **Is there any procedure for verifying that fuel assemblies in the spent fuel pool are in the correct locations after fuel movements have ceased?**

Response

NNECO believes that the existing controls for proper fuel assembly placement in the SFP are sufficient, and coupled with the requirement for 800 ppm boron concentration in the SFP whenever fuel is stored in the SFP, reduce the probability of an inadvertent criticality to an appropriately low value.

Verification of correct fuel assembly location in the SFP after fuel movements is currently accomplished by a combination of several proceduralized inspection and tracking processes. These practices provide reasonable assurance that each fuel assembly in the Millstone Unit No. 3 inventory, whether in the core or in the SFP, resides in its specified location. The processes and procedures used for the current SFP design will be revised for use with the proposed SFP modifications by expanding their application to three burnup-dependent fuel storage regions.

All fuel assembly movements are controlled as Special Nuclear Material (SNM) under the direct supervision of qualified Reactor Engineering or licensed Operations personnel. Procedural controls and physical equipment constraints limit fuel assembly movements in the SFP to only one fuel assembly at a time.

Fuel assembly movements into and out of the SFP are controlled in accordance with engineering procedure EN 31001, "Supplemental SNM Inventory and Control," which requires two personnel, the SNM Executor and the SNM Checker, for all fuel assembly movements. The following description illustrates the methodology that confirms the correct placement of fuel assemblies in the SFP.

From initial core fuel load to the present, the serial number of any new fuel assembly is verified prior to moving the fuel assembly to its assigned SFP storage rack location. When moved into the SFP, there is a second verification that each fuel assembly is being placed into its specified fuel storage location. This provides an initial baseline location for every fuel assembly brought into Millstone Unit No. 3.

For fuel assemblies loaded or reloaded into the reactor core, a serial number verification is again performed, in accordance with plant procedures EN 31001, "Supplemental SNM Inventory and Control," and EN 31007, "Refueling Operations," to ensure that each fuel assembly has been placed into its proper reactor core location. In the SFP, after the core load is complete, a verification by piece-count is performed. This piece-count verification in the SFP does not check fuel assembly serial numbers, but confirms that there is a fuel assembly in each designated fuel storage location, and that no fuel assembly is present in fuel storage locations that should be empty. This is a double verification process in that it is performed by two qualified personnel who survey the SFP and prepare survey sheets as verifier and reviewer.

During core offload, fuel removal is observed and supervised by a licensed Senior Reactor Operator who has no other concurrent responsibilities during this core alteration operation. As the spent fuel is being removed from the core and moved to the transfer canal, the person moving the fuel in containment (the SNM Executor) has a set of move sheets (currently called the Refueling Worklist Form) specifying the core location from which to remove each spent fuel assembly. There is a second person (the SNM Checker) performing a verification of the removal of each fuel assembly from the proper reactor core location. Therefore, there is a second verification that each fuel assembly is being removed from the specified reactor core location. The requirements for second verification are contained in procedures MC-5, "Special Nuclear Material Inventory and Control," EN 31001, and EN 31007.

Also during core offload, as the spent fuel is being removed from the transfer canal and placed in the SFP, the person moving the fuel in the SFP (the SNM Executor) has a set of move sheets (currently called the Refueling Worklist Form) specifying the SFP storage rack location in which to place each spent fuel assembly. There is also another person (the SNM Checker) with an identical set of move sheets performing a verification of the placement of each fuel assembly into the proper SFP storage location. Therefore, there is a second verification that each fuel assembly is being placed into the specified fuel storage location. The requirements for second verification are contained in procedures MC-5, EN 31001, and EN 31007.

Additional Information

NNECO is aware of the fact that fuel handling is a multi-faceted process that on an industry-wide basis has been subject to various errors. To preclude the occurrence of similar conditions at Millstone Station, NNECO utilizes an industry Operating Experience (OE) Program that is administered by the independent Nuclear Safety Engineering Group. This OE Program is the process by which Millstone Unit No. 3 identifies and assimilates the lessons learned from events, including fuel handling, which occur within the nuclear industry into the procedures and practices specific to Millstone.

The following information provides additional insight regarding the likelihood and probable consequences of a misloading event.

The proposed SFP is made up of three new Regions, designated Region 1, 2, and 3. Each is discussed separately below.

Once the core is reloaded, and fuel movement has been completed, the remaining fuel in the SFP is typically of low reactivity (i.e., highest measured burnup). In this case, there would not be any fuel left in the SFP that could cause a violation of either the proposed Region 1 or Region 2 TS burnup requirements. All of the fuel-assemblies (approximately 500 assemblies) currently in the Millstone Unit No. 3 SFP meet the proposed TS requirements for storage in either the proposed Region 1 or Region 2 racks. That is to say, every fuel assembly currently in the Millstone Unit No. 3 SFP is qualified to be stored in any of the proposed Region 1 or Region 2 SFP storage locations. Therefore, after refueling fuel movement is complete, to violate the proposed Region 1 or Region 2 TS burnup requirements, there would have to be: (1) a premature permanent discharge of a "very reactive fuel assembly" (such a fuel assembly currently does not exist in the Millstone Unit No. 3 SFP), and (2) that particular "very reactive fuel assembly" would have to be misloaded into Region 1 or 2, despite the double verification that each moved fuel assembly is loaded into the proper SFP location.

The proposed Region 3 racks are the existing spent fuel racks which will still contain boraflex as an active neutron absorber, but boraflex will no longer be credited for reactivity control. The proposed Region 3 is the region most likely to encounter an accidental misloading event, since fuel would normally be present in the SFP which is not qualified for this region. However, there is expected to be very little fuel movement into or out of the Region 3 racks since their primary intended purpose is for long term fuel storage. Furthermore, for a misloading event to occur, double verification that the fuel assembly is being properly located would have to fail. In addition, for a misloading event to have any impact on the SFP K_{eff}, a fuel assembly must be misloaded such that the fuel assembly is placed in a region for which it is not qualified. Given the minimal fuel movement activity associated with these racks, and the double verification requirements described above, the probability of a fuel misloading event having an impact on SFP K_{eff} is very low.

Even in the unlikely event that a single fuel assembly was misloaded in any region of the SFP, with no credit for soluble boron, criticality would not result, although the SFP K_{eff} limit of .95 could be exceeded. In the limiting case, where a single fresh (nominal enrichment of 5.00 w/o) fuel assembly is postulated to be misplaced or accidentally dropped in Region 3, the presence of >425 ppm soluble boron in the water ensures that K_{eff} is maintained <0.95. (Note: 425 ppm is a calculated value; a value of 800 ppm has been selected for conservatism in the proposed TS.) Furthermore, as described in the response to Question 1, NNECO maintains the SFP at ≥ 2600 ppm soluble boron at all times.

In summary, the following conditions will exist following completion of fuel movement during a refueling:

(1) The proposed Region 1 and Region 2 TS burnup limits are low enough that the fuel typically remaining in the SFP following fuel movement could be placed in any Region 1 or Region 2 storage location. Therefore, under normal conditions there should be no possible fuel misloading event that could impact the SFP K_{eff} for Region 1 or Region 2.

(2) The proposed Region 3 racks will have very little fuel movement into or out of these racks since their primary intended purpose is long term storage of spent fuel. To

have an impact on SFP K_{eff}, the misloading event must be such that a fuel assembly placed in a Region 3 storage location is not qualified for Region 3 storage. Given the minimal fuel movement activity associated with Region 3 racks, and the double verification requirements described above, the probability of a fuel misloading event having an impact on SFP K_{eff} is very low for Region 3.

(3) Even if a single fuel misload event should occur such that it impacted SFP K_{eff}, maintaining the SFP soluble boron concentration per the proposed TS as modified at a minimum of 800 ppm will preclude a criticality event. 800 ppm is almost double the concentration that is necessary to maintain the SFP K_{eff} <0.95 with a single fuel misloading. Per the proposed TS as modified, the SFP soluble boron concentration will be surveilled on a weekly basis.

5. Where are these procedures documented?

The controls discussed in responses to Questions 1 through 4 are maintained in approved plant procedures. The specific procedure numbers are included within the applicable responses.

Docket No. 50-423
B18025

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Response to Requests for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)

Responses to Draft RAI dated February 25, 2000

Responses to Draft RAI dated February 25, 2000

Reference:

Letter, dated March 19, 1999 from R. P. Necci, to U.S. NRC, "Millstone Nuclear Power Station, Unit No. 3 - Proposed Revision to Technical Specification – Spent Fuel Pool Rerack (TSCR 3-22-98)," Attachment 5 titled "Licensing Report for Spent Fuel Rack Installation at Millstone Nuclear Station Unit 3."

1. **You indicated in Chapter 6 of the Reference cited [above] that the structural analyses of the spent fuel racks for the required loading conditions were performed in compliance with the US NRC Standard Review Plan (SRP) and the former US NRC Office of Technology (OT) position paper related to spent fuel storage. With respect to your structural analyses using the DYNARACK computer code:**

   (a) **Explain how the target (design basis) response spectra (referred to in Section 6.4 of the Reference) was obtained.**

   (b) **You state in Section 6.9.1 of the Reference that the low value (i.e., 1.03 inches) of the maximum rack displacement (shown in the Table titled "Rack Displacement Results") indicates that rack overturning is not a concern. Justify this statement by providing the results of the rack overturning analyses that identify that the design criteria related to kinematic stability (i.e., minimum safety factors against rack overturning of 1.5 for OBE and 1.1 for SSE specified in SRP 3.8.5) are satisfied.**

Response to 1.(a)

The target response spectra referred to in Section 6.4 of the referenced Licensing Report were obtained by broadening and smoothing the plant response spectra for the fuel pool floor (Fuel Building Elevation 11'-0") in accordance with the requirements of Regulatory Guide 1.122 and Table 1.8-1 of the Millstone Unit No. 3 FSAR. This was accomplished by expanding the frequency range around. each peak from –15% to +15% of the peak's frequency value. The resulting curve was then smoothed by increasing the acceleration values so as to envelop the original spectrum curve.

Response to 1.(b)

In order to demonstrate that the spent fuel racks are kinematically stable, two single rack overturning runs were performed (Run No. 20 on page 6-22 and Run No. 33 on page 6-23 in the Licensing Report). Rack C1 and Rack D5 were selected for this overturning run because they have the highest aspect ratio (i.e., length/width ratio), which makes the rack prone to overturning. Furthermore, these overturning runs were each subjected to 1.5 times the Safe Shutdown Earthquake (SSE), which is greater than the 1.1 amplifier set forth in SRP 3.8.5.

From the results, the maximum computed displacements at the rack top for Run No. 20 is 0.492 inches (see page 6-26 of the Licensing Report) and 1.02 inches for Run No. 33 (see page 6-27 of the Licensing Report). To reach the incipient point of overturning, the top of rack C1 must displace nearly 54.24 inches (distance between pedestal centerlines) and top of rack D5 must displace nearly 60.0 inches (distance between pedestal centerlines). Therefore, the minimum safety factor against rack overturning for rack C1 is about 53 [= 54.24 in/1.03 in] and for rack D5 is about 58 [= 60.0 in/1.03 in]. These safety factors clearly satisfy the kinematic acceptance criteria stated in Chapter 2.0 (page 2-2) and Subsection 6.7.1 of the Licensing Report with a very large margin.

2. (a) **Section 7.4.2 "Deep Drop Events" in the Reference states that the "deep drop" through an interior cell does produce some deformation of the baseplate and localized severing of the baseplate/cell welds. You further indicate that the fuel assembly support surface is displaced by a maximum of 2.9 inches, which is less than the distance of 4-5/8 inches from the baseplate to the liner. Provide the design limit of the allowable deformation of the baseplate, and discuss the impact of the localized severing of the baseplate/cell wall welds on the integrity of the racks and the fuel assemblies.**

   (b) **In the same section on Deep Drop Events cited above, you state that the deep drop event whereby the impact region is located above the support pedestal produces a negligible deformation on the baseplate, and a maximum stress in a localized region is limited to only 25 ksi. Provide the maximum stress in the concrete slab, and the failure limits of the stresses in the liner and in the concrete slab, citing the references which give these failure limits.**

Response to 2.(a)

The design limit of allowable deformation of the baseplate is specified to be 4-5/8 inches for the mechanical accident which ensures that a fuel drop to the baseplate should not lead to a second impact between the baseplate and the spent fuel pool liner. The LS-DYNA simulation results for the "deep drop" accident indicate that the baseplate does not fail during the impact, but the baseplate/cell welds immediately adjacent to the impact location are partially severed (see Response Reference [2.1]). The maximum calculated Von Mises stress in the baseplate is 48.86 ksi which is less than the material failure stress limit (stainless steel SA240-304L) of 66.2 ksi. As described in Chapter 3.0 of the Licensing Report, there are four cell-to-baseplate welds for each cell and all cells are inter-connected to each other by cell-to-cell welds along the cell height. Localized damage to the welds in the rack honeycomb structure has little consequence to the structural integrity of the rack. The results also show that the stored fuel assemblies will remain separated by the cell walls after the postulated accident. It should be noted that the impactor (i.e., the dropped fuel assembly and the associated tools) is modeled as a rigid body, which conservatively channels all the impact energy into the target (i.e., the baseplate). Therefore, the baseplate/cell wall welds will not be severed to the extent as predicted by the LS-DYNA simulation.

Response to 2.(b)

The failure stress of the liner material (stainless steel SA240-304) is 71 ksi, which is given in Response Reference [2.2]. The static unconfined compressive strength of the pool slab concrete is 4000 psi. The concrete failure limits for a dynamic event should be much higher than the static limit, as suggested by many credible textbook references. Laterally confined and simultaneously subjected to water pressure of the spent fuel pool, the upper stratum of the pool slab exhibits a tri-axial compressive stress behavior, which also reduces the tendency of internal cracking. In the deep drop analysis, a nonlinear "piecewise-linear" stress-strain curve is used to characterize the behavior of the pool slab concrete under tri-axial compression. The curve is an extrapolation of the stress-strain curve experimentally obtained for the concrete with unconfined compressive stress of 3,660 psi and subjected to tri-axial compression. The latter is shown in Fig. 2.19 of the textbook "Reinforced Concrete Structures" by Park and Paulay (Response Reference [2.3]). This curve was further adjusted to coincide with the actual unconfined compressive strength of 4000 psi. Based on this stress-strain curve, the failure stress is 20.2 ksi.

The deep drop analysis results show that the concrete slab experiences a maximum localized (peak normal) compressive stress of 25.2 ksi, which exceeds the failure stress of 20.2 ksi. This indicates that the concrete slab would experience localized crushing. However, the result also indicates that the high stress region is located directly beneath the pedestal and is limited to a circular area whose diameter is less than 5 inches. The rest of the slab area is in tension with a maximum stress of 112 psi, a value that is easily supported by the concrete without cracking.

Response References

[2.1] Mechanical Accident Analysis for Millstone Unit 3, Holtec Report No. HI-81889.

[2.2] ASME, "Boiler & Pressure Vessel Code," Section II, Part D – Material Properties, 1995.

[2.3] R. Park and T. Paulay, "Reinforced Concrete Structures," Figure 2.19, John Wiley and Sons, 1975.

3. You indicated in Chapter 8 of the Reference that the design conditions described in SRP 3.8.4 and American Concrete Institute (ACI) Code 349-85 were used as guidance in the calculations of the spent fuel pool (SFP) capacity. With respect to the SFP capacity calculations using the ANSYS computer code discussed in Chapter 8 of the Reference, explain how the interface between the liner and the concrete slab is modeled, and also how the liner anchors are modeled; explain how such modeling accurately represents the real structural behavior.

Response

The pool liner is not included in the overall 3-D ANSYS structural model of the spent fuel pool. Any contribution to the pool structural support by the thin liner is conservatively neglected. The stress analysis of the liner is considered in a separate stress analysis, using the ANSYS computer code, focused on the in-plane stress distribution. The liner in the Millstone Unit No. 3 pool is assembled from austenitic steel plates which are seam welded along the contiguous edges of the plates resulting in a sealed container geometry to hold pool water. The seam weld lines are also locations of anchor. The stress analysis of the pool liner was evaluated against the following criteria, which were met:

1) In-plane stresses in the liner during the seismic event will not cause rupture in the liner from a single load application.

2) Repetitive loading during a seismic event will not cause fatigue failure in the liner (1 SSE and 20 OBEs occurring in sequence is the design basis).

To evaluate the stress field in the liner, it is modeled as a 2-D plate, which is fixed along its edges to simulate the weld seams. The liner anchors are assumed to be rigid, and therefore, are not explicitly modeled. A bounding geometry was utilized wherein the anchor lines are conservatively assumed to be nearest to the pedestal location. The finite element solution evaluated the stress distribution at the line of support representing the weld seam.

Thus, the finite element models conservatively predict stresses in the fuel pool structure and fuel pool liner.

4. **Provide a Table showing the maximum bulk pool temperature for the three discharge scenarios (Section 5.3 in the Reference), and discuss the basis for allowing the bulk pool temperature to exceed the code allowable temperature of 150°F for any of the scenarios, if such a condition exists.**

Response

Discharge Scenario	Temperature
1	150°F
2	150°F
3	148.8°F

The bulk spent fuel pool (SFP) temperature analysis performed for Millstone Unit No. 3 calculates the minimum core hold time by limiting the bulk pool temperature to 150°F for Scenarios 1 and 2. For Scenario 3, the maximum calculated bulk temperature at the end of a four hour loss of forced cooling is 148.8°F.

Therefore, the code allowable temperature limit of 150°F is not exceeded for any of the three scenarios.

It should be noted that as part of a separate plant design change and license amendment request related to full core off-load, a single active failure of the SFP cooling system was evaluated. The assumed event is coincident with the instant when the last fuel assembly of a full core off-load is transferred to the pool and the pool is postulated to be at its limiting 150°F initial temperature. A failure is assumed to disable the active train of cooling and 30 minutes is required to put the standby train into service. SFP bulk temperature would increase to approximately 155°F before cooling was restored and the bulk temperature returned to below 150°F. The design of the SFP structure and support systems were verified acceptable against this elevated temperature. However, since this evaluated event is conservatively assumed to occur at the completion of the off-load, it has no impact on the subject scenarios.

Northeast Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(860) 447-1791
Fax (860) 444-4277

The Northeast Utilities System

Reference 4

JUN 16 2000

Docket No. 50-423
B18113

Re: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 3
Response to Request for Additional Information,
Spent Fuel Pool Rerack (TAC No. MA5137)

In a letter dated March 19, 1999,[1] Northeast Nuclear Energy Company (NNECO) submitted a proposed revision to the Millstone Unit No. 3 Technical Specifications for Spent Fuel Pool Rerack. The proposed changes modify the Technical Specifications to allow for additional racks to be installed in the Millstone Unit No. 3 spent fuel pool (SFP) in order to maintain full core reserve capability.

On May 2, 2000,[2] the Nuclear Regulatory Commission (NRC) requested additional information on various radiological considerations associated with the installation and long term operation of the Millstone Unit No. 3 SFP. The answers to those questions are presented in Attachment 1 to this letter.

A telephone conference between NNECO and the NRC staff was held on May 8, 2000, to discuss the basis for the proposed revisions to Technical Specification (TS) 3.9.1.2, including modifications submitted by NNECO on April 17, 2000.[3] At that time, the Staff

(1) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Proposed Revision to Technical Specification, Spent Fuel Pool Rerack (TSCR 3-22-98)," dated March 19, 1999.

(2) Memorandum from Victor Nerses to James W. Clifford, "Millstone, Unit No. 3, Draft Request for Additional Information, Spent Fuel Rerack Amendment (TAC No. MA5137)," dated May 2, 2000.

(3) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98)," dated April 17, 2000.

provided its position that the proposal be further modified to include remedial actions in the event the soluble boron concentration is reduced below the proposed acceptance limit. NNECO concurred with the Staff position and it was agreed that the proposed change would be modified such that the ACTION requirements contained in the current NRC approved version of Specification 3.9.1.2 would be retained. On this basis, this supplemental modification does not impact the safety assessment or the no significant hazards determination provided with the original submittal. Attachment 2 provides the revised marked-up TS page. Attachment 3 provides the associated retyped TS page.

An additional telephone conference was held on May 25, 2000, between representatives of NNECO and the NRC Staff. At that time, the Staff requested clarifications regarding heavy load handling information provided in the March 19, 1999,(1) submittal. Attachment 4 provides NNECO's response to that request.

There are no regulatory commitments contained within this letter.

If the NRC Staff should have any questions or comments regarding this submittal, please contact Mr. David Dodson at (860) 447-1791, extension 2346.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 16th day of June, 2000

Notary Public

Date Commission Expires: Jun 30 2004

Attachments (4)

cc: See next page

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Docket No. 50-423
B18113

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Response to Request for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)
Responses to Draft Request for Additional Information Dated May 2, 2000

Responses to Draft RAI Dated May 2, 2000

1. **Discuss how the increased number of spent fuel assemblies stored in the Millstone Unit No. 3 SFP will affect the dose rates in any accessible areas below the refueling deck and adjacent to the SFP walls (including any accessible areas below the SFP). State whether the storage of an increased number of spent fuel assemblies in the Millstone Unit No. 3 SFP will necessitate any radiation zoning changes to any of the surrounding areas.**

Response

The rerack shielding analysis calculated that dose rates at the Millstone Unit No. 3 spent fuel pool (SFP) wall outer surface due to stored fuel assemblies in the reracked pool will be a maximum of 2.5 mR/hr. This result is considered conservative because it is based on the following conservative assumptions:

- All fuel assemblies have a burnup of 60,000 MWD/MTU
- All fuel assemblies have decayed only 100 hours
- Core power is 3,636 MW(t), vs. actual rated power of 3,411 MW(t)
- The source consists of multiple fuel assemblies all located at the fuel pool wall

This calculated dose rate value represents an increase in the maximum dose rate from current negligible values, however, this value is well within the design basis values for the original SFP design. Therefore, the increased number of spent fuel assemblies stored in the SFP will not require radiation zoning changes in any accessible areas surrounding the SFP.

Regarding dose rates underneath the fuel pool, the SFP sits on bedrock. Thus, there are no accessible areas below the SFP.

2. **Provide a description of any sources of high radiation, other than spent fuel assemblies, that may be in the Millstone Unit No. 3 SFP during any diving operations needed to remove underwater appurtenances and to install the new fuel racks. Discuss what precautions (such as fuel shuffling, removal of high radiation sources, use of TV monitoring, diver tethers, use of physical or visual barriers, etc.) will be used to ensure that the divers will maintain a safe distance from any high radiation sources in the SFP.**

Response

Sources of high radiation in the SFP other than fuel assemblies and fuel assembly inserts such as burnable poison rod assemblies (BPRAs) include:

- Lock tabs (stored on pool floor, northeast corner).
- Thimble plugs (stored on pool floor, northeast corner).

- Vacuum filters (stored in the cask pit area, which is far removed from diving operations).

Precautions to reduce exposure to diving personnel include:

- The installation contractor will be Underwater Construction Corporation (under the direction of Holtec International) which is very experienced in safe diving evolutions for SFP reracks.
- Diver tethers with tenders will be used to keep divers within prescribed areas.
- Diver exposure will be minimized as the result of a spent fuel shuffle that has already been performed.
- The Millstone Unit No. 3 SFP has much open floor space, and most diving operations will be performed with a significant distance between the diver and existing fuel racks.
- While lock tabs and thimble plugs would not produce significant diver exposure in their present locations, they will be moved farther away from the planned diving area to further reduce diver exposure.
- Visual contact with the divers will be maintained during all diving operations using underwater TV cameras.

3. Discuss the need for any additional lighting in or above the SFP to ensure that both the diver work area is adequately illuminated and the dive tenders above the SFP can maintain visual surveillance of the divers in the SFP at all times.

<u>Response</u>

The following activities will provide adequate lighting of the diving work area, and ensure that dive tenders can maintain visual surveillance of the divers at all times:

- The permanent overhead and underwater lighting in the SFP has been evaluated by NNECO and determined to provide adequate general illumination for most anticipated diving and spent fuel rack installation operations.
- The installation contractor is tasked with providing and installing additional portable lighting to locally support diving and rack installation operations as necessary.
- The project specific diving procedure requires that the diver and the responsible Health Physics technician concur that the underwater lighting level is adequate for each underwater diving operation.

4. **Describe how you plan to monitor the doses received by the divers during the reracking operation (e.g., use of extremity or multiple TLDs, alarming dosimeters, remote readout radiation detectors). Describe how you plan to maintain continuous communication with the divers while they are in the SFP.**

Response

Doses received by divers will be monitored using a multiple dosimetry package to include extremity monitoring, alarming dosimetry, and teledose. Continuous voice communication with the divers will be maintained while they are in the SFP using dedicated communication equipment. This equipment will be provided by Holtec International and approved for use by NNECO Health Physics.

5. **Describe how you plan to survey the portions of the SFP where divers may be used to ensure that you have an accurate dose rate map of these underwater areas. Verify that you will perform updated dose rate surveys in the SFP any time that there is a change in location of the high radiation sources in the SFP.**

Response

NNECO Health Physics Operations Procedure RPM 2.2.8, "Underwater Radiological Surveys," is used to perform SFP underwater surveys. Accurate pre-diving dose maps are ensured by the use of two independent underwater survey meters and the recording of dose rates on survey maps containing specified grid points. In accordance with Health Physics Operations Procedure RPM 2.5.1, "Health Physics Requirements for Diving Evolutions," if the work area radiological survey is greater than 24 hours old or any fuel or high radiation component has been moved within the underwater work area, a pre-dive work area survey must be verified prior to a diving evolution.

Assessment surveys were taken during the rerack project ALARA planning period. As identified in response to Question 2, fuel assemblies and BPRAs affecting the rerack work area have already been moved, and other high radiation sources near the work area have been identified for relocation prior to diving operations. There are no plans to move high radiation sources in the SFP during the scheduled rerack diving period.

6. **Discuss your plans to use a vacuum to remove any crud or other debris from the floor of the SFP before and during the SFP re-racking project to maintain diver doses ALARA.**

Response

Recent radiological surveys of the planned diving areas in the SFP indicate that exposure levels are low, and there is no significant crud or discernable debris. Normal SFP maintenance practices will provide assurance that prior to starting the reracking project, the pool floor will remain free of any significant crud or debris.

NNECO plans to vacuum the pool floor after divers complete the removal of underwater appurtenances, primarily to support Foreign Material Exclusion control. NNECO anticipates little or no debris generation from other portions of the rerack installation process, particularly since existing fuel racks will not be removed from the fuel pool or otherwise disturbed.

Health Physics will perform underwater surveys during the periods of diving operations, and will require pool vacuuming should it become necessary to maintain diver doses ALARA.

7. **The re-racking of the SFP will result in storage space for roughly 1100 additional fuel assemblies. Discuss what effect the storage of additional fuel assemblies in the SFP will have on the overall evaporation rate from the SFP area and whether this increased evaporation rate will result in an increase in the amount of gaseous tritium released from the SFP.**

Response

Increases in SFP bulk water temperature result in a corresponding increase in SFP evaporation rate. The storage of additional fuel assemblies in the SFP has the potential to increase bulk water temperature and thus increase overall evaporation rate. However, for the proposed rerack change there will be no increase in the design evaporation rate for the SFP, since the design storage capacity of the SFP is not being changed from the current limit of 2169 assembly locations as approved in License Amendment No. 60.[(1)] The rerack will result in an increase in the total number of physical storage locations from the present 756 locations to 1860 locations. Because the total actual storage locations will remain below the design number of locations, SFP evaporation rate and SFP cooling will remain within current design parameters.

Tritium in the SFP water comes primarily from the pool's connection to the reactor coolant system during refueling operations. Should the SFP evaporation rate increase due to the storage of additional fuel assemblies, there would be a corresponding increase in gaseous tritium release rate as well. Tritium release from buildings other than the containment is an input to the plant design for radiological effluent controls to meet the requirements contained in 10 CFR Part 20 and 10 CFR Part 50, Appendix I. Emission of residual tritium from spent fuel is a contributor to this input, and any increased emission in the SFP due to additional assemblies from a refueling would be within design basis as long as the design capacity of 2169 assemblies is not exceeded. Because the number of stored assemblies proposed by this TS change will not exceed the design capacity of 2169, any release of radioactivity, including tritium, to the environment will not exceed current design bases for radiological effluents.

---

(1) D. H. Jaffe (USNRC) letter to E. J. Mroczka, "Issuance of Amendment (TAC No. 77924)," dated March 11, 1991.

Millstone Station is required to maintain a monitoring program for radiological effluents. This monitoring program includes measurements of radioactivity in effluents and in the environment. It also includes on-going evaluations of changes in patterns of radioactive releases in order to assess the need to make changes to the program. It is for this reason that NNECO continues to monitor and evaluate the Millstone Unit No. 3 SFP as a specific source of tritium releases to the environment. If the magnitude of release of tritium from the SFP should become significant, changes would be initiated to ensure releases to the environment remain acceptable.

8. **Discuss how the storage of the additional spent fuel assemblies will affect the releases of radioactive liquids from the plant.**

Response

The storage of additional spent fuel assemblies in the SFP will have negligible effect on the releases of radioactive fluids from the plant. NNECO does not anticipate the generation of significant additional liquid radwaste as a result of this modification, either as a direct result of the rack installation process or from the operation of the reracked SFP with additional stored spent fuel assemblies.

Docket No. 50-423
B18113

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Response to Request for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)
Revised Marked-up Technical Specifications Page

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be maintained uniform, and ~~sufficient to ensure that the boron concentration is~~ greater than or equal to ~~1750~~ 800 ppm.

Applicability

Whenever fuel assemblies are in the spent fuel pool.

Action

a. With the boron concentration less than ~~1750~~ 800 ppm, initiate action to bring the boron concentration in the fuel pool to at least ~~1750~~ 800 ppm within 72 hours, and

b. With the boron concentration less than ~~1750~~ 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to ~~1750~~ 800 ppm every ~~72 hours~~ 7 days.

Docket No. 50-423
B18113

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Response to Request for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)
Revised Retyped Technical Specifications Page

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be maintained uniform, and greater than or equal to 800 ppm.

Applicability

Whenever fuel assemblies are in the spent fuel pool.

Action

a. With the boron concentration less than 800 ppm, initiate action to bring the boron concentration in the fuel pool to at least 800 ppm within 72 hours, and

b. With the boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 800 ppm every 7 days.

Attachment 4

Millstone Nuclear Power Station, Unit No. 3

Response to Request for Additional Information
Spent Fuel Pool Rerack (TAC No. MA5137)
Additional Questions Regarding Heavy Load Handling

Additional Questions Regarding Heavy Load Handling

1. **Regarding lifting devices described in Section 3.3 of the Holtec Licensing Report, provide additional detail with respect to the use of installed equipment and its interface with vendor supplied lifting devices, and the design and qualification standards applied to vendor supplied lifting devices.**

Response

The installed 10-ton new fuel receiving and 10-ton new fuel handling cranes will be used to manipulate the new storage racks upon delivery. Section 9.1.4 of the Millstone Unit No. 3 Final Safety Analysis Report (FSAR) provides a description of these load handling systems and their design capabilities. Section 9.1.5 of the FSAR discusses the degree to which these systems conform to the requirements of NUREG-0612, "Control of Heavy Loads," and NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment."

Information related to vendor supplied lifting rigs is provided in Section 3.3 of Attachment 5 to the March 19, 1999,[1] submittal. Further details are provided herein as Enclosure 1, which contains five figures depicting the rigging arrangements to be used in handling the storage rack assemblies. These figures are excerpted from the NNECO approved Millstone Vendor Procedure entitled "Onsite Handling & Installation Procedure" to be used by Holtec, and are identified as Exhibits 6.5.1 through 6.5.5. Additional information regarding the required ratings of the components to be utilized is also provided on these figures.

Additionally, all lifting devices employed in this evolution are required to be certified in accordance with Millstone Common Maintenance Procedure C MP 713B, "Lifting and Handling Equipment - Identification and Certification of Contractor Supplied Equipment." Compliance with the requirements of this procedure is required by the bid specification for Holtec rack installation services. As specified within the procedure;

> Contractor-supplied equipment for use at Millstone Station must meet the requirements of the following applicable ANSI standards, procedures, and Federal regulations:
>
> - B30.10c-1992, "Hooks"
> - B30.21c-1992, "Manually Lever Operated Hoists"
> - B30.16-1992, "Overhead Hoists (Underhung)"
> - B30.9b-1993, "Slings"
> - 29 CFR 1910.184, "Slings" - 7/1/92

---

(1) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98)," dated March 19, 1999.

Holtec is required to provide suitable documentation of compliance with the above standards for all equipment prior to its installation and/or use at the Millstone site.

2. Identify the industry guidelines utilized to establish training standards for use of lifting, upending, and all other aspects of the rack installation process.

Response

NNECO's training program for personnel conducting rigging operations at Millstone is documented in the Millstone Rigging and Handling Program Manual. The program addresses the requirements of 29 CFR 1926.251, "Rigging Equipment for Material Handling," 29 CFR 1910.184, "Slings," and ASME B30.1, "Jacks." This program includes both classroom and practical exercises conducted over a one week period. Successful completion of this course is required in order to perform rigging evolutions at Millstone. Vendor personnel are required to either successfully complete the course or demonstrate proficiency against the course requirements through a test-out process.

3. Provide the weight of the heaviest rack module.

Response

The weight (calculated bounding value) of the heaviest rack module is 18,050 pounds.

4. Clarify the consequences of the rack drop event, particularly with respect to the consequences to the liner and estimated leakage if a liner puncture occurs. If liner puncture occurs, describe sources of makeup and their capacity with regards to the estimated leakage rate.

Response

In NNECO's March 19, 1999, submittal, it is identified on page 9 of Attachment 3 that the SFP liner is punctured and the concrete underlying the puncture zone suffers a small indentation as a consequence of the rack drop event.

The Millstone Unit No. 3 SFP is a stainless steel lined reinforced concrete structure. The liner is approximately 0.25 inches in thickness and is supported by the reinforced concrete slab which is approximately 8 feet thick. Based on information contained in the detailed Holtec report entitled "Mechanical Accident Analysis For Millstone Unit 3," the area of the puncture is roughly equivalent to that of the rack pedestal dimension (i.e., approximately 5 inches in diameter) with a corresponding indentation in the underlying reinforced concrete slab of approximately 2.7 inches. While the concrete is damaged as a result of the event, it retains its structural integrity thereby preventing a significant loss of SFP inventory.

This damage estimate is based on a quarter rack finite element analysis of the stresses induced in the liner and concrete as a consequence of the event. As such, for a single rack, this would correspond to four separate impact areas. However, due to the highly localized nature of the induced stresses, the consequences are considered to be bounded by the quarter rack analysis conclusions. Additionally, the Holtec analysis is based on a 40-foot drop in water. The maximum lift height of the rack assembly will be approximately 43.5 feet in order to clear the curb surrounding the SFP. This difference in lift height is not considered to significantly affect the outcome of the 40-foot drop evaluation.

The actual flow from the liner puncture is not estimated because the flow would essentially be limited to that being absorbed by the concrete itself, which is negligible compared to the SFP makeup capability. Any flow to the area between the liner and concrete would be significantly restricted due to the limited clearance between these elements. An impact rupture of the liner over a weld seam would be collected in the leak chase channels which are normally isolated. In the event that a significant loss of volume should occur, low level alarms in the control room would alert plant operators to the conditions and prompt entry into the appropriate emergency procedure. This procedure has provisions for gravity makeup or forced makeup to the SFP. Additional information regarding SFP makeup sources is described in FSAR Section 9.1.3.2.

In addition to the above, a contingency procedure has been prepared to effect repairs to the liner should a rack drop event of this magnitude occur.

5. Clarify item 5 of Table 3.5 regarding use of "non-customer" lifting devices.

Response

The reference in this entry is to vendor supplied lifting devices. These lifting devices are depicted on the figures provided as Enclosure 1 to this submittal.

6. Clarify the discussion at the beginning of Section 10.5 of the Holtec Licensing Report regarding upending operations.

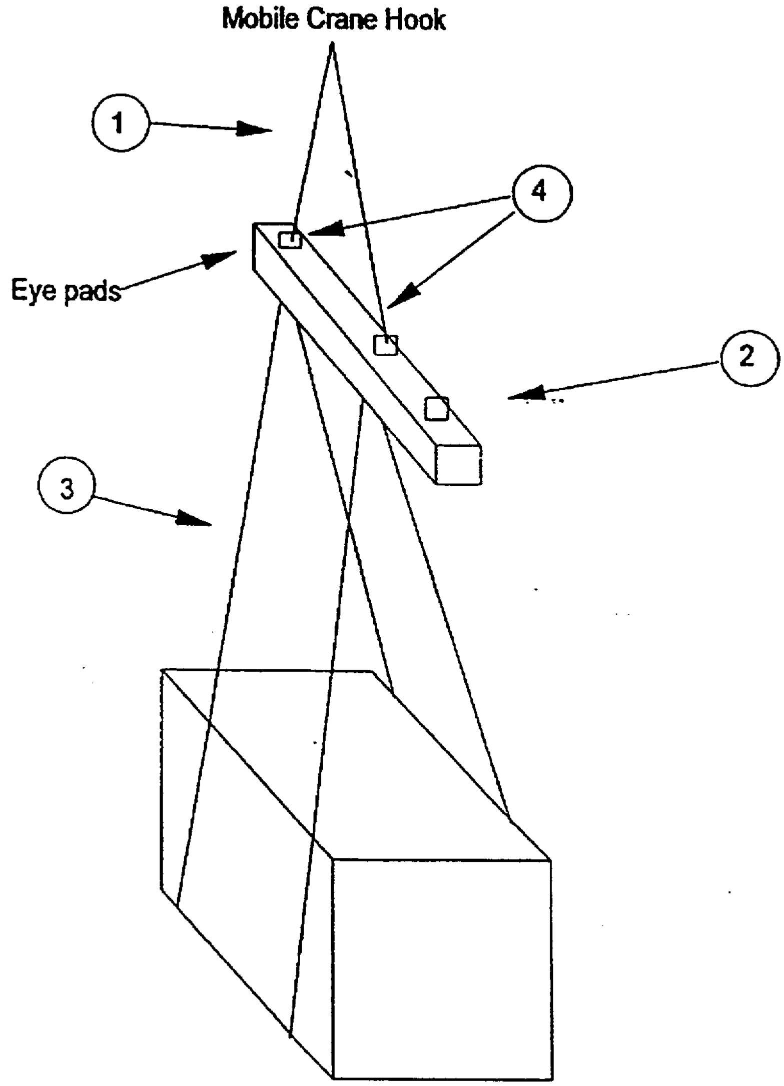
Response

The Exhibit 6.5.3 of Enclosure 1 illustrates the rigging arrangement for the upending process as well as the designated cranes to be used in this process.

Enclosure 1

Additional Questions Regarding Heavy Load Handling
Rigging Configurations - Exhibits 6.5.1 through 6.5.5

EXHIBIT 6.5.1
RACK HORIZONTAL LIFT (Storage Area)



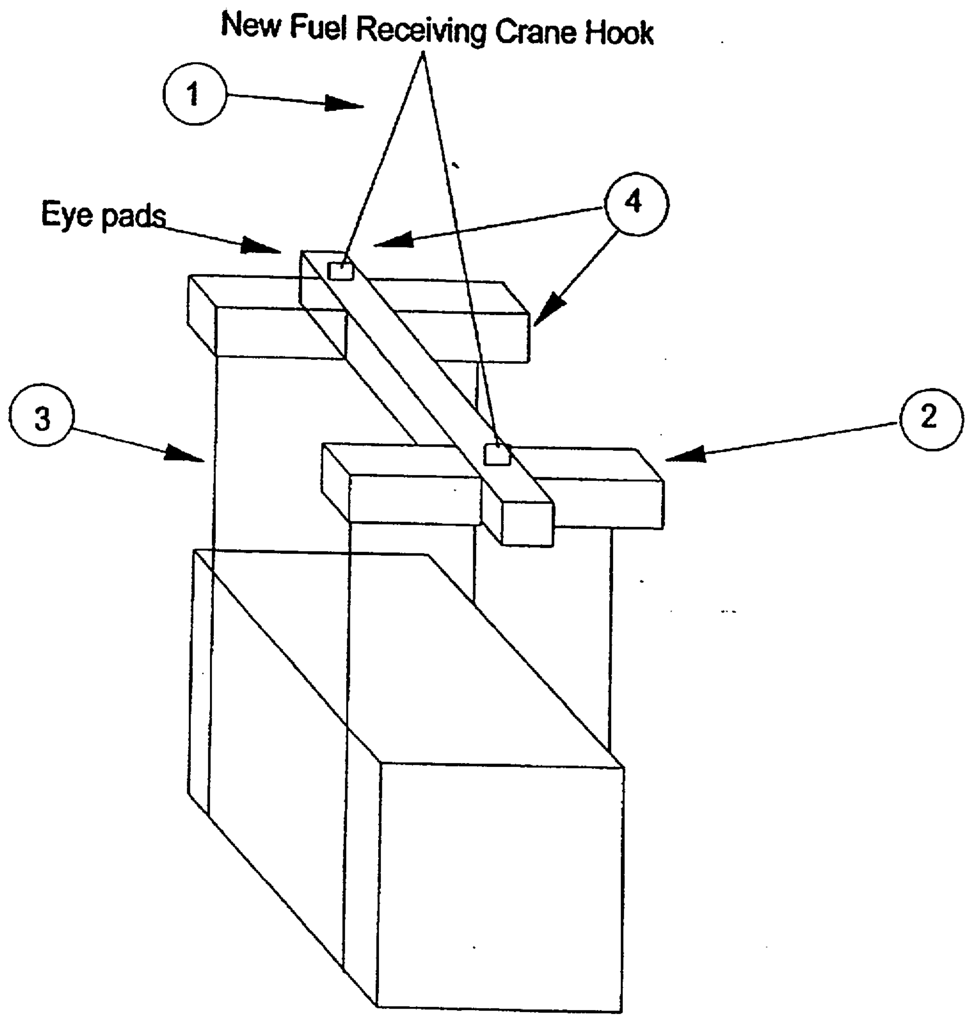
Max. rack weight = 18,050#

ITEM	QUANTITY	DESCRIPTION	MIN. RATING
1	2	NYLON SLINGS	7 TON
2	1	SPREADER BEAM	14 TON
3	2	NYLON SLINGS	10 TON BASKET
4	8	SCREW PIN SHACKLES	12 TON

NOTE:

1. All angles are a minimum of 45 degrees.
2. Additional/alternate rigging may be used as necessary as long as the minimum ratings of each piece of additional/alternate rigging meets the requirements of the above table.

EXHIBIT 6.5.2
ALTERNATE RACK HORIZONTAL LIFT



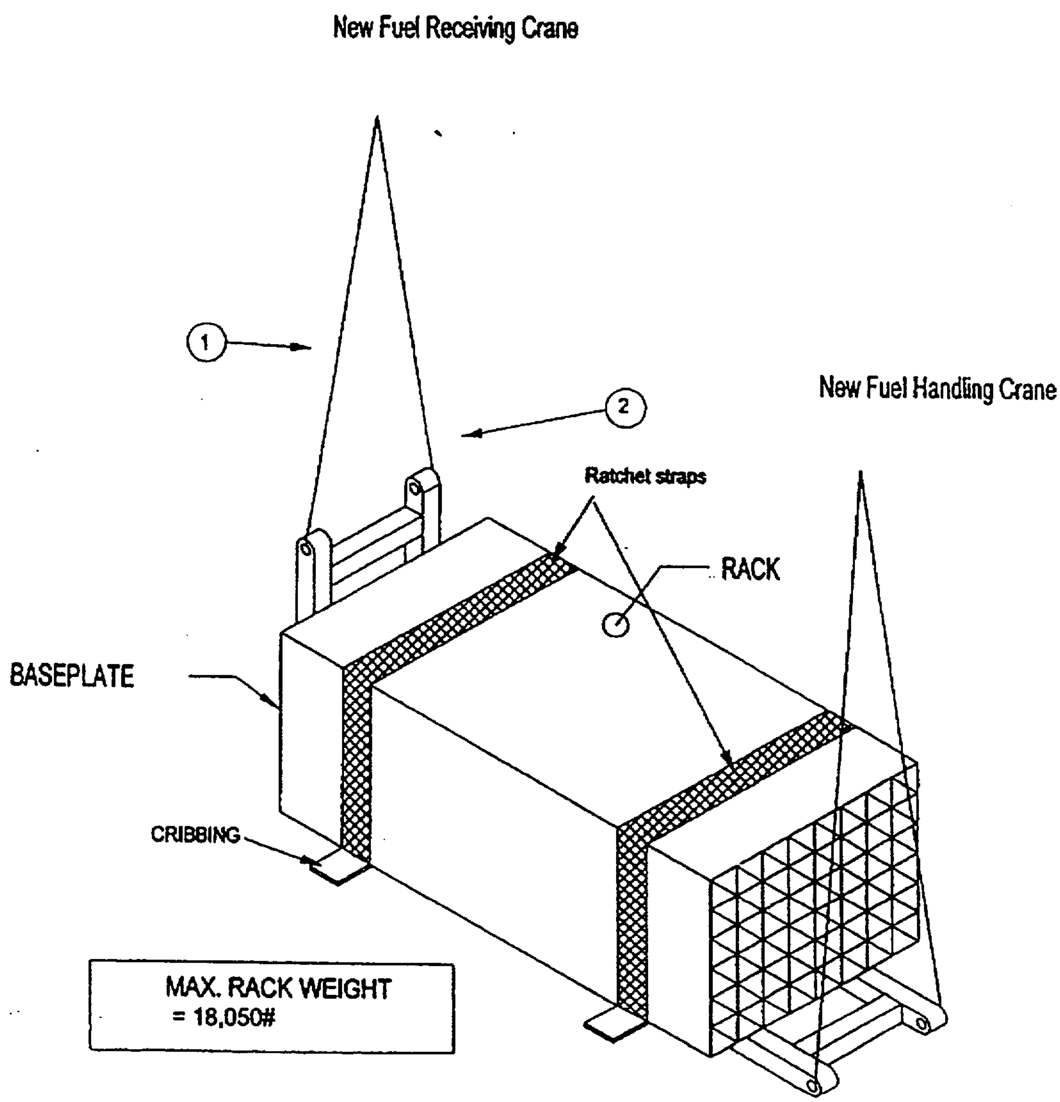
Max. rack weight = 18,050#

ITEM	QUANTITY	DESCRIPTION	MIN. RATING
1	2	NYLON SLINGS	7 TON
2	1	H-SPREADER BEAM	14 TON
3	2	NYLON SLINGS	10 TON BASKET
4	6	SCREW PIN SHACKLES	12 TON

NOTE:

1. All angles are a minimum of 45 degrees.
2. Additional/alternate rigging may be used as necessary as long as the minimum ratings of each piece of additional/alternate rigging meets the requirements of the above table.

EXHIBIT 6.5.3
RIGGING CONFIGURATION FOR RACK UPENDING

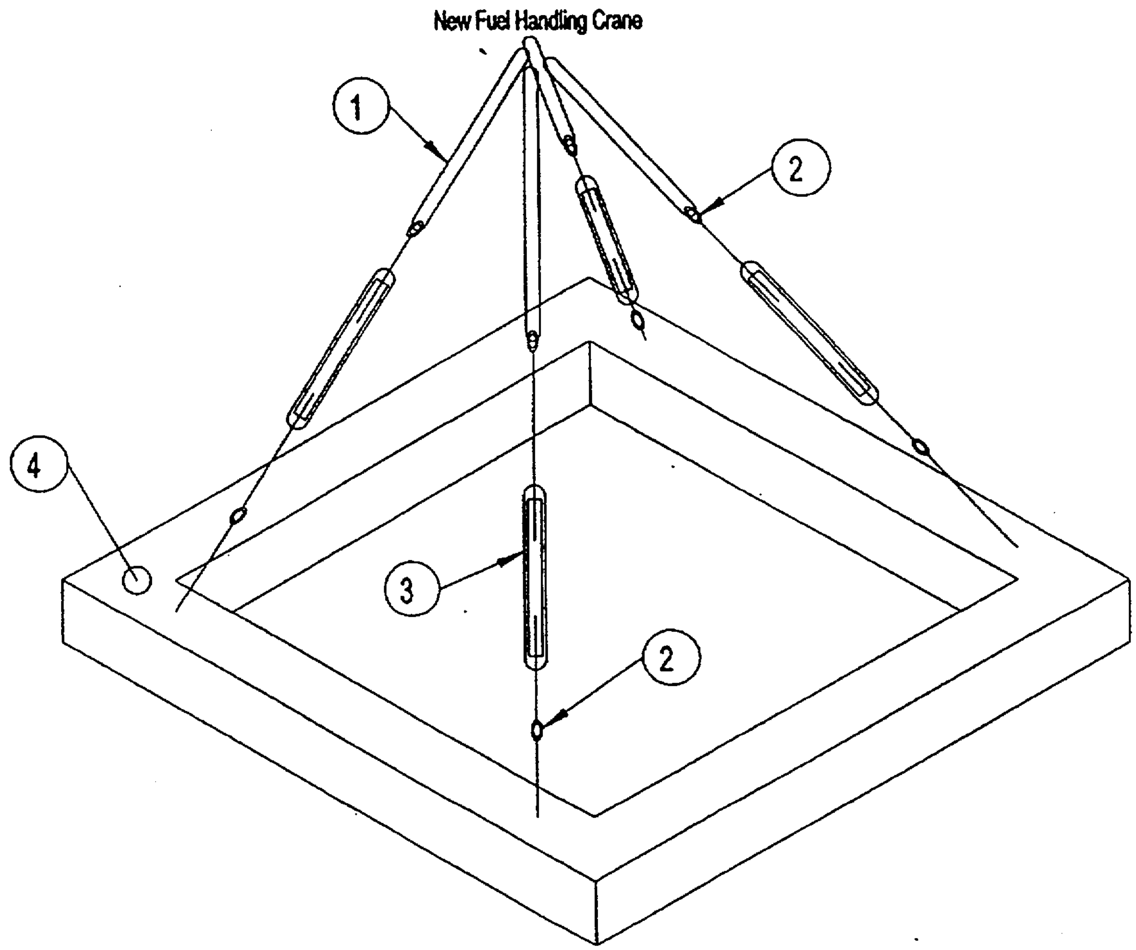


ITEM	QUANTITY	DESCRIPTION	MIN. RATING
1	4	NYLON SLINGS	8 TON
2	4	SHACKLES	12 TON

NOTE: 1. Shackles in Upender (4 min.) rated for 12T min (based on eye pad dimensions).
2. Additional/alternate rigging may be used as necessary as long as minimum ratings of each piece of additional/ alternate rigging meets the requirements of the above table.
3. Minimum angle on all rigging is 45 degrees.

EXHIBIT 6.5.4

NEW RACK LIFT RIG CONFIGURATION



MAX. RACK WEIGHT = 18050#

ITEM	QUANTITY	DESCRIPTION	MIN. RATING
1	4	NYLON SLINGS	10 TON
2	8 (or 4)	SHACKLES	10 TON
3	4	TURNBUCKLES	10 TON
4	1	HOLTEC LIFT RIG	NUREG 0612 QUALIFIED

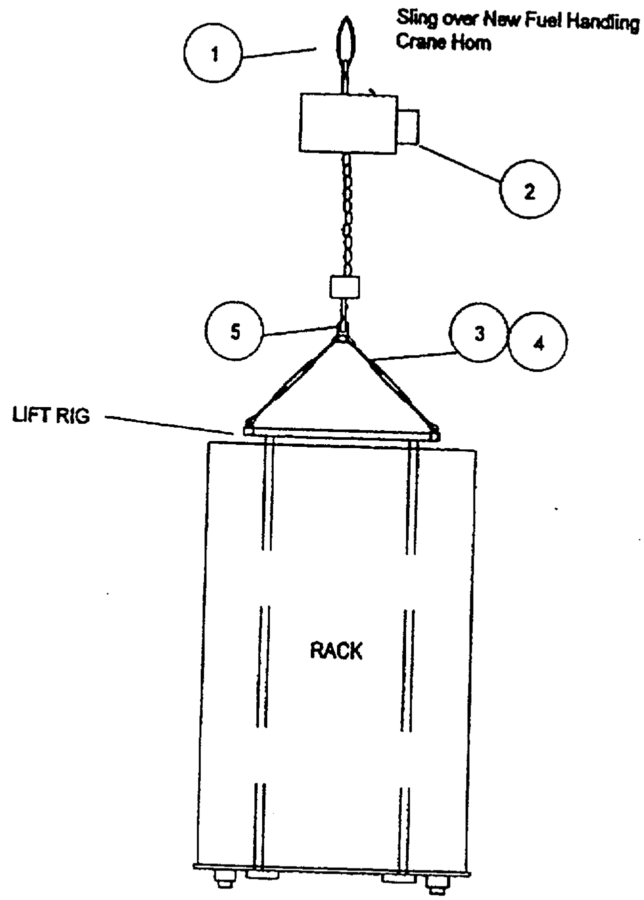
NOTES:

1) MINIMUM RIGGING ANGLE IS 45 DEGREES.

2) ADDITIONAL/ALTERNATE RIGGING MAY BE USED AS LONG AS MINIMUM RATINGS OF EACH PIECE OF RIGGING MEETS THE REQUIREMENTS OF THE ABOVE TABLE; i.e. TURNBUCKLES/SLING ARRANGEMENT MAY BE REPLACED BY JUST SLINGS OF MINIMUM REQUIRED RATING.

EXHIBIT 6.5.5

NEW RACK RIGGING CONFIGURATION



MAX RACK WEIGHT =18050#

ITEM	QUANTITY	DESCRIPTION	MIN. RATING
1	1	NYLON SLING	12 TON
2	1	ELECTRIC HOIST	10 TON
3	4	TURNBUCKLES	10 TON
4	As required	SHACKLES	10 TON
5	2	D-RING	14 TON

NOTES:

1. Minimum rigging angle is 45 degrees.
2. Rigging items may be changed as required as long as the minimum load rating shown above in the particular load path is maintained.
3. A single designed pin/pin connection device can be used in place of item 1 above. The connector will require a rating of 28 ton minimum.

ORIGINAL

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Reference 5

BEFORE THE ATOMIC SAFETY AND
LICENSING BOARD

- - - - - - - - - - - - - - - - - - x

In the matter of:	Docket No. 50-423-LA-3
NORTHEAST NUCLEAR ENERGY COMPANY	ASLBP No. 00-771-01-LA
(Millstone Nuclear Power Station, Unit No. 3)	

- - - - - - - - - - - - - - - - - - x

DEPOSITION OF: DAVID LOCHBAUM

Taken before Robin L. Balletto, Registered Professional Reporter, a Notary Public in and for the State of Connecticut, at the Holiday Inn, New London, Connecticut, on May 10, 2000, commencing at 8:35 a.m.

APPEARANCES:

For the Nuclear Regulatory Commission

Ann P. Hodgdon, Esq.
United States Nuclear Regulatory Commission
Washington, D.C. 20555

For Northeast Nuclear Energy Company

WINSTON & STRAWN
David A. Repka, Esq.
Donald P. Ferraro, Esq.
1400 L Street, N.W.
Washington, D.C. 20005-3502

For the Intervenors

Nancy Burton, Esq.
147 Cross Highway
Redding Ridge, Connecticut 06876

Also Present:
Laurence Kopp
Tony Attard
Victor Nerses
Gordon Thompson

Index of examination at conclusion of transcript.

A During the development of that coded Grand Gulf for that methodology, as an independent reviewer I ran a few codes to benchmark the individual, but he -- I don't think we could have traded places. His knowledge was -- we didn't trade places, I think I could have done that if given that task, but the answer to your question is no, I don't think I'm an expert in that methodology.

Q Are you familiar with the contractor known as Holtech?

A Yes, I am.

Q Have you reviewed their criticality calculations before for various applications in the nuclear industry?

A I have seen the nonproprietary versions of their criticality margins, I've just reviewed them. That is not one of my major concerns, so I haven't looked at those in the same depth as I've looked at the thermohydraulics or some of the other analyses.

Q Do you recognize Holtech as having particular expertise in this area?

A They've done a number of them, but I haven't formed an opinion, I haven't graded them A, B, C or anything like that.

Q Have you looked at at least the nonproprietary versions of the criticality calculations that have been submitted to support the Unit III spent fuel rack amendment that we're here talking about today?

A I reviewed the original submittal and the attachment, which was prepared by Holtech or developed by Holtech, so I reviewed all of that information.

Q Do you intend through your testimony to comment on those criticality calculations?

A No. As far as Contention 6, which is the criticality issue, as I envision my role is to look at the controls that are being proposed or in the submittal as far as preventing criticality, because my experience is more in implementing the controls, not in the criticality margin calculations and defining what the margins are. Mine is more the implementation phase, so I would review whatever Northeast Utilities plans to do from that aspect, and Dr. Thompson would look at the margins, the methodology and the analysis and that part of it.

Q So you perceive Contention 6 as being focused on criticality and including a component of handling controls?

A Contention 6, the handling controls comes into options. Are there options available to what Northeast Utilities is proposing to do on the Contention 6, so my role would be to discuss those options and explain why those might be better or less reliance on administrative project, and I was also working on a -- it was a vertical slice done in the spent fuel pool cooling system, spent fuel pool and spent fuel pool cooling system, and I was on that team to go through and verify everything was in the FSNR and license and basis broader was being done. So part of that looked at the surveillance procedures, but that was probably less than half a day out of the whole project, so it wasn't a huge effort.

Q Have you ever personally done a chemistry surveillance related to the spent fuel pool anywhere?

A No, but also as an engineer, generally we would write those procedures or review those procedures. We're not or I'm not a technician, I never take those kinds of results, that wasn't my job function.

Q Do you have an impression as to whether that particular surveillance is relatively complicated or relatively simple or somewhere in between?

A I think my impression, again, it is a relatively simple procedure to do, and that impression is based on licensee event reports. To my knowledge there haven't been a huge number of reports saying people are not doing this right or having trouble doing this right.

Q Do you have an opinion as to whether relying on soluble boron or taking credit for soluble boron as a criticality control measure is legal or not?

A No, I don't really have an opinion. That goes back to the Contention 6. I guess the answer is still the same.

Q Do you have an opinion as to whether administrative control is too complex to be relying upon apart from the law as a practical matter?

A No, because on Contention 5 we were concerned that surveillance was not going to be done except during the period of fuel movements, and it would be discontinued. That was not consistent with the standard technical specifications for pressurized water reactors, so it seemed to be less stringent or less protective than the standard tech spec, so we thought the surveillance, or I thought the boron surveillance was a necessary thing to continue doing.

Q At all times throughout the -- whenever there is fuel in the pool, that's what you mean?

A At all times.

Q Not just during fuel movements?

A That's correct.

Q Now, are you familiar with the supplemental submittal the company made to revise the proposed tech spec to require surveillance at all times?

A The one on April 17, I believe?

Q I think that is the correct date.

A Around that date. Yes, I've seen that.
Q Does that particular proposal resolve your concern on Contention 5?
A If it is implemented the way it was submitted, it would address my concerns about Contention 5. When Nancy Burton faxed me that submittal, or actually I received the one you mailed me before I got the fax, but when I saw that and talked to Nancy, my advice was to continue going to Contention 5, because the submittal could be withdrawn or the NRC could elect to do something different, so that if it were implemented the way it is submitted, my concerns about Contention 5 would go away. It is whether that will happen or not is why it is still on the table in my mind.
Q Do you have any reason to believe it won't be implemented that way, and when you say implement, I assume you mean that that tech spec will be incorporated by the NRC the way it's been written?
A I would just, whether than withdrawing the contention I would wait, unless the ASLB issued an order saying it had to be done that way, I wouldn't want to withdraw the contention because there is too many things that could happen down the road.
Q Like what?
A It could be withdrawn. You could issue a

letter tomorrow withdrawing the contention going back to the original submittal.

Q You are a distrustful sort.

A Just cautious. I would prefer cautious to distrustful.

Q But if this is implemented in the amendment as issued by the NRC, if this tech spec is incorporated, then you would have, or your Contention 5 would be satisfied?

A My concerns about Contention 5 would be satisfied, that's correct.

Q Have you had an opportunity to reread the supplemental submittal the company made on May 5, 2000, which is a response to a request for additional information made of the company by the NRC staff on the license and application?

A No, I don't even know that I had them.

Q Probably missed you in transit. This submittal, among other things, describes some of the fuel movement procedures as they currently exist and how they will be adapted for the proposed new racks. Well, it is probably not efficient if I -- I'll hand it to you. It is a submittal from Northeast Nuclear Energy Company dated May 5 to the NRC, and I guess what I was looking for was any reaction you might have to that submittal, but if you haven't read it --

A I guess that goes a little bit further than what I was trying to say, because even if you had one rack where you could put any fuel assembly in any location, you still have administrative controls because you have to assure you pick up the right assembly in the reactor core and so on. So the safest path is one that reduces, to the extent practical, the reliance on administrative controls.

I don't contend that you can reduce that to zero, that is not my position, but a pathway that increases reliance in administrative controls in lieu of physical configurations is not a preferred path.

Q But you have not done a study or come up with a specific proposal for the Millstone Unit III spent fuel pool that would satisfy that desire or objective?

A No, other than point out -- no, I have not done that study.

Q Earlier this morning we talked about the attributes of fuel handling procedures and controls that would be related to putting fuel in the pool and including regional storage, based on what you've seen in your review of the RAI response, do you see anything there that looks like it fails to meet what you would expect to see in those kinds of procedures, putting aside that any procedure has the possibility of human error?

A Yes, on page 4 of Attachment 1, the last paragraph, there is a discussion of two plant procedures, EN 31001 and EN 31007, and then it talks about a verification that is done, but I'm not sure which procedure requires that verification because it is not clear from the writeup, but in any event, it says that in the spent fuel pool after the core load is complete a verification by piece count is performed, this piece count verification in the spent fuel pool does not check fuel assembly serial numbers, but confirms there is a fuel assembly in each designated fuel storage location.

If you have regional requirements that certain burn-up fuel can only go in certain racks and those kinds of administrative controls, then the piece count check would not identify a situation where two fuel assemblies were swapped and the right number was in the spent fuel pool, the right number might even have been in the regional racks, but you had, for example, a region 1 fuel assembly that was in region 3, and a region 3 fuel assembly that was in region 1, piece count checks would not have identified that.

Earlier this morning I talked about Browns Ferry videotaping the spent fuel pool, we actually verified the serial numbers were in the right storage locations, which is above and beyond the piece count check, so that was the one -- and I guess also, when I

you looked at, do you recollect?
A Yes, they were going to put in high density racks which would have allowed any fuel new or radiated to be put into those racks. I don't recall any limits on -- basically any hole in the storage rack that passed the physical size test, it wasn't bowed or anything like that, anything that was passed the qualification test could be used to store any fuel that they had, with the exception of fail fueled, if something was failed there is separate canisters for that, but with that I don't recall any restrictions.
Q And did you have any complexity issue there with what you looked at, with what you were asked to look at?
A No. In fact, my recommendations to the activist group was that there wasn't any grounds for intervening, and they didn't, as far as I recall.
Q With respect to this complexity issue, I know this specific proposal you described to me as being more complex than others you've seen, and you've also described that you have a concern that administrative controls, people make mistakes and they can fail. Is your concern -- I mean, is that the scope of your concern, or do you have an additional concern that this is Northeast Utilities and Northeast Utilities can't be trusted to

implement administrative controls?

A Northeast Utilities has not the most enviable record in this area, but I don't think that is a factor in my conclusion on this issue, because in June of '98 Millstone III was thought or was going for a restart, and UCS presented to the commission that we didn't think the plant was ready for restart. No matter what their test plan was, it wasn't that we thought Northeast Utilities had not the wherewithal to do a good job, we didn't think the Nuclear Regulatory Commission had fixed all the things that it needed to do to ensure that any problems that came up in the future they would be corrected.

Q Your issue was their oversight process not the operational?

A That's correct. If it was an issue of lack of trust with Northeast Utilities, in June of '98 we would have opposed restart for that as well.

Q But you didn't oppose restart?

A No, and I think our presentation either in the written testimony, or I can't recall the oral remarks themselves, I think we clearly indicated that it wasn't an issue with Northeast Utilities, they seemed to have done a lot of issues, but still, I guess the other way to answer that question would be if I reviewed Fitzpatrick's submittal in June of '98 and it had this amount of complexity, I would have advised the local group that there are some administrative control issues here. Whether they would intervene or not I can't presuppose, and that wouldn't have been nuclear power authority, it is a complexity issue, not company per se.

Q One of the issues that has come up in the context of these issues is Draft for Common Reg. Guide 1.13 from December of 1981. Are you familiar with that document?

A Yes, I am.

Q And I think there is something referred to there as the double contingency principle.

A Yes.

Q Is that something you intend to address in this proceeding?

A Only in the aspect I described earlier, I'll be supporting Dr. Thompson in what could be the possible consequences or why would the proposal not lead to the right consequences or lead to the right outcome. Dr. Thompson is much better versed in why that is or is not a good idea.

Q So are you going to be offering any opinion as to what the double contingency principle requires or doesn't require?

A At this time I have no intention to do that.

ORIGINAL

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND
LICENSING BOARD

- x

In the matter of:	Docket No. 50-423-LA-3
NORTHEAST NUCLEAR ENERGY COMPANY	ASLBP No. 00-771-01-LA
(Millstone Nuclear Power Station, Unit No. 3)	

- x

DEPOSITION OF: GORDON THOMPSON, PhD

Taken before Robin L. Balletto, Registered Professional Reporter, a Notary Public in and for the State of Connecticut, at the Holiday Inn, New London, Connecticut, on May 10, 2000, commencing at 11:10 a.m.

APPEARANCES:

For the Nuclear Regulatory Commission

Ann P. Hodgdon, Esq.
United States Nuclear Regulatory Commission
Washington, D.C. 20555

For Northeast Nuclear Energy Company

WINSTON & STRAWN
David A. Repka, Esq.
Donald P. Ferraro, Esq.
1400 L Street, N.W.
Washington, D.C. 20005-3502

For the Intervenors

Nancy Burton, Esq.
147 Cross Highway
Redding Ridge, Connecticut 06876

Also Present:
Laurence Kopp
Tony Attard
Victor Nerses
David Lochbaum

Index of examination at end of transcript.

Probability and Consequences of Criticality Events in Fuel Pools. This has, I believe, one or two appendices.

Q Now, in that document, that analysis, did you look at the probability and consequences of criticality as a result of fuel mishandling events, or was it more broad than that?

A That discussion focused on fuel mishandling and boron dilution, and there are other mechanisms that could cause criticality events, for instance, the dropping of a heavy object, for example, but that is mentioned in Appendix C in a fairly cursory way.

Q Have you ever been involved in moving fuel in a nuclear power plant?

A No.

Q Have you ever reviewed procedures related to fuel movements?

A Not in a professional capacity in the nuclear industry. I will be reviewing the procedures that have been supplied to us in this case.

Q But you haven't done that to date?

A That's correct.

Q Have you ever done criticality calculations related to determining, for example, K effective?

A No.

Q Have you reviewed the analyses that Northeast

Utilities submitted in conjunction with this license amendment application, the criticality analyses prepared by Holtech?

A I've done an initial review of the license application and will be doing a detailed review in the course of preparing a brief. I should say that for the purpose of this proceeding I do not expect to challenge the calculated results on K effective provided by Holtech, but will focus on whether the assumptions underlying those calculations are appropriate to cover all of the scenarios that should be faced.

Q So if they calculated for a scenario X involving boron one mishandling or whatever, you would accept the result they came up with, the number?

A For the purpose of this, yes, I would accept it.

Q Have you reviewed the licensing board's decision in the Shearon Harris case since it was issued?

A Not yet.

Q So far we've talked this morning about what you would contribute to Contention 6, which is the GDC62 contention, and you said you would also contribute to Contention 4. Can you describe for me where and what it is you will contribute to Contention 4?

A Contention 6 is framed as a legal matter.

Underlying this legal question is what you might describe as more of an engineer's perspective than a lawyer's perspective, and the applicant's proposed course of action reduce the level of safety or increase the level of risk, and the underlying technical issues are really the same, just that in one case it's phrased from a lawyer's point of view and the other from an engineer's point of view.

Q Contention 6 is the proposal of a legal, and Contention 4, if I might characterize it as, is it is as good as it needs to be from an engineering perspective?

A That's a fair accusation.

Q So the focus of Contention 4 would be on the complexity, I gather, and do you have an opinion as to the complexity to the proposal as it currently exists?

A My general opinion about criticality in fuel pools is that there should be no reliance on burn-up, fuel aging or soluble boron under normal or accident conditions. I believe that reliance should be placed on spacing and on fixed boron or other fixed neutron absorber, and that any criticality arrangement in the fuel pool that relies to any extent on the credit for burn-up, soluble boron or aging is in my view a mistake from an engineering point of view, quite aside from whether it is legal.

Q That's what I was going to ask you. Contention

6 is premised on the theory that it would be unlawful. Contention 4 is premised on your professional engineering opinion that it is not --

A That it's a mistake.

Q It's a mistake, okay. Now, is your opinion in that regard based upon particular experiences or is it based upon your conceptual analysis of the issue?

A It is not based on any personal experience. It is based on my analysis that this problem is a generic problem, and that is illuminated by the base of experience with incidents that I've identified, incidents of fuel mispositioning and boron dilution.

Q So, again, it's a generic position related to reliance on soluble boron and reactivity restrictions and decay time restrictions?

A But in the course of preparing our brief on Millstone we may be saying additional things that are specific to most of it. I can't determine that at present because our discovery is ongoing.

Q With respect to that, what kinds of additional things with respect to Millstone, the procedures?

A Mr. Lochbaum and I will be carefully reviewing the procedures, we will be seeing if we can identify boron dilution events that are specific to Millstone. Beyond that I can't say what we might find.

Q That is something you have not done to date?
A Correct.
Q So your concerns, though, relate to the choice of hardware, equipment or controls employed and the possibility of boron dilution. Do you intend to offer testimony related to Millstone's performance history issues?
A I would not address that point. To the extent that the Coalition's brief addresses that, then Mr. Lochbaum and others will contribute, but not myself.
Q Do you have any particular knowledge of the issues that have existed at Millstone in the past?
A No.
Q You don't have any specific basis to conclude that Northeast Nuclear can't implement administrative controls any differently than any other licensee?
A No, I have no such knowledge.
Q I want to explore a little bit the issues of your preference, or your preference is probably too weak a term, but your view that criticality should be controlled by geometric spacing and fixed neutron absorption materials or absorbers. Could you identify for me what you believe the universe of ways to control criticality would be, not just the ones you prefer, but what is the complete set?

A Assuming the spent fuel is intact, then the options are spacing, sold neutron absorbers, soluble neutron absorbers, burn-up/enrichment limitations and age limitations. I believe that's the universe for intact fuel.

Q Are age limitations a reactivity issue or a cooling issue?

A Yes, it is a reactivity issue because the decay of plutonium 241 produces a daughter product that is an absorber of neutrons, therefore, more fuel with a longer decay time is less reactive.

Q Are you aware that the use of soluble boron or soluble absorbers Millstone would not be unique in that regard, are you aware of that fact?

A That's correct.

Q And I guess are you also aware that Millstone is not unique in its proposal to rely upon reactivity limitations?

A That's correct.

Q So would it be your position that all the plants that rely upon those techniques are not in compliance with GDC62?

A That's correct.

Q With respect to soluble boron, do you disagree that the reactivity effect of soluble boron is a physical effect?

A The effect is certainly physical as is the effect on reactivity of age and burn-up. All of those mechanisms effect reactivity through the physics involved, and therefore, they are physical processes in that sense, but I believe do not meet the intent of GDC62.

Q And what is the distinction you draw between geometric spacing and fixed absorbers on one hand versus soluble boron and reactivity limitations on the other?

A Because spacing and fixed boron does require some administrative measures, but they are of finite duration. In simple terms they're one-time administrative measures, and once taken do not need to be repeated.

The preservation of a specified level of soluble boron, or the meeting of some requirement on burn-up or age requires ongoing administrative measures repeated on numerous occasions, and I believe the history of formulation of GDC62 shows that the intent was to exclude measures of that nature.

Q And the history would be that history that you've referred to in the Orange County brief?

A Correct.

Q With respect to fixed neutron absorbers, is it your view that there are no ongoing administrative controls related to those physical systems?

A When Boraflex -- the experience with Boraflex is shown that to ensure its efficacy as an absorber has required ongoing administrative attention. With Boral, which is now the preferred absorber material, experience has shown that only a very modest level of ongoing attention is necessary to ensure that the Boral maintains its function or serves its function, and therefore, again, in simple terms, the use of Boral really requires a one-time administrative measure and then a very modest of ongoing oversight.

Q So what makes Boral different from soluble boron is that it is not that it doesn't have any ongoing administrative controls, it is that they're very modest?

A They're very modest, and any failures would be likely to be manifested very slowly with lots of warning time, lots of opportunity to make the necessary corrective changes.

Q And Boraflex, on the other hand, requires more than modest surveillance?

A Well, the deterioration of Boraflex is what has led to its replacement.

Q So would Boraflex be precluded as a matter of law because it is no longer a physical system?

A That's a fine point of law, regulatory law that I wouldn't speak to at the moment, but --

Q It would seem to follow from the idea that if it involves more than modest ongoing controls, it must be not physical?

A If I were an NRC commissioner asked on a regulation by stating that Boraflex does not meet GDC62, then I would vote for it, but I can't give you a more precise answer.

Q But that would be as a matter of law as opposed to as a matter of engineering preference?

A Yes.

Q I mean, clearly as an engineer, you prefer Boral over Boraflex?

A Yes, but as a matter of law, I would vote that way.

Q Now, with respect to boron, I take it that soluble boron, the effect there your position would be that it is more than a modest administrative control?

A Correct.

Q What administrative controls are you referring to with respect to soluble boron?

A The boron content has to be measured periodically, if there is dilution for any reason, then the boron content has to be increased. The management of the pool, which includes the management of the intake and the outgoing of water from the pool has to ensure that there is no significant dilution event, and there are at any point a variety of possible dilution scenarios which are plant specific, and at this point I can't say what we will be saying about that, but the occurrence of any boron dilution scenarios would involve the failure of one or more administrative controls somewhere in the plant operations.

Q With respect to monitoring the boron technique and completing the surveillance, in your view that is more than a modest operation?

A Yes.

Q Have you ever done one of those surveillances?

A No, and then let me elaborate a little bit. As a chemical procedure, the sampling of the water and the measurement of boron is a relatively simple procedure that's been done many, many times. That, however, misses the point. The point is that the occurrence of a boron dilution event, an undetected boron dilution event or boron dilution event that is undetected up to the point which it becomes associated with a criticality accident is a reflection of multiple administrative actions.

The tech specs in the Millstone call for, I believe, a seven-day surveillance, so in looking at boron dilution events, your first cut at this will be to look at events that can occur within a seven-day time frame, and at your second cut you might postulate one missed surveillance event, or one incorrectly performed surveillance, and then look at 14 days and see what that does to your boron dilutions, so the actual act of sampling and measuring is properly seen in a wider context of actions.

Q Have you identified any particular boron dilution events in the industry which you are going to rely upon?

A There is one event at McGuire that is described in our Harris filing, and at this point I don't know if we identify more at Millstone. That is an actual event, by the way, rather than a hypothetical.

MR. REPKA: As an aside, I would say that we have asked on discovery for a list of all events, including boron dilution events and other types of events that the Coalition is going to rely upon. The discovery period is scheduled to close on the 30th of May. If you, in fact, your experts have not done the inquiry they need to do, this puts us in a bit of an awkward position in that we would expect a response to our discovery that would include any boron dilution scenarios that you might identify.

I would ask you Ms. Burton, do you have plans to supplement your prior responses to address that?

MS. BURTON: Yes, as the discovery is proceeding, we do.

MR. REPKA: And that will occur prior to the end of May, whether it is May 30th or 31st, I forget the exact date.

MS. BURTON: I think it better.

MR. REPKA: And that would certainly be my expectations, and I'll rely on that expectation.

THE WITNESS: Can I ask for a point of clarification. One matter is, the record of boron dilution events that have actually occurred, and then a separate matter is the development of scenarios for boron dilution events that might occur at Millstone.

Am I to understand that after the end of discovery we cannot come up with a fresh scenario for the latter? Because the reason I ask is that it is not until nearing the end or perhaps after discovery closes that we will be doing a really thorough reading of the documents that we have received, and I can't exclude the possibility that we will identify some scenario at that time.

MR. REPKA: We have discovery requests that go to hypothetical scenarios as well as actual incidents that you will rely upon. I think we would be entitled to know at the close of discovery what those events are, because just like you, we will then be in a position of writing our paper, which is due on June 30th. So we cannot be put into the position of trying to address something that we don't know what it is until days or weeks prior to the due date.

The point is, we do need to know what you're going to rely upon by the end of the discovery period, that's why we have the discovery period, and each party will know what the other side is going to rely upon and write their paper, so my answer is yes, we would expect you to complete that work by the end of discovery so you could tie in performance.

Would your understanding be any different?

MS. HODGDON: That is particularly true in a Subpart K proceeding where everyone files simultaneously and there are no responses. So we have a motion to compel outstanding on that very matter. We ask what documentation, et cetera, you would rely on in your filing and your answer was you'll see it in our brief, and we said, no we won't, we'll see it before that. Actually, that was promised to us some weeks ago, and we still don't have it, and we now have a motion to compel, which the board, I'm not sure whether it is included in the things that they will actively decide on, but definitely we're entitled to know those things and we were entitled to have known them some time ago.

THE WITNESS: Another point of clarification. If we identify a scenario in general terms, we give a one-paragraph description by the end of discovery and then in the next month we elaborate on that in our brief, is that a problem?

MR. REPKA: Well, I would reserve comment on the issue depending upon what the words are, but I think some description of a scenario would be acceptable. Obviously we would have to be on notice as to what it is so that we could address it and that is the question. It has to be sufficiently descriptive.

MS. HODGDON: Your attorney has perhaps told you that the purpose of discovery is to eliminate surprise, so the element of surprise is still lurking here as we've asked questions and they have not been answered as the licensee has, and so I should hope that the licensing board will sustain our objections to find the element of surprise is something that we're going to find in your brief, in the Intervenor's brief on June 30.

MR. REPKA: Are we on board here? Are we

in agreement as to how we're going to proceed?

MS. BURTON: Dr. Thompson, I don't know if his answer has been --

THE WITNESS: I'm clear about what the applicant and the staff say on this.

MR. REPKA: I just want to emphasize, this isn't about playing games in discovery and doing things that are legalistic, this is about getting to the real issue here, and the real issue is if the intervenors and the Coalition and their experts have a problem, we want to know and we're entitled to know what this problem is so we can address it either by explaining why it is not a problem or fixing the problem that may exist. So this is not about games, this is about getting to the issues.

THE WITNESS: I would like to make a point on that that the FSAR is an important thing that we could really benefit from seeing, and from where we sit it is an extremely difficult exercise to get ahold of an up-to-date, complete, legible copy of the FSAR for any facility.

The Washington public document room has microfiche, which is practically illegible, hard copy you have to ask for them to ship it in from a warehouse, it is incomplete, so if we're talking fairness here, I think you ought to let us see the complete, up-to-date --

MR. REPKA: We discussed that earlier, and we're going to hear back from you all on if there are specific sections. We already have provided the spent fuel pool sections, and we'll wait to hear back from you as to what other sections you may like to see. Once we see that proposal we can discuss how we'll deal with that.

BY MR. REPKA:

Q We were talking about dilution events in Contention 6. Before I leave those kinds of issues I want to ask, in the area of the criticality analyses I think you've already -- your position seems to be that there should be no credit for boron; is that accurate?

A There should be no credit for soluble boron under either normal or accident conditions.

Q Are there other weaknesses in other areas in the assumptions related to the criticality analysis that Northeast Nuclear has submitted that you disagree with or find to be inadequate?

A Well, the same point applies to taking credit for burn-up or age.

Q With respect to the issue of boron dilution, are you aware of any operational reason why a licensee

of what the brief says on that.
Q Have you read 50.68?
A Not recently.
Q Let me hand you a copy of 50.68. Take a minute to glance over that.
A Okay.
Q 50.68B4 specifically speaks to storage of spent fuel.
A Right.
Q Would you agree or disagree that that regulation specifically contemplates fuel assembly reactivity, and it says, "Fuel assembly reactivity must not exceed 0.95 at a 95 percent probability, 95 percent confidence level," do you see that language?
A Yes, it is actually a bad use of the word reactivity.
Q Why so?
A It's the neutron multiplication factor that they're referring to. Reactivity is the change in that factor depending on the changes of some other parameter.
Q So would it be your position that the terminology there does or does not include consideration of enrichment, burn-up, aging considerations?
A In this section I don't see any segment one way or another about taking credit for burn-up or aging

effects on the fuel assembly reactivity.

Q Do you agree that burn-up has a reactivity effect?

A It does.

Q And how about enrichment, does it have a reactivity effect?

A It does.

Q And aging?

A It does.

Q Now, in there it also talks about, a little further up in section 50.68B4 it speaks to if credit is taken for soluble boron, K effective, it goes on.

A Right.

Q So does that or does that not contemplate soluble boron credit?

A It does contemplate it, yes.

Q Do you have any other views related to how 50.68 is intended to imply, or is that something you're leaving to your attorney?

A I'll leave that to the attorney. I should say that looking at this from a technical point of view I find an ambiguity in the phrase maximum fuel assembly reactivity, and I would like to see a clarification of that phrase, and I would interpret the phrase maximum fuel assembly reactivity as implying to a fresh fuel assembly with a maximum enrichment permitted by the plant's tech specs. That's for PWR fuel, for PWR fuel it is a little more complicated because fresh fuel is not the most reactive.

Q Fresh fuel is not the most reactive for BWR?

A BWR.

Q But for PWR it is.

A Yes. And I think it is unfortunate that the regulation sets forth this phrase without defining it.

Q Did you comment on the rule making of 50.68?

A No.

Q Do you have any intent to ask the commission to clarify its rules?

A Not at present.

Q Let me change subjects for you a little bit. In the statement of the contention in bases as originally submitted in the Coalition's supplemental position, and I don't have the date in front of me, but I'm thinking it is like November 16 or 30, somewhere along there, there is a discussion of Reg. guide 1.113 in the double contingency principle. Are you familiar with that?

A I'm looking for the Coalition's submission. I have a November 17, 1999 supplemental petition to intervene.

Q Correct.

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

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

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

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

Sincerely,

Brian Grimes

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

Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

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

I. BACKGROUND

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

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

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

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

II. REVIEW DISCIPLINES

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

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

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

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

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

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

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

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

1. Neutron Multiplication Factor

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

1.1 Normal Storage

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

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

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

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

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

1.2 Postulated Accidents

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

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

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

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

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

1.3 Calculation Methods

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

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

1.4 Rack Modification

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

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

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

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

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

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

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

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

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

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

(b) The storage lattice pitch; and,

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

1.5 Acceptance Criteria for Criticality

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

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

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

(2) Decay Heat Calculations for the Spent Fuel

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

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

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

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

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

(4) Potential Fuel and Rack Handling Accidents

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

(5) Technical Specifications

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

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

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

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

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

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

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

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

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

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

(2) Applicable Codes, Standards and Specifications

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

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

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

(3) Seismic and Impact Loads

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

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

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

**American Institute of Steel Construction, Latest Edition.

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

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

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

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

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

(4) Loads and Load Combinations:

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

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

(5) Design and Analysis Procedures

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

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

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

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

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

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

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

(6) Structural Acceptance Criteria

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

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

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

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

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

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

TABLE

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

Notes:

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

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

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

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

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

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

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

(8) Testing and Inservice Surveillance

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

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

V. COST/BENEFIT ASSESSMENT

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

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

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

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

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

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

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

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

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

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

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

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

(c) shipment to another reactor site,

(d) shutting down the reactor.

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

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

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

V.2. RADIOLOGICAL EVALUATION

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

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

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

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

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

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

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

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

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

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

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

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

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

V.3 ACCIDENT EVALUATION

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

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

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

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

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

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

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

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

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

VI. REFERENCES

1. Regulatory Guides

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

   1.29 - Seismic Design Classification

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

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

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

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

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

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

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

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

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

5. The Aluminium Association, Specification for Aluminium Structures

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

Reference 7

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing Criterion 61.

B. DISCUSSION

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

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

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 protectio requirements are acceptable for the water makeup source and its deliver ystem, the pool structure, the building housing the pool, and the filtratior tilatio system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Cr ia for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System ltration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," egulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal V lation 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 or 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 tc applicants regar ing the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participatio 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 <u>not</u> 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 multi .cation factor, k_a; i.e.,

$$k_s \leq k_a$$

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

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

where

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

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

Δk_u = the uncertainty in the benchmark experiments, and

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

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

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

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

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

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

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

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

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

4. STORAGE RACK ANALYSIS ASSUMPTIONS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

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

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

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

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

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

(2) Periodically verify continued presence of absorber.

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

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

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

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

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

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

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

1.2 Need for Proposed Action

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

1.3 Value/Impact of Proposed Action

1.3.1 NRC

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

1.3.2 Government Agencies

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

1.3.3 Industry

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

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

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

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objective 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.

August 19, 1998

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

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

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

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

Attachment:
As stated

DISTRIBUTION

PDR	File Center
SRXB R/F	E. Kendrick
T. Marsh	T. Collins
G. Hubbard	E. Weiss
H. Richings	L. Kopp
A. Attard	G. Holahan
M. Chatterton	

SRXB:DSSA	SRXB:DSSA
LKOPP LK	EWEISS
8/6/98	8/17/98

OFFICIAL RECORD COPY G:\FUELSTOR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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

1. INTRODUCTION

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

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

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

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

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

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

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

a. fuel rod parameters, including:

   1. rod diameter

   2. cladding material and cladding thickness

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

b. fuel assembly parameters, including:

   1. assembly length and planar dimensions

   2. fuel rod pitch

   3. total number of fuel rods in the assembly

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

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

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

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

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

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

2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

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

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

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

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

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

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

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

where

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

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

3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

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

4. NEW FUEL STORAGE FACILITY (VAULT)

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

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

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

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

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

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

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

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

5. SPENT FUEL STORAGE RACKS

A. Reference Criticality Safety Analysis

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

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

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

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

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

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

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

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

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

   d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1.0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations: i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.

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

---

[1] [*] is the boron concentration required to maintain the 0.95k_{eff} limit without consideration of accidents.

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

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

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

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

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

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

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

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

B. Additional Considerations

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

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

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

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

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

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

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

6. REFERENCES

1. NRC, "Standard Review Plan," NUREG-0800, Rev.2, Section 9.1.1, "New Fuel Storage," July 1981.

2. NRC, "Standard Review Plan," NUREG-0800, Rev. 2, Section 9.1.2, "Spent Fuel Storage," July 1981.

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

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

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

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

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

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

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

NNECO Operating Experience (OE) Matrix*

Millstone-Related Experiences

Reference 9

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
12/20/76	MNP-1	Letter from Ernst Volgenau (Director, Office of Inspection and Enforcement) to D.C. Switzer (President, NNECO) "Order to Show Cause and Order Suspending License"	This letter refers to a special inspection (due to an unplanned criticality) that occurred on November 12 - 19, 1976. Violations include: (1) an unplanned criticality and automatic reactor trip from high flux on four IRM channels; (2) control rods were moved for shutdown margin testing without proper verification by an operator or engineer; (3) the shift Supervisor dismissed the issue.	This event is not relevant to the spent fuel pool criticality analysis. This happened to the Millstone 1 reactor core, not the spent fuel pool. The testing involved withdrawal of control rods for Shutdown Margin (SDM) testing, and was designed to show that there was sufficient SDM for certain control rod configurations. There is no equivalent of this testing for the SFP. This testing places you in a situation where a single failure can take the reactor critical. The operator pulled the wrong control rod and took the reactor critical with the head off. The only relevance this shows for the SFP is that a single failure can occur. In this case for the reactor, this single failure allowed the reactor to go critical. NRC regulations for the SFP are to ensure that no single failure event in the SFP will take the reactor critical (and further Keff will be less than 0.95). We comply with these regulations; therefore, there is no credible single failure that can take the SFP critical.
2/11/85	Hatch-1, MNP-2, Monticello, Palisades, TP-4, Cook-1	NRC Information Notice 85-12: Recent Fuel Handling Events	Describes events in which: (1) fuel was dropped because of failures or deficiencies in hoist equipment; and (2) other incidents involving deficiencies or misoperation of fuel handling equipment or procedures.	These events are bounded by the Millstone 3 criticality analysis. This notice is for a single fuel assembly handling event at various nuclear plants, or in the case of Millstone 2, a single fuel pin handling event. These events all involve an event where (at most) a single fuel assembly was placed in an unexpected condition, that condition was immediately evident, and actions were taken to resolve the situation. We acknowledge that single fuel assembly handling events can and do occur. A possible fuel handling accident/event is part of the criticality analysis for the MP3 re-rack. Therefore, the effect on the criticality analysis for a single fuel handling event is accounted for.

* This OE Matrix compiles events and incidents identified by the Intervenors and by NNECO during discovery in this proceeding. NNECO's intent here is to respond to those events that Connecticut Coalition Against Millstone and the Long Island Coalition Against Millstone may rely on in support of Contentions 4, 5, and 6 in this proceeding. NNECO makes no representation regarding the "completeness" of this list (*i.e.*, where the Intervenors may apparently invoke <u>any</u> fuel handling or boron concentration discrepancy). NNECO included events specifically listed in discovery responses by either CCAM/CAM or NNECO. This OE Matrix is sponsored in the sworn testimony of Mr. Joseph Parillo and Mr. Michael Jensen.

Millstone-Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
3/18/85	MNP-2	Plant Incident Report 85-39	A fuel assembly was lowered into contact with another assembly located in the fuel upender.	This event is bounded by the Millstone 3 criticality analysis. This event is a single fuel assembly handling event at Millstone 2. This event has no criticality implications. It occurred before fuel assembly burnup was ever credited. Also, it did not actually result in a circumstance that changed the spent fuel pool K-effective. Further, with regard to criticality, the fuel upenders are fully qualified for 5 w/o fresh fuel. Regardless, a possible fuel handling misplacement/event is part of the criticality analysis for the MP3 re-rack. Therefore, the effect on the criticality analysis for a single fuel handling event is accounted for.
10/2/85	MNP-2	Plant Incident Report 85-101	A fuel assembly was moved to an incorrect location in the spent fuel storage pool and lowered until it came in contact with an assembly already placed in that location.	This event is bounded by the Millstone 3 criticality analysis. This event is a single fuel assembly handling event at Millstone 2. This event has no criticality implications. It did not actually result in a circumstance that changed the spent fuel pool K-effective. Even if the fuel assembly had been placed in the incorrect location, the fuel assembly was fully qualified for that location. Regardless, a possible fuel handling misplacement/event is part of the criticality analysis for the MP3 re-rack. Therefore, the effect on the criticality analysis for a single fuel handling event is accounted for.
6/12/87	MNP-1	Licensee Event Report 87-19-00	A fuel assembly in the reactor core was found to be 90 degrees out of the proper orientation.	This event has no relevance to the spent fuel pool. Improperly rotated fuel is of significance in the reactor core, but is not relevant to the spent fuel pool. Spent Fuel Pool criticality analyses do not require a specific fuel rotation. Therefore, there is no such thing in the spent fuel pool as a misrotated fuel assembly that affects the criticality analysis.
5/9/88	MNP-1, Oskarshamn, VY	NRC Information Notice 88-21: "Inadvertent Criticality Events at Oskarshamn and at U.S. Nuclear Power Plants"	Inadvertent criticality pertaining to movement of control rods. Plant operators failed to observe indications on the instruments that could have prevented or mitigated the event.	Same as 12/20/76 event at Millstone 1, which was discussed above.

Millstone-Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
4/22/92	MNP-2	NRC Information Notice 92 -21, Supplement 1: Spent Fuel Pool Reactivity Calculations	Updates information initially supplied by ABB Combustion Engineering and incorporated in Information Notice 92-21 (describes errors which were discovered in reactivity calculations for spent fuel pools).	This event would not occur in the Millstone 3 criticality analysis since multiple independent computer code methods have been used to validate the results. The events involve errors in calculating SFP K-effective, including an event at Millstone 2. Corrective actions for these events involve ensuring that vendors calculate K-effective with more than 1 computer code, and that the benchmarks to qualify the codes involve strong absorbers. For the Millstone 3 re-rack, as documented in our NRC re-rack submittal, the KENO and MCNP computer codes are used to provide an independent check of the K-effective results, to ensure that an error in 1 code would be identified by a difference in KENO and MCNP results. Also, as documented in our NRC re-rack submittal, the benchmark results for KENO and MCNP include high worth absorber critical experiments. Hence the causes of this event are not present in the MP3 re-rack. Also, this IN is not directly relevant to the contentions since the ability to calculate K-effective is NOT a contention.
6/25/92	MNP-2	Licensee Event Report 92-003-01	A calculation error in the criticality analysis for the spent fuel pool.	Same event as addressed below in IN 92-21.
4/27/94	MNP-3	Plant Information Report 3-94-079	A fuel assembly was moved from one location in the spent fuel storage pool to an incorrect location and lowered until it came in contact with an assembly already placed in that location.	This event is bounded by the Millstone 3 criticality analysis. This event is a single fuel assembly handling event at Millstone 3. This event has no criticality implications. It did not actually result in a circumstance that changed the spent fuel pool K-effective. Even if the fuel assembly had been placed in the incorrect location, the fuel assembly was fully qualified for that location. Regardless, a possible fuel handling misplacement/event is part of the criticality analysis for the MP3 re-rack. Therefore, the effect on the criticality analysis for a single fuel handling event is accounted for.

Millstone-Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
4/26/95	MNP-3	Adverse Condition Report - 710	While transferring fuel in the spent fuel storage pool, the crane operator inadvertently brought an assembly to the wrong location; the error was detected before the assembly was lowered.	This event is bounded by the Millstone 3 criticality analysis. This event is a single fuel assembly handling event at Millstone 3. This event has no criticality implications. It did not actually result in a circumstance that changed the spent fuel pool K-effective. Regardless, a possible fuel handling misplacement/event is part of the criticality analysis for the MP3 re-rack. Therefore, the effect on the criticality analysis for a single fuel handling event is accounted for.
8/31/95	MNP-1	GENE 523-A085, "Independent Assessment of Spent Fuel Pool Cooling at Millstone 1 Nuclear Power Station," D. Saxena and G. Stramback, General Electric Company	General Electric review of Millstone 1 spent fuel practices which includes a description of issues related to decay heat removal.	This is not relevant to any of the contentions. This document has no criticality implications. See IN 95-54.
11/14/95	MNP-1	Adverse Condition Report - 6385	A fuel assembly was placed in the spent fuel storage pool in the wrong orientation.	This event has no relevance to the spent fuel pool. Improperly rotated fuel is of significance in the reactor core, but is not relevant to the spent fuel pool. Spent Fuel Pool criticality analyses do not require a specific fuel rotation. Therefore, there is no such thing in the spent fuel pool as a misrotated fuel assembly that affects the criticality analysis.

Millstone-Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
12/1/95	MNP-1, Cooper	NRC Information Notice 95-54: "Decay Heat Management Practices During Refueling Outages"	NRC assessments of licensee control of refueling operations and the methods for removing decay heat produced from the irradiated fuel stored in the spent fuel pool.	This is not relevant to administrative controls used for criticality in the Millstone 3 spent fuel pool. This IN is directly involved with heat removal from the SFP, and has nothing directly to do with criticality. This event involved Millstone 1 not having consistency between its SFP cooling analysis and operations with respect to full core offload during refueling. If the intention is to argue that this administrative failure in the MP1 SFP shows compliance problems with administrative controls to be used at Millstone 3 for criticality control, the following should be noted: The process of implementing administrative limits for criticality involves 3 basic tasks: (1) calculating the criticality limits correctly. This is not a contention. (2) translating the criticality analysis limits (burnups/enrichments/decay times) into plant procedures correctly, and (3) putting the correct fuel assemblies into the correct locations. The first item above is not a contention. The second and third items are administrative in nature, and are the subject of the contentions. Items (2) and (3) are directly measurable by both Millstone and industry experience. Therefore, it is this directly applicable experience that is relevant to determine the reliability of the administrative controls to be used for criticality compliance. The success or failure of other administrative controls have significantly less meaning when you can directly measure and assess the adequacy of the controls that are to be used for criticality compliance.
12/14/95	MNP-1	Licensee Event Report 95-009-02	A determination at Millstone 1 that portions of the spent fuel pool cooling piping, which had been used with a maximum operating temperature of 150° F, actually was only designed for 85° F.	Same as IN 95-54.

Millstone-Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
3/1/96	MNP-1	NRC Daily Event Report No. 30050, "Unanalyzed Heavy Load Path for Moving Gates Outside the Spent Fuel Pool"	Potential movement of spent fuel pool gates over irradiated fuel.	This is not relevant to any of the contentions. NNECO personnel identified the potential for this event in a pre-job brief prior to moving the gates. NNECO revised the procedure to preclude this event from occurring.
3/6/96	MNP-1	Licensee Event Report 96-023-00	Determined that new fuel assemblies moved over irradiated fuel assemblies in the spent fuel storage pool.	This is not relevant to Millstone 3 spent fuel pool. The Millstone 3 new fuel elevator is located in the transfer canal, and there are crane interlocks to prevent new fuel movement over the spent fuel pool.
7/25/96	MNP-1	Licensee Event Report 93-011-02	Determined that during refueling outages the Millstone 1 spent fuel pool cooling system, by itself, would have been incapable of maintaining pool temperature below the 150 ° F design limit, under certain conditions. Conditions in question involve the transfer of a full reactor core into the spent fuel pool.	Same as IN 95-54, above.
10/3/96	MNP-1	Adverse Condition Report M1-96-0646	Determined that a spent fuel assembly in the spent fuel storage pool was not fully seated in the storage rack.	This is not relevant to Millstone 3 spent fuel pool. This event is particular to BWRs which have channel fasteners which can cause fuel to hang up on the top of the fuel racks. There are no corresponding circumstances for PWRs.
1/14/97	MNP-1	Adverse Condition Report M1-97-0082	Determined that an irradiated fuel assembly, stored in a damaged fuel container in a control rod storage rack, may have been an unanalyzed configuration.	This is not relevant to Millstone 3 spent fuel pool. This event concerns a single BWR fuel assembly which was dropped and damaged in the 1970s. The fuel assembly is stored in a special container segregated away from other fuel assemblies in a special control rod storage rack. In 1997, NU determined that there was no documented, readily available information that documented that the single fuel assembly could not by itself pose a criticality threat, due to the potentially damaged mechanical structure. The fuel assembly had been and continued to be appropriately isolated from other fuel assemblies. Analysis was then performed and documented to show that the fuel assembly could not pose a criticality threat.

Millstone-Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
6/27/97	MNP-3	Preliminary Notification of Event or Unusual Occurrence PNO-I-97-039	At Millstone 3 the reactor plant component cooling water system was lined up to the wrong spent fuel pool heat exchanger. This system misalignment and subsequent increase in the spent fuel pool temperature went unnoticed for approximately 28 hours.	Same response as IN 95-54. The Millstone 3 event was a SFP heatup event. This is not relevant to the contentions. As discussed in the response to IN 95-54, the failure to have consistency between analysis and practice can be directly assessed for the criticality issues that are of contention. The success or failure of other administrative controls is not relevant. The intervenors are trying to argue that because an administrative control failed in one particular area in the past, that the administrative controls in question for the SFP criticality controls will fail in the future. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.
1/28/99	MNP-2	Condition Report M2-99-0304	Approximately 2,370 gallons of SFP water transferred to CW system.	Same response as 6/27/97 event. The SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
10/9/80	Browns Ferry	Licensee Event Report 80-037-01	Two fuel assemblies were misoriented 90°.	This event has no relevance to the spent fuel pool. Improperly rotated fuel is of significance in the reactor core, but is not relevant to the spent fuel pool. Spent Fuel Pool criticality analyses do not require a specific fuel rotation. Therefore, there is no such thing in the spent fuel pool as a misrotated fuel assembly that affects the criticality analysis.
12/18/86	Cooper	Licensee Event Report 86-034-00	Fuel loaded in the spent fuel pool with a U-235 loading higher than allowed by the Technical Specifications.	This event is not relevant to the Millstone 3 spent fuel pool. The Millstone 3 criticality analysis considers the highest possible fuel enrichment, 5 w/o, that can be received from a manufacturing facility. This event involved Cooper having a 14.5 gram U-235/axial cm TS limit for fuel enrichment. Cooper actually received fuel with up to 14.6 gram U-235/axial cm. This was solely due to nominal conditions vs. manufactured conditions. The criticality analysis correctly considered the manufacturing tolerances, but the TS was actually based on nominal conditions. Since the Millstone 3 criticality analysis includes manufacturing tolerances, and fuel cannot be physically made greater than the 5 w/o Millstone 3 TS limit, the corresponding event could not happen here.
2/24/87	Oyster Creek 1	Licensee Event Report 87-006-00	A misplacement of fuel in the spent fuel pool based on exceeding the average planar enrichment of U-235.	This event is not applicable to Millstone 3 since the spent fuel pool criticality analysis addresses fuel up to the maximum enrichment of 5 w/o U-235. The event involved allowing higher enrichments in the spent fuel pool than the TS allowed. The criticality analysis was bounded by the higher enrichments used, but the TS were not revised to reflect the higher allowed enrichments, since personnel did not realize the new fuel would be temporarily stored in the spent fuel pool.
9/4/87	CY	Letter from R.L. McGuinness, NU, to D.B. Miller, NU, "Reportability of Spent Fuel Pool Cooling"	Reportability determination for loss of spent fuel pool cooling described in LER 87-0015.	Same as IN 95-54.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
9/11/87	CY	Licensee Event Report 87-015-00	A loss of spent fuel pool cooling.	Same as IN 95-54.
9/11/87	D.C. Cook	Licensee Event Report 87-015-00	Malfunction of the refueling Manipulator Crane load cell which led to the discovery that the load cell had not been adequately calibrated.	Not relevant to criticality contentions. Same response as IN 97-68.
12/1/87	Crystal River 3	Licensee Event Report 87-026-00	A misplacement of a 3.85% enriched fuel assembly in a spent fuel pool that was limited to storage of 3.5% enriched fuel.	This event involved 1 fuel assembly which was incorrectly moved and did not qualify for regional storage. This event is bounded by the single fuel misload event assumed in the Millstone 3 criticality analysis. Even if this event did occur at Millstone 3, the reactivity effects would be easily bounded by the Millstone 3 criticality analysis limiting single fuel misload event.
5/5/88	Turkey Point-3, Braidwood	NRC Information Notice 88-20: "Unauthorized Individuals Manipulating Control and Performing Control Room Activities"	Potential problems resulting from unauthorized persons manipulating controls and performing control room activities.	This is not relevant to any of the contentions. A non-licensed individual performed a control room manipulation (correctly) under licensed supervision. However, the individual did not qualify for the exemption to manipulate controls without a license, since he was not formally enrolled in a training program.
4/19/90	Catawba 1	Licensee Event Report 90-016-00	Missed sample from Refueling Water Storage Tank (RWST).	There was a missed TS surveillance on the RWST boron concentration. The relevance is unclear to the Millstone 3 re-rack amendment. If anything, the proposed Millstone 3 re-rack SFP boron monitoring surveillance is designed to reduce the possibility of a missed surveillance by aligning the surveillance interval with the chemistry department's normal weekly SFP monitoring schedule.
8/91	N/A	NUREG/CR-5819: Probability and Consequences of Rapid Boron Dilution in a PWR	Describes probability and consequences of rapid boron dilution in PWR.	This document is not relevant to the Millstone 3 spent fuel pool criticality analyses. This document concerns the dilution of the Reactor Coolant System of a PWR. This document has nothing to do with SFP dilutions. It is, therefore, not relevant.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
11/25/91	McGuire 1	Licensee Event Report 91-016-00	The vacant row requirement not satisfied for 11 fuel assemblies.	This event involves inadequate controls at the interface of 2 regions of fuel storage. While the fuel was properly placed within the required regions, the interface of the 2 regions was not properly controlled. This same event could not happen for the proposed Millstone 3 configuration. For all regions except the Region 1 3-out-of-4 configuration, the racks have been analyzed to show that no interface problem exists with adjacent fuel storage regions, so that there are no additional requirements due to interface concerns. For Region 1 3-out-4 storage, the Technical Specification requirements specify the interface requirements to other regions. These interface requirements are essentially unchanged from the current requirements. Further, NNECO's rerack design is such that the Region 1 3-out-of-4 cell blocker placement has already been fixed, and this cell blocker placement precludes any interface requirement problem. Therefore, for the proposed Millstone 3 rerack, fuel meeting the requirements for regional storage, ensures by itself, with no additional actions, that there are no interface problems.
6/92	N/A	NUREG/CR-5771: "Probability and Consequences of Misloading Fuel in a PWR"	Probability and consequences of misloading fuel in a PWR.	Not relevant to the Millstone 3 criticality analysis. This document concerns the probability and consequences of misloading fresh fuel in the reactor core of a PWR. The only relevance to the SFP is that the document uses some fault trees of SFP operation to contribute to the probability of fresh fuel misloads to the core causing criticality. A minimum of 5 fresh fuel misloads together were necessary in the core to be critical. The report assumes that there is only 1 person in the SFP verifying the correct fuel storage locations. Therefore, for that reason alone, this report has no relevance to MP3, which requires 2 people to verify the correct location of fuel.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
11/4/92	Univ. of Michigan	NRC Information Notice 92-73: "Removal of a Fuel Element from a Research Reactor Core While Critical"	Issued as a result of an event in which licensed operators at a research reactor inadvertently removed a fuel element from a reactor core that was critical.	The inadvertent removal of a fuel assembly from a critical research reactor core is not directly relevant to a spent fuel pool criticality analysis. Indirectly, it would show that a single fuel mishandling event should be considered, as it already has been in the Millstone 3 SFP criticality analysis.
10/21/93	Vermont Yankee	NRC Augmented Inspection Team (AIT) Report 50-271/93-81	Two fuel handling events that took place at Vermont Yankee. It was determined that the grapple had not properly closed on the fuel assembly handle, and that the grapple light was not energized, resulting in the drop of the assembly on September 3, 1993. The second event was due to an inadvertent operator error.	Same response as IN 94-13 below.
2/22/94	VY, Peach Bottom, Susquehanna, Nine Mile Point	NRC Information Notice 94-13: "Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment"	Potential problems resulting from inadequate oversight of refueling operations and inadequate performance on the part of refueling personnel.	These events are bounded by the Millstone 3 spent fuel pool criticality analysis. The fuel assembly events cited involve single fuel assembly handling events. Regardless of how the events are initiated, whether mechanical failure, human failure, or inadequate oversight, it is still a single fuel assembly handling or misplacement event. The events cited here have only 1 fuel assembly, at any one time, in an unexpected or inappropriate location. A possible fuel handling accident/event is part of the criticality analysis for the MP3 re-rack, therefore the effect on the criticality analysis for a fuel handling event is accounted for.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
3/17/94	River Bend, Indian Point-3, Catawba, Summer, Turkey Point, Braidwood, Comanche Peak-2	Memorandum to Jack E. Rosenthal to Sanford L. Isreal - Review of Mispositioned Equipment	Several plants between 1990 and 1993 that had mispositioned equipment events.	Not relevant to the Millstone 3 spent fuel pool criticality analysis. This report is an NRC staff report on mispositioned equipment, mostly valves. The reason this report is probably cited is to show that independent verification checks can fail, in this case on valve positions. The circumstances on valve position verification are quite different then verifying that a fuel assembly is being placed/removed from the proper SFP location. Valve mispositioning and verification can be subject to many issues that are not present in the SFP. Valve position mis-identification can be a result of the location of valves, the number of valves to be verified in a very short time, each valve may have unique indications, or left vs. right action. In contrast, SFP verification is a slow process that is consistent in terms of its verification environment. In short, the administrative success/failure of independent verification of fuel movement should be judged on its own history, not an extrapolation of success/failure of other administrative controls that have no bearing on the specifics of fuel movement verification. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.
6/28/94	Waterford	NRC Information Notice 94-13, Supp. 1	Unanticipated and unintended movement of fuel assemblies due to improper operation of refueling equipment.	Same response as IN 94-13, above.
7/11/94	McGuire 1	Licensee Event Report 94-005-00	Reduction in spent fuel pool boron concentration from 2105 ppm to 1957 ppm due to a dilution event.	This was a small reduction in boron concentration as a result of a transfer canal draindown. The dilution was quickly detected and criticality safety was never even potentially compromised. NNECO has procedural cautions in place to preclude a similar event from occurring at Millstone.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
8/15/94	Byron	Licensee Event Report 94-006-00	A fuel assembly in the spent fuel pool in the wrong position based on not meeting minimum burnup requirements.	This event involved 1 fuel assembly which had attained about 3000 mwd/mtu burnup less than required for regional storage. This event is bounded by the single fuel misload event assumed in the Millstone 3 criticality analysis. The effect on K-effective to this event is not large. Even if this event did occur at Millstone 3, the reactivity effects would be easily bounded by the Millstone 3 criticality analysis limiting single fuel misload event.
10/14/94	Zion Unit 1 & 2, Indian Point 3	NRC Information Notice 94-75: "Minimum Temperature for Criticality"	Technical Specifications for minimum temperature for criticality were not supported by the safety analyses at the following plants: Zion Units 1 and 2, Indian Point Unit 3.	Not relevant to the Millstone 3 spent fuel pool criticality analysis. The event concerns minimum temperature for criticality in the reactor, and whether there is consistency between the safety analysis and plant requirements. This is not relevant to the contentions. It is inappropriate to conclude that because an administrative control fails in one particular area, that the administrative criticality controls for the SFP will also fail. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.
2/7/96	Oconee	Licensee Event Report 96-001-00	Inadvertent suspension of a fuel assembly inside the spent fuel pool mast.	This is the same event listed below as 3/5/96 NOV. This is a single fuel assembly handling event and, as such, is bounded by the Millstone Unit 3 criticality analysis. There were, in fact, no changes in SFP K-effective due to this event.
3/5/96	Oconee	Notice of Violation and Proposed Imposition of Civil Penalty 50-269/96-02	A spent fuel assembly was inadvertently left withdrawn from the Units 1 and 2 spent fuel pool rack. The violation involved the failure to provide adequate procedures to control fuel assembly movement in the spent fuel pool.	This is a single fuel assembly handling event and is bounded by the Millstone Unit 3 criticality analysis. There were, in fact, no changes in SFP K-effective due to this event.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
3/25/96	Hope Creek	Licensee Event Report 95-042-00	A visual inspection of the reactor core revealed a fuel bundle that was apparently 180° out of proper orientation.	This event has no relevance to the spent fuel pool. Improperly rotated fuel is of significance in the reactor core, but is not relevant to the spent fuel pool. Spent Fuel Pool criticality analyses do not require a specific fuel rotation. Therefore, there is no such thing in the spent fuel pool as a misrotated fuel assembly that affects the criticality analysis.
5/22/96	Several	NRC Press Release 96-74: NRC Staff Completes Survey of Refueling Practices at Nation's Nuclear Power Plants	15 power plants at nine sites that might have violated license commitments when fuel was moved from the reactor to the spent fuel storage pool during refueling.	This has to do with SFP decay heat removal, not criticality. Same response as Information Notice 95 - 54.
6/25/96	Byron	Licensee Event Report 96-008-00	Fuel assemblies in the spent fuel pool in the wrong position based on not meeting minimum burnup requirements.	This event involves 3 fuel assemblies which had attained a small amount of burnup less than required for regional storage. This event involves improperly qualifying these 3 fuel assemblies due to lack of timely independent verification. The effect on K-effective to this event was negligible. This event should not occur at Millstone 3 due to our requirement for independent verification prior to fuel qualification for regional storage. Even if this event did occur at Millstone 3, the reactivity effects would be easily bounded by the Millstone 3 criticality analysis limiting single fuel misload event.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
7/15/96	Braidwood 1	Licensee Event Report 96-007-00	A fuel misposition in the spent fuel pool because of failure to consider effects on lower burnup fuel in adjacent storage locations.	This event involves inadequate controls at the interface of 2 regions of fuel storage. While the fuel was properly placed within the required regions, the interface of the 2 regions was not properly controlled. This same event could not happen for the proposed Millstone 3 configuration. For all proposed Millstone 3 regions, except the Region 1 3-out-of-4 configuration, the racks have been analyzed to show that no interface problem exists with adjacent fuel storage regions, so that there are no additional requirements due to interface concerns. For Region 1 3-out-4 storage, the Technical Specification requirements specify the interface requirements to other regions. These interface requirements are essentially unchanged from the current requirements. Further, NNECO's rerack design is such that the Region 1 3-out-of-4 cell blocker placement has already been fixed, and this cell blocker placement precludes any interface requirement problem. Therefore, for the proposed Millstone 3 rerack, fuel meeting the requirements for regional storage, ensures by itself, with no additional actions, that there are no interface problems.
8/5/96	Braidwood 1	Licensee Event Report 96-008-00	A fuel assembly not in the required checkerboard configuration based on burnup vs. initial enrichment.	This event is bounded by the MP3 rerack criticality analysis. This is a single fuel handling event involving 1 misloaded fuel assembly. Further, procedures require independent verification for qualification of fuel which would prevent this event from happening at Millstone 3.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
9/30/96	Cooper	Notice of Violation 50-298/95-18	Failure by licensed operating personnel to follow procedural requirements, including: (1) failure to insert control rods in the proper sequence; (2) failure to notify the shift supervisor of a mispositioned control rod; and (3) failure to obtain the concurrence of the shift supervisor and reactor engineer for a recovery plan of the mispositioned control rods.	Not relevant to the Millstone 3 spent fuel pool criticality analysis. This involves improper control rod positioning and failure to follow procedures. It is inappropriate to conclude that because an administrative control fails in one particular area, that the administrative criticality controls for the SFP will also fail. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.
11/14/96	N/A	Briefing on Spent Fuel Pool Study - Public Meeting	NRC assessment of the likelihood and consequences of an extended loss of spent fuel pool cooling inventory.	This has to do with SFP decay heat removal, not criticality. Same response as IN 95 - 54.
3/24/97	Beaver Valley	Notice of Violation 50-334/96-10, 50-412/96-10	(1) Failure of staff to follow procedures and implement appropriate work practices and controls; (2) operators inadvertently deenergized the waste gas decay tank; and (3) failure to take appropriate corrective action.	Not relevant to the Millstone 3 spent fuel pool criticality analysis. This involves operator valve mispositioning and failure to follow procedures. It is inappropriate to conclude that because an administrative control fails in one particular area, that the administrative criticality controls for the SFP will also fail. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.
9/2/97	Zion	Notice of Violation and Proposed Imposition of Civil Penalties 50-295/97-006	Violation issues, including: (1) reactivity management; (2) command, control, and communication; (3) corrective actions - reactivity management event; (4) corrective actions - reactor voiding event; (5) failure to comply with a limiting condition for operation; (6) undetected displacement of reactor coolant; and (7) failure to report the accumulation of gas in the reactor coolant system.	Not relevant to the Millstone 3 spent fuel pool criticality analysis. This event involves reactivity management of the reactor core due to control rod operation. It is inappropriate to conclude that because an administrative control fails in one particular area, that the administrative criticality controls for the SFP will also fail. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
9/3/97	Calvert Cliffs	NRC Information Notice 97-68: "Loss of Control of Diver in a Spent Fuel Storage Pool"	Inadequacies in license control work which resulted in a diver crossing about 4.6 meters (15 feet) of unsurveyed fuel transfer area floor and coming within a few feet of radiation dose rate ranging from 120 to 200 Gy/hr (12,000 to 20,000 rad/hr).	This is not directly relevant to any of the contentions. The only commonality is that an administrative failure occurred in a spent fuel pool. It has nothing to do with criticality issues, which are the concern of the contentions. It is inappropriate to conclude that because an administrative control fails in one particular area, that the administrative criticality controls for the SFP will also fail. These SFP criticality controls can be directly assessed based on Millstone and industry experience. There is no need to infer success/failure based on the success/failure of other administrative controls.
4/3/98	TMI - 1	Licensee Event Report 98-002-01	Failure to take a sample of spent fuel pool following addition of water.	Same response as 4/19/90 Catawba 1 event.
6/4/98	Hope Creek	Notice of Violation 50-354/98-05	Violations issues, including: (1) the residual heat removal system was not maintained available during a 1990 refueling outage while the reactor core was fully offloaded for the purpose of augmenting fuel pool cooling; (2) check valves and gas bottle regulators were not properly tested; and (3) an inconsistency in a design basis assumption related to chiller temperature was found, but not acted upon.	This has to do with core/SFP decay heat removal, not criticality. Same response as IN 95 - 54.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
8/11/98	Braidwood 1	Licensee Event Report 96-010-02	Boraflex configurations not consistent with criticality analysis, and also Boral configuration was not consistent with criticality analysis.	Neither of the 2 issues should concern the MP3 License amendment. For Boraflex degradation, one of the purposes of the MP3 re-rack license amendment is to go to a no-Boraflex reactivity credit condition in the existing racks. This LER concerns long term Boraflex degradation, which eventually results in the Boraflex material condition falling outside of the bounds of the criticality analysis. "On-going" administrative controls (*i.e.*, long term Boraflex surveillance testing), which are used in implementing any physical criticality system, identified this Braidwood deficiency. It is this long-term Boraflex degradation that MP3 is trying to avoid by going to a no-Boraflex credit in the criticality analysis. Concerning the second issue in this LER, the Boral racks to be installed at MP3 have Boral panels on all 4 sides of the stored fuel assemblies, including the rack exterior cells. Therefore, the LER condition is not applicable to the Boral racks to be installed at MP3.
6/25/99	Salem-1, Diablo Canyon-1, Vogtle-2, San Onofre-2	NRC Information Notice 99-21: "Recent Plant Events Caused by Human Performance Errors"	Human performance weaknesses resulting in: (1) the Salem 1 reactor automatically shutting down because of a low bearing oil pressure turbine trip; (2) the Diablo Canyon 1 annunciator alarmed resulting in spent fuel pool pump 1-2 not operating as expected; (3) Vogtle 2 "steam flow /feed flow mismatch" annunciator spare alarm sounded resulting in unexpected closing of loop 3 valve; and (4) a loss of shutdown cooling recorded at San Onofre 2.	The second event was a SFP heatup event. This is not relevant to the contentions. Same response as IN 95 - 54.

Non-Millstone Related Experiences

DATE	PLANT	DOCUMENT	DESCRIPTION	NNECO POSITION
3/2/00	McGuire	Licensee Event Report 00-003-00	Certain modeling methods used to perform SFP criticality analyses determined to be non-conservative in that k_{eff} may exceed 0.95 for postulated off-normal conditions with 0 ppm boron in the SFP.	The Millstone 3 rerack criticality analyses do not have this problem. This issue concerns whether adequate reactivity uncertainties due to axial burnup distributions have been applied. For the Millstone 3 rerack, specific limiting axial burnup distributions for various burnups were analyzed in both 2 and 3 dimensions to determine bounding axial burnup penalties. This analysis was performed with bounding Millstone 3 specific data. Axial burnup uncertainties are specifically discussed in the licensing report submitted for the proposed amendment.
3/15/00	Yankee	Licensee Event Report 00-002-00	Discovered that past practice of moving fuel assemblies over lower tier storage racks resulted in assemblies being lifted 13 inches above the racks, which is outside the design basis.	This event is not applicable to Millstone Unit 3. The new Millstone 3 racks have been designed to be the same height as the existing racks, to avoid these types of problems.
3/23/00	Farley	Licensee Event Report 00-004-00	Determined that three fuel assemblies were loaded into SFP in configurations contrary to Technical Specifications.	Addressed in J. Parillo Affidavit, ¶¶ 43-46.

Rules and Regulations

Federal Register

Vol. 63, No. 218

Thursday, November 12, 1998

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50 and 70

RIN 3150-AF87

Criticality Accident Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to give licensees of light-water nuclear power reactors greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area in which the material is handled, used, or stored. This action is taken as a result of the experience gained in processing and evaluating a number of exemption requests from such licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

EFFECTIVE DATE: The final rule is effective on December 14, 1998.

FOR FURTHER INFORMATION CONTACT: Michael T. Jamgochian, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone: (301) 415-3224; e-mail: mtj1@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Background

The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to give persons licensed to construct or operate light-water nuclear power reactors the option of either meeting the criticality accident requirements of paragraph (a) through (c) of 10 CFR 70.24 in handling and storage areas for SNM, or electing to comply with certain requirements that are set forth in a new Section 50.68 in 10 CFR Part 50. The requirements in Section 50.68 are generally the requirements that the NRC has used to grant specific exemptions from the requirements of 10 CFR 70.24. In addition, the NRC is deleting the current text of Section 70.24(d) concerning the granting of specific exemptions from Section 70.24 because it is redundant to 10 CFR 70.14(a). Section 70.24(d) is rewritten to provide that the requirements in paragraphs (a) through (c) of 10 CFR 70.24 do not apply to holders of a construction permit or operating license for a nuclear power reactor issued under 10 CFR Part 50, or combined licenses issued under 10 CFR Part 52, if the holders comply with the requirements of 10 CFR 50.68(b).

II. Discussion

On December 3, 1997 (62 FR 63825), the NRC published a direct final rule in the **Federal Register** that would have provided persons licensed to construct or operate light-water nuclear power reactors with the option of either meeting the criticality accident requirements of paragraph (a) of 10 CFR 70.24 in handling and storage areas for SNM, or electing to comply with requirements that would be incorporated into 10 CFR Part 50 at 10 CFR 50.68. A direct final rule (62 FR 63825) and a parallel proposed rule (62 FR 63911) amending Parts 70 and 50 were published in the **Federal Register** on December 3, 1997. The statement of considerations for the direct final rule and the proposed rule stated that if significant adverse comments were received on the direct final rule, the NRC would withdraw the direct final rule and would address the comments in a subsequent final rule. Significant adverse comments were received from the public, and on February 25, 1998, the NRC published a notice withdrawing the direct final rule and revoking the regulatory text. Since the direct final rule had an effective date of February 17, 1998, it was necessary for the February 25, 1998 notice to revoke the regulatory text which became effective on February 17, 1998, as well as to withdraw the direct final rule. With the withdrawal and revocation, the proposed rule is the only regulatory proposal remaining. The NRC has determined to modify the proposed rule to address public comments and to make several editorial clarifications. The analysis of and response to the public comments to the proposed rule are set forth below.

III. Comments on the Proposed Rule

The NRC received comments on the December 3, 1997, proposed rule (62 FR 63911) from Commonwealth Edison, Carolina Power & Light Company, Southern Nuclear Operating Company, Nuclear Energy Institute, Northern States Power Company, Trojan Nuclear Plant, and Detroit Edison. Copies of the letters are available for public inspection and copying for a fee at the Commission's Public Document Room, located at 2120 L Street, NW. (Lower Level), Washington, DC. Many of the comment letters suggested editorial type changes, some of which have been incorporated into this final rule. The comments are classified into nine general comments and are addressed as follows:

Comment 1: The proposed rule should not prohibit licensees from applying for exemptions under the guidelines of 10 CFR 70.14 and should contain provisions to note that any existing approved exemptions remain valid.

Response: Even though the wording of paragraph (d) in the current version of 10 CFR 70.24, which provides for applying for exemptions should "good cause" exist, is being deleted, licensees are not prohibited from applying for such exemptions under the guidelines of paragraph (a) of 10 CFR 70.14, "Specific Exemptions."

The standard for issuance of exemptions under Section 70.14 is essentially the same as the "good cause" criterion in paragraph (d) of Section 70.24. Therefore, its removal from Section 70.24(d) will not change the standard for, or otherwise serve to limit the granting of, exemptions to Section 70.24.

This rulemaking does not affect the status of exemptions to the requirements of Section 70.24 that were previously granted by the NRC. A licensee currently holding an exemption to Section 70.24 may continue operation under its existing exemption (including any applicable conditions imposed as part of the granting of the exemption) and its current programs and commitments without any further action. Alternatively, a licensee currently holding exemptions to Section 70.24 may elect to comply with the new alternative provided under Section 50.68(b), but if it does so, its exemption would be inapplicable and would not serve as a basis for avoiding compliance with the criteria listed in Section 50.68(b). A licensee whose exemption was issued as part of its operating license and whose exemption contained conditions imposed as part of the granting of the exemption, need not apply for a license amendment to delete the exemption conditions as a prerequisite for complying with Section 50.68(b).

Comment 2: For many BWRs, optimum moderation calculations are not performed for the fresh fuel storage racks because administrative controls are in place to preclude these conditions. In accordance with vendor recommendations, compensatory measures have been established to preclude an optimum moderation condition in the fresh fuel storage racks. The rule should contain a provision that exempts this requirement if adequate controls have been established to preclude an optimum moderation condition.

Response: The NRC agrees and has added the following provision to 10 CFR 50.68(b)(3): "This evaluation need not be performed if administrative control and/or design features prevent such moderation, or if fresh fuel storage racks are not used."

Comment 3. The rule should eliminate the reference to General Design Criterion 63 (GDC 63) and should describe the underlying monitoring requirements.

Response: The reference to GDC 63 was initially incorporated to ensure that licensees receiving an exemption to 10 CFR 70.24 would not erroneously view the exemption as the basis for removing from the spent fuel pool area radiation monitors that were installed to meet other monitoring requirements, such as those contained in 10 CFR 20.1501 and GDC 63. This rule change does not affect these other monitoring requirements; therefore, referencing GDC 63 has been deleted.

Comment 4. Placing a limit on enrichment offers no direct safety benefit and should not be included.

Response: The NRC disagrees with the comment. The maximum allowable nominal enrichment of reactor fuel is currently limited to 5-weight percent on the basis of possible criticality concerns even in a dry environment, as well as currently approved extensions to 10 CFR 51.52 based on an environmental impact study for enrichments higher than 5-weight percent. Any future approved enrichment extension can be readily handled by modifying this criterion.

Comment 5. Replace "may not permit" with "shall prohibit the" in Criterion (1).

Response: The NRC agrees and has used the phrase suggested by the commenters.

Comment 6. Use of "pure water" and "unborated water" should be consistent.

Response: The NRC agrees. The final rule uses the term "unborated water."

Comment 7. Criteria (2) and (3) should not be applicable if the licensee does not use the fresh fuel storage racks.

Response: The NRC agrees and has added the following provision to 10 CFR 50.68 (b)(2) and (b)(3): "This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

Comment 8. The meaning of "transportation" in criterion (1) is unclear.

Response: The NRC agrees and has deleted the term.

Comment 9. The phrase "maximum permissible U-235 enrichment" in Criteria (2), (3), and (4) should be replaced by the phrase "maximum fuel assembly reactivity."

Response: The NRC agrees and has used the phrase suggested by the commenter.

IV. Section-by-Section Analysis

10 CFR 50.68

Paragraph (a) of Section 50.68 allows a nuclear power plant licensee (including a holder of either a construction permit or a combined operating license) the option of complying with Section 70.24 (a) through (c), or complying with the requirements in paragraph (b) of Section 50.68. The corresponding provision in Section 70.24 is paragraph (d).

Paragraph (b) sets forth eight specific requirements which a licensee must comply with so long as it chooses under the provisions of Section 50.68 to avoid compliance with the requirements of Section 70.24 (a) through (c).

A licensee currently holding an exemption to Section 70.24 may elect to comply with the new alternative provided under Section 50.68, but if it does so, its exemption to Section 70.24 is inapplicable to, and would not serve as a basis for avoiding compliance with the eight criteria in Section 50.68(b).

10 CFR 70.24

Paragraph (d)(1) of Section 70.24 allows a nuclear power plant licensee (including a holder of either a construction permit or a combined operating license) the option of complying with Section 70.24 (a) through (c), or complying with the requirements in 10 CFR Section 50.68. This paragraph is the corresponding provision to Section 50.68(a).

Paragraph (d)(2) clarifies that the status of exemptions to the requirements of Section 70.24 that were previously granted by the NRC continue unaffected by this rulemaking. A licensee currently holding an exemption to Section 70.24 may continue operation under its existing exemption (including any applicable conditions imposed as part of the grant of the exemption) and its current programs and commitments without any further action.

A license that seeks an exemption from the requirements of Section 70.24 must meet the criteria for an exemption under Section 70.14. The standard for issuance of exemptions remains unchanged from the old rule, since the Commission regards the former "good cause" criterion under the previous version of Section 70.24(d) as being essentially the same as the standard for issuance of exemptions under Paragraph 70.14.

V. Metric Policy

On October 7, 1992, the Commission published its final Policy Statement on Metrication. According to that policy, after January 7, 1993, all new regulations and major amendments to existing regulations were to be presented in dual units. The new addition and amendment to the regulations contain no units.

VI. Finding of No Significant Environmental Impact

The NRC has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, would not be a major Federal action significantly affecting the quality of the human environment; and therefore, an environmental impact statement is not required. The final rule provides an alternative to existing requirements on criticality monitoring. The alternative method contained in the final rule in the new Section 50.68 represents a codification of the criteria currently used by the NRC for granting exemptions from the criticality monitoring requirements in 10 CFR 70.24(a). These criteria provide an acceptable alternative for assuring that there are no inadvertent criticality events of special nuclear material at nuclear power reactors, which is the purpose of the criticality monitoring requirements in Section 70.24(a). Experience over 15 years has demonstrated that the alternative criteria have been effective in preventing inadvertent criticality events, and the NRC concludes that as a matter of regulatory efficiency, there is no purpose to requiring licensees to apply for and obtain exemptions from requirements of Section 70.24(a) if they adhere to the alternative criteria in the new Section 50.68. Since the alternative contained in Section 50.68 provides an equally effective method for preventing inadvertent criticality events in nuclear power plants, the NRC concludes that the final rule will not have any significant impact on the quality of the human environment. Therefore, an environmental impact statement has not been prepared for this regulation. This discussion constitutes the environmental assessment for this rulemaking.

VII. Paperwork Reduction Act Statement

This final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget, approval numbers 3150-0009 and 3150-0011.

VIII. Public Protection Notification

If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

IX. Regulatory Analysis

The current structure of the current 10 CFR 70.24 is overly broad and places a burden on a licensee to identify those areas or operations at its facility where the requirements are unnecessary, and to request an exemption if the licensee has sufficient reason to be relieved from the requirements. This existing structure has resulted in a large number of exemption requests.

To relieve the burden on power reactor licensees of applying for, and the burden on the NRC of granting exemptions, this amendment permits power reactor facilities with nominal fuel enrichments no greater than 5-weight percent of U-235 to be excluded from the scope of 10 CFR 70.24, provided they meet specific requirements being added to 10 CFR Part 50. This amendment is a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests which concluded that the likelihood of criticality was negligible.

The only other viable option to this amendment is for the NRC to make no changes and allow the licensees to continue requesting exemptions. If no changes are made, the licensees will continue to incur the costs of submitting exemptions and NRC will incur the costs of reviewing them. Under this rule, an easing of the burden on licensees results from not having to request exemptions. Similarly, the NRC's burden will be reduced by avoiding the need to review and evaluate these exemption requests.

This rule is not a mandatory requirement, but an easing of burden action which results in regulatory efficiency. Also, the rule does not impose any additional costs on existing licensees and has no negative impact on public health and safety, but will provide savings to future licensees, and may provide some reduction in burden to current licensees whose current exemption includes conditions which are more restrictive than the requirements in Section 50.68. There will also be savings in resources to the NRC as well. Hence, the rule is shown to be cost beneficial.

The foregoing constitutes the regulatory analysis for this final rule.

X. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the NRC hereby certifies that this rule, if adopted, will not have a significant economic impact on a substantial number of small entities. This rule affects only the licensees of nuclear power plants. These licensee companies that are dominant in their service areas, do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act, 5 U.S.C. 601, or the size standards adopted by the NRC (10 CFR 2.810).

XI. Backfit Analysis

The NRC has determined that this rule does not impose a backfit as defined in 10 CFR 50.109(a)(1), since it provides an alternative to existing requirements on criticality monitoring. Accordingly, the NRC has not prepared a backfit analysis for this rule.

XII. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a "major rule" and has verified this determination with the Office of Information and Regulatory Affairs, Office of Management and Budget.

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

10 CFR Part 70

Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

For the reasons stated in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, the National Environmental Policy Act of 1969, as amended, and 5 U.S.C. 553, the NRC is adopting the following amendments to 10 CFR Parts 50 and 70:

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

The authority citation for 10 CFR part 50 continues to read as follows:

1. **Authority:** Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951, as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 and 50.81

also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.68 is added under the center heading "Issuance, Limitations, and Conditions of Licenses and Construction Permits" to read as follows:

§ 50.68 Criticality accident requirements.

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under Part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures shall 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 moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

(5) The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

(6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).

PART 70—DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

The authority citation for 10 CFR part 70 continues to read as follows:

1. **Authority:** Secs. 51, 53, 161, 182, 183, 68 Stat. 929, 930, 948, 953, 954, as amended, sec. 234, 83 Stat. 444, as amended, sec. 1701, 106 Stat. 2951, 2952, 2953 (42 U.S.C. 2071, 2073, 2201, 2232, 2233, 2282, 2297f); secs. 201, as amended, 202, 204, 206, 88 Stat. 1242, as amended, 1244, 1245, 1246, (42 U.S.C. 5841, 5842, 5845, 5846).

Sections 70.1(c) and 70.20a(b) also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161). Section 70.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 70.21(g) also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Section 70.31 also issued under sec. 57d, Pub. L. 93-377, 88 Stat. 475 (42 U.S.C. 2077). Sections 70.36 and 70.44 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234).

Section 70.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237). Section 70.62 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

2. In § 70.24, paragraph (d) is revised to read as follows:

§ 70.24 Criticality accident requirements.

* * * * *

(d)(1) The requirements in paragraphs (a) through (c) of this section do not apply to a holder of a construction permit or operating license for a nuclear power reactor issued under part 50 of this chapter or a combined license issued under part 52 of this chapter, if the holder complies with the requirements of paragraph (b) of 10 CFR 50.68.

(2) An exemption from § 70.24 held by a licensee who thereafter elects to comply with requirements of paragraph (b) of 10 CFR 50.68 does not exempt that licensee from complying with any of the requirements in § 50.68, but shall be ineffective so long as the licensee elects to comply with § 50.68.

Dated at Rockville, Maryland this 28th day of October, 1998.

For the Nuclear Regulatory Commission.

William D. Travers,

Executive Director for Operations.

[FR Doc. 98-30253 Filed 11-10-98; 8:45 am]

BILLING CODE 7590-01-P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 98-NM-217-AD; Amendment 39-10880; AD 98-23-13]

RIN 2120-AA64

Airworthiness Directives; British Aerospace Model Viscount 744, 745, 745D, and 810 Series Airplanes

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule.

SUMMARY: This amendment supersedes an existing airworthiness directive (AD), applicable to all British Aerospace Model Viscount 700, 800, and 810 series airplanes, that currently requires repetitive inspections to detect cracks and corrosion in the inboard and outboard engine nacelle structures on the wings; replacement of any cracked fittings and mating struts; and treatment or replacement of any corroded fittings or struts. This amendment requires repetitive inspections to detect cracking or corrosion of the eye end fittings of the outboard engine lower support or of the bore of the taper pin holes, and repair, if necessary. This amendment also limits the applicability of the existing AD. This amendment is prompted by reports of cracked and separated lower eye end fittings. The actions specified by this AD are intended to detect and correct cracking of the eye end fittings of the outboard engine lower support, which could result in reduced structural integrity of the engine nacelle support structures.

DATES: Effective December 17, 1998.

The incorporation by reference of certain publications listed in the regulations is approved by the Director

DOCKETED
USNRC

'98 JAN -6 A11 :2

OFF... RULEMAKING... ADJUDICATIONS STAFF

DOCKET NUMBER
PROPOSED RULE PR 50 + 70
(62 FR 63825)
(62 FR 63911)

January 2, 1998

Secretary
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attn: Rulemaking and Adjudications Staff

The following comments are respectively submitted in response to the proposed changes to Criticality Accident Requirements, 10 CFR 50.68 and 70.24, published in Federal Register Volume 62, Number 232, Page 63825, December 3, 1997.

The phrase "as required by GDC 63" of proposed 10 CFR 50.68 (b) (6) should be removed for the following reasons. First, some plants were licensed before the General Design Criteria were promulgated and their licensing bases address the GDC on a case-by-case basis; the phrase in question infers that the General Design Criteria as stated in 10 CFR Part 50 Appendix A are part of every licensees' design basis. Second, the phrase does not add any substance since proposed 50.68 (b) (6) simply restates the relevant portion of GDC 63; omitting the reference would be consistent with proposed 50.68 (b) (1) through (5) which implement GDC 62 without specific reference to that GDC. Third, a person unfamiliar with 10 CFR 50 Appendix A would not recognize the reference to GDC 63 as stated.

Proposed 10 CFR 50.68 (b) (7), which places a five (5.0) weight percent limit on U-235 enrichment, should be eliminated and the phrase "maximum permissible U-235 enrichment" in proposed 50.68 (b) (2), (3), and (4) should be replaced by the phrase "maximum fuel assembly reactivity" for the following reasons. First, the discussion in the Federal Register announcement does not indicate that the enrichment limitation is the basis for a safety analysis; it is simply a statement of current practice. Second, the safety issue is fuel assembly reactivity of which enrichment is only one parameter; burnable poison, material selection, and geometry are major factors affecting reactivity that could compensate for higher enrichments. Third, by modifying 50.68 (b) (2), (3), and (4) as proposed, the reactivity limitation objective of fuel storage racks can be achieved without placing a limitation on fuel enrichment.

We appreciate the opportunity to comment on this proposed rule change.

Marcus H. Voth
Project Manager - Licensing
612-271-5116, marcus.h.voth@nspco

* Letter received by electronic mail on January 2, 1998 --- ATB

DS10

Northern States Power Company
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

Reference 12

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

The Birmingham 1,200-foot transition area described in § 71.181 (32 F.R. 2148 and 3765) would be altered as follows: "* * * thence southwest along the southeast boundary of V-209 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VOR; thence clockwise along this arc to longitude 87°30'00" W.; thence north along longitude 87°30'00" W. to point of beginning, excluding that portion that coincide with R-2101 and the Gadsden, Ala., transition area * * *." would be deleted and "* * * thence southwest along the southeast boundary of V-209 to longitude 88°00'00" W.; thence north along longitude 88°00'00" W. to the north boundary of V-18; thence northeast along the north boundary of V-18 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VORTAC; thence clockwise along this arc to longitude 87°30'00" W.; thence north along longitude 87°30'00" W. to point of beginning, excluding that portion that coincides with R-2101 and the Gadsden, Ala. transition area * * *." would be substituted therefor.

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

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

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

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

JAMES G. ROGERS,
Director, Southern Region.

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

[14 CFR Part 71]

[Airspace Docket No. 67-SO-64]

TRANSITION AREA

Proposed Designation

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

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

The Camden transition area would be designated as:

That airspace extending upward from 700 feet above the surface within a 7-mile radius of Woodward Field (latitude 34°17'03" N., longitude 80°33'53" W.); within 3 miles each side of the 040° bearing from the Camden RBN (latitude 34°17'02" N., longitude 80°33'42.8" W.), extending from the 7-mile radius area to 8 miles northeast of the RBN.

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

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

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

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

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

[14 CFR Part 71]

[Airspace Docket No. 67-EA-1]

FEDERAL AIRWAYS

Supplemental Proposed Alteration

On March 1, 1967, a notice of proposed rule making was published in the FEDERAL REGISTER (32 F.R. 3402) stating that the Federal Aviation Agency was considering amendments to Part 71 of the Federal Aviation Regulations that would realign V-1 from Cape Charles, Va., via the INT of Cape Charles 013° and Salisbury, Md., 206° True radials; to Salisbury; that would designate a segment of V-139 from Norfolk, Va., via Cape Charles; to Snow Hill, Md., including a west alternate from Norfolk to Snow Hill via INT of Norfolk 350° and Snow Hill 226° True radials; and that would revoke the segment of V-194 from Norfolk to INT of Norfolk 001° and Cape Charles 313° True radials. Floors of 1,200 feet above the surface were proposed for these airway segments. These actions were proposed to simplify air traffic control procedures and flight planning in the Norfolk area.

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

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

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

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

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

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

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

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

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

ATOMIC ENERGY COMMISSION

[10 CFR Part 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plant Construction Permits

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

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:

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS[1]

TABLE OF CONTENTS

[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.

Group and title	Criterion No.
V. Reactivity control:	
Redundancy of Reactivity Control	27
Reactivity Hot Shutdown Capability	28
Reactivity Shutdown Capability	29
Reactivity Holddown Capability	30
Reactivity Control Systems Malfunction	31
Maximum Reactivity Worth of Control Rods	32
VI. Reactor coolant pressure boundary:	
Reactor Coolant Pressure Boundary Capability	33
Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention	34
Reactor Coolant Pressure Boundary Brittle Fracture Prevention	35
Reactor Coolant Pressure Boundary Surveillance	36
VII. Engineered safety features:	
A. General requirements for engineered safety features:	
Engineered Safety Features Basis for Design	37
Reliability and Testability of Engineered Safety Features	38
Emergency Power for Engineered Safety Features	39
Missile Protection	40
Engineered Safety Features Performance Capability	41
Engineered Safety Features Components Capability	42
Accident Aggravation Prevention	43
B. Emergency core cooling systems:	
Emergency Core Cooling Systems Capability	44
Inspection of Emergency Core Cooling Systems	45
Testing of Emergency Core Cooling Systems Components	46
Testing of Emergency Core Cooling Systems	47
Testing of Operational Sequence of Emergency Core Cooling Systems	48
C. Containment:	
Containment Design Basis	49
NDT Requirement for Containment Material	50
Reactor Coolant Pressure Boundary Outside Containment	51
Containment Heat Removal Systems	52
Containment Isolation Valves	53
Containment Leakage Rate Testing	54
Containment Periodic Leakage Rate Testing	55
Provisions for Testing of Penetrations	56
Provisions for Testing of Isolation Valves	57
D. Containment pressure-reducing systems:	
Inspection of Containment Pressure-Reducing Systems	58
Testing of Containment Pressure-Reducing Systems	59
Testing of Containment Spray Systems	60
Testing of Operational Sequence of Containment Pressure-Reducing Systems	61
E. Air cleanup systems:	
Inspection of Air Cleanup Systems	62
Testing of Air Cleanup Systems Components	63
Testing of Air Cleanup Systems	64
Testing of Operational Sequence of Air Cleanup Systems	65
VIII. Fuel and waste storage systems:	
Prevention of Fuel Storage Criticality	66
Fuel and Waste Storage Decay Heat	67
Fuel and Waste Storage Radiation Shielding	68
Protection Against Radioactivity Release from Spent Fuel and Waste Storage	69
IX. Plant effluents:	
Control of Releases of Radioactivity to the Environment	70

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

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

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

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

I. OVERALL PLANT REQUIREMENTS

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

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

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

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

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

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

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

Criterion 7—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.

terion 8—Overall Power Coefficient (*…gory 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 pre…ribed operating ranges.

…*riterion 13—Fission Process Monitors and …trols* (*Category B*). Means shall be pro…ded 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 (*Catergory B*). Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Criterion 20—Protection Systems Redundancy and Independence (*Category B*). Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection system. 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 reactor 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.]

Reference 13

Act of February 2, 1903, as amended the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, the reference to the State of Ohio in the introductory portion of paragraph (e) and paragraph (e)(9) relating to the State of Ohio are deleted.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended.)

Effective date. The foregoing amendment shall become effective upon issuance.

The amendment excludes a portion of Clinton County, Ohio, from the areas quarantined because of hog cholera. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not apply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the excluded area. No areas in Ohio remain under the quarantine.

The amendment relieves certain restrictions presently imposed but no longer deemed necessary to prevent the spread of hog cholera and must be made effective immediately to be of maximum benefit to affected persons. It does not appear that public participation in this rule making proceeding would make additional information available to this Department. Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and unnecessary, and good cause is found for making it effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 16th day of February 1971.

F. J. MULHERN,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2380 Filed 2-19-71;8:46 am]

[Docket No. 71-526]

PART 76—HOG CHOLERA AND OTHER COMMUNICABLE SWINE DISEASES

Areas Quarantined

Pursuant to provisions of the Act of May 29, 1884, as amended, the Act of February 2, 1903, as amended, the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, in paragraph (e)(13) relating to the State of Texas, subdivision (xvi) relating to Smith County is deleted, and new subdivisions (xxii) and (xxiii) relating to Bexar County are added to read:

(13) *Texas*. * * *

(xxii) That portion of Bexar County bounded by a line beginning at the junction of Interstate Highway 410 and Farm-to-Market Road 78; thence, following Farm-to-Market Road 78 in a northeasterly direction to Farm-to-Market Road 1518; thence, following Farm-to-Market Road 1518 in a southeasterly and then southwesterly direction to U.S. Highway 87; thence, following U.S. Highway 87 in a northwesterly direction to Interstate Highway 410; thence, following Interstate Highway 410 in a northwesterly direction to its junction with Farm-to-Market Road 78.

(xxiii) That portion of Bexar County bounded by a line beginning at the junction of the Bexar-Medina County line and State Highway 16; thence, following State Highway 16 in a southeasterly direction to Farm-to-Market Road 471; thence, following Farm-to-Market Road 471 in a southwesterly and then northwesterly direction to Farm-to-Market Road 1957; thence, following Farm-to-Market Road 1957 in a southeasterly and then southwesterly direction to the Bexar-Medina County line; thence, following the Bexar-Medina County line in a northerly direction to its junction with State Highway 16.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended)

Effective date. The foregoing amendments shall become effective upon issuance.

The amendments quarantine portions of Bexar County, Tex., because of the existence of hog cholera. This action is deemed necessary to prevent further spread of the disease. The restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will apply to the quarantined portions of such county.

The amendments also exclude a portion of Smith County, Tex., from the areas quarantined because of hog cholera. No areas in Smith County, Tex., remain under the quarantine. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not comply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the area excluded from quarantine.

Insofar as the amendments impose certain further restrictions necessary to prevent the interstate spread of hog cholera, they must be made effective immediately to accomplish their purpose in the public interest. Insofar as they relieve restrictions, they should be made effective promptly in order to be of maximum benefit to affected persons.

Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendments are impracticable, unnecessary, and contrary to the public interest, and good cause is found for making them effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 16th day of February 1971.

F. J. MULHERN,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2339 Filed 2-19-71;8:46 am]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

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

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

(i) The principal design criteria for the facility

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

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

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

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

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised criteria have been reduced to 55 in number, include definitions of important terms, and have been rearranged to increase their usefulness in the licensing process. Additional criteria describing specific requirements on matters covered in more general terms in the previously proposed criteria have been added to the criteria. The Categories A and B used to characterize the amount of information needed in Safety Analysis Reports concerning each criterion have been deleted since additional guidance on the amount and detail of information required to be submitted by applicants for facility licenses at the construction permit stage is now included in § 50.34 of Part 50. The term "engineered safety features" has been eliminated from the revised criteria and the requirements for "engineered safety features" incorporated in the criteria for individual systems.

Further revisions of these General Design Criteria are to be expected. In the course of the development of the revised criteria, important safety considerations were identified, but specific requirements related to some of these considerations have not as yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion in the criteria at this time. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(i) Consideration of the need to design against single failures of passive components in fluid systems important to safety.

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem and the required interconnection and independence of the subsystems have not yet been developed or defined.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, DC.

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

§ 50.34 Contents of applications; technical information.

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

* * * * *

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[1] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units:

* * * * *

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

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

INTRODUCTION

DEFINITIONS

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

INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

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

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

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

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

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the envirionmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electrical power systems. An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

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

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

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

Criterion 18—Inspection and testing of electrical power systems. Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

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

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

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

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

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

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

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

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

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

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

(Secs. 161, 182, 68 Stat. 948, 953; 42 U.S.C. 2201, 2232)

Dated at Washington, D.C., this 10th day of February 1971.

For the Atomic Energy Commission.

W. B. McCool,
Secretary of the Commission.

[FR Doc.71-2370 Filed 2-19-71;8:48 am]

Title 14—AERONAUTICS AND SPACE

Chapter I—Federal Aviation Administration, Department of Transportation

[Docket No. 71-EA-13; Amdt. 39-1158]

PART 39—AIRWORTHINESS DIRECTIVES

American Aviation Corp.

The Federal Aviation Administration is amending § 39.13 of Part 39 of the Federal Aviation Regulations so as to issue an airworthiness directive applicable to

Reference 14

DOCKET NUMBER PR
PROPOSED RULE 50

OAK RIDGE NATIONAL LABORATORY

OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37830

September 6, 1967

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.

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.

<u>Criterion 24</u> - 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.

<u>Criterion 25</u> - 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.

<u>Criterion 26</u> - Protection Systems Fail-Safe Design

This criterion places a requirement not only on the protection system but on the plant as well. For example, a plant design could be such that operation of the protection mechanism when not needed would be highly undesirable. (An illustration is the closure of the steam stop valves in a

BWR.) Criterion 26 requires the plant to be able to accept operation of the protection system when not needed. We believe this is a good objective and we support this criterion.

Section V - Reactivity Control

1. The title of this section should be "Reactivity Control for Reactor Shutdown".

2. This group of criteria should distinquish more clearly between functions of reactivity control; namely, the dynamic reactivity reduction process and the static holddown functions. The first function must be performed at such times as in power transients and loss-of-coolant accidents with the objective of preventing exceeding "acceptable fuel damage limits" referred to in Criteria 28 and 29. Margins expressed in terms of shutdown parameters are inappropriate and inadequate for the dynamic function.

   The reliability with which each function must be carried out depends upon the seriousness of the consequences of failure of that function.

Criterion 27 - Redundancy of Reactivity Control

This criterion is not clear. It does not state whether the two reactivity control systems (1) should both be capable of both increasing and decreasing reactivity for operation, or (2) should both be capable of fast shutdown, or (3) should one be for fast shutdown and one for holddown. We recommend that the word "shutdown" be substituted for "control" in this criterion. These systems should also meet the requirements of Criteria 28, 29, 30, 31, and 32.

Criteria 28, 29, and 30 taken together indicate that one of the shutdown systems is not required to cope with positive transients and is essentially a method of obtaining reactivity holddown capability. However, reactors that must be shut down rapidly to allow the containment system to function need two separate and fast shutdown systems. A single fast or "primary" shutdown system together with a "holddown", or slow, "secondary" shutdown system is not satisfactory in this case.

Criterion 29 - Reactivity Shutdown Capability

As stated in our comments on Criterion 27, some reactors require a shutdown to allow the containment to function. In such cases, this criterion should require that two shutdown systems be applied. Each such system should be capable of preventing an unacceptable situation.

This criterion carries a reference to shutdown margin that could well be made a separate criterion as the shutdown requirements are a function of the number of rods, reactor operating conditions and function desired (e.g., reduction of nuclear power level or holddown of the subcritical reactor). Although we have not addressed ourselves to these conditions in detail, we believe that a margin much greater than the worth of the most effective control rod is needed for reactors having many rods.

Criterion 30 - Reactivity Holddown Capability

In cases requiring the reactor to be shut down in order to achieve containment, two of these systems should be required. See comments on Criteria 27 and 29.

Criterion 31 - Reactivity Control Systems Malfunction

This criterion should be expanded to include all failures of the plant operating system that are capable of increasing reactivity. In particular this criterion should not be limited to the unplanned withdrawal of only one control rod since a failure of the control rod operating system may not be restricted to the withdrawal of only one rod. All failures that may affect the performance of the control rod operating system must be considered. Of a more general nature, all failures that can introduce reactivity increases must be considered. In addition to control rods, there are coolant temperature changes, and perhaps even void effects that need analysis.

Criterion 33 - Reactor Coolant Pressure Boundary Capability

We agree with the intent of the criterion but it is not clear what is meant by "positive mechanical means" for preventing a rod ejection. A definition is needed.

Section VII - Engineered Safety Features

With the exception of reactor shutdown systems, all other engineered safety features are discussed in this section. These are: emergency power system, emergency core cooling system, containment enclosure system, containment pressure-reducing system (including containment heat removal), and air cleaning systems.

For each of these systems, there should be criteria for design of the system and their components as well as criteria for testing and inspection.

The objective of these criteria would be clearer if each system were treated in separate subsections and the criteria for each were set up in parallel form. Thus, there would be criteria for the inspection and testing of emergency power system (now covered in only Criterion 39) as well as the inspection and testing criteria for the other engineered safety features. Criterion 52, "Containment Heat Removal Systems," would be grouped with Criteria 58-61 with which it is generally associated. Such a rearrangement raises questions on other points of apparent inconsistancy, e.g., Criterion 60 is seen to be but a special case of Criterion 61, etc.

Criterion 37 - Engineered Safety Features Basis for Design

Again a definition of engineered safety features is necessary. For example, if the scram must work in order that the containment not be overstressed, then the scram system must be considered part of an engineered safety feature.

Criterion 38 - Reliability and Testability of Engineered Safety Features

We agree with this criterion. However, its title and inclusion in Section VII, both of which pertain only to engineered safety features, does not reflect its more general applications which include "inherent" as well as "engineered safety features". It would more appropriately be included in Section I.

Criterion 39 - Emergency Power for Engineered Safety Features

A difficult point in the application of this criterion is that of redundancy in the offsite power system. For example, a plant failure that results in shutting off the electric generator driven by the reactor could produce the loss of all offsite power. The probability of this consequential loss of offsite power varies widely as a result of changes in the power system and of variations in power system load. As a result of this wide variation in the reliability of offsite power, we recommend that this criterion require that redundant and independent onsite power system be required such that onsite power alone be capable of supplying the needs of the engineered safety features after a failure of a single active component in the onsite power system. We do not believe that the offsite power is really independent of the power from a main generator operated from the reactor to be safeguarded.

Criterion 40 - Missile Protection

Analysis shall be made to show that fragments and components that could be ejected from highly pressurized system's rotating equipment would not impair the function of an engineered safety feature. Typical missiles requiring analyses are such items as primary system valves, flanges, instrumentation, etc. When rotating equipment is not completely contained, such as in a concrete vault, a missile map should be provided for rotating equipment (e.g., main turbines, pumps, etc.)

Criterion 41 - Engineered Safety Features Performance Capability

We agree with this criterion as far as it goes. In particular the detailed requirements for the emergency core cooling system as contained in Criterion 44 illustrate the desired amplification (but for that system only). Thus, it could be generalized and added to Criterion 41 as follows: "The performance of each engineered safety feature shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident."

Criterion 42 - Engineered Safety Features Components Capability

We see no need to limit this criterion to the loss-of-coolant accident and suggest that . . . "by the effects of a loss-of-coolant accident" be changed to read "the effects of the accident for which the function is required."

Criterion 43 - Accident Aggravation Prevention

It is not obvious what purpose this criterion is intended to serve. If something specific is in mind here it should be stated, i.e., are we worried about the core becoming critical again, or inducing a thermal shock, etc. Perhaps this should not even appear here but be in the general discussion.

Criterion 44 - Emergency Core Cooling Systems Capability

As noted in the discussion on Criterion 41, we would restrict this criterion to the first two sentences (having already included the remainder of this criterion as a general requirement in Criterion 41). However, as we interpret the intent of these sentences, each of the two emergency cooling systems should cover the whole range of pipe break conditions up to the maximum. To make this point clearer, it might be better to rephrase the second sentence defining the cooling system requirements as follows: "For each size break in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe, at least two emergency core cooling systems, preferably of different design principles and each with a capability for accomplishing abundant emergency core cooling, shall be provided."

Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling Systems

We agree with the intent of this criterion and suggest that in addition to "the transfer to alternate power sources" the operation of the reactivity control system (which must shutdown the reactor and then provide holddown in the cold condition after the loss-of-coolant accident) should be mentioned.

Criterion 49 - Containment Design Basis

We agree with the intent of this criterion but feel that the following need some elaboration:

Line 10: "Considerable Margin" should be defined in some manner.

Line 13: What degree of failure of the emergency core cooling system is assumed?

Criterion 50 - NDT Requirement for Containment Material

This criteria needs further clarification. The temperature of the steel members in question under normal operating and testing conditions should be defined, i.e., the temperature of the component when the ambient temperature is at its lowest recorded (or perhaps expected) value. Furthermore, the requirement of NDT + 30° F has no meaning in the eyes of the stress analyst although it has found some usage. This temperature is half way between NDT and FTE and unless there is adequate justification of which we are unaware, we recommend using NDT + 60° F which defines the transition, e.g., temperature at which cracks won't propagate at stresses less than yield.

Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment

The intent of this criterion is not clear. It would appear that Criterion 53 which requires redundant valving would also cover reactor containment coolant boundaries outside containment. If, however, it is intended to require extensions of the containment, it should be specifically stated. In any event . . . delete "appropriate" and "as necessary" in lines 4 and 5 and the entire last sentence which begins, "Determination of . . .". These words do not materially contribute to the sense of the statement of the criterion and therefore should be omitted.

Criteria 54, 55, and 56 - Containment Leakage Rate Testing, Containment Periodic Leakage Rate Testing, and Provisions for Testing of Penetrations

Following the words "design pressure" it is suggested that "defined by Criterion 49" be inserted.

Criterion 56

This criterion is not sufficiently inclusive. The types of penetrations which should be tested should NOT be limited to the two that are mentioned, but for instance should also include electrical penetrations and piping penetrations that do not require expansion joints. The penetration testing is usually done at greater than design pressure.

Criterion 66 - Prevention of Fuel Storage Criticality

We do not understand the implication of "or processes" at the end of the first sentence, nor do we believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. Hence, the last sentence of this criterion should be changed to read as follows: "Such means as geometrically safe configuations shall be used to insure that criticality cannot occur."

Criterion 67 - Fuel and Waste Storage Decay Heat

To the extent that removal of decay heat is a function necessary to prevent escape of fission products, decay heat removal systems should be designed to the same requirements for redundancy, inspectability, and testability as engineered safety features on reactors. This should include facilities for supplying additional coolant fluid in the event of accidental loss.

Reference 15

OFFICIAL USE ONLY

OFFICIAL USE ONLY

January 28, 1971

SECY-R 143

AMENDMENT TO 10 CFR 50 - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Note by the Secretary

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

W. B. McCool

Secretary of the Commission

DISTRIBUTION	NO. OF COPIES
Secretary	10
Chairman Seaborg	4
Commissioner Ramey	2
Commissioner Johnson	2
Commissioner Larson	2
Commissioner	1
General Manager	2
Deputy Gen. Mgr.	1
Director of Regulation	1
Deputy Dir. of Regulation	1
Asst.Dir.of Reg.for Admn.	2
Tech. Adv. to the Dir.of Reg.	1
Asst. Dir.of Reg. for Reactors	1
Asst. Gen. Mgr.	1
Exec. Asst. to GM	1
Spec. Asst. to GM	1
Asst. GM for Admn.	1
Asst. GM for Operations	1
Asst. GM for Plans	1
Asst. GM for Reactors	1
General Counsel	6
Compliance	6
Congr. Relations	1
Controller	5
Inspection	1
Materials Licensing	2
Operational Safety	2
Plans & Reports	2
Public Information	2
Reactor Dev. & Tech.	10
Reactor Licensing	2
Reactor Standards	2
State & Lic. Relations	2
Chairman, AS&LBP	2

LEG.! 1-3 Part 50

OFFICIAL USE ONLY

OFFICIAL USE ONLY

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

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

THE PROBLEM

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

BACKGROUND AND SUMMARY

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

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

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

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

OFFICIAL USE ONLY

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

a. Published Criterion 1 - Quality Standards

Comment - It should not be necessary for each applicant to show that an applicable code or standard is sufficient. A showing of sufficiency should be required only for those items not covered by an applicable code or standard.

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

b. Published Criterion 11 - Control Room

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

Resolution - The criterion has been rewritten to make it clear that only one control room is required and reference to Part 20 has been deleted (see New Criterion 19). It should be noted that we have discussed control

OFFICIAL USE ONLY

room requirements with industry representatives in order to understand better their views. One reactor manufacturer, supported by several utilities, made a presentation to the regulatory staff on this subject. The new wording of the criterion is in agreement with the industry position expressed in these discussions.

c. Published Criterion 28 - Reactivity Hot Shutdown Capability

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

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

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

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

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

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

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

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

f. Published Criterion 44 - Emergency Core Cooling Systems Capability

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

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

g. Published Criterion 49 - Containment Design Basis

Comment - Functioning of the emergency core cooling system is required for containment integrity; therefore,

OFFICIAL USE ONLY

it is inconsistent to require that the containment design be based on the assumed failure of emergency core cooling systems.

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

5. The staff met in February 1970 with an ad hoc AIF group, which included representatives of reactor manufacturers, utilities and architect engineers to discuss the revised General Design Criteria. The comments of this group were reflected in a June 4, 1970 draft of the revised General Design Criteria that was forwarded to the AIF for comment. The AIF forwarded comments and stated it believed the criteria should be published as an effective rule after reflecting its comments. These comments have been reflected in the General Design Criteria in Appendix "A".

6. The revised criteria establish minimum requirements for the design of water-cooled nuclear power units and provide guidance for the design of other nuclear power units whereas the previously proposed criteria provided guidance for applicants for construction permits for all types of nuclear power plants.

7. The revised criteria include definitions in accordance with comments received from industry that certain crucial terms should be defined. In addition, the criteria have been rearranged to increase their usefulness to designers and evaluators.

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

OFFICIAL USE ONLY

9. The revised criteria do not include the term "engineered safety features." The requirements in the previously proposed criteria for these features have been incorporated in the revised criteria for the individual systems which are used for this purpose.

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

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

STAFF JUDGMENTS

12. The Divisions of Reactor Licensing and Compliance and the Office of the General Counsel concur in the recommendation of this paper. The draft public announcement was prepared by the Division of Public Information. The Office of Congressional Relations concurs in the draft letter to the Joint Committee on Atomic Energy.

RECOMMENDATION

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

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

b. Note that the amendment to 10 CFR Part 50 set forth in Appendix "A" will be published in the Federal Register to be effective 90 days after publication.

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

d. Note that a public announcement such as Appendix "D" will be issued when the amendment is filed with the Federal Register.

LIST OF ENCLOSURES

APPENDIX "A"

TITLE 10 - ATOMIC ENERGY

CHAPTER 1 - ATOMIC ENERGY COMMISSION

PART 50 - LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

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

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

(i) The principal design criteria for the facility

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

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

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

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

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised criteria have been reduced to

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem and the required interconnection and independence of the subsystems have not yet been developed or defined.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, non-random, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

[illegible] of [illegible], as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U. S. Atomic Energy Commission, Washington, D. C., 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C.

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

§ 50.34 Contents of applications; technical information.

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

*   *   *   *   *

(3) The preliminary design of the facility including:

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

* * * * *

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

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

* * * * *

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

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. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) [illegible] failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26 and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

LOSS OF COOLANT ACCIDENTS

Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

SINGLE FAILURE

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

ANTICIPATED OPERATIONAL OCCURRENCES

Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited

---

[1]Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2]Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. OVERALL REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS AND RECORDS

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

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

CRITERION 3 - FIRE PROTECTION

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

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

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

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

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

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

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

CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

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

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

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

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially 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 (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

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

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

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

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

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

CRITERION 19 - CONTROL ROOM

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

Equipment [illegible] the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable [illegible] the protection system can [illegible] demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

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

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

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

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

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

CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

[illegible] AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. FLUID SYSTEMS

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

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

[illegible] 32 [illegible]

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

CRITERION 33 - REACTOR COOLANT MAKEUP

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

CRITERION 34 - RESIDUAL HEAT REMOVAL

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

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

CRITERION 35 - EMERGENCY CORE COOLING

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

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

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in [illegible] pressure vessel, [illegible], and piping, to assure the integrity and capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

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

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

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and 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, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

CRITERION 44 - COOLING WATER

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

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

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. REACTOR CONTAINMENT

CRITERION 50 - CONTAINMENT DESIGN BASIS

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

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

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

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

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

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

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

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

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

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment. or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment. or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

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

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment. or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment. or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. FUEL AND RADIOACTIVITY CONTROL

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

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

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

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

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

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

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

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

Dated at ______________________________ this __________

day of ________________ 1971.

FOR THE ATOMIC ENERGY COMMISSION

____

W. B. McCool
Secretary of the Commission

APPENDIX "B"

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

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

APPENDIX "C"

DRAFT LETTER TO THE JOINT COMMITTEE ON ATOMIC ENERGY

1. Enclosed for the information of the Joint Committee is a copy of a notice of rule making amending the Commission's regulation "Licensing of Production and Utilization Facilities," 10 CFR Part 50 to add an Appendix A, General Design Criteria for Nuclear Power Plants. Proposed criteria were published for comment on July 11, 1967. The criteria in the notice of rule making reflect consideration of the comments received on the proposed criteria published for comment and subsequent developments in the technology and in the licensing process.

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

3. The amendment will be effective 90 days after publication in the Federal Register.

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

APPENDIX "D"

DRAFT PUBLIC ANNOUNCEMENT

AEC PUBLISHES GENERAL DESIGN CRITERIA
FOR NUCLEAR POWER PLANTS

The AEC is publishing a revised set of general design criteria for use in establishing the principal design criteria for nuclear power plants.

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

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

The amendment to Part 50 of the Commission's regulations fixes minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units previously approved by the Commission for construction. It provides guidance, also, for establishing the principal design criteria for other

PROPOSED RULE PR-50
Gen. Design Criteria

AI ATOMICS INTERNATIONAL

A Division of North American Rockwell Corporation

September 25, 1967

In reply refer—
67AT-5374

DOCKETED
USAEC
SEP 29 1967
Office of the Secretary
Public Proceedings
Branch

Secretary
U. S. Atomic Energy Commission
Washington, D.C. 20545

Gentlemen:

The revised set of proposed General Design Criteria, which were published in the Federal Register on July 11, 1967, for public comment, represents the results of a great deal of very fruitful effort to develop standards to assist in the preparation of applications for nuclear power plant construction permits. The early release of the first set of criteria developed by the regulatory staff, with the request for comments, initiated the extensive efforts recognized as necessary for effective evolution and development of the criteria. These resulting criteria, which reflect the public comments and suggestions, represent a significant improvement, both in organization and format and in content, over the initial criteria published in 1965. They offer considerably more and better guidance for the preparation of applications for nuclear power plant construction permits and operating licenses.

Our review has resulted in a number of comments and recommendations which are outlined below. Our more general comments are followed by those specifically directed to the individual criteria by number.

First we recommend that in adoption of the proposed criteria as a part of 10 CFR 50, they be more specifically directed to and required of large pressurized and boiling water reactors. This approach in the application would reduce the possibility of ritualistic adherence by reviewers to the requirements of the criteria when considering reactor types other than those for which the criteria were specifically developed. Detailed implementation of the criteria for other reactor types, and particularly for the advanced reactors now receiving major attention, can then proceed in whatever manner is most appropriate for the reactor without preconceived conclusions from the results of application to the water reactors. Also this more specific application to water reactors will reduce the possibility of their misuse by intervenors in public hearings for other reactor types.

P. O. Box 309, Canoga Park, California 91304
Cable Address: ATOMICS

The proposed criteria appear to be extremely qualitative in a number of areas. For example, we note the use of words and phrases such as: "impairing of safety" (Criterion 4), "acceptable fuel damage limits" Criteria 6 and 14), "appropriate margins" (Criterion 6), "exceedingly low probability" (Criterion 9), "high functional reliability" (Criteria 19 and 38), "sufficient" (Criterion 20), "necessary" (Criterion 20), "considerable margin" (Criterion 32), "limited allowances" (Criterion 33), "abundant" and "negligible" (Criterion 44), "considerable margin" (Criterion 49), "as close to design as practicable" (Criteria 61 and 65), "reliable" (Criterion 67), "undue amounts" (Criterion 69), and "high population density for very large cities" (Criterion 70). While we recognize that development of effective definitions of these types of terms is a very difficult task, we wish to encourage a strong continuing effort to define the terms quantitatively and then to include a section on definitions as an integral part of the criteria.

Our specific comments on the individual criteria are identified below by each criterion number.

2. Some quite specific criteria have been developed and applied to such natural phenomena as tornadoes and earthquakes in previous reactor application reviews. Including examples of this kind of guidance would be helpful to applicants. We also recommend that, in addition to the two items cited, the design bases established as a result of this criterion reflect the results of analyses which include not only the quantitative severity of the natural phenomena but also their probability of occurrence.

4. The implication that any degradation or impairment of safety is unacceptable and should be removed.

5. It might be noted that the records should be accessible subsequent to the occurrence of an accident.

8. We believe that it is unnecessary to require the overall power coefficient to be not positive in the power operating range. It is quite possible for the overall coefficient to be positive, and there be no unacceptable safety problem. For example, in a sodium graphite reactor, the coefficient has a prompt negative component together with a positive component with a long time constant. This results in an overall positive coefficient, but the negative portion of the coefficient is large enough and fast enough to assure

satisfactory control and safety. In fact, the lack of an overall negative coefficient is an advantage, since compensation for a large temperature and power defect in the reactivity is not required.

10. It is entirely conceivable that containment, as used today for water reactors, may not be required for other types of reactors currently under development. It would seem appropriate to give some recognition now to this in this criterion.

11. The basic requirement here is the provision of a control room that will remain habitable and will provide capability to shut the reactor down and maintain it in a safe condition. Application of the radiation exposure limits in 10 CFR 20 in this criterion is unduly stringent and is unnecessary. The 10 CFR 20 limits are for normal operations and should not be required in "accident conditions."

13. The requirement for monitoring the fission process for "... all conditions that can ... cause variations in reactivity" is too inclusive in this context. The examples given are simple and of external origin. More subtle conditions could be, e.g., fuel motion during life, changes in core geometry, etc. It may not be possible to monitor these conditions directly. What is important is monitoring of reactivity, and a predictive analysis by means of which observations and predictions can be compared, and any anomalies identified.

14. We submit that it is unnecessary for all core protection systems "to act automatically."

16. This criterion should require monitoring for leakage of reactor coolant; monitoring the "reactor coolant pressure boundary" is unnecessary.

20. The bases for determining when two different operating principles are necessary should be included here.

28. It is not necessary for two reactivity control systems to act fast enough to prevent exceeding acceptable fuel damage. Hence, we recommend deletion of "... including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits."

29. Shutdown margins greater than the worth of the most effective control rod appear inconsistent with the fact that reactors now being licensed have in excess of 100 such rods. We suggest the criterion be directed to providing shutdown margins greater than the maximum worth of any one gang of rods which can be driven or controlled by an operator or the control system.

36. We would point out that, except for financial risk, the requirements of this criterion are unnecessary if failure of the coolant boundary does not result in loss of coolant and subsequent core failure. Hence, application of this to low pressure coolant systems can be relaxed significantly.

39. Requirements for offsite power should be deleted, since adequate onsite power systems must always be required for emergency operation of the engineered safety features.

42. Here, it should be recognized that the loss-of-coolant accidents may not be design basis accidents for other power reactors for which these criteria are generally applicable.

44. We believe that the extent of independence and redundancy outlined here for the emergency core cooling systems is not necessary for low pressure systems. Also we question the necessity for "preferably of different design principles."

66. The second sentence should be replaced with "Inherent means should be used where practicable."

67. The criterion should be revised to require the design to be based on preventing exposures in excess of 10 CFR 20 limits.

69. The criterion should require that containment be provided if radioactivity releases due to accidents lead to public exposure in excess of 10 CFR 20 limits.

We believe your consideration of our comments will lead to further improvements in the General Design Criteria. If there are questions, or if we can provide further clarification, we shall be pleased to do so.

Sincerely yours,

Sidney Siegel, for

J. J. Flaherty
President
Atomics International Division

Reference 17

Those Listed Below

October 7, 1966

G. A. Arlotto
Facilities Standards Branch, SS

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

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

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

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

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

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

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

Attachment:
As Stated Above

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

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

Form AEC-318 (Rev. 9-53) U. S. GOVERNMENT PRINTING OFFICE 16—62761-2

Those Listed Below

October 7, 1966

G. A. Arlotto
Facilities Standards Branch, SS

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

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

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

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

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

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

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

Attachment:
As Stated Above

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

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

Form AEC-318 (Rev. 9-53)

U. S. GOVERNMENT PRINTING OFFICE 16—62761-1

Revised Draft
10/6/66

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

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

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

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

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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

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

CRITERION 1 (Category A) QUALITY AND PERFORMANCE STANDARDS

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

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

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

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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

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

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

CRITERION 10 (Category B) FUEL AND WASTE STORAGE SYSTEMS

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

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

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

OFFICE ▶						
SURNAME ▶						
DATE ▶						

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

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

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

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

FACILITY

CRITERION 1

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

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

CRITERION 6

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

CRITERION 7

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

CRITERION 8

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

CRITERION 9

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

CRITERION 14

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

CRITERION 15

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

CRITERION 16

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

CRITERION 19

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

CRITERION 20

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

CRITERION 21

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

CRITERION 22

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

CRITERION 26

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

CRITERION 27

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

VIII. FUEL AND WASTE STORAGE SYSTEMS

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

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

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

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

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

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

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

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

IX. PLANT EFFLUENTS

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

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

February 8, 1967

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

Dear Mr. Palladino:

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

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

Sincerely yours,

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

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

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

OFFICE ▶	SS:ADIR	RL				
SURNAME ▶	DiNunno:jjb	PAM				
DATE ▶	2/8/67	2/8/67				

GENERAL DESIGN CRITERIA

FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS

February 6, 1967

Table of Contents