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May 8, 2000

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Subj: Northeast Nuclear Energy Company
(Millstone Nuclear Power Station, Unit 3)
Docket No. 50-423-LA-3

Administrative Judges:

On March 24, 2000, the Nuclear Regulatory Commission (NRC) Staff issued Board Notification 2000-02 to the Atomic Safety and Licensing Board (Licensing Board) and to the parties to this proceeding. The Board Notification addressed the Staff's request for additional information (RAI) related to the license amendment application at issue in this proceeding.

By this letter, Northeast Nuclear Energy Company (NNECO) is now providing the Licensing Board with a copy of the company's response to the Staff's RAI addressed in the Board Notification. NNECO's response is potentially relevant and material to the issues in this proceeding. The NNECO response was dated May 5, 2000 and a copy was sent directly to the Intervenors.

Sincerely,

David A. Repka

David A. Repka
Counsel for Northeast Nuclear Energy
Company

Enclosure
cc: Service List

Template = SECY-043

SECY-02

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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

RPNecci

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

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(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.

Docket No. 50-423
B18025

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

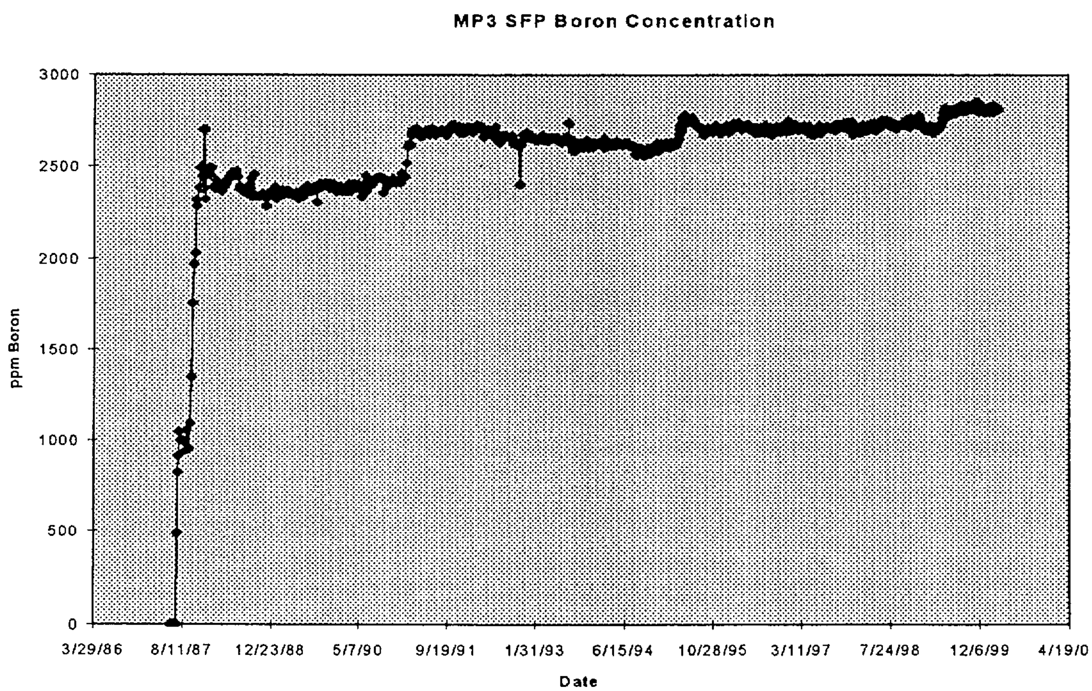
1. **What will be the minimum and maximum boron concentrations in the spent fuel pool specified by chemical procedures if your submitted amendment is approved?**

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:



(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 $\geq$ 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.