Docket No. 50-423
B13641

Attachment 1

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specifications

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5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water.

b. A nominal 10.35-inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Fuel assemblies stored in Region I of the spent fuel pool may have a maximum nominal fuel enrichment of up to 5.0 weight percent U_{235}. Region I is designed to permit storage of fuel in a 3-out-of-4 array with the 4th storage location blocked as shown in Figure 3.9-2.

d. Fuel assemblies stored in Region II of the spent fuel pool may have a maximum nominal fuel enrichment of up to 5.0 weight percent, conditional upon compliance with Figure 3.9-1 to ensure that the design burnup of the fuel has been sustained.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.5.3 The spent fuel storage pool contains 756 storage locations of which a maximum of 100 locations will be blocked.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

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Attachment 2

Millstone Nuclear Power Station, Unit No. 3
Description and Evaluation of Proposed Change
to Technical Specification

November 1990

Millstone Nuclear Power Station, Unit No. 3
Description and Evaluation of
Proposed Change to Technical Specifications

Description of Change

Section 5.6, Fuel Storage

Design Feature Section 5.6.1.1.e has been deleted. This section specified that actual plant analysis was performed on a Cycle 3 specific basis due to considerations on the spent fuel pool cooling and piping systems and pool structure. The beyond Cycle 3 thermal hydraulic analysis has now been completed on these structures and components for spent fuel with a maximum enrichment of 5.05 weight percent extrapolated out to the end of life and consequently this restriction is no longer necessary.

Evaluation

A. Spent Fuel Pool Bulk Temperature Analysis

The spent fuel pool bulk temperature was calculated to account for spent fuel with a maximum enrichment of 5.05 weight percent U-235 with the number of stored fuel assemblies in the spent fuel pool increased to the projected final capacity. In addition, the decay heat load program has modeled the actual fuel bundle transfer rate from the reactor (3 bundles/hr) and has revised the time to the start of fuel movement into the pool based upon refueling outage experience.

The decay heat loads were calculated for a number of pool operating conditions at the end of pool life. These are:

1. Normal refueling--one refueling load core (1/2 core) transferred from the reactor vessel to the pool starting 11 days after shutdown.

2. Normal plant operation--one refueling load (1/2 core) in the pool after 25 days of decay time. This conservatively models the decay heat load in the spent fuel during normal plant operation (reactor cooling system operating, turbine on line).

3. End of cycle full core offload--one full core offload after one year of power operation. Fuel transfer from the reactor starts 11 days after shutdown.

4. Emergency full core offload--a full core offload commencing 36 days after the start of the previously refueling. This corresponds to an "abnormal maximum heat load" as defined in Standard Review Plan (SRP) 9.1.3 and is used as the decay heat load for

all full core offloads occurring less than one year after a previous refueling shutdown.

The decay heat load for the older spent fuel in the pool was modeled as one refueling batch one year old, one refueling batch at 3 years old and the remaining refueling batches at 18-month intervals for cases 1, 2, and 3 above. For case 4, the older spent fuel was modeled as one refueling batch 18 months old and the remaining refueling batches at 18-month intervals over the life of the pool. The analysis is conservative because the present fuel plan is evaluating up to 24-month operating cycles beginning with Cycle 5. The fuel burnup rates used were consistent with 24-month operating cycles which adds more conservatism to the decay heat load.

A summary of the pool temperatures using these heat loads under a variety of normal and accident conditions is found in Table 1.

It should be noted that for Millstone Unit No. 3, a full core discharge is considered an emergency event with respect to the design basis analyses. The current Millstone Unit No. 3 analyses limit the allowed number of full core discharges to (6) for the (40) year life of the plant. Millstone Unit No. 3 has discharged the full core after both Cycle 1 and 2 operations and is currently planning on performing a full core discharge after Cycle 3. These full core discharges have been performed to provide greater flexibility during the refueling outages. If NNECO decides to continue this practice as a routine event beyond Cycle 3, then a reanalysis will be performed to classify this as a normal routine evolution prior to exceeding the current (6) full core discharge limitation.

Table 1
Millstone Unit No. 3 End of Life
Spent Fuel Pool Temperatures
Results from Analysis (5.05% Enrichment)

Steady State Operation

Plant Condition	Heat Load (BTU/hr.)	Maximum Pool Temp	Cooling Water Temperature
Normal Refueling (1/2 Core Offload)	21.77×10^6	129.08°F	95°F
Normal Operating	18.59×10^6	124.16°F	95°F
End of Cycle Full Core Offload (more than 1 yr. of operation)	34.79×10^6	149.46°F	95°F
"Emergency" Refueling (1/2 Core + Full Core) Less than 1 yr. of operation.	35.05×10^6	149.86°F	95°F
Loss of Fuel Pool Cooling. Peak Temperatures Following a Single Failure			
Normal Refueling	21.77×10^6	140°F(1)	95°F
Normal Operating	18.59×10^6	140°F(2)	95°F
End of Cycle Full Core Off-Load	34.79×10^6	154.5°F(3)	95°F
Loss of Power Concurrent with a Design Basis Accident(4)			
Normal Operating	18.59×10^6	145°F	95°F

(1) Loss of pool cooling for approx. 1.9 hrs. with pool at initial temperature of approximately 129°F.
(2) Loss of pool cooling for 4 hours with pool at initial temperature of approximately 125°F.
(3) Loss of pool cooling for 1/2 hour with pool at initial temperature of approximately 150°F.
(4) LOP with CDA, or loss of power following safety injection and recirculation mode. See FSAR Table 8.3-1. Pool cooling lost for four hours.

B. Summary of Analysis

All analyses were performed assuming only one of two trains of spent fuel pool cooling is in operation.

For all normal refuelings, as defined in SRP 9.1.3, the spent fuel pool temperature is maintained below the SRP limit of 140°F. In the event of a single active failure, the alternative train of cooling can be initiated in sufficient time to keep the pool temperature below 140°F.

For an emergency full core offload, the spent fuel pool bulk pool temperature is maintained below 150°F which is the maximum long-term temperature limit per FSAR Section 9.1.3, Table 9.1-2. This limit is based on the limitations of the concrete structure at high temperatures. The predicted temperature is also below the no boiling criteria required by SRP 9.1.3. Single failures are not required to be assumed during this event per Standard Review Plan 9.1.3

During an end of cycle full core offload, the spent fuel pool bulk temperature is also maintained below 150°F. This meets the FSAR limits described above. The SRP does not have any temperature requirements for this event. Table 1 contains a postulated single active failure occurring during the end of cycle full core offload. During this event, the spent fuel pool bulk temperature will rise to approximately 155°F until the alternate train of cooling is initiated. With the other train of cooling on line, the pool temperature will decrease to less than 150°F.

During a design basis accident (containment depressurization actuation signal), coincident with a loss of power, cooling water to the spent fuel pool coolers is lost for four hours. During this time, the pool bulk water temperature rises to approximately 145°F and then decreases when cooling is restored. This is within the maximum analyzed spent fuel pool bulk water temperature for the system.

The design temperature of the spent fuel pool cooling system components and piping is 200°F. None of the analyzed scenarios exceed this design temperature.

For purposes of the concrete and spent fuel pool liner design only, the spent fuel pool cooling is assumed to be lost during both a normal and emergency full core offload. The pool temperature is assumed to increase until it reaches 200°F. At 200°F, pool cooling is reinitiated and the cooldown time to 150°F was determined. This time-temperature transient is used in the qualification of the concrete and spent fuel pool liner. While there are no credible failure modes which will produce this transient, it conservatively envelopes all

credible failures. The impact of these transients is addressed in part D, below.

C. SFP Cooling System and Piping

NNECO has reviewed the impact of this proposed license amendment on the SFP Cooling System and associated piping and has concluded that this change does not involve a significant hazards consideration.

The spent fuel pool cooling system design temperatures and the system piping stress temperature limits will not be exceeded under any operating scenarios. As such, this change does not increase the probability of occurrence or consequences of any accident previously evaluated.

The operation of equipment in the spent fuel pool cooling system has not been modified in any way by this change, and as such, it does not create the possibility for a new or different kind of accident than previously evaluated.

The maximum spent fuel pool fluid temperature predicted under any scenario is still significantly below the spent fuel pool cooling design temperature of 200°F. As such, it does not reduce the margin of safety.

D. SFP Structure Evaluation

The spent fuel pool is located in the fuel building at Millstone Unit No. 3. The building, approximately 112 feet by 92.5 feet, is supported on compacted fill and/or rock. The fuel building has a ground floor at grade elevation and a basement 13 feet below grade. The spent fuel pool portion of the building is reinforced concrete construction from approximately 24 feet below to 28 feet above grade. The spent fuel areas are protected from tornado missiles by a reinforced concrete superstructure. The spent fuel pool is "L" shaped, with the bottom of the pool approximately 13 feet below grade. The floor is 8-foot thick reinforced concrete. The walls are 6-foot thick reinforced concrete. The walls and floor of the pool are lined with 1/4-inch thick stainless steel plate.

The spent fuel pool was designed for the loads and load combinations specified in the "Millstone Nuclear Power Station Unit 3 Final Safety Analysis Report," Section 3.8.4. The loading analysis was performed in two parts. The first part was a mechanical load analysis. The mechanical loads included dead loads, live loads, hydrostatic loads, hydrodynamic loads, and seismic loads. The second part was a thermal load analysis. The thermal loads included normal operating and accident loads. The thermal and mechanical load analyses were combined

and the design analysis was performed. The original analysis assumed the storage of 3.85 w/o U-235 fuel in the pool.

A calculation entitled "MP3 SFP Analysis for Temperature Profiles of 5.05 Enrichment Fuel" was prepared to address the effect of the change of fuel enrichment from 3.85 weight percent U-235 to a maximum fuel enrichment level of 5.05 weight percent U-235 on the spent fuel storage pool. The existing mechanical load analysis was determined not to be affected due to the fact that the total weight of the fuel assembly is the same regardless of fuel enrichment levels. The change from 3.85 weight percent U-235 to a maximum fuel enrichment level of 5.05 weight percent U-235 does result in revised normal operating and accident thermal loads. The above referenced calculation addressed both changes in steady state and transient temperature curves resulting from the change of fuel enrichment. The results of the calculation are summarized as follows:

- Both the maximum normal refueling and full core offload steady state temperature effects on the concrete structural elements for 5.05 weight percent U-235 fuel are less severe than that originally analyzed for effects of 3.85 weight percent U-235 fuel temperature transients, considering either normal heat load with loss of cooling or normal heat load, maximum To, with loss of cooling.

- The temperature transient cases associated with the 5.05 weight percent U-235 fuel are time dependent functions which happen at a quicker rate than the originally analyzed for 3.85 weight percent U-235 fuel transients. Because of this, the new transients impose less of a thermal gradient on the concrete structural elements and result in less concrete stresses and strains.

The original concrete analyses for 3.85 weight percent U-235 fuel bound the effects that the 5.05 weight percent U-235 fuel would impose on the structure.

The above calculation also determined that the existing liner design is still valid for 5.05 weight percent fuel because of the following:

- The mechanical load analysis, as stated previously, remains the same.

- The thermal load analysis for the 5.05 weight percent U-235 fuel is bounded by the 3.85 weight percent U-235 fuel analysis. This is due to the fact that the liner analysis is governed by the maximum temperature that is applied. The maximum temperature that the liner is subjected to is the same for either fuel enrichment.

In conclusion, the storage of 5.05 weight percent $U\text{-}23^5$ fuel will not adversely affect the spent fuel pool concrete and liner system and, based on the above, does not involve a significant hazards consideration since it does not:

(1) Significantly increase the probability of occurrence or consequences of an accident previously evaluated; or,

(2) create the possibility for a new or different kind of accident from any evaluated; or

(3) Significantly reduce the margin of safety

E. Conclusion

NNECO has reviewed the proposed change in accordance with 10CFR50.92 and has concluded that it does not involve a Significant Hazards Consideration because this change will not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed. The proposed change qualifies the SFP Cooling System, piping and pool structure out to end of pool life utilizing the original design basis acceptance criteria. Since SFP temperatures remain as before, there is no adverse impact on the results of any previously analyzed accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed accident. Since there are no changes in the way the plant is operated or in the operation of the equipment credited in the design basis accidents, the potential for an unanalyzed accident is not created. Also, no new failure modes are introduced.

3. Involve a significant reduction in the margin of safety. The proposed change qualifies the SFP cooling system, piping and pool structure through the end of the pool life. Since there is no impact on the consequences of any accident previously analyzed, there is no reduction in the margin of safety.