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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 99 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1. INTRODUCTION

In Reference 1, Northeast Nuclear Energy Company (NNECO) submitted a license amendment request and the preliminary Reload Safety Analyses (RSA) in support of the Millstone Unit No. 2, Cycle 7 reload. The final Reload Safety Analysis was provided in Reference 2. As indicated in these submittals, the bases on which the Cycle 7 reload was analyzed were documented in a "Basic Safety Report" (BSR) (Ref. 3), and in the Cycle 6 Reload Safety Analysis (Reference 4). The BSR, as supplemented by Reference 5 serves as the reference fuel assembly and safety analysis report for the use of Westinghouse fuel at Millstone 2 (a Combustion Engineering plant). Reference 6 documents the NRC staff's review and acceptance of the BSR. The analysis and evaluation of the reload was accomplished using the methodology of Reference 7. This methodology was approved in Reference 8.

In Reference 1, NNECO informed the Staff that due to the elevated levels of radioactive iodine and other fission products identified during Cycle 6 operation, NNECO anticipated the discovery of a number of fuel assemblies with leaking fuel rods during the refueling outage for Cycle 7.

Since that time, NNECO performed fuel sipping identifying 16 fuel assemblies with failed fuel rods. In addition, visual examination revealed several fuel assemblies to have broken holddown springs. NNECO is replacing all leaking fuel assemblies with a combination of reconstituted and previously discharged fuel assemblies. These changes have necessitated a revised loading pattern (Reference 2) for Cycle 7 operation.

1.1 General Description

The Millstone 2 reactor core is comprised of 217 fuel assemblies. Each fuel assembly has a skeletal structure consisting of five (5) Zircaloy guide thimble tubes, nine (9) Inconel grids, a stainless steel bottom nozzle, and a stainless steel top nozzle. One hundred seventy-six fuel rods are arranged in the grids to form a 14x14 array. The fuel rods consist of slightly enriched uranium dioxide ceramic pellets contained in Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel.

Nominal core design parameters utilized for Cycle 7 are as follows:

Core Power (MWt)	2,700
System Pressure (psia)	2,250
Reactor Coolant Flow (GPM)	350,000
Core Inlet Temperature (°F)	549
Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stack height of 136.4 inches)	6.065

The feed fuel for the Millstone 2, Cycle 7 core consists of twenty-four (24) zoned-enrichment interior feed assemblies, each containing sixty (60) fuel rods at 2.62 w/o U235 and one-hundred sixteen (116) fuel rods at 2.91 w/o U235, and forty-eight (48) zoned-enrichment peripheral assemblies, each containing sixty (60) fuel rods at 2.91 w/o U235 and one-hundred sixteen (116) fuel rods at 3.29 w/o U235. The zoned-enrichment assembly configuration contains 12 lower enrichment fuel pins around each of the five control rod water holes. The feed fuel will replace twenty (20) Combustion Engineering (CE) Batch A assemblies, one (1) CE Batch B assembly, and fifty-one (51) Westinghouse Batch F assemblies. An additional five (5) Westinghouse Batch F assemblies will be discharged from the end of Cycle 6, and will be replaced by five (5) Westinghouse Batch F assemblies which were removed from the core

at the end of Cycle 5. Due to fuel defects in Cycle 6, and subsequent symmetry considerations, fourteen (14) Westinghouse Batch G assemblies, seven (7) Westinghouse Batch F assemblies (these Batch F and G assemblies were removed from the core at the end of Cycle 5), and four (4) CE Batch A assemblies (discharged at the end of Cycle 1) are needed as well. As a result of fuel reconstitution, the fuel rods from seven (7) Westinghouse reload assemblies to be used in Cycle 7 have been placed in Combustion Engineering (CE) skeletons. Also, twenty-one (21) fuel rods have been replaced with stainless steel rods in Cycle 7. The twenty-one stainless steel rods are distributed among eleven (11) fuel assemblies, with the number of stainless steel rods in each of these assemblies ranging from one to five. A summary of the Cycle 7 fuel inventory is given in Table 1.

TABLE 1
Millstone Unit 2 Cycle 7
Core Loading

Region	Type	Number of Assemblies	Initial Enrichment w/oU235	%Theoretical Density	BOC** Burnup Average (MWD/MTU)
A	CE	4	1.93	95.0	15960
F1	W	4	2.70	94.5	25200
F2	W	5	3.30	94.9	22200
F2	W*	3	3.30	94.9	21560
G1	W	19	2.72	95.0	23470
G2	W	32	3.19	94.7	19290
G2	W*	4	3.19	94.7	9970
H1	W	30	2.73	95.2	13790
H2	W	44	3.22	94.8	9560
J1	W	24	2.62/2.91	95.2/95.1	0
J2	W	48	2.91/3.29	95.1/95.2	0

*Westinghouse fuel reassembled using CE skeletons.

**EOL Cycle 6 burnup assumed: 11,500 MWD/MTU.

2. FUEL SYSTEM DESIGN

The fuel system design for Millstone Unit 2, Cycle 6 is the same as that approved (Ref. 6) for Cycles 4, 5, and 6. That is, approval of the BSR constituted approval of the use of a mixed core of Combustion Engineering and Westinghouse fabricated fuel assemblies. The replacement of CE fuel with Westinghouse fuel at each reloading would eventually lead to a core with all Westinghouse fuel.

As described in Reference 2, the reload redesign utilizes a combination of reconstituted and previously discharged fuel assemblies to replace leaking fuel assemblies. Since this redesign uses previously approved fuel assembly types, and since the redesign and the reinserted CE assemblies will not receive greater than design exposure, the redesign is acceptable from the fuel system point of view.

At the end of Cycle 5, NNECO identified broken holddown springs on 15 fuel assemblies. Initial plans were to effect replacement of the broken holddown springs. The procedure developed was utilized successfully on one fuel assembly. However, NNECO decided that the irradiated fuel repair procedure involved a high risk with the potential for damaging fuel assemblies, particularly fuel pins, during the repair.

NNECO therefore reached the conclusion and provided supporting analysis (Ref. 10) that operation of Cycle 6 with 9 fuel assemblies, each with a single broken holddown spring, was acceptable and prudent. The analysis provided by NNECO characterizes the breaks to the holddown springs, provides justification that the breaks were caused by excessive vibratory motion during reactor operation, discusses fretting wear, loose parts, control rod jamming and the probability of multiple fractures, and concludes that operation of Cycle 6 with the 9 assemblies having broken holddown springs would be acceptable. This is primarily because the number of active turns of the springs is only slightly decreased by the types of breaks observed. Future new fuel would have newly designed springs. We found this acceptable.

Nine assemblies identified to have broken holddown springs at the end of Cycle 6 will be reloaded for Cycle 7 operation. In addition 4 assemblies from Cycle 5 needed to provide symmetry in the loading pattern and which have broken holddown springs will be utilized. We find this acceptable based upon the finding for Cycle 6 and the lack of any problems observed in operation of Cycle 6 with 9 fuel assemblies having broken holddown springs. No broken holddown springs were identified in Batch H fuel at the end of Cycle 6 operation. Batch H fuel had a new top nozzle design intended to eliminate the problem, which has proven to be the case for the one cycle of exposure of the 74 assemblies in Batch H.

3. NUCLEAR DESIGN

The nuclear design procedures and models used for the analysis of the Millstone Unit 2 Cycle 7 reload core (References 1 and 2) are the same as those used for Cycle 6. These are documented in the Millstone Unit 2 Basic Safety Report (BSR), (Reference 3) and have been approved (Reference 6) for the analysis of the Millstone Unit 2 core using Westinghouse reload fuel beginning with Cycle 4. In addition, the methods described in Reference 7 document the methodology used by Westinghouse for performing this as well as other reloads. This methodology was approved in Reference 8.

The physics analysis of the reload specifically included the zoned-enrichment fuel assemblies, the 21 stainless steel rods in reconstituted fuel assemblies, and the loading pattern of the various fuel types described in Section 1.1 above, in order to determine maximum linear heat rates achievable in normal operation, control rod worths for the shutdown margin evaluation, and the Cycle 7 kinetics characteristics for use in the accident evaluation. Also included in the analysis is substitution of full strength control rods for part strength control rods in the lead control element assembly (CEA) bank. This hardware change was implemented during the refueling outage. Because these calculations were performed with approved methods, they are acceptable.

In Reference 2, Table 2, the kinetics parameters for the Cycle 7 reload redesign are given. These are all within the current limits with a small exception in the

least negative above 30% power Doppler temperature coefficient and in the maximum delayed neutron fraction. Both of these parameters had the same values in Cycle 6. The conclusion there was that no reanalysis was necessary because the potential effects were small. This was found acceptable for Cycle 6 and continues to be acceptable for Cycle 7. Two accidents were reanalyzed for other reasons, and are discussed under Accident Analysis, Section 5.

The control rod worths and shutdown requirements for the Cycle 7 design are presented in Table 3 of Reference 2 and compared with previous Cycle 6 values. At EOC 7, the reactivity worth with all control rods inserted assuming the highest worth rod is stuck out of the core is 6.26% and assuming a 10% reduction to allow for uncertainty. The reactivity worth required for shutdown, including the contribution required to control the steamline break event at EOC 7, is 5.89%. Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline break at the worst time in core life allowing for the most reactive control rod stuck in the fully withdrawn position and also allowing for calculational uncertainties. We have reviewed the calculated control rod worths and the uncertainties in these worths based upon comparison of calculations with experiments presented in the BSR and in previous Westinghouse reports. On the basis of our review, we conclude that the NNECo's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive control rod is stuck in the fully withdrawn position.

4. THERMAL-HYDRAULIC DESIGN

Millstone 2 Cycle 7 utilized the Basic Safety Report (Ref. 3) which was approved by the staff in Reference 6. The Basic Safety Report was also used as the basis for Cycle 4, 5, and 6 operation.

As discussed in the BSR, the Westinghouse fuel assemblies have been designed and shown through testing to be hydraulically compatible with all resident Millstone 2 fuel assemblies. The stainless steel rods in the reconstituted fuel assemblies were treated as heated rods in the THINC DNB analysis. This is conservative since it results in higher subchannel enthalpy predictions.

No significant variations in thermal margins result from the Cycle 7 reload. The Cycle 7 analysis takes a partial credit of 3.0% of the net conservatism which exists between convoluting and summing the uncertainties of various measured plant power parameters in terms of power. This partial credit was applied in previous cycles and its approval is discussed in more detail in the Cycle 4 Reload Safety Evaluation Report (Ref. 9); therefore, we find operation of Cycle 7 acceptable.

5. ACCIDENT ANALYSIS

As a result of the change to full strength CEAs in the lead CEA bank, the value of the ejected rod worth for the HFP ejected rod accident for Cycle 7 increased to 0.28% $\Delta k/k$. The licensee therefore provided a reanalysis of this event. The results show that the energy deposition increased from 171 cal/gm for the reference analysis to 185 cal/gm for the Cycle 7 analysis. This is below the criterion of 200 cal/gm established as a limit for this accident in the BSR, and is therefore acceptable.

The split enrichment fuel assembly design flattens the power peaking by placing slightly lower enrichment fuel pins around the large water holes in the fuel assembly. In order to assess the effect of this flattening on a limiting DNB event, the loss of flow accident for Cycle 7 was reanalyzed. The results show the MDNBR to be 1.30, which is acceptable.

In Reference 11, the licensee provided a reanalysis of the small break LOCA. This was done because there was an inconsistency between the Technical Specification requirement on axial shape index (ASI) and the ASI assumptions used in the approved small break LOCA analysis. The inconsistency was documented in Millstone Unit 2 Licensee Event Report 85-001-0.

The most negative ASI input to the approved analysis was an ASI of 0.14. The Technical Specifications allow the 100% power ASI to be no more negative than -0.10. If the 0.06 ASI uncertainty is introduced into the analysis, the most negative upper bound to ASI becomes -0.16, which is inconsistent with the -0.14 ASI input to the small break LOCA model. Reference 11 provides the results of a small break LOCA analysis which allowed the ASI value to be -0.16. The calculated peak clad temperature increased from 1971°F to 2035°F. This is below the acceptance criterion of 2200°F for the small break LOCA, and is therefore acceptable.

6. TECHNICAL SPECIFICATIONS

Technical Specification changes proposed by the licensee in Reference 1 and as clarified in reference 12 are acceptable as follows:

The main Technical Specification change proposed by the licensee trades range in radial peaking for more range in axial shape index (ASI). For monitoring of the power distribution with excore detectors, the maximum radial peak is specified in the Technical Specifications by a limit on the total planar radial peaking factor, F_{xy}. The maximum axial peak is specified by limits on the ASI. The product of the radial and axial peaks is the core peaking factor, which is proportional to the maximum peak linear heat rate. The maximum allowable peak linear heat rate in turn is limited as a result of the LOCA analysis to 15.6 kw/ft in normal operation of the powerplant. A decrease in the allowable value of F_{xy} can be offset with an increase in the allowable value of ASI without changing the limiting achievable linear heat rate.

The current Millstone Unit 2 Technical Specifications define an allowable ASI envelope for $F_{xy} \leq 1.791$. Also defined is a power derate curve if the F_{xy} limit cannot be met. The proposed Technical Specification change defines an expanded allowable ASI envelope and an appropriate power derate envelope if $F_{xy} \leq 1.62$. The licensee has indicated in Reference 1 that an analysis was performed to verify that the current envelope is unaffected by the hardware change in the

lead CEA bank. The licensee further indicated that an additional analysis was performed to verify this for the new derate envelope for $F_{xy} \leq 1.62$, and that the appropriate ASI envelope for this F_{xy} was calculated with the approved methods of Reference 3.

The licensee also proposed to delete the indexing parameters M and N. These parameters specify allowable power levels when less than all reactor coolant pumps are used and when excore detectors are used for monitoring and the F_{xy} limit is exceeded. Reference 12 contained a page which was inadvertently left out of the Reference 1 submittal. This page deleted reference to the indexing parameters M and N. These parameters had previously been deleted on another page submitted with Reference 1. An additional technical specification page was included with Reference 12 which corrected a typo contained in the Reference 1 submittal. The way the revised Specifications have been written makes the change administrative, and it is therefore acceptable.

Since the proposed Technical Specification changes were calculated and evaluated with approved methods, and since they do not alter the maximum peak linear heat rate achievable in normal operation of the powerplant, the changes are acceptable.

7. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8. CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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8. REFERENCES

1. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), with "Cycle Refueling-Preliminary Reload Safety Analysis," February 6, 1985.

2. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), with "Cycle 7 Refueling-Reload Safety Analysis," June 5, 1985.

3. "Basic Safety Report," Westinghouse proprietary report for Millstone Unit 2, Docket Number 50-336, submitted via letter, W. G. Counsil (NU) to R. Reid (NRC), March 6, 1980.

4. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), November 17, 1983.

5. W. G. Counsil (NNECO), letter to R. A. Clark, November 17, 1981.

6. L. S. Rubenstein (NRC), memorandum for T. M. Novak, "SER Input on Millstone Unit 2 BSR," February 16, 1982.

7. Bordelon, F. M. et.al., "Westinghouse Reload Safety Methodology", WCAP-9272, March 1978.

8. C. O. Thomas (NRC), letter to E. P. Rahe, Jr. (W), "Acceptance for Referencing of Licensing Topical Report WCAP-9272(P)/9273(NP)", May 28, 1985.

9. W. G. Counsil (NNECO), letter to R. A. Clark, June 3, 1980.

10. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), December 1, 1983.

11. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), April 11, 1985.

12. W. G. Counsil (NNECO), letter #B11569 to J. R. Miller (NRC), "Proposed Revisions to Technical Specifications", June 11, 1985.