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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING LICENSE AMENDMENT NO. 73 TO PROVISIONAL OPERATING

LICENSE NO. DPR-21

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-245

1.0 INTRODUCTION AND DISCUSSION

By letter dated September 9, 1980 (Reference 1) Northeast Nuclear Energy Company (NNECo) proposed Technical Specification changes to allow core reload with 128 General Electric fuel assemblies containing pressurized fuel rods with 2.82% enriched uranium oxide (UO_2) and 40 General Electric fuel assemblies containing pressurized fuel rods with 2.65% enriched UO_2 and burnable poison. All 168 new assemblies, replace older depleted fuel assemblies and contain the fuel rods in an 8X8 matrix. The assemblies have drilled lower tie plates and finger springs to regulate by pass flow and have been used in other Boiling Water Reactors. This makes a total of 316 fuel assemblies with drilled lower tie plates and finger springs for Reload 7 compared to 148 for Reload 6. NNECo representatives explained during a telephone conversation with NRC representatives that the increased uranium enrichment would extend the rated power production capability between core reloadings from the present period of 12 months to 18 months to match the main turbine overhaul schedule. The increased fuel enrichment provides a sufficient increase in fissile uranium (U_{235}) to achieve this. At this time NNECo does not intend to produce more power per bundle, measured in megawatt days per metric ton of uranium (MWd/MTU) and, therefore, the inventory of heavy radioactive metals accumulated in the core during power production is not changed significantly.

Additional information was provided by NNECo letter dated February 25, 1981 in response to various NRC questions related to the September 9, 1980 Reload 7 NNECo submittal. An earlier request by NNECo, dated April 2, 1980, to extend Maximum Average Planar Linear Heat Generation (MAPLHGR) limits beyond 30,000 MWd/t average planar exposure was superseded by the revised MAPLHGR limits of the September 9, 1980 NNECo submittal.

NNECo letter dated September 10, 1980, submitted a General Electric Report entitled, "Millstone 1 Segmented Test Rod Bundle - Supplement 5" NEDO-20592-5P as an Appendix to the Reload No. 7 request.

By letter dated September 30, 1980, NNECo proposed Technical Specification changes related to torus water volume limits.

By letter dated November 6, 1980, NNECo requested Technical Specification changes related to the Standby Liquid Control System (SLCS) boron concentration and minimum sub critical reactor (shutdown) requirements.

The staff has not completed the evaluation of all of the proposed Technical Specification changes in the above NNECo submittals. Evaluation of the remaining requests is continuing and the results of that review will be addressed in a safety evaluation that is expected to be completed shortly.

2.0 EVALUATION

2.1 Minimum Critical Power Ratio (MCPR)

2.1.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 5, the MCPR which may be allowed to result from core-wide or localized transients is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid boiling transition. There has been no change in the safety limit MCPR for Millstone-1 from Cycle 7 to Cycle 8.

2.1.2 Operating Limit MCPR (OLMCPR)

Various transients could reduce the MCPR below the intended safety limit MCPR during Cycle 8 operation. The most limiting of these operational transients have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR).

The transients evaluated were the limiting pressure and power increase transient (in this case, the load rejection transient without turbine bypass to the main condenser), the limiting coolant temperature decrease transient (loss of feedwater heater), the feedwater controller failure transient, the control rod withdrawal error transient and the fuel loading error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 2 were assumed.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Results for 100% power/100% flow core conditions were as follows:

Transient	Exposure	ϕ (% NBR)	Q/A (%NBR)	P_{SL} (psig)	P_v (psig)	ΔCPR 8x8	8x8R	P8x8R
Load Rejection w/o Bypass	EOC8	305	116	1193	1226	0.30	0.30	0.32
	EOC8-1800 MWd/t	252	111	1183	1217	0.22	0.22	0.24
Loss of 100^{o}F Heater Feedwater	--	118	117	1033	1072	0.15	0.15	0.15
Feedwater Controller Failure	--	111	106	1033	1072	0.06	0.07	0.07
Rod Withdrawal Error @ 107% RBM Set Point						0.24	0.14	0.14
Fuel Loading Error Rotated Bundle								0.19

ϕ Neutron Flux

Q/A Surface Heat Flux

NBR Nuclear Boiler Rating

P_{sl} Steam line pressure

P_v Reactor Vessel Pressure

ΔCPR Change in Critical Power Ratio

Addition of the most severe CPR to the safety limit CPR (1.07) gives the appropriate operating limit for each fuel type. This will assure that the safety limit MCPR is not violated due to transients. Using the above table, the licensee has proposed the following operating limits:

Fuel Type	OLMCPR EOC8-1800 MWd/t to EOC8	BOC8 to EOC8-1800 MWd/t
8x8	1.37	1.31
8x8R	1.37	1.29
P8x8R	1.39	1.31

Operating limit for the same fuel types (8x8 and 8x8R) for cycle 8 is 1.37 compared to 1.34 for cycle 7.

Based on comparison of the void coefficient (less negative in Cycle 8) and scram worth (more negative in Cycle 8) with Cycle 7 analyses, we would expect the OLMCPR to be lower. However, changes in the axial power distribution for Cycle 8 cause a delay in scram effectiveness and more than offset the changes in void coefficient and scram worth.

Since the higher OLMCPR obtained from the analyses is more conservative and since these limits will preclude violation of the safety limit MCPR in the event of any anticipated operational occurrence, we find these limits to be acceptable.

Millstone-1 transient analyses were performed with the REDY code. All future reload analyses are to be performed with the ODYN transient analysis code. Generic BWR/3 transient analyses reported in Reference 7 show the REDY analysis to be conservative relative to the ODYN analysis for the limiting transients; therefore, the REDY analyses are acceptable for use without penalty until the change to ODYN analyses can be implemented.

2.2 End-of-Cycle Power Coastdown

The licensee desires to coast down (Reference 8) to 70% rated power after the end of the normal cycle. The licensee is applying a generic approach for the power coast down analysis described in Section 5 of Reference 5. This approach concluded that the coastdown operation beyond full power operation is conservatively bounded by the analysis at the end of Cycle 8 conditions. Therefore, we conclude that operation in the proposed coastdown mode is acceptable.

2.3 Fuel Loading Error

The licensee has analyzed the rotated bundle loading error event and the mislocated bundle loading error event for the Reload No. 7 core based on the new analysis procedure described in Reference 6. Analyses show that the worst-case fuel loading error for a rotated bundle results in a MCPR of 1.26. Since this MCPR is greater than the safety limit of 1.07, we find this analysis to be acceptable.

2.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 5. As specified in Appendix C of Reference 5, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is acceptable.

2.5 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 2) show that the channel hydrodynamic and reactor core decay ratios at the natural circulation and 105% rod line intersection (which is the least stable point of operation) are below the stability limit. Decay ratio for Cycle 8 was 0.61 as compared to 0.70 for Cycle 7. Because the operation in the natural circulation mode will be prohibited by the Technical Specifications, there will be added margin to the stability limit and this is acceptable to the staff.

2.6 MAPLHGR Limits

The licensee has proposed to replace or renumber the current Technical Specification curves (Figures 3.11.1.a through 3.11.1.h) with new curves (Figures 3.11.1.a through 3.11.1.g) for the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size based on the analyses (Reference 3) done by the previously approved methods for all the fuel types in the core. These MAPLHGR curves have been extended from 30,000 MWd/t to 40,000 MWd/t. Although the methodology used is generally applicable for the extension of these curves from 30,000 MWd/t to 40,000 MWd/t, the staff believes the effects of enhanced fission gas release in high burnup fuel (above 30,000 MWd/t) are not adequately accounted for in References 2, 3 and 4. To compensate for this deficiency, the staff has estimated the amount the MAPLHGR Limits in Figures 3.11-1.a to 3.11-1.g of the proposed Technical Specifications should be reduced. The reduction imposed is based on the results of comparative calculations of fuel volume average temperature performed by General Electric using GEGAP III with and without an NRC correction for enhanced fission gas release and the relationship between peak cladding temperature and MAPLHGR increase presented in Reference 10. In estimating the MAPLHGR reduction, the staff conservatively assumed that the change in volume average temperature can be translated directly into a peak cladding temperature change. Table 1 gives the percent reduction in MAPLHGR as a function of exposure above 30,000 MWd/t for all the fuel types for Millstone-1 Cycle 8 in References 2 and 3. We have limited the extension of the MAPLHGR to 36,000 MWd/t to account for the uncertainties in enhanced fission gas release above this exposure.

TABLE 1 - REDUCTION IN MAPLHGR AS A FUNCTION OF EXPOSURE

Exposure MWd/t	30,000	32,000	34,000	36,000
Reduction MAPLHGR, %	9.0	12.5	16.25	20.5

These MAPLHGR reductions to the licensee's proposed Technical Specifications in Figures 3.11-1.1.a to 3.11-1.g assure that the cladding temperature and local cladding oxidation would remain below the 2200°F (peak cladding temperature) and 17 percent (local cladding oxidation) limit allowed by 10 CFR 50.46 when the effects of enhanced fission gas release above 30,000 MWd/t are conservatively accounted for.

The licensee has agreed to the reductions in MAPLHGR Limits given in Table 1 and to the 36,000 MWd/t exposure limit as per discussions on February 9, 1981. These limitations are applicable for all the fuel types for Millstone-1 for Cycle 8 and for the Subsequent cycles until additional anslysis is provided with approved GESTER Models.

Another area having safety implications which requires consideration is the 1 percent plastic strain criterion of the Zircaloy fuel rod cladding as the safety limit below which fuel damage due to overstraining is not expected to occur. At extended exposures (i.e., beyond 30,000 MWd/t peak pellet exposure) this safety limit has not been calculated. However, the probability of a high exposure bundle achieving power levels that would challenge the 1 percent plastic strain limit is extremely small, based on analysis performed in accordance with the approved methodology of NEDO-20566, "Loss-of-Coolant Accident Analysis Generic Report."

Tables S-3 and S-4 of 10 CFR 51.20 are based on an average fuel burnup of 33,000 MWd/t for irradiated fuel from the reactor. Therefore, even though this amendment establishes MAPLHGR limits for fuel burnup out to 36,000 MWd/t, the average burnup level of fuel from the reactor should not exceed 33,000 MWd/t.

Accordingly, we find the proposed MAPLHGR versus average planar exposure values acceptable when modified as stated.

2.7 Standby Liquid Control System

The current Technical Specifications for the Millstone Unit 1 reactor require that the Standby Liquid Control System (SLCS) be capable of bringing the core to shutdown in the cold xenon-free state. Traditionally, a calculated shutdown margin of 3% Δk has been used to account for various uncertainties and assure that shutdown could be accomplished. Recently, new calculation procedures and more precise definition of uncertainties have led to the conclusion that an adequate calculated shutdown margin is 2.6% Δk.

Currently, the SLCS is required to be capable of producing a boron concentration in the core coolant of 600 parts per million. The calculated value of the shutdown margin for Cycle 8 of Millstone 1 is 2.88% Δk for this concentration. This margin meets the requirement. However, in order to provide extra margin for possible future cycles Northeast Utilities proposes to require the SLCS to be capable of producing a boron concentration of 660 parts per million. At this concentration the calculated shutdown margin is 3.88% Δk.

We have reviewed the submittal of Northeast Utilities and find the changes to the Technical Specifications to be acceptable. This finding is based on the following considerations:

a. The total change is reactivity from hot full power to the cold, xenon-free state is of the order of 15 to 20 percent Δk. Thus, a shutdown margin of 2.6 percent Δk implies a calculational uncertainty of at least 13 percent for the boron worth. This is a conservative value for the uncertainty.

b. The implies cold clean boron worth of 60 parts per million per 1 percent Δk is consistent with similar worths from other reactor calculations.

c. The increase of end point boron concentration to 660 parts per million offers additional margin to safe shutdown.

2.8 Increased Torus Water Volume

The licensee has proposed as part of the Mark I Containment Long Term Program (LTP) modification that the Technical Specification for the torus water volume be changed from a range of 92,000 ft^3 to 94,000 ft^3 to a new range of 98,000 ft^3 to 100,400 ft^3; and that the downcomer submergence be changed to result in a range of submergence from 3.0 ft to 3.33 ft with a 1.0 psia pressure between the drywell and wetwell air space. The previous range of submergence was 4.7 tO 4.9 ft.

The licensee has indicated that the increased water volume in the torus will provide an increased heat sink for both a Loss of Coolant Accident (LOCA) and for Safety Relief Valve (SRV) steam condensation. However, this increased volume will result in a reduced volume for storage of non-condensible gases. The licensee notes that GE Topical Report NEDO-24575 indicates that peak pressures of 42 psig and 32 psig will occur in the drywell and wetwell air spaces, respectively, during a LOCA which are well below the original design pressure of 62 psig. The licensee has also noted that the increased pressure in the wetwell air space tend to slightly increase the available NPSH for the pumps that draw from the torus. It is also pointed out by the licensee that the decreased downcomer submergence reduces the pool swell loads that could be imposed on the torus in the initial stages of a LOCA.

The staff's evaluation report on the Mark I containment LTP, NUREG-0661, addresses the potential for uncovering the downcomers in Section 3.12.1 (seismic slosh assessment). This assessment was performed at the most extreme conditions that could potentially lead to uncovering the dowmcomers and was predicted for a minimum three-foot downcomer submergence.

Seismic motion induces suppression pool waves which can (1) impart an oscillatory pressure loading on the torus shell; and (2) potentially lead to uncovering the ends of the downcomers, which could result in steam bypass of the suppression pool and potential overpressurization of the torus, should the seismic event occur simultaneously with a LOCA. To assess these effects, the Mark I Owners Group undertook the development of an analytical model which would provide plant-specific seismic wave amplitudes and torus wall pressures. This model was based on 1/30-scale "shake test" data for a Mark I torus geometry as reported in GE Report NEDO-21471-2.

Based on the results of plant-specific analyses, using the analytical model, the Mark I Owners Group concluded that (1) the seismic wave pressure loads of any Mark I torus are insignificant in comparison with the other suppression pool dynamic loads; and (2) the seismic wave amplitudes will not lead to uncovering of the downcomers for any Mark I plant. This conclusion was based on the maximum calculated pressure loads and the minimum wave trough depth relative to the depth of the downcomer exit.

We have reviewed comparisons of the analytical predictions with scaled-up test data, the small-scale test program, and the seismic spectrum envelope used in the plant-specific analyses. Based on this review, we conclude that the seismic slosh analytical predictions will provide reasonably conservative estimates of both the wall pressure loading and the wave amplitude, for the entire range of Mark I plant conditions.

Since the maximum local wall pressures were found to be less than 0.8 psi at a 95% upper confidence limit, the Mark I Owners Group has proposed that the seismic slosh loads may be neglected in the structural analysis. We agree that the seismic slosh loads are insignificant in comparison with the other suppression pool dynamic loads. On this basis, we conclude that neglecting seismic slosh loads for the plant-unique analyses is acceptable.

The results of the slosh wave amplitude predictions indicate that, within the local area of maximum amplitude and with maximum suppression pool drawdown (resulting from ECCS system flows), the slosh waves will not cause uncovering of the downcomers exits. We have reviewed the assumptions used in these analyses and conclude that they are sufficiently conservative. Based on the above discussion, we find the proposed change for downcomer submergence acceptable.

The condensation capability of the suppression pool as a result of reducing the downcomer submergence is discussed in Section 3.12.6 of NUREG-0661.

Condensation capability of the suppression pool is a function of the local pool temperature in the vicinity of the downcomer exit. Full Scale Test Facility (FSTF) test results (NEDE-24539-P) and foreign test data (NEDE-21885-P) have shown that thermal stratification occurs, and becomes more severe as the downcomer submergence is reduced. The most severe thermal stratification has been observed in low flow tests with a quiescent pool. However, in actual plant conditions, the effects from operation of the Residual Heat Removal (RHR) system and SRV discharge action provide sufficient long-term pool mixing to minimize thermal stratification. We have determined that even with vertical thermal stratification, the high energy deposition is accompanied by an increased flow and mixing, which prevent overpressurization of the torus. In addition, the analytical predictions of the torus pressure and bulk temperature response have been found to be conservative when compared with FSTF test data for plant simulated initial conditions. The local temperature variation in the pool which has been observed in the test data is not significant to the structure, and, therefore, need not be considered in the structural analysis.

Based on this assessment, we conclude that a minimum initial downcomer submergence of three feet is acceptable; and that there is sufficient conservatism in the containment response analysis techniques to accommodate the effects of thermal stratification.

The proposed changes are likely to increase the pressure in the drywell and wetwell air space in the event of a LOCA. However, since there exists a 50% margin between calculated and design pressure in the containment, we are approving the proposed changes. These pressures will be confirmed by the analysis required in NUREG-0661 during the program which is now in place for plant unique analyses.

The licensee has also proposed that the following sentence be added to the Technical Specifications: "Tests done in the FSTF showed complete condensation with bulk pool temperature as high as 185°F with a corresponding surface temperature of 230°F. Regarding condensation of steam released through the SRVs and quenchers, test data have shown complete condensation beyond the 200°F limit of the NRC Acceptance Criteria." The licensee further proposes to add the words "bulk" and "local" to the temperature description where appropriate. All of these changes to the Technical Specifications are seen in Enclosure 1.

NUREG-0661, in Section 3.10.7.1, describes the local and bulk pool temperature difference.

Local temperature denotes an average water temperature in the vicinity of the discharge device and represents the relevant temperature which controls the behavior of the condensation process occurring at the pipe exit. In general, this temperature will differ from both the temperature of water in contact with the steam and the bulk temperature of the entire suppression pool. The latter, of course, is a calculated value based on the total energy and mass release into the pool, assuming the pool acts as a uniform heat sink. Because bulk temperature is used in plant transient analyses, the difference between the bulk and local value must be specified so that the analysis can demonstrate operation within the prescribed limits.

In a test facility, the volume of water associated with a single discharge device is only a small fraction of the volume which would exist under prototypical conditions. In such a confined pool, differences between local and bulk conditions are minimal. Tests indicate that temperature distributions in a confined pool are relatively uniform, with generally no more than a 2°F to 3°F variation. Thus, under test conditions, the measured temperature can generally be interpreted as local temperature.

To determine the difference between bulk and local conditions for the quencher device, the Mark I Owners Group relied on the in-plant tests at Monticello. Test results indicated that the difference between bulk and local temperature is 43°F for the test without the RHR system in operation and 38°F for the tests with the RHR in operation. The test with the RHR in operation was conducted with only one RHR loop operating in the pool recirculation mode.

In late 1978, the Mark I Owners Group conducted an adjunct series of tests at the same facility. The purpose of the tests was to investigate methods to improve thermal mixing in the suppression pool and reduce the bulk to local pool temperature difference. These methods include modifications of T-quencher design and the RHR discharge configuration. The T-quencher was modified by adding a number of holes at the tips of one of the quencher arms. The RHR system was modified by installing a 90° elbow, with a 10 x 8-inch reducing nozzle at the end of the existing discharge lines. These modifications were intended to promote mixing in the suppression pool during SRV discharge. Test results show a substantial improvement in the pool mixing. The difference between bulk and local temperature was reduced to approximately 15°F for the test, with one RHR operating in the pool recirculation mode.

Results from these two series of tests clearly indicate that the quencher design and RHR discharge line configuration influence the difference between bulk and local pool temperature to a great extent. The Mark I Owners Group has not presented a generic method for determining the pool temperature difference. Consequently, the staff requires that each plant establish the pool temperature difference, supported by the appropriate data base and with consideration for the plant specific SRV discharge and RHR system arrangement.

NUREG-0661, in Section 3.10.7.2, describes the basis for the local pool temperature limit.

A local pool temperature limit of 200°F has been established generically for quencher devices based on small-scale and in-plant tests. Small-scale tests on selected quencher devices were performed by Kraftwerk Union AG (KWU) in West Germany. The results of these tests indicate that the hole pattern in a perforated pipe quencher is the controlling parameter for effective steam condensation. Using a quencher device with an optimized hole pattern, KWU conducted tests at elevated pool temperatures. The phenomenon of steam condensation instability did not occur, even as the local pool temperature approached the boiling point.

In-plant tests were also performed in a European BWR plant. The discharge devices tested were four-arm quenchers with an optimized hole pattern. The results of the tests indicate that smooth steam condensation is achieved over a wide range of reactor pressures (100 psia to 1100 psia) and pool temperatures (140°F to 176°F). These tests also showed good pool mixing, which was attributed to the bulk pool motion induced by the air or steam jets discharging through special holes at the end of two adjacent quencher arms. The maximum variation of pool temperature was not more than 10°F.

Based on its evaluation of these test data, the staff finds that:

(1) The hole (i.e., perforation) pattern is the primary design feature for achieving smooth steam condensation. Therefore, the 200°F local pool temperature limit applies to all quencher devices designed with the same hole pattern as that tested. Based on its review of the available data, the staff concludes that the 200°F local pool temperature limit also applies to the generic Mark I T-quencher and similar devices with an equal hole diameter and an equal or greater hole spacing.

(2) The small-scale test results showed that steam condensation instability did not occur when the maximum local temperature reached 210°F. The judgment of the staff is that a 200°F temperature limit will provide additional conservatism and will ensure that unstable steam condensation will not occur when a quencher device is utilized.

(3) Plant-unique analyses of the pool temperature response to transients involving SRV operations will be necessary to demonstrate that the suppression pool can be maintained within the limit of a 200°F local temperature.

It must be emphasized that the above limit on maximum suppression pool local temperature was established on the basis of test data that are currently available to the staff. As additional data become available, an increase in this temperature limit may be justifiable.

2.9 Automatic Depressurization System (ADS) Valve

The licensee has reported that an additional one, of the six existing safety/relief valves, will be added to the automatic depressurization system (ADS), also known as automatic pressure relief system (APR), by modification of the actuation logic. The result is that four of six S/RV's will open for the ADS function instead of three of six.

The reason for the change is to improve ECCS response to a small break LOCA by causing a more rapid vessel depressurization if ADS if required. This is only needed in the event of a loss of feedwater combined with a loss of FWCI.

We have concluded that the addition of the fourth ADS valve is adequate compensation for planar heat generation limits that had been imposed on Millstone 1 operation during the last fuel cycle. According to the General Electric "Loss of Coolant Accident Analysis Report for Millstone Unit 1 Nuclear Power Station, NEDO 24085-1", dated July 1980, the fourth ADS valve reduces pressure fast enough to allow low pressure coolant injection into a broken recirculation loop in the event of small break 0.10 ft^2 or less and failure of the gas turbine generator without exceeding the peak clad temperature limit of 2200^oF. NRC Amendment No. 67 dated May 8, 1980, allowed credit for the isolation condenser used in the sequence of events considered by this analysis.

2.10 Jet Pump Baseline Data

The licensee has proposed to delete the requirements to obtain single-loop flow Jet Pump baselien data since single-loop operation is not licensed nor permitted at Millstone 1. We agree that the Technical Specification requirement should be removed for the reason stated.

2.11 Segmented Test Rod Bundle (STR)

We have reviewed "STR Bundle Submittal, Millstone 1 Segmented Test Rod Bundle (Supplement 5)", dated June 1980, which was attqched as an Appendix to the Reload No. 7 License Amendment submittal (Reference 1). This updates the original report submitted on October 3, 1974, as part of Reload No. 2 License Amendment submittal. We are in agreement, as we have been in past reviews, with the licensee conclusions that the STR bundle core irradiation program at Millstone 1 does not effect the health and safety of the public.

2.12 Primary Containment Isolation Valves

The licensee has stated that the proposed changes are administrative in nature and correct several typographical errors. The staff agrees that the proposed corrections to the Technical Specifications should be made.

2.13 Acoustic Monitors for Safety/Relief Valves

Acoustic monitors were installed on the discharge line of all six safety/relief valves in December 1979 consistent with NRC NUREG-0578. We agree with the licensee that this post TMI plant improvement provides additional capability to verify safety/relief valve operation. The appropriate Technical Specifications proposed by the licensee are acceptable and should be made.

3.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of the environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

We have concluded, based on the consideration discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 11, 1981

REFERENCES

1. Letter, W. G. Council (NNECO) to Director of Nuclear Reactor Regulation (NRC), dated September 9, 1980.

2. "Supplemental Reload Licensing Submittal for Millstone Unit 1 Reload 7", Y1003J01A09, June, 1980, enclosure of Reference 1.

3. "Loss-of-Coolant Accident Analysis Report for Millstone Unit 1 Nuclear Power Station", NEDO-24085-1, July 1980, enclosure of Reference 1.

4. "Proposed Technical Specification Changes" along with the technical descriptions and bases for such changes, September 1980, enclosure of Reference 1.

5. "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-1, July, 1979.

6. Safety Evaluation Report (letter), D. G. Eisenhut (NRC) to R. E. Engel (GE), MFN-200-78, dated May 8, 1978.

7. Letter to P. S. Check (NRC) from R. H. Buchholz (GE), MFN-155-80, "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

8. Letter, P. A. Blasioli (NNECO) to J. J. Shea (NRC), "EOC 8 to 70 % Coastdown", December 17, 1980.

9. Telex, P. A. Blasioli (NNECO) to J. J. Shea (NRC) "NRC Response, Reload-7 MCPR changes," January 16, 1981.

10. R. B. Elkins, Fuel Prepressurization, NEDE-23786-1-P, March, 1978.