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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 12

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated September 9, 1987 (Ref. 1), the Northeast Nuclear Energy Company (NNECo) (the licensee) proposed to amend Appendix A of Facility Operating License No. NPF-49. The requested amendment furnished information to support authorization for Millstone Unit 3 operation during Cycle 2.

The Millstone 3 Cycle 2 (hereafter referred to as M3C2) reload will consist of 84 fresh assemblies in place of 65 Region 1 and 19 Region 2 assemblies from the previous cycle. In support of the M3C2 reload NNECo submitted a topical report (Ref. 2) which summarizes the reload scope, the plant transient analyses, and the design and safety analyses. In addition, Technical Specification (TS) changes were proposed to allow operation with a positive moderator coefficient at power levels less than 70 percent, ramping down to zero at 100 percent power. The licensing reports for the latter TS changes are provided in Reference 3. Additional TS changes for Cycle 2 startup were submitted in Reference 4.

In a related matter, NNECo submitted a Radial Peaking Factor Limit Report for Cycle 2 of operation in Reference 6. Also, in response to a staff request for additional information (Ref. 7), the licensee provided additional information (Ref. 8) regarding the expected moderator temperature coefficient for Millstone 3 during the upcoming Cycle 2.

2.0 EVALUATION

2.1 Cycle 2 Reload Description

The M3C2 reload will retain 109 Westinghouse (W) assemblies from the initial cycle and will add 84 W assemblies into an established scatter pattern. The fresh assemblies have minor mechanical differences in pellet, sleeve, and end plug characteristics which do not impact the design bases and do not require TS changes. The new fuel has a higher enrichment than the initial fuel. The Cycle 2 core loading inventory is given in Table 2.1 of this SE. The reload evaluation is based on the Cycle 1 exposure of 17500 to 19000 MWD/MTU and a Cycle 2 exposure of 16000 MWD/MTU. The reload evaluation was performed using accepted methodology described in WCAP-9273 (Ref. 5). Necessary changes to the TS for the Cycle 2 reload are limited to those related to the increased enrichment (TS Section 5.3.1) and the proposed positive moderator coefficient as discussed in the following SE Item 2.2.

Submittal of the Radial Peaking Factor Report (Ref. 6) is required by Technical Specification 6.9.1.6 and identifies nuclear aspects of the design calculations for Millstone 3 Cycle 2. The report identifies F_{xy} limits to provide assurance that the initial conditions assumed in the LOCA analysis are met. The determination of the limits is performed in accordance with acceptable procedures described in WCAP-8403 "Power Distribution Control and Load Following Procedures" which is part of the approved reload methodology for Westinghouse core reloads. We find the identified limits acceptable,

Table 2.1

Millstone 3 Cycle 2 Core Loading Inventory

Region Designation	Number of Assemblies	Initial Enrichment (w/o U235)	BOC Burnup Ave (MWD/MTU)	EOC Burnup Ave (MWD/MTU)
2	45	2.899	19600	35600
3	64	3.395	14200	30200
4A	56	3.5	0	16000
4B	28	3.8	0	16000

2.2 Moderator Temperature Coefficient

The proposed amendment includes a request to change the Millstone 3 moderator temperature coefficient Technical Specification. The request proposes to increase the upper bound of the moderator coefficient given in Specification 3.1.1.4 from zero pcm/deg F to +5 pcm/deg F for power levels below 70 percent of rated power (One pcm is equal to a reactivity change of 10^{-5} delta-k/k). Above 70 percent power the allowed value would decrease linearly to zero pcm/deg F at full power. Approval of the change is requested for both N and N-1 loop operation. The licensee's justification is provided in a Licensing Report (Ref. 3) prepared by Westinghouse Electric Corporation.

The licensee has assessed the impact of a positive moderator coefficient on the accident analyses presented in Chapter 15 of the Millstone Unit 3 FSAR. Those incidents which were found to be sensitive to positive or near-zero moderator coefficients were reanalyzed. These incidents are limited to transients which cause the reactor coolant temperature to increase. Accidents not reanalyzed include those resulting in excessive heat removal from the reactor coolant system and those for which a large negative moderator coefficient is more limiting. We agree with the licensee's conclusions about which transients did and did not require reanalysis.

The transients not reanalyzed are:

A. RCCA misalignment

B. Startup of an inactive reactor coolant pump

C. Excessive heat removal due to feedwater system malfunctions

D. Excessive load increase event

E. Steam generator tube rupture

F. Main steam line depressurization

G. Feedwater system pipe break

H. Inadvertent operation of the ECCS at power

The incidents reanalyzed, with two exceptions, used a +5 pcm/deg F moderator temperature coefficient, assumed to remain constant for variations in temperature. The reanalyses were done with the same computer codes used in the Millstone 3 FSAR analyses and with the same or more conservative uncertainty allowances. The incident exceptions are the rod ejection and rod withdrawal from a subcritical condition, for which the computer code cannot accept a constant coefficient.

The incidents reanalyzed and their results are:

A. Uncontrolled RCCA bank withdrawal from a subcritical condition

The results of the reanalysis of this transient produced a peak heat flux which did not exceed the full power nominal value presented in the FSAR. Therefore the conclusions presented in the FSAR are still applicable.

B. Uncontrolled RCCA bank withdrawal at power

The results of the reanalysis of this transient show that the nuclear flux and overtemperature delta-T trips prevent the core minimum DNBR ratio from falling below 1.3 for this incident, so that the conclusions presented in the FSAR are still valid.

C. Loss of Reactor Coolant Flow

The most severe loss of flow transient is caused by the simultaneous loss of power to all four reactor coolant pumps. For this case a minimum DNBR was obtained which is above the limit of 1.3.

D. RCP Shaft Seizure

The locked rotor event was reanalyzed because of the potential effect of the positive MTC on the nuclear power transient and thus on the peak RCS pressure and fuel temperature. A positive MTC will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. For three loops initially operating the peak clad temperature was 1828 deg F and the peak RCS pressure was 2613 psia; these values do not exceed the accepted safety limits.

E. Loss of External Electrical Load/Turbine Trip

The turbine trip event was reanalyzed for the beginning of cycle (BOC) and at end of cycle (EOC). Four cases for both N-loop and N-1 loop operation were analyzed: reactor at beginning-of-life with operation of the pressurizer spray and pressurizer power operated relief valves (PORV) and the reactor at end-of-life with no credit for pressurizer spray or PORVs. The result of a turbine trip is a core power which momentarily exceeds the secondary system power removal causing an increase in RCS coolant temperature. The reactivity addition due to a positive MTC causes an increase in both nuclear power and RCS pressure. Since the DNB ratio does not drop below its initial value and the peak RCS pressure increases slightly but is less than 110 percent of design the staff's acceptance criteria is met and the conclusions presented in the FSAR are still applicable.

F. RCCA Ejection

The results of reanalysis of this incident show the fuel and clad temperature do not exceed the limits specified in the FSAR, so that the conclusions presented in the FSAR are still valid.

G. Loss of Normal Feedwater

The results of reanalysis of this incident show minimal changes in the pressurizer water volume and reactor coolant system temperature so that the conclusions presented in the FSAR are still valid.

H. Inadvertant Opening of a Pressurizer Safety or Relief Valve

The results of reanalysis of this incident show that the minimum DNBR stays above the accepted safety limit and the conclusions presented in the FSAR are still valid.

I. Anticipated Transients Without Scram

Although the Millstone 3 Cycle 2 Technical Specification moderator temperature coefficient (MTC) is being increased to + 5 pcm/deg F below 70% power and to a linear variation from + 5 pcm/deg F at 70% power to zero pcm/deg F at 100% power, the licensee stated, in its response (Ref. 8) to a staff request for additional information (Ref. 7), that the Cycle 2 core design, hot full power (HFP) MTC will be more negative than - 7.6 pcm/deg F with equilibrium xenon conditions. In addition, the MTC for Cycle 2 will become more negative as cycle burnup increases. This Cycle 2 HFP MTC at equilibrium xenon conditions is more conservative than the - 5.5 pcm/deg F value used in generic ATWS studies (Ref. 9) performed for four-loop Westinghouse plant designs and which yielded a limiting peak pressure of 3200 psig. Since Millstone 3 is, from an ATWS point of view, similar to the four-loop class of Westinghouse plants for which the studies were performed, the staff concludes that ATWS considerations are not significantly impacted. In addition the Millstone 3 Cycle 2 physics startup tests will provide a hot, zero power (HZP) MTC reference point which can be used to assess the core design HFP MTC with equilibrium xenon conditions.

Results

Since the reanalysis of plant transients did not result in exceeding any of the limits specified in the existing analyses for the Millstone Unit No. 3 reactor, we conclude the proposed Technical Specification change will not result in any significant loss of safety margins, and is therefore acceptable.

2.3 Increase in Allowable Value of Boron Concentration in Borated Water Sources

The incorporation of a positive moderator coefficient and the anticipated longer fuel cycle necessitates increasing the required boric acid in the Emergency Core Cooling System (ECCS) accumulators and the refueling water storage tank (RWST). The licensee's submittal includes a safety evaluation by Westinghouse of the impact of raising the boron concentration on the Loss of Coolant Accident (LOCA) and non-LOCA analyses and design considerations (Ref. 3). The proposed amendment would increase the RWST boron concentration from 2000 ppm to the range 2300 to 2600 ppm and the ECCS accumulator boron concentration from 1900 ppm into the range 2200 to 2600 ppm. The changes are necessary to assure that the reactor will remain subcritical in cold shutdown following a LOCA when higher enrichment fuel is used for anticipated extended fuel cycles for Millstone Unit 3. This increase also has implications for the analysis of several non-LOCA events and for the system chemistry through the mechanism of pH change in the post accident containment sump liquid inventory.

Non-LOCA Safety Analyses

The only non-LOCA events which are affected by the increased boron concentration are those for which the ECCS is actuated. Each of these events has been examined to determine the effect of increased boron concentration. The results show that the increased boron has a generally helpful effect on the individual event and in no case does it have a harmful effect.

LOCA Analysis

The small break LOCA analysis makes no assumption about the boron concentration in the ECCS water (shutdown is achieved and maintained by control rods). Thus the increased boron concentration has no effect on the small break LOCA analysis.

During the initial portion of the large break LOCA analysis subcriticality is maintained by voids in the core and the increased boron concentration has no effect on this portion of this event. Since the peak clad temperature occurs during this portion of the event there is no effect on the results of the LOCA analysis. Based on the similarity of assumptions to the LOCA analyses, the increase in ECCS water boron concentration would also have no effect on the long term mass and energy releases to the containment.

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For post LOCA shutdown no credit is taken for control rods in the Millstone 3 analysis. Thus the boron in the ECCS water is relied upon to maintain shutdown. The ECCS water, when mixed with other sources (reactor coolant water, etc.) must produce a boron concentration sufficient to maintain the reactor in a shutdown state. The adequacy of the increased concentration to achieve this goal is to be addressed for each reload.

Other Considerations

Increasing the ECCS boron concentration reduces the pH of the containment spray and recirculating core coolant solutions. The reduction in pH can lead to reduction of the iodine spray removal coefficient and decontamination factor (DF), an increase in the rate of hydrogen production due to zinc corrosion and an increase in the potential for chloride induced stress corrosion cracking of stainless steel. These effects have been examined by the licensee with the following results.

Since the FSAR analysis of radiological consequences of the large-break LOCA does not assume containment spray iodine removal, there is no change in the FSAR conclusion in this area.

Examination of zinc corrosion rate data shows that the corrosion rate for the minimum pH resulting from the increased boron concentration is less than that assumed in the FSAR. This is acceptable. Corrosion of other materials (e.g., aluminum) decreases monotonically with decreasing pH.

The pH range resulting from the boron increase would be outside the range recommended to minimize chloride stress corrosion cracking of stainless steel. To bring the post LOCA recirculation sump pH into an acceptable range, the sodium hydroxide concentration in the chemical addition tank (CAT) has been increased. The proposed TS changes (LCO 3.6.2.3) reflect this concentration increase along with a reduction in solution volume and are acceptable.

2.4 Additional Changes to Support the Cycle 2 Reload

In Reference 4 the licensee provided a safety evaluation for the following TS changes which are intended for incorporation in this amendment.

(1) Section 3/4 4.1.6 - The LCO and Surveillance Requirement for boron concentration in an isolated loop is revised to limit the required concentration to a maximum of 2300 ppm.

The present LCO requires that an isolated loop be at a boron concentration greater than or equal to the unisolated portion of the reactor coolant system (RCS). The proposed change would allow the isolated loop to be brought into service as long as the water used to fill the isolated loop has a boron concentration greater than the rest of the RCS. The licensee has evaluated the relevant design basis events for consequences of the change and has concluded that the existing FSAR analyses are still bounding or are not affected. The staff finds this change acceptable.

(2) Section 3.9.1 2 - The LCO and Surveillance Requirement for boron concentration during refueling operations were changed to require 800 ppm boron in the spent fuel pool subcriticality such that a dropped or misplaced fuel assembly will not cause the k-effective of the pool to exceed 0.95. The licensee has reviewed the relevant design basis events and has concluded that the increase in boron will not adversely affect the consequences. This change is therefore acceptable.

(3) Section 5.3.1 - The Design Features description of the fuel assemblies was revised to delete the maximum total weight limitation on individual fuel rods and clarifies the maximum enrichment for future core reloads. The change identifies the maximum nominal enrichment as 3.8 weight percent U-235. This is consistent with the proposed Cycle 2 design and is acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

The following Millstone Unit 3 Technical Specification changes have been proposed for operation during Cycle 2:

A. Changes to identify and support a positive moderator coefficient

TS Page	TS LCO or BASES	Title
3/4 1-4	3.1.1.3	Moderator Temperature Coefficient
3/4 1-11	3.1.2.5	Borated Water Source - Shutdown
3/4 1-12	3.1.2.6	Borated Water Sources - Operating
3/4 5-1	3 5.1	Accumulators
3/4 5-9	3.5.4	Refueling Water Storage Tank
3/4 6-14	3.6.2.3	Spray Additive System
3/4 9-1	3.9.1	Boron Concentration
B 3/4 1-3	3/4.1	Boration Systems
B 3/4 9-1	3/4.9.1	Boron Concentration

B. Other Technical Specification Changes

5-5	5.3.1	Fuel Assemblies
3/4 9-1c	3.9.1.2	Boron Concentration
3/4 4-8	3.4.1.6	Isolated Loop Startup

The above replacement TS pages identified in the licensee's request (Attachment 2 to letter B12661 and Attachment 1 to letter B12692) are acceptable as proposed.

4.0 SUMMARY

We have reviewed the reports submitted for the Cycle 2 operation of Millstone Unit 3. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. Sufficient basis has been provided to allow reload of 109 W assemblies and the use of a positive moderator coefficient. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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.

6.0 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.

Dated: January 20, 1988

Principal Contributors:

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7.0 REFERENCES

1. Letter, B-12661, E. J. Mroczka, NNECo, to U. S. NRC, dated September 9, 1987, with attachments

2. Reload Safety Evaluation for Millstone Unit 3 Cycle 2, Westinghouse Electric Corporation, August 1987

3. Positive Moderator Coefficient Licensing Report for Millstone Unit No. 3, Westinghouse Electric Corporation, August 1987

4. Letter, B-12692, E. J. Mroczka, NNECo, to U. S. NRC dated September 30, 1987, with attachments

5. WCAP-9293-A, "Westinghouse Reload Safety Methodology", dated July 1985

6. Letter, MP-11104, S. E. Scace, NNECo, to W. Russell, Region I Administrator, dated November 10, 1987

7. Letter, R. L. Ferguson (NRC) to E. J. Mroczka (NNECo), dated December 1, 1987.

8. Letter, E. J. Mroczka (NNECo) to R. L. Ferguson (NRC), dated January 7, 1988.

9. Letter (NS-EPR-83-2833 from E. P. Rahe W to S. J. Chilk (NRC), dated October 3, 1983.

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ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 12

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated September 9, 1986 (Ref. 1), the Northeast Nuclear Energy Company (NNECo) (the licensee) indicated that the reactor coolant temperature measurement system for the hot and cold legs for Millstone Unit 3 will be modified and requested changes to the plant's Technical Specifications (TS). This modification is to eliminate the Resistance Temperature Device (RTD) bypass manifold to improve availability, reduce radiation exposure, and reduce maintenance. However, the new hot leg temperature measurement method has the disadvantage of a slightly longer response time. Reference 1 included the proposed TS changes and also WCAP-11496 (Proprietary), the licensing report (Ref. 2) with RCS nozzle and thermowell locations, results of the reanalysis of affected FSAR Chapter 15 accidents, and a flow measurement uncertainty analyses.

2.0 BACKGROUND

The current method of measuring the hot and cold leg reactor coolant temperatures uses an RTD bypass system. This system was designed to address temperature streaming in the hot legs and, by use of shutoff valves, to allow replacement of the direct immersion narrow-range RTDs without draindown of the Reactor Coolant System (RCS). For increased accuracy in measuring the hot leg temperatures, sampling scoops were placed in each hot leg at three locations of a cross section, 120 degrees apart. The flow from the scoops is piped to a manifold where a direct immersion RTD measures the average temperature of the flow from the three scoops. This bypass flow is routed back to the RCS downstream of the steam generator. The cold leg temperature is measured in a similar manner with piping to a bypass manifold except that no scoops are used, as temperature streaming is not a problem due to the mixing action of the RCS pump.

The new method proposed for measuring the hot and cold leg temperatures uses narrow-range fast response RTDs manufactured by Weed Instruments, Inc. The RTDs are placed in thermowells to allow replacement without draindown. The thermowells, however, increase the response time.

The RTDs for the hot legs in loops B and C are placed within the existing scoops for Millstone Unit 3. A hole will be drilled through the end of each scoop so that water will flow through the existing holes in the leading edge of the scoop, past the RTD, and out through the hole. The RTD measures the temperature at one point in the new method. This is in contrast to the temperature measurement of the average of the flow from the five sample holes from the hot leg scoops used in the RTD bypass flow method. However, in the new method, the radial location of each RTD measurement is at the same radius as the center hole of the scoop. Therefore, the licensee states, it is the equivalent of the average scoop sample if a linear radial temperature gradient exists in the pipe.

The RTDs in the hot legs for loops A and D have two of the three RTDs mounted in thermowells as described above. The third thermowell cannot be installed in the existing scoop location due to structural interference. Therefore, this thermowell is located downstream of the existing scoop in independent bosses. Although these RTDs are not in scoops, the sensor will be at the same radial location (in line with the center hole) as the other RTDs which are mounted inside the existing scoops.

The design for the measurement of the cold leg temperature has also been modified. A single thermowell with one fast response, narrow range, dual-element RTD is mounted in each cold leg. One of the dual elements is a spare. This is in place of the original method in which the measurement was by an external RTD in the cold leg bypass manifold.

An electronic system is used to perform the averaging of the reactor coolant hot leg signals from the three RTDs in each hot leg and then to transmit the signal for the average hot leg temperature to protection and control systems. There is a routine for performing a quality check of the three temperature signals for each hot leg, which are used to get an average value of the temperature variation. Also, there is a capability to add a bias to the averaging calculation, if needed, to compensate for the loss of one of the three RTD sensor inputs. The bias considers the past history of the previous hot leg readings.

Reference 1 provided the TS changes regarding the new reactor coolant system (RCS) temperature measurement system modifications required because of the elimination of the RTD bypass loop. Reference 2 provided the results of the reanalysis of several Chapter 15 non-LOCA accidents. The accuracy of the hot leg temperature was included in an RCS flow measurement uncertainty analysis submitted in Reference 2. Since Millstone Unit 3 has N-1 loop capability, information was presented for both four and three loop operation where applicable.

3.0 ANALYSIS

We questioned the licensee regarding the response time and uncertainty effects of the new measurement system. The licensee responded to our questions in a letter dated November 25, 1987 (Ref. 3). The increased response time has the primary impact on the results of the accident analysis. The uncertainty of the hot leg temperature measurement affects the accident analysis and is the principal contributor in the analysis for calculating the RCS flow measurement uncertainty.

3.1 RTD Response Time

The overall response time of the new thermowell RTD temperature system is one second longer than the former RTD bypass system (7.0 vs 6.0 seconds). The 7.0 second overall response time for the new RTD system is a result of adding a 1.5 second electronics delay to the 5.5 second response time for the RTD sensor. The single RTDs in loops A and D not inside of scoops are expected to have a slightly faster response time. However, no credit is taken for this. Because of the increased channel response time, there are longer delays from the time when fluid conditions in the reactor coolant system (RCS) require overtemperature delta-T or overpower delta-T reactor trips until a trip is actually generated. The licensee presented information in Reference 2 concerning the FSAR Chapter 15 non-LOCA accidents that rely on the above mentioned trips and which were evaluated for the longer response time.

As noted in NUREG-0809, (Ref. 4), extensive RTD testing has revealed RTD time response degradation with aging. In view of this, surveillance tests are needed. The approved in situ method for measuring RTD response time is the Loop Current Step Response (LCSR) method. The licensee cannot tell at present if there is margin from the manufacturer's stated response time for the RTD sensor of 5.5 seconds. This will be determined after the initial installed response time test and succeeding tests which will show any effect of possible drift in response time as mentioned in Reference 4.

3.2 RTD Uncertainty

The new method of measuring each hot leg temperature with three thermowell RTDs manufactured by Weed Instruments, Inc., used in place of the RTD bypass system with three scoops, has been analyzed to be slightly more accurate. The new RTD thermowell with measurement at one point may have a small streaming error relative to the former scoop flow measurement because of a temperature gradient over the 5-inch scoop span. However, this gradient has been calculated to have a small effect. Also, since possible temperature uncertainties from imbalanced scoop flows are eliminated, the overall result is more effective. In addition, since the new method uses three RTDs for each hot leg temperature measurement, it is statistically a more accurate temperature measurement than the former method which used only one RTD for each hot leg temperature measurement. Although the new hot leg RTD temperature measurement is initially slightly more accurate than with the former RTD manifold method, it becomes less accurate because of the additional uncertainties introduced when the signals are processed for averaging before being sent to the 7300 processing system. System uncertainty calculations were performed by the licensee that verify that sufficient allowance has been made in the reactor protection system setpoints to account for an increased initial RCS average temperature error. Therefore, the current values of nominal setpoints for the Millstone Unit 3 TS are still valid.

The licensee has stated (Ref. 10) that during the Cycle 2 startup test program they will collect the new hot leg and cold leg narrow range RTD temperature measurements at 100% of power and compare the Tavg and delta T measurements to those values obtained during the Cycle 1 startup test program before elimination of the RTD bypass system. This comparison will be utilized to verify that the new method of temperature measurement is accurate. Similar temperature measurements are also being taken at the Catawba plant to compare their values before and after a similar modification.

In response to questions regarding the monitoring of the RTDs for failure, the licensee responded (Ref. 3) that the following three types of alarms are in place:

	HI/LO Deviation Alarm
1. Tavg/Auctioneered Tavg	± 2° F
2. Tref/Auctioneered Tavg	± 3° F
3. Delta T/Auctioneered Delta T Deviation	± 2° F

These alarms are monitored continuously and a channel operability check is performed every 12 hours. Information is also obtained on individual RTDs to obtain bias values that can be used to compensate for a failed RTD in the hot leg averaging calculation. This information is obtained on a monthly basis.

In response to a question on RTD drift, the licensee explained (Ref. 3) their method for calibrating the RTDs at each refueling prior to startup. This method uses a Westinghouse recommended in-situ calibration known as the "Incore Thermocouple and Resistance Temperature Detector (RTD) Cross Calibration". Data is collected during plant heat-up following refueling. At RCS temperatures of approximately 320° F, 390° F, 450° F, and 515° F, the resistance of each of the 32 RTDs is recorded, as well as the indicated temperature from each incore thermocouple. These readings are recorded 4 times at each temperature within a span for 10 minutes. The calibration of the digital voltmeter used to measure resistance is verified and the data is validated prior to recommencement of heat-up. Prior to reactor startup, the calibration of each RTD and each incore thermocouple is verified by comparison to the average temperature indicated by the RTDs. If any RTD deviates from its calibration curve, a new calibration curve is generated based upon the data obtained during the cross-calibration. In addition, each incore thermocouple is compared to a calculated average temperature and must be within 2° F to be considered operable. The average temperature of the RTDs is compared to the incore thermocouples which are considered to be very stable and to have no drift. If a systematic, long-term drift in RTD calibration occurred, it would mean that an abnormally large number of incore thermocouples would be found with a deviation greater than 2° F compared to the RTDs. Steps would then be taken to correct the RTDs for drift.

The platinum resistance temperature sensors (RTDs) are believed to be very stable and to have relatively small calibration drifts. However, according to several sources (Refs. 5, 6, 7) RTDs have been known to shift in calibration. Therefore at the time of taking the calorimetric heat balance at refueling, necessary steps for correction (recalibration in a lab) should be made for any appreciable calibration drifts encountered or the RTD(s) declared inoperable and replaced. For small deviations found in their in-situ cross calibration method, the licensee stated (Ref. 3) that the calibration of the resistance to voltage converters of the affected RTD(s) will be adjusted to account for the shift. The stated drift of the RTDs by the manufacturer is ± 0.2° F per year. This is significantly less than the value assumed for the uncertainty calculations for the protection system.

3.3 Non-LOCA Accidents Reanalyzed

The primary impact of the RTD bypass system elimination is the increased RTD response time. Thus, only those events which rely on the overtemperature and overpower delta-T (OTDT and OPDT) reactor trips are impacted. The accidents in FSAR Sections 15.1 to 15.6 were examined by the licensee (Ref. 1 and 2) and the following non-LOCA accidents affected by the longer RTD response time were reanalyzed: (1) the Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal; (2) Loss of Load/Turbine trip; (3) Inadvertent Opening of a Pressurizer Safety or Relief Valve; and (4) the Steamline Rupture at Power. These accidents are described in Chapter 15 of the FSAR.

The licensee stated that the LOFTRAN computer code was used for the analysis of these events. For each event reanalyzed the basic assumptions regarding initial conditions, instrument errors, and setpoint errors that are not directly related to RTD Bypass Elimination remain largely the same as those in Chapter 15 of the FSAR. However, certain additional changes were made. The ± 30 psi allowance on pressurizer pressure in FSAR Section 15.0.3.2 was increased to ± 45 psi for more conservatism. Also increased uncertainties were applied to pressurizer and steam generator water levels which were increased from 5% to 5.73% and 5.53% respectively. These increased uncertainties were incorporated to bound calculated increases in the associated transmitter uncertainties. Also, changes were made in the reactor protection system setpoints to account for the new thermowell mounted RTDs. The time constant for lead/lag compensation of delta-T was increased from 8.0 to 12.0 seconds for the non-LOCA transient analysis.

The first accident, Uncontrolled RCCA Withdrawal, is described in Section 15.4.2 of the FSAR. For this event, the High Neutron Flux and Overtemperature delta-T reactor trips are assumed to provide protection against DNB. This event was analyzed with the increased time constants and the lead/lag changes of delta-T. Plots of DNBR versus time were provided which showed that the DNBR criterion was met for this accident.

The Loss of Load/Turbine Trip event is described in Section 15.2.3 of the FSAR. This event is affected by the increase of RTD response time which provides input to the over temperature delta-T trip. Both N and N-1 cases were reanalyzed and for all combinations of reactivity feedback and pressure control. The DNBR limit of 1.3 was met and the primary system pressure remained below 110% of the design value of 2,500 psi.
The Inadvertent Opening of a Pressurizer Safety Relief Valve event is described in Section 15.6.1 of the FSAR. The reanalysis was done considering both N and N-1 loop operation. The positive moderator coefficient causes nuclear power to increase as pressure decreases until reactor trip occurs at Overtemperature delta T. The DNBR remains above 1.30 throughout the transient and is therefore acceptable.

For the Steamline Rupture at Power event the analysis included the increased response time and was performed for both N and N-1 loop operation. The analysis showed that the design basis as described in WCAP-9226-RI has been met.

In response to a question regarding the impact of the RTD response time on the uncontrolled boron dilution at power event the licensee responded (Ref. 3) that the event would not be affected. The licensee stated that boron dilution analyses at power are performed to show that sufficient operator action time is available to terminate the dilution prior to loss of the shutdown margin requirement specified in the TS (1.6% delta-k/k). The calculated operator action time is not affected by the replacement of the RTDs since it is defined as starting simultaneously when reactor trip occurs. Therefore, the available operator action time would not change. For the case with manual rod control and no operator action to terminate the dilution, the power and temperature rise will result in a trip on an over temperature delta-T signal. The boron dilution transient results in a positive reactivity insertion and is essentially equivalent to an uncontrolled rod withdrawal at power transient. The reactivity insertion rate is within the range analyzed in WCAP-11496. Those analysis results are applicable to the boron dilution transient at power and thus show that the acceptance criteria for the boron dilution event are met with the increased RTD response time.

In summary, the impact of the RTD bypass elimination for Millstone Unit 3 on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated. For the events impacted, it was demonstrated that the conclusions presented in the FSAR (including DNBR remains above 1.30) remain valid for both N and N-1 loop operation.

3.4 LOCA Evaluation

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The licensee stated in Reference 2 that the magnitude of the uncertainties are such that RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will not be affected. Past sensitivity studies concluded that the inlet temperature effect on peak clad temperature is dependent on break size. As a result of these studies, the LOCA analyses are performed at a nominal value of inlet temperature without consideration of small uncertainties. The RCS flow rate and steam generator secondary side temperature and pressure are also determined using the loop average temperature (Tavg) output. These nominal values used as inputs to the analyses are not affected due to the RTD bypass elimination. It is concluded that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint for both N and N-1 loop operation without requiring any reanalysis.

3.5 Flow Measurement Uncertainty

The licensee provided a new flow measurement analysis in Reference 2 that accounts for changes due to the RTD bypass removal. The methodology used was the same as that used for the Shearon Harris Unit 1 plant as provided in WCAP-11169, Rev. 1, October 1986. This analysis used the plant-specific instrumentation for the Millstone Unit 3. The results of the analysis indicated that for four loop operation the flow measurement uncertainty can be reduced from the current value of ± 2.4% (not including a 0.1% penalty for feedwater venturi fouling allowance) to a new value of ± 1.8% (including the cold leg elbow taps and excluding feedwater venturi fouling). For three loop operation the results of the analysis indicated that the flow measurement uncertainty can be reduced from the current value of ± 2.76% to ± 2.00%, excluding the 0.1% penalty for feedwater venturi fouling. Our review has found the results of the analysis to be acceptable.

The licensee provided information in TS section 3/4.2.3 which showed that the minimum indicated RCS flow rate is 385,210 gpm for four loop operation and 304,780 gpm for three loop operation. The corresponding flow measurement uncertainties are ± 1.8% and ± 2.0% for four and three loop operation respectively. The minimum indicated RCS flow rate values are obtained by increasing the thermal design flows (TDFs) of 378,400 gpm (4 x 94,600 gpm/loop) and 298,800 gpm (3 x 99,600 gpm/loop) for four and three loop operation respectively by their corresponding flow measurement uncertainties.

The licensee is installing inspection ports (Ref. 8) upstream of the venturis. This is to be similar in design to that provided in another plant (Ref. 9). The inspection port is about 13 inches upstream of the low pressure tap of the venturi meter. This places the inspection port about midway between the low pressure and high pressure taps. With a 4-3/4" diameter opening of the inspection port, the observer will have an unobstructed view of the inside pipe wall opposite the port and can see the inlet contour of the venturi along the opposite wall. With the aid of an inspection mirror and light, the entire circumference of the inside pipe can be viewed as well as the converging section of the venturi. To inspect the diverging section of the venturi a flexible fiber-scope would be used.

As stated above, for the new analysis there is a further reduction in flow measurement uncertainty for four loop operation of Millstone Unit 3 from ± 2.4% to ± 1.8%. It is the staff's position that the additional ± 0.1 penalty for venturi fouling should be applied unless the venturi meters are cleaned at each refueling. The minute buildups on the venturi that could affect the flow measurement cannot be accurately quantified by just visual means.

When the 0.1% venturi fouling factor is added, the resulting flow measurement uncertainties are ± 1.9 for four loop operation and ± 2.1% for three loop operation. The TS minimum indicated RCS flow rate for these conditions is 385,590 gpm for four loop operation and 305,075 gpm for three loop operation.

The licensee has stated (Ref. 3) that the latest Millstone Unit 3 RCS measured flow rate is 415,294 gpm based on three of the loops having a flow rate of 110% of thermal design flow and one loop at 109% of thermal design flow. This is equivalent to an indicated RCS flow rate (415,294 gpm) that is 109.75% of thermal design flow (1.0975 x 378,400 gpm). Therefore the existing Millstone Unit 3 RCS flow rate is well above the required thermal design flow rate.

TS sections 4.2.3.1.6, 4.2.3.2.6 and the bases for TS section 3/4.2.4 (page B 3/4 2-6) will need to be modified to state that t[illegible] penalty for undetected fouling of the feedwater venturis of 0.1% will be added to the flow measurement uncertainty values if the venturis are not cleaned. This is to be done before the precision heat balance is made to calibrate the RCS flow rate indicators (approximately once per 18 months). The licensee has stated that the feedwater venturis have been cleaned for the Cycle 2 operation. The licensee has stated (Ref. 10) that the above TS's will be modified to reflect the requirement of 0.1% penalty if the venturis are not cleaned and submitted for NRC approval. The staff requires this modification prior to Cycle 3 operation.

3.6 Instrumentation

A. Current System

Currently, the hot and cold leg RTD's are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTD's and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTD's. The RTD's are located in manifolds and are directly inserted into the reactor coolant bypass loop without thermowells. Each RTD manifold (one hot leg and one cold leg per reactor coolant loop) contains two narrow-range RTD's: one for protection and control system inputs and one as a spare. Flow into each bypass loop is provided by three scoops located at 120° intervals around the hot leg and a tap into the corresponding cold leg.

Each loop's pair of RTD's (one in the hot leg and one in the cold leg) is used to provide inputs for protection system functions based on the average loop temperatures ($Tavg = T_{HOT} + T_{COLD})/2$) and the loop differential temperature $\Delta T = T_{HOT} - T_{COLD}$). Protection functions based on these inputs are: overtemperature ΔT and overpower ΔT reactor trips with their associated (non-protection) rod stop and turbine runback actions, low Tavg main feedwater isolation, and low-low Tavg (P-12) steam dump block signals.

Each loop's pair of RTD's is also used to provide inputs for control system functions based on the average loop temperature and the loop differential temperature. Control functions based on these inputs are: turbine loading stop from auctioneered low Tavg; rod, steam dump and pressurizer level control from auctioneered high Tavg; rod insertion limit alarms from auctioneered high ΔT and Tavg.

B. Modified System

The hot leg temperature inputs from each reactor coolant loop will be developed from three fast response, narrow range single-element RTD's mounted in thermowells located within the three existing RTD bypass manifold scoops (except for loops A and D where the two of the three thermowells will be mounted in the scoops with the third thermowell located downstream in an independent boss). One fast response, narrow range dual-element RTD per loop will be mounted in a thermowell located at the existing penetration for the bypass loop into the cold leg. Both elements of the dual-element RTD will be wired to the process instrumentation cabinets with one element per RTD serving as a spare. In the event of an element failure, switchover to the spare element can be readily accomplished.

Each hot leg temperature input for protection system functions will be developed by electronically averaging the signals from the three new fast response, narrow range RTD's. This averaged input will replace the single input from the currently installed hot leg RTD. Each cold leg input for protection system functions will be provided by the new fast response, narrow range RTD which replaces the currently installed cold leg RTD. In the event of a hot leg RTD failure, the electronics allow a bias developed from historical data for the failed RTD to be manually added via a potentiometer to the remaining two RTD signals in order to obtain an average value comparable to the three-RTD average prior to failure of the one RTD. If an element in the dual-element cold leg RTD fails, the spare element can be used instead.

Inputs for the control system functions will be provided, through isolators, from the average loop temperatures and loop differential temperatures calculated by the protection system. This aspect of the design has not been changed; only the use of three hot leg RTD's instead of one per loop to provide a hot leg temperature is different.

Our review and evaluation is based upon Sections 7.2 and 7.3 of the Standard Review Plan (SRP). Those sections state that the objectives of the review are to confirm that the reactor trip and engineered safety features actuation system satisfy the requirements of the acceptance criteria and guidelines applicable to the protection system and will perform their safety function during all plant conditions for which they are required. Since our review indicates that the modified system does not functionally change (except three hot leg RTD's are utilized instead of just one) the reactor trip and engineered safety features actuation systems, the staff's original evaluation conclusions for these systems, as documented in Section 7.2 and 7.3 of the SER for Millstone Nuclear Power Station Unit 3, (NUREG-1031), remain valid. Based on this and the licensee's statement that the new hardware for the RTD bypass elimination has been qualified to WCAP-8587, "Methodology to Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," we find the plant modifications to eliminate the RTD bypass manifold and to install fast response RTD's directly in the reactor coolant system hot and cold legs to be acceptable.

3.7 Mechanical Design

The removal of the RTD bypass loop and the installation of the RTD thermowells require modifications to the hot leg scoops, the hot leg piping, the crossover leg bypass return nozzle, and the cold leg bypass manifold connection. The licensee proposed that welding and non-destructive examinations will be per ASME Code Section XI requirements. Fabrication will be in accordance with the ASME Code Section III. By the letter of December 23, 1987, the licensee has verified that during installation there was no deviation from the proposed RTD configurations in the referenced report. We find the welding, non-destructive examinations and fabrication to be acceptable.

4.0 EVALUATION OF TECHNICAL SPECIFICATIONS

As a result of the modifications associated with the removal of the existing RTD bypass manifold and replacement by fast response RTDs, changes to the plant's TSs (Ref. 3) were proposed to allow operation of Millstone Unit 3 with the RTD bypass manifolds removed and to incorporate the reduced flow measurement uncertainty values for N and N-1 operation.

Change 1 - Bases TS Section 2.1, page B 2-5, for Overtemperature delta-T - The statement "transit delays from the core to the temperature detectors (about 4 seconds)." The about 4 seconds was deleted as the bypass piping is removed. This is acceptable as an editorial change to represent the present condition.

Change 2 - Bases TS Section 2.1, page B 2-6, for Overpower delta-T - The statement "compensation for piping delays from the core to the loop temperature detectors" was eliminated. This change is an editorial change to reflect the present condition.

Change 3 - TS Section 3.2.3.1, page 3/4 2-15 for four loop operation - The indicated RCS total flow rate was changed from 387,500 gpm to 385,210 gpm. The flow measurement uncertainty was changed from 2.4% to 1.8%. These changes are acceptable as discussed in the Analysis Section.

Change 4 - TS Section 3.2.3.2, page 3/4 2-18 for three loop operation - The indicated RCS total flow rate was changed from 307,050 gpm to 304,780 gpm. The flow measurement uncertainty was changed for 2.76% to 2.0%. These changes are acceptable as explained in the Analysis Section.

Change 5 - Table 3.2-1, page 3/4 2-24, DNB Parameters - The indicated RCS Tavg values for four loop and three loop operation were changed from 589.2° F and 581.7° F respectively to 591.2° F and 583.4° F respectively. The indicated pressurizer values for four loop and three loop operation were changed from 2,220 psia to 2,226 psia. These changes reflect the changes in uncertainty from replacing the RTD bypass system and are acceptable.

Change 6 - Table 3.3-2, page 3/4 3-8, Reactor Trip System Instrumentation Response Times - The response times for Function Unit 7, Overtemperatures delta T and Functional Unit 8, Overpower delta-T was changed from 4.0 to 7.0 seconds. These changes are due to the removal of the RDT bypass system and the longer response time of the new RTD thermowell system. They are acceptable as explained in the Analysis Section.

Change 7 - Table 4.3-1, page 3/4 3-10, Reactor Trip System Instrumentation Surveillance Requirements - Under "channel calibration" for Functional Unit 7, Overtemperature delta T, the reference to note 12 was deleted. Also note 12 was revised to read "(NOT USED)". There changes are acceptable as they are editorial changes to reflect the present condition.

As a result of the new instrumentation associated with the removal of the existing RTD bypass manifold and replacement by fast response RTD's, the following changes to the plant's TS were proposed:

Change 1 - Change the entries for Z and Sensor Error for Functional Unit 7.a, Overtemperature ΔT, Four Loops Operating, in Table 2.2-1 from "5.9" to "2.3" to "5.76" and "1.67 + 1.17 (Temp + Press)" respectively.

Change 2 - Change the entries for Z and Sensor Error for Functional Unit 7.b, Overtemperature ΔT, Three Loops Operating, in Table 2.2-1 from "5.9" and "2.3" to "5.77" and "1.73 + 1.17 (Temp + Press)" respectively.

CHANGE 3 - Change the entries for Z and Sensor Error for Functional Unit 8, Overpower ΔT, in Table 2.2-1 from "1.43" and "0.11" to "1.22" and "1.67" respectively.

CHANGE 4 - Change the entries for Z, Sensor Error and Allowable Value for Functional Unit 12, Reactor Coolant Flow-Low, in Table 2.2-1 from "1.74," ".8" and "89.3" to "1.52," "0.78" and "89.1" respectively.

CHANGE 5 - In Note 1 to Table 2.2-1, change "ΔT = Measured ΔT by RTD Manifold Instrumentation" to "ΔT = Measured ΔT by Reactor Coolant System Instrumentation."

CHANGE 6 - Change the entry for τ_1 in Note 1 to Table 2.2-1 from "8s" to "12s."

CHANGE 7 - Change the allowable value (for Overtemperature ΔT) in Note 2 to Table 2.2-1 from "4.1%" to "3.6%" for three loop operation.

CHANGE 8 - Change the allowable value (for Overpower ΔT) in Note 4 to Table 2.2-1 from "3.4%" to "2.8%."

CHANGE 9 - Change the response times for Functional Unit 7, Overtemperature ΔT and Functional Unit 8, Overpower ΔT, in Table 3.3-2 from "4.0" to "7.0."

CHANGE 10 - Under "CHANNEL CALIBRATION" for Functional Unit 7, Overtemperature ΔT, in Table 4.3-1, delete reference to Note 12. Also reivse Note 12 to read "(NOT USED)."

CHANGE 11 - Change the entries under Allowable Value for Functional Units 5.d.1 and 5.d.2 from "562°F" to "560.6°F."

CHANGE 12 - Change the entry under Allowable Value for Functional Unit 9.b, Low-Low Tavg, P-12, in Table 3.3-4 from "549.7°F" to "549.6°F."

CHANGE 13 - Change the entry for the response time for Functional Unit 12.a from "6.8" to "12.".

Changes 1, 2, 3, 4, 7, 8, 11, and 12 above are new values based on revised instrumentation uncertainties resulting from the bypass manifold elimination. The values were calculated using the Westinghouse setpoint methodology as previously approved by the staff for Millstone Unit 3 TS (see SER for Millstone 3, NUREG-1031, Section 7.2.2.2). We find these changes acceptable.

Changes 6, 9, and 13 are values based on revised safety analyses submitted as part of the licensee's letters. We find these changes acceptable.

Change 5 and 10 above are editorial changes resulting from the removal of the RTD bypass manifold. On the basis that these changes add clarification and conciseness to the plant's TS, we find them acceptable.

5.0 SUMMARY

The impact of the RTD bypass elimination for Millstone Unit 3, on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated and found to be acceptable. For the events impacted by the increase in the channel response time, it has been demonstrated that the conclusions presented in the FSAR remain valid. For the remaining Chapter 15 non-LOCA events, the effect of the increased initial RCS average temperature error allowance has been ascertained by separate evaluations. In all instances, the conclusions presented in the Millstone Unit 3 FSAR remain valid under this error allowance assumption and the DNBR limit value is met. The licensee's analysis to support an RCS flow measurement uncertainty value, which includes the new hot leg RTD temperature accuracy, was provided in Reference 2. This analysis was evaluated by the staff to include an RCS flow measurement uncertainty value in the Technical Specifications and was found to be acceptable after some changes were made. Because of possible degradation in RTD response time and calibration, periodic testing is needed to assure that the values assumed are consistent.

The licensee has cleaned the feedwater venturi meters used for the RCS flow measurement calibration prior to the forthcoming Cycle 2 operation. They have committed (Ref. 10) to modify the Technical Specification to indicate that a 0.1% penalty will be applied to the flow measurement uncertainty if the venturi meters are not cleaned. The staff requires that the modifications to the Technical Specifications be submitted and approved by the staff prior to Cycle 3 operation.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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.

7.0 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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