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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113

TO FACILITY OPERATING LICENSE NO. DPR-36

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER PLANT

DOCKET NO. 50-309

1.0 INTRODUCTION

By letter dated December 28, 1988, Reference (1), and as clarified by letter dated May 30, 1989, Reference (16), the Maine Yankee Atomic Power Company (MYAPC) submitted an application to modify the Technical Specifications for Maine Yankee to permit operation of the eleventh cycle at an uprated power level of 2700 MWt. Operation of Maine Yankee at 2630 MWt has already been approved by the staff, Reference (2). A report providing justification for operation of Cycle 11 at 2700 MWt, Reference (3), was submitted with the above letter along with attachments providing a power escalation program, and environmental assessment.

The evaluation of this uprate application is presented herein. In addition, those transients and accidents for which a new or revised analysis has been performed are evaluated in the Safety Analyses Section. The attachments listed above were reviewed and an evaluation of the proposed Technical Specification changes is also presented.

2.0 EVALUATION OF FUEL DESIGN

2.1 Fuel Loading Pattern

The Cycle 11 fuel loading pattern was previously reviewed and approved by the staff, Reference (2). There are no changes to the loading pattern as a result of this uprate.

2.2 Fuel Mechanical Design

The Cycle 11 fuel mechanical design was previously reviewed and approved by the staff, Reference (2). There are no changes to the mechanical design as a result of this uprate.

2.3 Fuel Thermal Design

The licensee's analysis of the fuel thermal performance is the same as that used in the previously approved reload analyses for Cycle 11, Reference (2). The fuel thermal design analyses have been performed at the uprated power level of 2700 MWt using methodology previously approved by the staff and the results are acceptable. As a result, the fuel thermal design analyses for Cycle 11 at the uprated power level of 2700 MWt are acceptable. This finding includes both power-to-centerline melt and core average gap conductance calculations.

3.0 EVALUATION OF PHYSICS DESIGN

3.1 Core Characteristics

Core Characteristics for the uprated power level of 2700 MWt are unchanged with the exception of the cycle average full power lifetime, which is reduced from 12,750 MWD/MTU to 12,650 MWD/MTU.

3.2 Power Distributions

The power increase to 2700 MWt results in a slight increase in radial powers in the center of the core and a slight depression on the core periphery. The result is a maximum increase in peaking 0.3% for the limiting location. Such a small increase is accommodated in the expected core design margin allowance.

As a result, no change in the safety analysis or the resulting allowable unrodded radial peaking in the Technical Specifications is required for 2700 MWt operation. Therefore, the staff, Reference (3), finds this analysis to be acceptable.

3.3 Reactivity Coefficients and Kinetics Parameters

The moderator temperature coefficient (MTC), the fuel temperature coefficient (FTC), the soluble boron and burnable poison shim reactivity effects, and other kinetics parameters for the Cycle 11 core uprated to 2700 MWt are compared with the corresponding values of Cycle 3 (reference cycle) and Cycle 11 at 2630 MWt in the report justifying 2700 MWt operation of Maine Yankee for Cycle 11, Reference (3).

The MTC's at nominal operating HZP conditions are unchanged for 2700 MWt operation. MTC's at HFP are more negative by 0.01 and 0.02x10-4 delta rho per °F for BOC and EOC respectively. No change to the MTC Technical Specification limits are required for 2700 MWt operation. The Cycle 11 Doppler coefficients and defects versus fuel temperature do not change for 2700 MWt operation. Core average fuel temperature does increase by approximately 20°F which results in an increase in the Doppler defect from HZP to HFP of 0.03% delta rho and a decrease in the FTC of 0.01x10-5 delta rho per °F. The critical boron concentrations at HFP for 2700 MWt operation are decreased by 4 ppm at BOC and 8 ppm at EOC. Values of delayed neutron fraction and prompt neutron generation time change by less than 0.1% for 2700 MWt operation.

Since the above data have been calculated using approved methods, are used in the safety analysis with appropriate calculational uncertainties applied in a conservative manner, and are included in the Technical Specifications, the staff finds the data to be acceptable.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

4.1 Thermal-Hydraulic Analysis

The steady-state and transient departure from nucleate boiling (DNB) analyses were performed using the COBRA-III C computer program. COBRA-III C was developed by Battelle Northwest Laboratory for use in the thermal-hydraulic analysis of nuclear fuel elements in rod bundles. The application of COBRA-III C to the Maine Yankee thermal-hydraulic design is described in References 4 and 5. The computer program was also used on a one-eighth core assembly-by-assembly model to determine hot assembly enthalpy rise flow factors. This model was used to account for the difference in hydraulic characteristics between the CE and the ANF fuel assemblies. The inlet flow maldistribution imposed on the model was based on the results of flow measurements taken in scale model flow tests of the Maine Yankee reactor vessel as described in References 6 and 7. The resulting hot assembly flow factors for the CE assemblies was 0.972 for those assemblies with bottom peaked power distributions and 0.990 for those assemblies with top peaked distributions. A 0.95 enthalpy rise flow factor was applied to the ANF fuel assemblies because of the higher spacer loss coefficients relative to the CE fuel. These factors are applied to the inlet mass velocity in the hot channel model in predicting DNB performance. They are unaffected by the power level uprate.

4.2 Fuel Rod Bowing

A parameter which is considered in the thermal-hydraulic design is rod-to-rod bowing within fuel assemblies. The licensee has evaluated the maximum channel gap closure due to fuel rod bowing for both the CE and the ANF fuel assemblies with the highest burnup during Cycle 11. The results of this evaluation are unchanged for 2700 MWt operation. The maximum closure for the CE fuel was calculated to be 24.8%. For the ANF fuel assemblies, the maximum gap closure due to fuel rod bowing was predicted to be less than 33%. Allowances for rod pitch and clad diameter variations due to manufacturing tolerances result in an additional maximum channel closure of approximately 10% for the most adverse conditions. In accordance with the approved methodology Maine Yankee used, no penalty is to be applied to fuel if the predicted gap closure is less than 50%. Therefore, no rod bow penalty is required for any of the fuel in Cycle 11.

5.0 SAFETY ANALYSES

Maine Yankee has reviewed the parameters which influence the results of the transient and accident analyses for Cycle 11 operation at 2700 MWt to determine which events, if any, require a reanalysis. The parameters of importance are initial operating conditions, core power distributions, reactivity coefficients, shutdown CEA characteristics, and reactor protection system (RPS) trip setpoints and time delays. In addition, the effect of a maximum of 250 plugged tubes in each steam generator was evaluated for all events. For those events where the parameters for Cycle 11 are outside the bounds considered in previous safety analyses, a new or revised analysis was performed. These are:

(1) CEA Withdrawal
(2) Boron Dilution
(3) Excess Load

(4) Loss of Feedwater
(5) Loss of Coolant Flow
(6) Full Length CEA Drop
(7) Steam Line Rupture
(8) Seized Rotor
(9) Loss of Coolant Accident (LOCA)
(10) Loss of Load
(11) CEA Ejection

5.1 CEA Withdrawal Event

The CEA withdrawal is an anticipated operational occurrence (AOO) for which the RPS is relied upon to assure no violation of the specified acceptable fuel design limits (SAFDLs). The most severe CEA withdrawal transient occurs for combination of reactivity addition rate and time in core life that results in the slowest reactor power rise to the level just below the Variable Overpower Trip. The reference safety analysis parametric study covered the range of MTCs from +0.5x10-4 delta k/k°F to -3.0x20-4 delta k/k/°F and reactivity addition rates from 0 to 0.7x20-4 delta k/k/sec.

The minimum DNBR for a CEA withdrawal event for Cycle 11 operation at 2700 MWt occurs for a bank withdrawal from an initial power level of 62% of rated power. Protection against violation of the SAFDLs is assured by the Variable Overpower Trip. The minimum DNBR for this event is greater than 1.20 as calculated with YAEC-1 DNB correlation and the peak pressure is less than the American Society of Mechanical Engineers (ASME) design overpressure limit of 2750 psia.

This analysis, using approved methods and assumptions, assures that the SAFDLs are not violated and is, therefore, acceptable.

5.2 Uncontrolled Boron Dilution

An inadvertent boron dilution will reduce the boron concentration in the primary coolant which in turn will increase the reactor core positive reactivity. During power operation, the resulting reactivity insertion will increase the reactor power and automatic safety systems will act to shutdown the reactor and maintain the plant within safety limits. However, a boron dilution event during shutdown will not be mitigated by any automatic safety systems. If it is allowed to continue unmitigated it would result in reactor recriticality unless the operator takes appropriate corrective action to stop the dilution within the necessary time period.

The licensee indicated that the boron dilution event was analyzed for the following operating modes:

(1) refueling
(2) cold shutdown - filled RCS
(3) cold shutdown - drained RCS
(4) hot shutdown - filled RCS
(5) hot shutdown - drained RCS
(6) startup

(7) hot standby
(8) power operation
(9) failure to borate prior to cooldown.

The assumptions made in the Cycle 11 evaluation for 2700 MWt operation are consistent with those made in References 8 and 9. These events were evaluated using a mathematical model that has been previously reviewed and found to be suitably conservative.

For the refueling mode of operation, the limiting dilution was based on the maximum flow of the primary water makeup of 250 gpm. Based on the Cycle 11 core loading, the critical boron concentration under cold conditions (68°F during refueling) is 1346 ppm with the two most reactive CEAs withdrawn or 905 ppm with all CEAs inserted. The minimum initial reactor vessel boron concentration which will prevent an inadvertent criticality within 30 minutes is 1979 ppm with the two most reactive CEAs withdrawn or 1331 ppm for all CEAs inserted. There is, therefore, ample time for the operator to acknowledge the required audible count rate signal and take corrective action.

Dilution during shutdown conditions with the RCS partially drained was addressed in References 9 and 10. The licensee has shown the boron concentrations required to meet the 5% delta k/k Technical Specification subcriticality requirement for shutdown conditions as well as the required initial RCS boron concentrations to allow 15 minutes margin to criticality during filled RCS conditions. The concentrations required by the Technical Specifications conservatively bound those required to meet the 15 minute criterion for margin to criticality during boron dilution events from these conditions.

To evaluate the boron dilution event during hot standby, startup, and power operation for Cycle 11, the licensee indicated that the same assumptions were used as in the analysis in Reference 8 except for the inverse boron worth and higher critical boron concentration (1758 ppm) at hot standby. Based on the maximum reactivity insertion rate, it would take approximately 53 minutes of continuous dilution at the maximum charging rate to absorb the minimum Technical Specification shutdown margin of 3.2% delta k/k.

Failure to add boron during cooldown was evaluated based on conservative values of MTC, initial temperature, and maximum cooldown rate. In order to achieve criticality from these initial conditions, the temperature reduction requires approximately 61 minutes.

Based on the acceptability of the operator response times and the fact that the results haven't changed from the Cycle 11 analysis already approved Reference (2), the staff concludes that the results for Cycle 11 operation at 2700 MWt are acceptable.

5.3 Excess Load Event

The excess load event occurs whenever there is rapid increase in the heat removal from the reactor coolant without a corresponding increase of reactor power. This power-energy removal mismatch results in a decrease of the reactor coolant average temperature and pressure. When the moderator temperature coefficient of reactivity is negative, unintentional increases in reactor power may occur. Therefore, the excess load event as reported in Reference 8 was analyzed over a wide range of power levels and negative MTCs to determine the minimum margin to the linear heat generation rate (LHGR) and DNBR specified acceptable fuel design limits (SAFDLs). The most negative MTC value of -3.17x10-4 delta k/k°F used is slightly less negative than the value predicted for Cycle 11 operation at 2700 MWt of 3.19x10-4 delta k/k°F, including uncertainty. The minimum DNBR for this transient is 1.32 and corresponds to an event initiated from the positive edge of the symmetric offset band at full power and results in a power increase to the variable overpower trip setpoint. The closest approach to fuel centerline melt corresponds to an event initiated from the negative edge of the symmetric offset band near full power and results in a power increase to the variable overpower trip setpoint.

The results of the analysis meet the SRP 15.1.1 criteria and, therefore, are acceptable.

5.4 Loss of Feedwater Event

A loss of feedwater event could be caused by main feed pump failure or feed control valve malfunction. Loss of feedwater flow would result in a decrease in steam generator water level, increase in primary pressure and temperature and reduction in the secondary system capability to remove the heat generated in the reactor core. The event is a heatup transient. The minimum DNBR calculated for this event for Cycle 11 operation at 2700 MWt is 1.51 and peak RCS pressure is bounded by the loss of load transient of less than 2750 psia. For the loss of feed transient occurring from full power with the single failure of one auxiliary feedwater pump, the steam generator level reaches a minimum of 34.0% of the tube bundle height 18.5 minutes after the low level trip occurs. This level provides adequate heat sink throughout the transient.

The results of the analysis meet the SRP 15.2.7 criteria and are, therefore, acceptable.

5.5 Loss of Coolant Flow

The loss of coolant flow transient results are sensitive to initial overpower DNB margin, rate of flow degradation, low reactor coolant flow reactor trip setpoint, available scram reactivity, and MTC. For Cycle 11 operation at 2700 MWt, the thermal power margin for the 100% power Power Dependent Insertion Limit (PDIL) case is lower than the thermal margin for the FSAR design power distribution at full power conditions and, therefore, this event was calculated using the 100% power PDIL power distribution. The assumptions pertaining to rate of flow degradation, low flow trip setpoint, and MTC remain the same as in the reference safety analysis while the available shutdown margin assumed for Cycle 11 bounds the value assumed for the reference safety analysis. The minimum DNBR for the transient is 1.29.

The value meets the criterion as stated in SRP 15.3.1 and 15.3.2 and, therefore, the staff concludes that the results of a loss of coolant flow event occurring during Cycle 11 operation at 2700 MWt are acceptable.

5.6 Full Length CEA Drop Event

The drop of a full length CEA is an Anticipated Operational Occurrence (AOO) which relies on the provision of adequate initial overpower margin to assure no violation of the SAFDLs. The LCO symmetric offset band is designed to restrict permissible initial operating conditions such that the SAFDL for DNB and fuel centerline melt are not exceeded for this event.

In order to cover all potentially limiting conditions, the CEA drop for Cycle 11 operation at 2700 MWt was analyzed from power levels ranging from 0 to 100% of full power. Previous analysis (Reference 8) have shown that the worst full length CEA drop with respect to DNB is the minimum worth CEA that results in the maximum increase in power peaking. Therefore, the Cycle 11 CEA drop evaluation was based on a CEA worth of 0.10% delta k/k. The results of the Cycle 11 uprate DNB evaluation indicate that the limiting full length CEA drop is one initiated from the positive edge of the 100% power symmetric offset LCO band. The minimum DNBR for this event is 0.29, well above the limiting minimum value of 1.20.

With respect to fuel centerline melt, the worst case full length CEA drop is one initiated from power distributions at the edge of the symmetric offset LCO band at each power level. The maximum allowable steady-state linear heat rate required to assure that the maximum linear heat generation rate after the drop does not violate the SAFDL of 23.2 kw/ft (for the fresh fuel) is used in deriving the LCO band on symmetric offset for the RPS.

The safety analyses of the CEA drop event assumes that control of the turbine admission values is performed manually. However, it is possible for the core power to return to a level higher than the pre-drop power level during a CEA drop transient if the turbine admission values are in the automatic pressure control mode (IMPIN) of operation. Therefore, a separate Symmetric Offset operating band has been derived by assuming that the core power returns to the maximum level allowed by the Variable Overpower Trip Setpoint. This reduced operating band applies to the Symmetric Offset trip function whenever the IMPIN mode of turbine control is used.

The results of a CEA drop event meet the criteria stated in SRP 15.4.3 and are, therefore, acceptable.

5.7 Main Steam Line Break

The main steam line break accident was analyzed in detail for Cycle 9 (Reference 11). The analysis was performed with RETRAN-02 MOD 2, which has been approved for use by MYAPC. The analysis assumed a double-ended guillotine break in the main steam line coincident with the worst single failure, a feedwater regulating valve failure. The goal of the analysis was to determine if the core returns to criticality after the initial reactor trip. If the available trip reactivity and boron worth is larger than the reactivity due to moderator and Doppler defects at all times, adequate margin exists to prevent recriticality.

A reanalysis of the hot full power thermal-hydraulic response was performed for the Cycle 11 uprate to 2700 MWt because the thermal-hydraulic characteristics have changed. The zero power conditions for the Cycle 11 power uprate to 2700 MWt remain the same and the reference safety analysis (Reference 11) remains valid.

For Cycle 11 operation at 2700 MWt, the nominal trip reactivity needed to avoid recriticality for HFP and HZP cases at BOC and EOC were determined. In all cases, the required trip reactivities are within the required shutdown margin Technical Specification for Cycle 11.

Since no return to criticality is predicted, the consequences of a main steam line break during Cycle 11 operation at 2700 MWt are acceptable.

5.8 Seized Rotor Accident

The most significant safety parameters which affects the seized rotor accident are the initial overpower DNB margin, core power distribution, radial pin power census, assumed rate of flow degradation, low reactor coolant flow trip setpoint, MTC, and primary-to-secondary leakage flow rate. Most of these factors remain unchanged for Cycle 11 operation at 2700 MWt. The important difference for the Cycle 11 uprate is a reduction in the initial overpower DNB margin. The percentage of fuel experiencing DNB for Cycle 11 operation at 2700 MWt using the Cycle 11 power distribution and the Cycle 11 pin census was less than 13.9% as compared to 10.8% for Cycle 11 analysis performed at 2630 MWt. The radiological release analyses based on these figures would have consequences within the bounds of 10 CFR 100. The staff, therefore, finds this event to have acceptable consequences if occurring during Cycle 11 operation at 2700 MWt.

5.9 Loss of Coolant

For Cycle 11 operation at 2630 MWt, the break spectrum analysis performed for Cycle 10 was found to be applicable as approved by the staff (Reference 2). However, for the uprate to 2700 MWt the Cycle 10 analysis was no longer applicable. Therefore, a new break spectrum calculation for operation in Cycle 11 at 2700 MWt was performed. This new calculation used the same methodology and assumptions as that in the Cycle 10 analysis except; the core power level was increased to 2700 MWt, the Cycle 11 reactor kinetics parameters were used, and the staff approved steam cooling model (Reference 12) was used.

For reach of the limiting breaks, a LOCA calculation was performed with input data specifically for Cycle 11 operation at 2700 MWt. The results of the analysis for each axial power shape indicate that the cladding temperature, cladding oxidation, and hydrogen generation values are in compliance with 10 CFR 50.46 Appendix K criteria.

Previous analyses have shown that small break LOCAs for Maine Yankee are nonlimiting. The results of these previous analyses are determined primarily by the decay heat values which are insignificantly impacted by the increased power level. In addition, since the peak clad temperature and other parameters were calculated to be well below the 10 CFR 50.46 criteria, it would not be significantly affected by either slight differences in core configurations between cycles or power level. The staff, therefore, concludes that the results of previous small break LOCA analyses for Maine Yankee are applicable to Cycle 11 operation at 2700 MWt.

5.10 Loss of Load Event

The loss of load event is an undercooling transient that results from station separation from the grid, turbine trip or electrical generator malfunctions. Subsequent closure of the main steam stop valves causes a large mismatch between reactor power output and heat removal capacity.

The loss of load transient was reanalyzed for the uprate to 2700 MWt. The peak pressure for this event is less than the American Society of Mechanical Engineers (ASME) design overpressure of 2750 psia. The minimum DNBR for this event is greater than 1.20 as calculated with the YAEC-1 DNB correlation.

This analysis, using approved methods and assumptions, assures that the SAFDLs are not violated and is, therefore, acceptable.

5.11 CEA Ejection

As a result of the uprate to 2700 MWt, a complete reanalysis of the CEA ejection event occurring from both HZP and HFP for BOC and EOC core conditions was performed by MYAPC. All cases analyzed resulted in a radially averaged fuel enthalpy below the acceptance criterion prescribed in Regulatory Guide 1.77 (Reference 13). In addition, only 0.7% of the fuel rods were predicted to have clad damage. A bounding radiological release calculation has shown that the resulting off-site doses are within 10 CFR Part 100 limits.

This event, therefore, has acceptable consequences for Cycle 11 operation at 2700 MWt.

6.0 CONTAINMENT OVERPRESSURE ANALYSES

The containment overpressure analysis was reviewed by MYAPC for operation at 2700 MWt. The peak containment pressure is primarily a function of the total energy discharged into the containment during the blowdown phase of a LOCA. The allowed primary system conditions for operation at 2700 MWt are designed to maintain approximately the same or less energy discharged to the containment for this event. Specifically, for 2700 MWt operation the average temperature in the primary system has been designed to remain unchanged from previous cycles, hence the stored energy of the fluid remains unchanged.

This is accomplished by restricting the core inlet temperature to 551.3°F, a value which maintains a core average temperature equivalent to operation at 2630 MWt.

In total the changes due to the increase in power level to 2700 MWt amount to a maximum perturbation on energy discharge to containment of less than 0.5% and their overall impact is negligible.

Based on the restriction to core inlet temperature of 551.3°F to essentially negate the effect of the power level increase on the containment overpressure analysis, the staff concludes that the containment design pressure will not be violated and that the results for Cycle 11 operation at 2700 MWt are acceptable.

An NRC team performed a Safety System Functional Inspection from January 9 through February 10, 1989 at Maine Yankee Atomic Power Station (Inspection Report 50-309/89-80 dated March 27, 1989). During that inspection, it was observed that the plant license had been amended to allow a power uprate from 2440 MWt to 2630 MWt with no updated analyses to verify the Component Cooling Water (CCW) System capabilities at this new rating. The licensee, by letter dated May 30, 1989, Reference (16), submitted the results of a conservative bounding analysis which justified operation at 2700 MWt for the CCW Systems.

In that submittal, the licensee examined the CCW System, the Residual Heat Removal (RHR) system and the Service Water (SW) System and the containment temperature following a loss-of-coolant accident. The performance of those systems and the containment temperature was analyzed as a function of the SW System. The CCW System has two trains made up of two new CCW heat exchangers and of two older CCW heat exchangers. The analyses showed that, with the new CCW heat exchangers on line, the above systems and structure could perform their safety-related functions with SW inlet temperatures of 75° while with the older CCW exchangers on line, SW inlet temperatures should be limited to 60°F. The licensee committed to observe these SW limits pending completion of more detailed analysis of CCW performance.

The licensee engaged Stone and Webster Engineer Corporation (SWEC) to perform an independent review of the boundary analysis. The SWEC review consisted of a detailed review of the calculation (not including a number check) and independent verification of the results using the Heat Transfer Research, Inc. computer program. The results correlated well with the Maine Yankee analysis.

This resolves staff concerns regarding the CCW system in relation to the power uprate to 2700 MWt.

7.0 RADIOLOGICAL EVALUATION AND REVIEW

A review of the 21 Design Basis Accidents (DBAs) discussed in (References 8, 14, and 15) by MYAPC indicates that the design objectives for Pressurized Water Reactor DBAs specified in the SRPs will not be exceeded nor significantly impacted by the Maine Yankee power uprate to 2700 MWt. A review of the impact on normal operating radioactivity releases has also been performed, with no direct impact for the escalated power level. The effects on the Maine Yankee Equipment Qualification (EQ) Program was also conducted, with no equipment change requirements identified for Cycle 11 operation at 2700 MWt.

Based on our review of the above findings, the staff concludes that the radiological consequences of the increase in power to 2700 MWt are not significantly changed from Cycle 11 operation at 2630 MWt and are all within 10 CFR Part 100 guidelines, and therefore, are acceptable.

8.0 ENVIRONMENTAL EVALUATION AND REVIEW

A Notice of Issuance of an Environmental Assessment and Finding of No Significant Impact relating to the proposed uprate of the core power level for the Maine Yankee Atomic Power Plant was published in the *Federal Register* on June 22, 1989 (54 FR 26281).

9.0 TECHNICAL SPECIFICATION CHANGES

The licensee has proposed (Reference 1) several changes to the Technical Specifications for the Cycle 11 power level uprate to 2700 MWt. The staff's review and evaluation of these changes follows with numbering corresponding to that presented in Reference 1.

1. Technical Specification Definitions, page 2

   (a) The steady state reactor core output has been modified. This change is acceptable because the modification reflects the change in reactor core output associated with the uprate in power level to 2700 MWt.

2. Technical Specification 2.1

   (a) The equation for thermal margin/low pressure has been modified in the text and in the associated Figures 2.1-1a and 2.1-1b. This change is acceptable because it reflects the setpoint calculations results for the power uprate to 2700 MWt.

3. Technical Specification 2.2

   (a) The steady state peak linear heat rates have been modified. This change is acceptable because the modification reflects the Cycle 11 SAFDLs for the prevention of centerline melting for the power uprate to 2700 MWt.

4. Technical Specification 3.10

   (a) A reduction in the LHR limits has been removed as an action from Technical Specification 3.10.C.2.2.2 if the measured value of total radial peaking factor exceeds the value given in Figure 3.10-4. This change is acceptable as it simplifies the actions for the Technical Specification and is accounted for by a reduction in the allowable kw/ft limit in Figure 3.10-11.

   (b) The PDIL given by Figure 3.10-1 has been modified. This change is acceptable because it reflects the Cycle 11 CEA insertion limits produced by the uprate analysis.

(c) The allowable power level versus the increase in total radial peak, Figure 3.10-5, has been modified. This change is acceptable since it correctly reflects the Cycle 11 power distribution and RPS Setpoints for 2700 MWt operation.

(d) The allowable coolant conditions, Figure 3.10-6, have been modified. This change is acceptable because it is consistent with the allowable coolant conditions assumed in the 2700 MWt uprate analysis.

(e) The allowable linear heat generation rate versus core height, Figure 3.10-11 has been modified. This change is acceptable because it reflects the results of the uprate analysis and also allows for simplification of Technical Specification 3.10.C.2.2.2.

10.0 EVALUATION FINDINGS

The staff has reviewed the information in the Maine Yankee Cycle 11 uprate report and attachments. The staff finds the proposed power level uprate to 2700 MWt and the associated modified Technical Specifications acceptable.

11.0 ENVIRONMENTAL CONSIDERATION

Notice of Consideration by the staff of issuance of the proposed amendment was published in the Federal Register February 1, 1989 (54FR5165) and no comments or requests for hearing were received. The Commission also consulted with the State of Maine and no comments were received. An Environmental Assessment (EA) and Finding of No Significant Impact was published in the Federal Register on June 22, 1989 (54FR26281). Based upon the EA, the staff has determined not to prepare an environmental impact statement for the proposed license amendment, and has concluded that the proposed action will not have a significant adverse effect on the quality of the human environment.

12.0 CONCLUSION

We have concluded, based on the consideration discussed above, that (1) there is a 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.

13.0 REFERENCES

1. Maine Yankee letter to USNRC, MN-88-133, December 28, 1988.

2. USNRC letter to MYAPC, Amendment No. 107 to DPR-36, September 27, 1988.

3. "Justification for 2700 MWt Operation of Maine Yankee for Cycle 11," YAEC-1662, Yankee Atomic Electric Company, December 1988.

4. "Maine Yankee Core Reactor Protection System Setpoint Methodology," YAEC-1110, September 1976.

5. "Maine Yankee Core Thermal-Hydraulic Model Using COBRA III C," YAEC-1102, June 1976.

6. "The Hydraulic Performance of the Maine Yankee Reactor Model," TR-DT-34, Combustion Engineering, June 1971.

7. Maine Yankee Atomic Power Station Final Safety Analysis Report (FSAR).

8. "Justification for 2630 MWt Operation of the Maine Yankee Atomic Power Station," YAEC-1132, July 1977.

9. Maine Yankee letter to USNRC, WMY 78-2, January 5, 1978.

10. Maine Yankee letter to USNRC, MN-82-53, "Boron Dilution During Hot and Cold Shutdown (Mode 5 Operation)," March 18, 1982.

11. "Maine Yankee Cycle 9 Core Performance Analysis," YAEC-1479, April 1985.

12. USNRC letter to MYAPC, "Evalution of Maine Yankee Steam Cooling Model for Large Break," June 28, 1988.

13. "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," NRC Regulatory Guide 1.77, May 1974.

14. Maine Yankee Final Safety Analysis Report (FSAR), Chapter 14-Safety Analysis.

15. "Maine Yankee Cycle 11 Core Performance Analysis," YAEC-1648, Yankee Atomic Electric Company, July 1988.

16. Maine Yankee letter to USNRC, MN-89-74, May 30, 1989.

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DATE July 10, 1989

AMENDMENT NO. 113 TO DPR-36 - MAINE YANKEE ATOMIC POWER STATION

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