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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATING TO AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. DPR-36

## MAINE YANKEE ATOMIC POWER COMPANY

## MAINE YANKEE ATOMIC POWER STATION

## DOCKET NO. 50-309

## 1.0 INTRODUCTION

By letter dated January 12, 1987 (Ref. 1), the Maine Yankee Atomic Power Company (MYAPC) submitted an application to modify the Technical Specifications for Maine Yankee to permit operation for a tenth cycle. A Cycle 10 core reload report (Ref. 2) was also submitted with the above letter. The fuels, physics, and thermal-hydraulic evaluations of this reload report are 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. An evaluation of the proposed Technical Specification changes is also presented.

## 2.0 EVALUATION

### 2.1 Fuel System Design

The Cycle 10 reload application involves fuel designs similar to those previously considered for the Maine Yankee reactor. The Maine Yankee Cycle 10 core will consist of 217 fuel assemblies with fuel rods arranged in 14 by 14 arrays. Of the five fuel types proposed for use in the Cycle 10 core, three fuel types were fabricated by Combustion Engineering (CE) and two were fabricated by Advanced Nuclear Fuels Corp. (ANF). Two of the CE fuel types, Type E and N, consist of 73 previously irradiated fuel assemblies. The Type P CE fuel consists of 72 unirradiated assemblies with an increased enrichment of 3.5 weight percent U-235. This higher enrichment was previously reviewed and approved by the staff (Ref. 3). The ANF fuel, denoted Types L and M, consists of 72 fuel assemblies in the Cycle 10 core which were previously irradiated during Cycles 7, 8 and 9.

As in Cycle 9, the Cycle 10 core will contain 81 control element assemblies (CEA's) of which four are non-scrammable. In addition to the CEA's, the Maine Yankee Cycle 10 core will also contain burnable poison rods in selected assemblies. There will be 1072 standard $B_4C$-$Al_2O_3$ burnable poison rods in Cycle 10 compared to 1264 in the previous cycle.

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## 2.1 Fuel Mechanical Design

The mechanical design features of both CE and ANF fuel assemblies to be used in the Cycle 10 core are listed in Table 3.3 of Reference 2. The fresh reload fuel, Batch P, being inserted in Cycle 10 is similar to the previously supplied reload fuel with one exception, a modification to the poison pin design. Compared to the previous CE Batch N, the overall length of the poison rods is decreased to allow for additional rod growth clearance. Also, additional free volume was provided to reduce the calculated internal poison pin pressure. The free volume was provided by replacing the two solid internal spacers with two hollow spacers. Since Batches L and M, the ANF supplied fuel, will achieve exposures higher than previously encountered at Maine Yankee, an extended burnup analysis (Ref. 4) was performed in order to demonstrate compliance with the appropriate design criteria of these higher exposures. The results of the extended burnup analysis for the ANF fuel batches are reported in Reference 2. These include evaluations of fuel cladding collapse, irradiation induced dimensional changes, cladding strain and fatigue analysis, maximum fuel rod internal pressure and fuel rod corrosion. The results indicate the primary stress in the cladding will not exceed the design stress limit, the collapse resistance of the fuel rods is sufficient to preclude collapse during the projected lifetime of the fuel, and the predicted fuel rod internal gas pressure remains below reactor coolant system (RCS) pressure throughout the projected lifetime of the fuel. These results were calculated with methods which have been approved by the staff (Ref. 5) and are in conformance with the requirements of Standard Review Plan (SRP) 4.2. The CE fuel mechanical design was approved for earlier cycles and is applicable to Batch P fuel. The staff, therefore, concludes that the Maine Yankee Cycle 10 fuel mechanical design is acceptable.

### 2.2 Fuel Thermal Design

The licensee's analysis of the fuel thermal performance is the same as that used in previous reload analyses including the use of power history effects and burnup-dependent fission gas release. The Batch L fuel was not analyzed explicitly because its exposure and power histories were bounded by Batch M. The fuel thermal design analyses have been performed using methodology previously approved by the staff. As a result, the fuel thermal design analyses for Cycle 10 are acceptable. This finding includes both power-to-centerline melt and core average gap conductance calculations.

## 3.0 PHYSICS DESIGN

### 3.1 Core Characteristics

Maine Yankee Cycle 10 incorporates a low-leakage design, achieved by placement of fresh (unirradiated) fuel assemblies in selected core interior locations and burned (irradiated) fuel assemblies on the core periphery. In addition to reducing the irradiation exposure to the reactor pressure vessel, this low-leakage core design also produces a less severe moderator defect with cooldown at end-of-cycle (EOC), improves the stability of the core to axial xenon oscillations near EOC, and extends the achievable full power lifetime of the cycle. Cycle 10 is expected to attain a cycle average full power lifetime of 11,900 MWD/MTU.

### 3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power densities are given in Reference 2 for beginning-of-cycle (BOC)(50 MWD/MTU), middle-of-cycle (MOC) (6000 MWD/MTU), and EOC (13,000 MWD/MTU) conditions for both unrodded and rodded (CEA Bank 5 in) configurations. These results show that the unrodded maximum 1-pin radial peak power occurs at BOC when its value is 1.53. The proposed Technical Specification change, shown in Figure 3.10-4 and giving the allowable unrodded radial peaking, including 10 percent calculational uncertainty, as a function of average exposure for the Cycle 10 core, indicates radial peaks in the range from 1.761 to 1.752. Comparison of the radial peaks given in the above power distributions with the allowable values shown in the Technical Specifications demonstrates the adequacy of the results given in the core performance analysis (Ref. 2). The staff, therefore, finds this analysis to be acceptable.

### 3.3 Reactivity Coefficients and Kinetics Parameters

The moderator temperature coefficient (MTC), the fuel temperature (Doppler) coefficient, the soluble boron and burnable poison shim reactivity effects, and other kinetics parameters for the Cycle 10 core are compared with the corresponding values of Cycle 3 (reference cycle) and Cycle 9 (previous cycle) in the Cycle 10 Core Performance Analysis Report (Ref. 2).

The MTC's at nominal operating HFP and HZP, BOC conditions are more negative than the corresponding previous cycle (Cycle 9) values primarily because of the decreased BOC critical boron concentration resulting from less excess reactivity in the core. The EOC values are also more negative than in the previous cycle due primarily to the increased core average exposure.

Technical Spec'fication MTC limits are provided for Cycle 10 based on the LOCA analysis moderator density defect curve. This curve infers specific MTC values in the operating range which must not be exceeded for the LOCA analysis to remain valid. The Cycle 10 Doppler coefficients are quite similar to the Cycle 3 and Cycle 9 values. The critical boron concentrations for Cycle 10 at BOC are less than those of Cycle 9 because of the smaller amount of excess reactivity in the core. Values of the delayed neutron fraction and prompt neutron generation time for Cycles 3, 9, and 10 are comparable and the differences reflect the effects of core average exposure and power weighting.

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

## 3.4 Control Requirements

The value of the required shutdown margin is determined either from the steam line break analysis (EOC) or from other safety analyses (BOC). Based on these values of required shutdown margin and on calculated available scram reactivity including a maximum worth stuck rod and appropriate calculational uncertainties, sufficient excess exists between available and required scram reactivity for all Cycle 10 operating conditions. These results are derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable. The Power Dependent Insertion Limits (PDILs) for CEA's are given in the Technical Specifications and are required to provide for sufficient available scram reactivity at all power levels during the cycle. An allowance for the PDIL CEA worth is made when determining the available scram CEA worth.

## 3.5 Augmentation Factors

Axially dependent flux augmentation factors had previously been incorporated as a power spike penalty in the calculation of the core power-to-incipient fuel centerline melt. As shown in Reference 6, these factors are no longer required for modern design PWR fuel fods and the staff has previously approved their removal from Maine Yankee (Ref. 7).

## 4.0 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 8 and 9. 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 accounted 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 10 and 11. The resulting hot assembly flow factors for the CE assemblies was 1.0. 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.

### 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 10. The maximum closure for CE fuel was calculated to be 22%. For the ANF fuel assemblies, the maximum gap closure due to fuel rod bowing was predicted to be less than 32%. 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 10.

## 5.0 SAFETY ANALYSES

Maine Yankee has reviewed the parameters which influence the results of the transient and accident analyses for Cycle 10 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 250 plugged tubes in each steam generator was evaluated for all events. For those events where the parameters for Cycle 10 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 Load
(5) Loss of Feedwater
(6) Loss of Coolant Flow
(7) Full Length CEA Drop
(8) Steam Line Rupture
(9) Seized Rotor
(10) CEA Ejection

Proposed changes to the Maine Yankee loss of coolant accident (LOCA) methods for Cycle 10 have been reviewed in a separate safety evaluation and are not included herein.

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 (SAFDL's). The most severe CEA withdrawal transient occurs for a 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 MTC's from $+0.5 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ to $-3.0 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ and reactivity addition rates from 0 to $0.7 \times 10^{-4}$ $\Delta k/k/sec$.

The minimum DNBR for a CEA withdrawal event for Cycle 10 occurs for a bank withdrawal from an initial power level of 96.4% of rated power. Protection against violation of the SAFDL's is assured by the Variable Overpower Trip. The minimum DNBR for this event is 1.47 as calculated with the 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 SAFDL's 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 shut down 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 license 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 10 evaluation are consistent with those made in References 12 and 13. 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 10 core loading, the critical boron concentration under cold conditions (68°F) during refueling is 1109 ppm. The minimum initial reactor vessel boron concentration which will prevent an inadvertent criticality within 30 minutes is 1651 ppm, which, therefore, is required for refueling. There is, therefore, ample time for the operator to acknowledge the audible count rate signal and take corrective action.

Dilution during shutdown conditions with the RCS partially drained was addressed in References 13 and 14. The licensee has shown the boron concentrations required to meet the 5% Δk/k Technical Specification sub-critically requirement for shutdown conditions as well as the required initial RCS boron concentrations to allow 30 minutes margin to criticality during drained RCS conditions. The licensee has stated that administrative procedures ensure that the higher of these two values are used and, therefore, a minimum margin to criticality of 30 minutes would be available for the operator to take appropriate action in the event of a limiting boron dilution from drained conditions.

To evaluate the boron dilution event during hot standby, startup, and power operation for Cycle 10, the licensee indicated that the same assumptions were used as in the analysis in Reference 12 except for the inverse boron worth and higher critical boron concentration (1571 ppm) at hot standby. Based on the maximum reactivity insertion rate, it would take approximately 59 minutes of continuous dilution at the maximum charging rate to absorb the minimum Technical Specification shutdown margin of 3.2% Δk/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 72 minutes.

Based on the acceptability of the operator response times and comparison with Cycle 9 analysis, the staff concludes that the results for Cycle 10 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. The transient which causes the most severe power excursion has been identified by the licensee as the steam dump and bypass system malfunction at hot standby and at EOC where the MTC is most negative. The excess load transient had been reanalyzed for Cycle 4 in Reference 15. In that analysis, a MTC of $-3.17 \times 10^{-4}$ $\Delta k/k^{\circ}F$ was assumed. This value is more negative than that predicted for Cycle 10, including uncertainty. The minimum DNBR for this transient is 1.42 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 results of the analysis meet the SRP 15.1.1 criteria and, therefore, are acceptable.

5.4 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. System parameters which have a major impact on the severity of the pressure excursion are the initial power level, initial RCS pressure, steam generator pressure, primary and secondary safety relief valve capacities and setpoints, high pressurizer pressure reactor trip setpoint, and MTC. Except for one change, the Cycle 10 limiting values for these parameters are the same as or bounded by those assumed in the Cycle 9 analysis. For Cycle 10, the number of plugged steam generator tubes was increased to 250 from 180. The peak RCS pressure remains below 2750 psia and the minimum DNBR is 1.92.

Since these values are within the NRC acceptance criteria of SRP Section 15.2.1, the results of a loss of load event during Cycle 10 are acceptable.

5.5 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 10 is 1.61 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 36.7% of the tube bundle height 19.3 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.

5.6 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 10, the thermal power margin for the 100% power 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 10 bounds the value assumed for the reference safety analysis. The minimum DNBR for the transient is 1.38.

This 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 10 are acceptable.

5.7 Full Length CEA Drop Event

The drop of a full length CEA is an AOO which relies on the provision of adequate initial overpower margin to assure no violation of the SAFDL's. 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 10 was analyzed from power levels ranging from 0 to 100% of full power. Previous analyses (Ref. 12) 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 10 CEA drop evaluation was based on a CEA worth of 0.10% $\Delta k/k$. The results of the Cycle 10 DNB evaluation indicate that the limiting full length CEA drop is one initiated from the positive edge of the 100% power symmetric offset LCO alarm band. The minimum DNBR for this event is 1.42, 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 23.1 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 valves 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 valves 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.

5.8 Main Steam Line Break

The main steam line break accident was analyzed in detail for the previous cycle (Ref. 16). 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.

For Cycle 10, 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 10.

Since no return to criticality is predicted, the consequences of a main steam line break during Cycle 10 are acceptable.

5.9 Seized Rotor Accident

The most significant safety parameters which affect 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 10. The important differences are a reduction in the initial overpower DNB margin, differences in the radial pin power census, and a more limiting 100% power PDIL power distribution. The percentage of fuel experiencing DNB using the Cycle 10 power distribution and the Cycle 10 pin census was less than 10.3% as compared to 7.5% for Cycle 9. The licensee also states that the radiological release analyses based on these figures would have consequences within the bounds of 10 CFR Part 100. The staff, therefore, finds this event to have acceptable consequences if occurring during Cycle 10.

5.10 CEA Ejection Event

As a result of the higher ejected CEA worths and the increased post ejection power peaking, the CEA ejection physics parameters have become more limiting for Cycle 10 as compared to those assumed in the reference safety analyses. Therefore, a reanalysis of the CEA ejection event occurring from both HZP and HFP for BOC and EOC core conditions was performed by MYAPC. This reanalysis made use of recently approved revisions to the methodology (Ref. 17). All cases resulted in a radially averaged fuel enthalpy below the acceptance criterion of 280 cal/gm prescribed in Regulatory Guide 1.77 (Ref. 18). In addition, only 3.6% 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 10.

## 6.0 TECHNICAL SPECIFICATION CHANGES

The licensee has proposed (Ref. 1) several changes to the Technical Specifications for the Cycle 10 reload core. The staff's review and evaluation of these changes follows with the numbering corresponding to that presented in Reference 1.

1. 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 10 SAFDL's for the prevention of centerline melting.

   (b) The text has been slightly modified. This change is acceptable as it clarifies that each fuel type has its own LHGR limit.

2. Technical Specification 3.10

   (a) The Power Dependent Insertion Limit (PDIL) for CEA's, Figure 3.10-1, has been modified. This change is acceptable because it reflects the Cycle 10 CEA insertion limits produced by the reload analysis.

   (b) The Allowable Unrodded Radial Peak Versus Cycle Average Burnup, Figure 3.10-4, has been modified. The change is acceptable as it reflects Cycle 10 radial peaking.

   (c) The Allowable Power Level vs. Increase in Total Radial Peak, Figure 3.10-5, has been modified. This change is acceptable since it reflects Cycle 10 power distributions and RPS setpoints.

## 7.0 EVALUATION FINDINGS

The staff has reviewed the information presented in the Maine Yankee Cycle 10 reload report and in the MYAPC responses to the staff request for additional information. The staff finds the proposed reload and the associated modified Technical Specifications acceptable.

## 8.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

## 9.0 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.

## 10.0 REFERENCES


1. J. B. Randazza (MYAPC) letter to the U. S. Nuclear Regulatory Commission, MN-87-04, January 12, 1987.

2. "Maine Yankee Cycle 10 Core Performance Analysis," YAEC-1573, Yankee Atomic Electric Company, December 1986.

3. USNRC Letter to J. B. Randazza (MYAPC), Amendment No. 92 to DPR-36, February 9, 1987.

4. "Mechanical Design Report Supplement for EXXON Nuclear Maine Yankee XN-3 and XN-4 Extended Burnup Program," XN-NF-86-94, September 1986.

5. Safety Evaluation of the EXXON Nuclear Company Topical Report, XN-NF-86-06(P), "Qualification of EXXON Nuclear Fuel for Extended Burnup," July 1986.

6. MYAPC Letter to USNRC, MN-86-69, "Augmentation Factor Removal," May 20, 1986.

7. USNRC Letter to MYAPC, "Augmentation Factor Removal," with attached Safety Evaluation June 20, 1986.

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

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

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

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

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

13. Maine Yankee Letter to USNRC, WMY 78-2, January 5, 1978.

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

15. Maine Yankee Letter to USNRC, WMY 78-62, "Maine Yankee Proposed Change No. 64," June 26, 1978.

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

17. "Modified Method for CEA Ejection Analysis of Maine Yankee Plant," YAEC-1464, December 1984.

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


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