UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.48 TO FACILITY OPERATING LICENSE NO. DPR-36

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER STATION

DOCKET NO. 50-309

1.0 Introduction & Background

By letter dated December 5, 1979 (Reference 1), as supplemented February 15, 1980 (Reference 21), Maine Yankee Atomic Power Company (MYAPC or the licensee) requested amendment of Appendix A to Facility Operating License No. DPR-36 for the Maine Yankee Atomic Power Station. The proposed changes amend the Technical Specifications to permit operation of the Cycle 5 core.

The Cycle 5 core will consist of 72 fresh Exxon Nuclear Company (ENC) fuel bundles and 145 once and twice burnt Combustion Engineering (CE) fuel bundles. Five of these depleted fuel bundles have been reconstituted (see Section 2.1.1). The forthcoming fuel cycle is the first fuel cycle at Maine Yankee to utilize ENC fuel. All anticipated operational occurrences and accidents were assessed and new analyses performed where needed. The Cycle 5 analysis was performed by Yankee Atomic Electric Company (YAEC) for the licensee.

The Loss of Coolant Accident (LOCA) was reanalyzed, using NRC approved methods, to demonstrate compliance with acceptance criteria. Reanalysis was necessitated by the introduction of ENC fuel in the core for Cycle 5. The ENC fuel was found to be limiting with respect to LOCA. Additional licensee analysis was performed to assess the potential impact of recent NRC concerns related to the LOCA-ECCS

(Emergency Core Cooling System) fuel clad models as presented in our draft report NUREG-0630 (Reference 2) (see Section 3).

2.0 Evaluation of Modifications to Core Design

2.1 Fuel Systems

2.1.1 Exposed Fuel

Once and twice burnt fuel fabricated by CE has been previously reviewed. During Cycle 4, higher than normal coolant activity levels were detected and reported (References 3, 4, 5). During the Cycle 4/5 refueling operation, all 217 fuel assemblies which resided in the core during Cycle 4 were examined to detect any leaking fuel. The examination is termed "fuel sipping." Nine of 217 Cycle 4 fuel assemblies were found to contain leaking fuel rods. Of these nine, five assemblies were scheduled to be utilized during Cycle 5. Eddy current testing of the ı 1 pins within these 5 assemblies detected 10 leaking fuel pins. Hence less than 0.1% of the fuel pins were found defective. Once burnt, defective fuel pins have been replaced with 1.95 weight percent uranium-235 fresh fuel pins. Twice burnt, defective fuel pins have been replaced by solid zircaloy rods. The replacement rods will not significantly perturb either the local and core wide power distributions or the local or core wide reactivity and hence represent an acceptable replacement program.

The failure of a small fraction of the fuel rod cladding is not atypical. Nevertheless, such failures raise the concern that an incipient fuel clad failure mode of core wide significance might exist. This is not believed to be the case. Coolant activity levels indicative of clad failures were observed at Maine Yankee during Cycle 4 almost a year ago. Since then the plant has experienced load reductions and trips. If an unidentified incipient fuel clad failure mode had existed, these maneuvers should have resulted in further fuel clad failures. Coolant activity level increases associated with further fuel degradation were not observed.

Coolant activity levels will be closely monitored during Cycle 5. Plant Technical Specification 4.2 requires sampling tests at least every 14 days when coolant activity levels are within specification and the plant remains at power, and more frequently otherwise. Plant practice is to take almost daily samples. Hence early detection of further fuel clad failure, should it occur, is assured.

2.1.2 Fresh Fuel

Seventy-two ENC fabricated fuel assemblies will be used in Cycle 5. The fuel assemblies are similar in design to the CE fabricated fuel assemblies. The most significant differences are: (1) an increased fuel rod cladding thickness, 31 mils vice 28 mils and (2) the use of bimetallic grid spacers of Zircaloy-4 structure and Inconel springs vice an all Zircaloy-4 grid. Minor differences in the diameter, height, dishing, initial density of the fuel pellets and poison rod pellets, and initial fuel rod pressure also exist. Loading tests of the ENC spacer grids to confirm their strength, and out-of-pile flow testing to address fretting (the interaction of the fuel rod clad and grid spacer springs) have been performed. Hydraulic loss coefficients were also determined from the later tests (see Section 2.2). We conclude that the mechanical and hydraulic properties of the ENC fuel are not significantly different from the CE fuel, the design of which has been previously accepted by the staff.

2.1.3 Guide Tube Wear

All worn fuel assembly guide tubes have been modified by the installation of stainless steel guide tubes. Sleeves have also been inserted within guide tubes in assemblies that have not resided under control rods and in fresh fuel assemblies. The CE and ENC fabricated guide tubes are similar. The results of inspections of the Cycle 4 guide tubes and guide tube inserts indicate no additional wear. The problem was resolved by changing the diameter of the flow opening in the bottom of all the fuel assemblies.

2.2 Thermal Hydraulic Design

Fuel thermal performance calculations were performed by YAEC using previously approved methods (References 6, 7). Studies of the ENC fuel pellet and clad temperatures, and gap conductance, in conjunction with power and fuel rod exposure result in a 21 kw/ft limit on linear heat generation rate (LHGR) for avoiding fuel centerline melt. These studies had previously been performed for the CE fabricated fuel (References 8, 9).

Critical heat flux calculations were performed by YAEC using reviewed methods (References 10, 11, 12) for the ENC and CE fuel. The ENC fuel assemblies are predicted to exhibit a higher flow resistance than the CE fuel assemblies and hence less flow (95.7% of core average flow in ENC bundles and approximately 102% of core average flow in CE bundles). Credit has not been taken for the increased flow in the CE fuel assemblies. The relative decreased flow in the ENC bundles and in turn reduced margin to departure from nucleate boiling ratio (DNBR) was considered by YAEC in their safety analyses (Section 3) and reactor protective system setpoint analyses (Reference 10).

2.2.1 Fuel Rod Bowing

The NRC fuel rod bowing model applied to Maine Yankee (Reference 13) requires a DNBR reduction for bundle average exposure greater than 24 GWD/T. CE fuel bundles with exposure greater than 24 GWD/T are predicted to exhibit significantly lower powers and hence greater DNBR overpower margin than the limiting fuel assemblies in the core. Hence a DNBR penalty for rod bowing is not required for these high burnup bundles. Similarly ENC fuel bundles are predicted to obtain an end of Cycle 5 burnup of less than 16 GWD/T and hence are not predicted to exhibit rod bow. We conclude Cycle 5 fuel will not exhibit any manifestations of rod bow.

2.3 Nuclear Design

The core loading arrangement is typical of a conventional three batch fuel management scheme. One third of the core, 72 of 217 fuel assemblies, will be replaced with 3.00 w/o U^{235} fresh fuel assemblies. One hundred seventy six fresh burnable poison rods, displacing fuel rods in the fresh assemblies, have been employed to tailor the radial power distribution.

Small differences between Cycle 4 and Cycle 5 predicted control rod worths are an inherent consequence of fuel rearrangement. Reported differences in shutdown reactivity, rod group worths, the ejected rod worth, dropped rod worth, and associated peaking factors are typical of reload cores of all pressurized water reactors. Similarly, reported values, and differences with past cycles, of kinetics parameters are typical. Changes of kinetics parameters are due to changes in the core average enrichment, excess reactivity and hence soluble boron concentration, and core average burnup and hence neutron energy spectrum.

Predicted values were obtained from YAEC calculations, performed using previously approved methods. Predicted or expected values for Cycle 5 are less severe than bounding values of these parameters assumed in past safety analyses except where indicated in Section 3.0.

3.0 Safety Analyses

All postulated anticipated operational occurrences and accidents were reviewed, reassessed or reanalyzed as described below. The Cycle 3 analysis, the justification for 2630 MWt operation, i.e., power upgrade (Reference 14), has been used as the reference cycle. Where minimum predicted departure from nucleate boiling ratio (MDNBR) is used as an acceptance criteria, a reassessment was required to consider the reduced DNBR overpower margin of ENC fuel relative to CE fabricated fuel.

Except where indicated the design power distribution (F_Z=1.68, $F^n_{\Delta H}$= 1.49) for DNBR calculations, and the reference cycle scram reactivity have been used in the current assessment. The assumed normalized scram reactivity as a function of time, the power dependent rod insertion limits, and permissible steady state coolant conditions have not been changed. Calculational methods have previously been reviewed and accepted.

3.1 Anticipated Operational Occurrences

The Inadvertent Control Element Assembly (CEA) Withdrawal, Loss of Load, and Loss of Feedwater event system responses are predicted to be bounded by the reference cycle analyses, and the Excess Load Increase by the Cycle 4 analysis (Reference 15). Reassessment of DNBR due to the introduction of ENC fuel results in a reduction of MDNBR of less than 0.1 units of DNBR. These events are not limiting with respect to DNBR.

Idle Loop Startup and Part Length CEA Drop events were not analyzed for Cycle 5. Operation with less than all Reactor Coolant Pumps running is prohibited by the plant Technical Specifications. The Part Length Rods have been locked in the full out position.

The Boron Dilution event was reassessed to account for changes in the critical boron concentration during refueling and startup. Based on parametric studies presented in the reference safety analysis, starting at a boron concentration of 1490 ppm, a continuous dilution for 30 minutes would be required to achieve criticality during refueling, and a continuous dilution for 1.7 hours would be required to achieve criticality during startup. These results are acceptable. Boron dilution at hot standby and at power is bounded by the reference analysis.

The Loss of Coolant Flow and Full Length CEA Drop (Rod Drop) are limiting DNBR events and establish operating limits on required overpower margin.

The dropped CEA event was reanalyzed assuming a maximum increase in peaking due to the dropped rod of 15.8% (a typical value for PWRs) and a minimum rod worth of 0.05% $\Delta k/k$. The design power distribution as well as realizable power distributions within the axial offset alarm limits were considered in the assessment of DNBR and fuel centerline melt. The MDNBR at 100% power using the design power distribution is greater than 1.43. Realizable power distributions within the axial offset alarms result in greater DNBR overpower margin than the design power distribution. To insure that the Specified Acceptable Fuel Design Limit, 21 kw/ft linear heat generation limit to centerline melt, is not violated, the initial heat rate will be limited to 18.1 kw/ft. This restriction is imposed by compliance with the limiting condition for operation based on symmetric offset.

The Loss of Coolant Flow event was reanalyzed due to the introduction of ENC fuel. The design power distribution, normalized scram reactivity, pump coastdown flow vs time, low flow trip setpoint, and moderator temperature coefficient were assumed to be the same as the reference analysis. Degradation of the initial DNBR due to the ENC fuel, relative to the reference analysis, has been offset by credit for the larger scram reactivity predicted to be available in Cycle 5 relative to the reference analysis. The MDNBR for the three pump coastdown from 100% power is predicted to be 1.53. Use of realizable axial power distributions within the axial offset alarm limits in the analysis would result in a larger predicted MDNBR. The Cycle 5 scram reactivity will be confirmed by the start-up test program.

3.2 Accidents

3.2.1 Steam Line Break

The Steam Line Break accident was reanalyzed using the end of Cycle 5 moderator defect which is more adverse than the values used in previous analyses. Offsetting credit for a slightly larger predicted scram reactivity was assumed in this analysis. The analysis predicts that the rapid cooldown of the plant following a steam line break will not result in a return to power.

This analysis does not adequately consider the effect of opening of the feedwater bypass regulating valves due to a low steam generator pressure reactor trip signal and hence underpredicts plant cooldown (Reference 16). The Steam Line Break accident is most severe at end of cycle. At beginning of cycle the reactivity insertion associated with plant cooldown is approximately one third of the reactivity insertion at end of cycle. Hence this issue need not be resolved to permit resumption of power operation. It should be resolved prior to mid-cycle. A revised steam line break analysis is to be submitted within 30 days after Cycle 5 startup (Reference 17).

3.2.2 Steam Generator Tube Rupture

The Steam Generator Tube Rupture event for Cycle 5 is bounded by the Reference Safety Analysis and the Final Safety Analysis Report (FSAR).

3.2.3 Locked Rotor

The Locked Rotor event was reanalyzed due to (1) the introduction of ENC fuel and hence a reduction of initial overpower DNBR margin, and (2) a more severe predicted fuel pin power census for Cycle 5 than appears in the Reference Safety Analysis. Results of the analysis show that 7.2% of the fuel would experience a DNBR less than 1.3. The licensee has assumed that all fuel experiencing a DNBR less than 1.3 will fail. This is a conservative assumption. A generic radiological release calculation assuming 10% fuel failure has shown that releases will be well within the criteria of 10CFR100. Hence the consequences of a Locked Rotor accident during Cycle 5 are acceptable.

3.2.4 CEA Ejection

The predicted rod worth and post ejection peaking factor for the ejected rod from hot full power, end of Cycle 5, are more severe than previously analyzed. This case has been reanalyzed. Rod worths at hot zero power beginning and end of cycle and at hot full power beginning of cycle are within the bounds of the Reference 14 and 15 analyses. Hence the ejected rod from these conditions has not been reanalyzed.

Using approved analytical techniques, YAEC has predicted that the CEA ejection from full power, end of Cycle 5, will result in no clad damage (assumed to occur if the average fuel enthalpy exceeds 200 cal/gm), and that 5.3% of the fuel will experience incipient centerline melt (assumed to occur if the centerline enthalpy exceeds 250 cal/gm). None of the fuel is predicted to experience full molten centerline conditions. These results are acceptable.

3.2.5 Loss of Coolant Accident

The Loss of Coolant Accident was reanalyzed to assess the adequacy of the Maine Yankee ECCS with the core partially loaded with ENC fuel. Analyses were performed using NRC reviewed models (References 18, 19). Calculations were performed at beginning of Cycle 5 for the double ended cold leg guillotine and slot breaks with discharge coefficients of 1.0, 0.8 and 0.6. The double ended cold leg slot (DECLS) break with discharge coefficient, C_D, of 1.0 is predicted to be limiting. Maine Yankee is a steam cooling limited plant exhibiting a predicted peak clad temperature of 2041°F in the ENC fuel. An initial peak linear heat generation rate in the ENC fuel of 13.5 kw/ft was assumed for these calculations.

Additional calculations of the DECLS break, C_D=1.0, were performed parametric in fuel burnup, initial peak linear heat generation rate, and axial shape profile. These calculations provide the basis for the limiting condition for operation of the peak linear heat generation rate (PLHGR) as a function of core burnup, axial peak location and fuel type. YAEC has reported results of heatup (TOODEE 2) calculations assuming an initial PLHGR of 14 kw/ft for ENC fuel at 792 MWD/MTU average Cycle 5 burnup and at end of Cycle 5, and CE once burnt fuel at beginning and at end of Cycle 5. CE fabricated twice burnt fuel was analyzed at beginning of Cycle 5 assuming a PLHGR of 12.2 and 14.0 kw/ft and at end of Cycle 5 assuming a PLHGR of 12.2 kw/ft. These calculations were performed using a top peaked axial power profile, as were the beginning of Cycle 5 break spectrum calculations. Additional calculations assuming a cosine shaped axial power profile and a PLHGR of 16 kw/ft were performed for ENC fuel, CE once burnt fuel, and CE twice burnt fuel at beginning of Cycle 5.

Results of these calculations are in conformance with the criteria of 10CFR50.46.

The small break LOCA was reanalyzed to consider the effect of introduction of ENC fuel for Cycle 5 operation. Since the Cycle 5 core is to consist of two thirds CE fuel previous break spectrum sensitivity studies, which showed that the limiting break size is 0.5 $ft.^2$, are applicable for Cycle 5. Results of the 0.5 $ft.^2$ small break analysis, which was performed for ENC fuel at beginning of Cycle 5 at an assumed PLHGR of 16 kw/ft, predict a peak clad temperature of 1,348°F, and hence show conformance with the criteria of 10 CFR 50.46.

Additional analysis has been performed to assess the potential impact of recent concerns related to the LOCA-ECCS fuel clad models included in draft report NUREG-0630 (Reference 2). The limiting break, the DECLS break, C_D=1.0, at beginning of Cycle 5 was recalculated using the NRC data including the slow ramp correlation of burst node temperature versus hoop stress. Results of this analysis show a small reduction (approximately 60°F) in predicted rupture node and peak clad temperature node temperatures. The predicted changes are consistent with the YAEC use of WREM data for licensing applications.

4.0 Technical Specifications

MYAPC has proposed changes to the plant Technical Specifications enumerated below. These changes are: (1) necessary for plant safety incorporating the effect of introduction of the ENC fabricated fuel and Cycle 5 specific expected power distributions, or (2) of convenience to plant operation, or (3) administrative.

4.1 Technical Specification 2.1.1 and Figures 2.1-1a, 2.1-1b, and 2.1-2

Modification of the Thermal Margin/Low Pressure Trip setpoints have been proposed for Cycle 5. Associated calculational methodology (Reference 20) has been previously reviewed and approved. The set-point Pvar, is equal to AQ_{DNB} + B Tin + C, where Q_{DNB} is equal to $A_1 * QR_1$. The constants A, B, and C are determined from a reference set of iso-DNB thermal limit lines. QR_1 is a power dependent scaling factor. A_1 is a function of axial offset which models the degradation of DNBR overpower margin with increasing (absolute value) axial offset. The A_1 function also accounts for increases in peak enthalpy rise, $F_{\Delta H}$, associated with rod insertion. Rod insertions up to the Power Dependent Insertion Limit (PDIL) are considered. Changes to the constant A and function A_1 have been proposed for Cycle 5. Credit has been taken for the reduction, relative to Cycle 4, of the predicted value of $F_{\Delta H}$ with control bank 5 at the PDIL. The reduction of the rodded $F_{\Delta H}$ and associated increase of the overpower DNBR margin for Cycle 5, relative to Cycle 4, provide the basis for the decrease of the variable low pressure trip setpoint. At 100% of rated power, nominal inlet temperature of 554°F, and an axial offset of -0.11, the Cycle 5 low pressure setpoint will be 1868 psi. At the same conditions the

Cycle 4 low pressure setpoint was 2029 psi. Constants B and C, the QR_1 function, and the PDIL have not been altered for Cycle 5. The change provides increased operational flexibility.

Changes of the limiting condition for operation (LCO), of axial (symmetric) offset vs fraction of core power have also been proposed for Cycle 5. This LCO provides protection for the Loss of Coolant Flow and Full Length CEA Drop Anticipated Operational Occurrences (Section 3.1). The proposed change of the permissible symmetric offset below 80% power provides increased operational flexibility. No change is proposed to this LCO above 80% power.

4.2 Technical Specification 3.10.A

The proposed changes provide the flexibility to insert control rods 3 inches to accommodate potential CEA guide tube wear. Reference to two loop operation (which is not permitted) has been deleted.

4.3 Technical Specification 3.10.A.3

This proposed change increases the required available shutdown worth from 2.9% to $3.2\% \Delta\rho$. The higher worth was assumed in the Cycle 5 Steam Line Break accident analysis (see Section 3.2.1). This change is necessary to make the plant Technical Specification consistent with the supporting safety analysis.

4.4. Technical Specification 3.10.B.1

The proposed changes of the permissible peak linear heat rate LCO are based on the assumptions of the Cycle 5 LOCA-ECCS analyses (see section 3.2.5). These changes are necessary to make the plant Technical Specifications consistent with the supporting safety analyses.

4.5 Technical Specification 3.10 and Fig. 3.10-2

Fuel densification augmentation factors have been deleted from the text and Figure 3.10-2 of the plant Technical Specifications. Augmentation factors are included in the determination of the symmetric offset trip limits, Figure 2.1-2, of the plant Technical Specifications. Augmentation factors have been generically excluded from the LOCA analysis.

4.6 Technical Specification Fig. 3.10-3

Proposed revisions to the permissible symmetric offset vs power level (Figure 3.10-3) reflect the revised Cycle 5 LOCA-ECCS analyses and predicted values of PLHR vs symmetric offset during Cycle 5. The revision effectively reduces the permissible reactor power when operating on ex-core instrumentation alone (In-Core Monitors inoperable) to approximately 90% of rated power. This change is necessary to make the plant Technical Specifications consistent with the supporting safety analyses.

4.7 Technical Specification Fig. 3.10-4

The proposed change of Figure 3.10-4, the allowable unrodded radial peak versus cycle average burnup, is based on explicit Cycle 5 calculated values of unrodded peaking factors. Compliance with these values assures, in part, the validity of the assumptions of the Cycle 5 setpoint analysis; Figure 3.10-5, below, assures the remainder of the setpoint analysis.

4.8 Technical Specification Fig. 3.10-5

The proposed change of Figure 3.10-5, the allowable power level as a function of fractional (%) increase in unrodded total radial peaking factor above the values as a function of core burnup shown in Figure 3.10-4, is based on an envelope of Cycle 5 predicted values. Compliance with the envelope shown in Figure 3.10-5 assures, in part, the validity of the assumptions of the Cycle 5 setpoint analyses.

4.9 Technical Specification 3.15

This change is only to cross-reference to another TS and corrects an error.

We conclude all the above TS changes are acceptable.

5.0 Startup Test Program

Startup tests have been proposed by the licensee to provide assurance that Maine Yankee has been loaded as intended. This test program is similar to that used at Maine Yankee and other similar reactors. We have reviewed the test program and consider it acceptable.

6.0 Environmental Considerations

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

7.0 Conclusion

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

Dated: March 7, 1980

8.0 References

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2. D. Powers, R. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630, November 1979.

3. E. C. Wood, Maine Yankee Reportable Occurrence #79-010/03L-0, MYAPC, April 10, 1979.

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5. E. C. Wood, Maine Yankee Reportable Occurrence #79-030/99X-0, MYAPC, December 7, 1979.

6. "GAPEX: A Computer Program for Predicting Pellet to Cladding Heat Transfer Coefficients," ENC, XN-73-25, August 13, 1973.

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10. P. Bergeron, D. Denver, "Maine Yankee Reactor Protective System Setpoint Methodology," YAEC-1110, September 1976.

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

16. E. Wood, Letter discussing Maine Yankee Reportable Occurrence 80-001/01L-0, MYAPC, January 30, 1980.

17. D. E. Moody, "Low Power Steam Line Break," MYAPC, MWY 80-31, February 26, 1980.

18. "Application of YANKEE WREM-BASED Generic PWR ECCS Evaluation Model to Maine Yankee," YAEC-1160, July 1978.

19. "Nodalization Changes to Blowdown and Hot Channel Models of YAEC's WREM Based PWR Evaluation Model," MYAPC, WMY 78-20, March 19, 1979.

20. P. Bergeron, D. Denver, "Maine Yankee Reactor Protection System Setpoint Methodology," YAEC-1110, September 1976.

21. D. E. Vandenburgh to USNRC - Replaced Five TS Pages, dated February 15, 1980.