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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO LICENSE NO. DPR-36

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER STATION

DOCKET NO. 50-309

1.0 Introduction

By letter dated April 28, 1981 (Ref. 1), as supplemented May 15, 1981 and June 19, 1981, Maine Yankee Atomic Power Company (MYAPC or the licensee) requested an amendment to 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 6 core.

In addition, Section 5.9 of this SE addresses our evaluation of other Technical Specification changes.

The Cycle 6 core will consist of the insertion of 72 fresh Exxon Nuclear Company (ENC) fuel assemblies, one burned Type E assembly from Core 2 with 72 Cycle 5 ENC assemblies and 72 Combustion Engineering (CE) assemblies, which had been in the core during Cycles 2, 4 and 5, remaining in the core.

The licensee has reviewed all relevant anticipated operational occurrences (AOOs) and postulated accidents, except the steam line break (SLB) for which only a bounding evaluation has been provided at this time (see Section 3.4). The reanalysis of the SLB accident was necessitated by the replacement of motor drives for the main feedwater pumps with steam turbine drives.

2.0 Evaluation of Core Design

2.1 Fuel Systems

2.1.1 Exposed Fuel

The exposed fuel types proposed for use in the Cycle 6 core consists of two fuel types which were fabricated by CE and one type that was fabricated by ENC. The first CE fuel type, Type E, consists of a single fuel assembly previously irradiated during Cycle 2 and now loaded at the core centerline position for Cycle 6. The second CE fuel type, Type I, consists of 72 assemblies previously irradiated during Cycles 4 and 5. The maximum burnup assembly for Cycle 6 is a Type I assembly, which is expected to accumulate a maximum assembly average burnup of less than 40 GWd/MtU. Included in the Cycle 6 core are three reconstituted Type I assemblies containing low-enrichment (1.95 percent U^{235}) and solid Zircaloy replacement rods.

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Reconstitution was accomplished before the previous Cycle 5 irradiation of these assemblies. The ENC fuel denoted as Type J consists of 72 assemblies irradiated during Cycle 5. The ENC fuel exhibited minor mechanical differences from that previously supplied by CE. The differences were discussed in References 4 and 14.

Use of the CE fuel assemblies was previously approved for Cycles 2, 4, and 5. The use of the ENC fuel assemblies was approved for Cycle 5 (Ref. 14).

2.1.2 Fresh Fuel

The ENC fuel, denoted as Type K, consists of 72 fuel assemblies in the Cycle 6 core. The ENC fuel exhibits minor mechanical differences from that previously supplied by CE. These differences include thicker fuel rod cladding and slightly modified fuel pellet density and geometry. The 72, fresh, Type K fuel assemblies are identical to those previously supplied by ENC and approved for Cycle 5 by Reference 14. The core loading by fuel type is given in Table 3.2 of Reference 2.

In addition to 77 full-length control elements assemblies (CEAs), the Maine Yankee Cycle 6 core will also contain burnable poison rods in selected assemblies. Sixty-eight assemblies will contain standard $B_4C\text{-}AL_2O_3$ burnable poison rods and four assemblies will contain test rods containing borosilicate glass. The test rods which were previously approved for irradiation in the Cycles 4 and 5 cores have essentially no remaining reactivity effect in this cycle of operation.

2.1.3 Fuel Mechanical Design

The mechanical design features of both CE and ENC fuel assemblies to be used in the Cycle 6 core are listed in Table 3.3 of Reference 2. Although these design features are identical to those previously approved for use in Maine Yankee, we have given this area additional review because the licensee has relied upon fuel mechanical design analyses (Ref. 3) provided by the fuel supplier, ENC, which are unavailable to the staff. In order to complete our review, we have relied upon a summary description of the fuel mechanical design analyses provided by the licensee as part of the Cycle 5 reload report (Ref. 4). In addition, we have verbally obtained fuel design data from the fuel supplier, ENC, and confirmed these values with licensee, MYAPC, for Cycle 6 fuel. The licensee has further agreed to supply the staff with a copy of Reference 3. Based on available information, we consider the submittal of this report confirmatory and its receipt is not required prior to startup. Therefore, we conclude that the Maine Yankee Cycle 6 fuel mechanical design is acceptable.

2.1.4 Fuel Thermal Design

As discussed in Sections 2.1.1 and 2.1.2 of this report, the fresh Type K fuel in the Maine Yankee Cycle 6 core is identical to that previously irradiated in the reactor. The licensee's analysis of the fuel thermal performance is also the same as that used in previous reload analyses with two exceptions: (1) the analysis now considers a number of power history effects and (2) the analysis now considers burnup-dependent fission gas release as prescribed in NUREG-0418 (Ref.5).

The power history effects relate only to the calculation of rod internal pressure, and fuel centerline melt limits. In the past, MYAPC considered a lead rod in which the power rating and burnup bounded the expected values of these parameters for the core in question. In the revised analysis, MYAPC has recognized the fact that maximum power and maximum burnup do not, in practice, occur in the same rod when previously irradiated fuel is present. To consider this feature in the fuel thermal analysis, it is necessary to consider a number of power histories. Each power history is limiting for power or burnup (not both) for each fuel type in the Cycle 6 core. Because it is no longer obvious which history will produce maximum fuel temperatures or rod internal pressures, all results must be examined to find the maximum conditions. The licensee has performed such an analysis for Cycle 6 operation, considering previous cycle exposures and uncertainties. The results show (1) calculated internal fuel rod pressures are less than system pressure for normal operation throughout Cycle 6, (2) a 21 kw/ft limit on linear heat generation rate is a conservative limit for avoiding center-line melt, and (3) a conservative value for use in transient analysis assumptions for gap conductance is 600 Btu/ft^2-hr-°F.

In addition to examining the results of the licensee's analyses, we have audited (Ref. 6) the results for the limiting temperature case. A review of both our own and the licensee's calculations indicate that the analysis has been performed in an acceptable manner.

The licensee's use of a burnup-dependent fission gas release model is the result of an NRC request (Ref.7) to all U.S. fuel vendors to consider this effect. Because the licensee has elected to use a method provided by the staff (Ref. 5) to consider this effect, we consider further review of this change unnecessary. We, therefore, conclude that the fuel thermal design analysis for Maine Yankee Cycle 6 is acceptable.

2.2 Thermal Hydraulic Design

2.2.1 Thermal Hydraulic Analysis

The steady-state and transient Departure from Nucleate Boiling (DNB) analyses were performed using the COBRA-IIIC computer program. COBRA-IIIC was developed by Battelle Northwest Laboratory for use in the thermal-hydraulic analysis of nuclear fuel elements in rod bundles. The application of COBRA-IIIC to the Maine Yankee thermal-hydraulic design is decribed in References 21 and 22. Critical Heat Flux (CHF) calculations were performed for ENC fuel and CE fuel. Since the ENC fuel assemblies have higher pressure drop characteristics, the minimum average flow factor to any ENC assembly in the limiting case would be 97.5% of the core average. This flow factor was determined from an eighth core assembly-by-assembly COBRA analysis. Similarly a minimum flow factor of 0.988 for the CE fuel assembly was predicted by the eighth core COBRA analysis. It should be noted that the 0.975 and 0.988 flow factors are the minimum flow factors for the two types of fuel. The reason that these assemblies do not have a flow fraction of 1.0 is that they are surrounded by a number of other assemblies which receive the remaining flow. As an example the CE assembly having the minimum flow factor is surrounded by four "hot" ENC assemblies which receive more that 100% flow. When these five assembly flows are averaged, the result is the core average flow or a flow factor of 1.0. The 0.975 flow factor for the ENC fuel results from the fact that in the limiting case this is what the ENC assembly would experience.

The decrease in flow for the ENC and CE bundles and in turn the reduced margin to DNB were appropriately considered by Yankee Atomic in their safety analyses and reactor protection system setpoint analyses.

Table 1 contains a list of the Cycle 6 thermal-hydraulic operating parameters. A comparison of these parameters with those of Cycle 5 is included. The operating parameters for the two cycles are comparable.

2.2.2 Fuel Rod Bowing

A significant parameter which influences the thermal-hydraulic design is rod-to-rod bowing within fuel assemblies. For the CE fuel assemblies the staff has developed criteria for evaluating the effects of rod bow on DNBR. The resultant reduction in DNBR due to rod-to-rod bowing is given by:

Bundle Average Burnup (MWD/MTU)	DNBR Penalty (%)
0-15000	0
15000-24000	0
24000-33000	3.0

The fuel management scheme at Maine Yankee is such that the limiting DNB assembly will never be a CE fuel assembly.

The licensee has cited References 8 and 9 as a basis for calculating rod bow magnitude for CE and ENC fuel, respectively. The reference (Ref. 8) used for CE fuel is an NRC directive to Maine Yankee giving departure from nucleate boiling ratio (DNBR) reductions as a function of burnup. These reductions are based on an interim safety evaluation (Ref. 10) of fuel rod bowing effects which was limited in application to bundle average burnups below 33 GWd/MtU (gigawatt days per metric ton of uranium). Since the maximum expected assembly average burnup for a CE fuel assembly at end of Cycle 6 will approach 40 GWd/MtU, the licensee has stated that the minimum difference between the DNB limiting location (ENC fuel) and the peak pin in high burnup assemblies (CE fuel) in the Cycle 6 core is greater than 9.3 percent. The assumption is that the DNBR reduction for CE fuel at 40 GWd/MtU is less than 9.3 percent. Maine Yankee referenced tests performed at Columbia University (Ref. 23) which show that there is no DNB degradation for channel closures below 50%. Based on these facts the licensee stated that a flow factor of 1.0 was justified for the ENC fuel. By use of a more recent staff position (Ref. 11) we are able to state that the maximum, bow-induced channel closure for any assembly in the Maine Yankee Cycle 6 core is less than 50 percent.

The reference (Ref. 9) cited for the rod bowing analysis of ENC fuel is an Exxon Nuclear Company topical report which is currently under staff review. The maximum channel gap closure (21 percent) calculated by the licensee for ENC fuel is not consistent with that calculated by the staff using the methods given in Reference 11. We are able to conclude, however, that the maximum channel closure for any ENC assembly in the Maine Yankee Cycle 6 core is less than 50 percent. This conclusion is based on an anticipated maximum assembly average burnup of 26 GWd/MtU for any ENC assembly.

We conclude that Maine Yankee used previously approved methods to do their thermal-hydraulic analysis and that they adequately justified the exclusion of rod bow compensation for the CE and ENC fuel assemblies. Therefore, the Cycle 6 thermal-hydraulic design is acceptable.

2.3 Operating Experience

2.3.1 Guide Tube Wear

Fretting wear has been observed in irradiated fuel assemblies from a number of operating CE reactors. These observations revealed an unexpected degradation of guide tubes that were under control element assemblies. Similar wear has been found in Maine Yankee fuel assemblies that were previously discharged. It was concluded that coolant turbulence was responsible for vibration of the normally fully withdrawn control rods and, when these vibrating rods were in contact with the inner surface of the guide tubes, a wearing of the guide tube walls took place.

CE has provided the staff with a report (Ref.12) that describes a stainless-steel guide tube sleeve which has been fitted on nearly all CE fuel in the Maine Yanke Cycle 6 core. Four demonstration test assemblies are exceptions. Two demonstration assemblies are being used to assess a revised upper end post design and two provide for reduced guide tube flow. In Cycle 6 these four assemblies will be in symmetric, single control element assembly (CEA) locations. These locations have been selected as preferred sites on the basis that the observed local guide tube wear in earlier cycles was measured to be low in relation to core average guide tube wear for Maine Yankee. The results of previous inspections of guide tubes and sleeves indicate that the chrome-plated stainless steel sleeves appear to essentially eliminate guide tube wear as a fuel problem. The fuel assemblies provided by ENC (Cycle 5 and Cycle 6) already incorporate chrome-plate wear sleeves to prevent guide tube wear. This design, which is similar to the CE design, was discussed in Reference 13.

In order to verify the acceptability of chrome-plated wear sleeves at Maine Yankee, our evaluation (Ref. 14) of the Cycle 5 Reload Report (Ref. 4) contained a provision for end-of-Cycle 5 surveillance of guide tube wear. We have been informed by the licensee that preliminary results of these inspections, on both CE and ENC fuel, are favorable and that a formal submittal describing the results will be made after Cycle 6 startup. While stainless-sleeves have precluded guide tube wear, they have probably increased the cladding wear that occurs on the control elements themselves. To date, however, no inspections have revealed CEA cladding wear rates that would indicate a potential for the loss of CEA hermiticity in the near future. Nevertheless, it remains uncertain as to whether wear degradation to CEAs could ultimately reduce the CEA design lifetime. Therefore, during the Cycle 6 outage, eddy current testing was also performed on some CEAs. These results will be discussed in the licensee's formal report. Because these results deal with the long-term wear problem, we conclude that the issue of guide tube wear has been adequately addressed for Cycle 6 operation.

2.3.2 Fuel Failures

Our evaluation (Ref. 14) of the Cycle 5 Reload Report (Ref. 4) also discusses fuel failures detected and reported during Cycle 4 operation. Ten leaking fuel pins in CE assemblies scheduled for reinsertion were replaced with low-enrichment or solid Zircaloy rods. Based on low coolant activity levels observed during Cycle 5 operation, the licensee has concluded that the fuel failure problem did not continue into Cycle 5 and that a similar problem is not expected to occur during Cycle 6. We find this conclusion acceptable.

TABLE 1

Thermal-Hydraulic Parameter

	Cycle 5	Cycle 6
I. Performance Characteristics:		
Total Heat Output Mw(t)	2630	2630
System Pressure Nominal psia	2235	2235
Minimum in Steady-State	2060	2060
Maximum in Steady-State	2260	2260
II. Coolant Flow:		
Total Flow Rate 10^6 lb/hr	136.0-134.6	136.0-134.6
Coolant Flow through Core 10^6 lb/hr	132.1-130.7	132.1-130.7
Pressure Drop Across Core psi	9.9	10.18
Average Mass Velocity 10^6 lb/hr-ft^2	2.47-2.444	2.47-2.444
III. Coolant Temperature:		
Design Inlet Temperature °F	546-554	546-554
Average Core Enthalphy Rise BTU/lb	68.7	68.7
IV. Heat Transfer:		
Total Heat Transfer Area ft^2	49,555*	49,188*
Average Heat Flux BTU/hr-ft^2	176,305*	176,830*
Average Linear Heat Rate (KW/ft)	5.95*	6.0*

* Allows .3% axial shrinkage due to fuel densification.

2.4 Reload Core Design

2.4.1 Core Fuel Loading

Four different types of assemblies will be used in the Cycle 6 core. These are given in Table 3.1 of Reference 2. All fuel types have been reviewed and approved in earlier reloads.

Of the 217 fuel assemblies constituting the Maine Yankee Cycle 6 core, 145 have been exposed in earlier cycles. Thus, one CE assembly was loaded in Cycle 2, 72 CE assemblies had been irradiated in Cycles 4 and 5, 72 ENC bundles were introduced at the beginning of Cycle 5 (BOC-5). The 72 fresh ENC assemblies scheduled to be loaded in the Cycle 6 core are of the same type as those inserted at BOC-5. The ENC fuel has been reviewed and accepted by Reference 14.

2.4.2 Burnable Poison Loading

The burnable poison shim rods are located in selected assemblies forming an octant symmetric pattern in the core. Two basic types of boron shim rods are used: (a) standard B_4C - Al_2O_3 shim rods and (b) borosilicate glass shim rods. Both types have been used in fuel introduced in earlier cycles.

2.4.3 Core Loading Pattern

The Cycle 6 fuel loading pattern is shown in Figure 3.3 of Reference 2 in a quarter core representation. The 72 new ENC fuel bundles are shown to be located on the periphery of the core. In the refueling scheme used by MYAPC, fresh fuel is loaded along the core-reflector interface and is, in subsequent cycles, moved inward. With the exception of 16 bundles which have been rotated in the new core, all assemblies maintained the same orientation with respect to the previous assembly position. The loading and orientation of the assemblies are such that mirror symmetry exists relative to the boundary lines of the quadrant. Since the methods used in the development of the reloaded core have been reviewed and approved earlier, we conclude that the Cycle 6 core loading configuration is acceptable.

2.5 Physics Analysis

Using approved methodology, static calculations have been carried out for a quarter core model of the Maine Yankee Cycle 6 core. The expected average exposure of 10,800 MWD/MT for this cycle was based on an end of Cycle 5 (EOC-5) exposure of 11,000 MWD/MT.

2.5.1 Core Characteristics

Basic nuclear characteristics for the Cycle 6 core are given in Table 4.1 of Reference 2 and compared with those of Cycles 3 and 5. Because of a safety analysis performed for Cycle 3, that cycle has been treated as a reference cycle. Control characteristics of Cycle 6, such as Doppler temperature coefficient, moderator temperature coefficient, kinetics and boron worth data are very similar to those of Cycle 5. The differences are attributed to the different core loadings, exposures and exposure histories, and not to the ENC fuel design (cladding, shorter pellet length). The only major distinction between the loadings of Cycles 5 and 6 lies in the proportion of the CE and ENC fuel. Since the NRC has previously reviewed and approved the new fuel design (Ref. 14) and because of the similarity of the core characteristics (Ref. 2) between Cycles 5 and 6, we find the CE and ENC fuels to be compatible.

2.5.2 Core Power Distribution

Hot, full power (HFP), axially averaged, relative assembly powers for BOC-6, middle of Cycle 6 (MOC-6) and EOC-6 quarter cores are given in Reference 2 for an all rods out (ARO) condition and for a configuration using the Bank 5 rods. These power distributions have been based on an EOC-5 exposure of 11,000 MWD/MT. These results show that the unrodded maximum 1-pin radial peak power occurs at EOC-6 when its value is 1.469. The proposed TS 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 6 core, indicates radial peaks in the range from 1.600 to 1.665. Comparison of radial peaks given the above power distributions with the allowable values shown in the TS demonstrates the adequacy of the results given in the core performance analysis (Ref. 2). We, therefore, find this analysis to be acceptable.

2.5.3 Reactivity Coefficients

The moderator temperature coefficient (MTC), the fuel temperature coefficient and other kinetics parameters for the Cycle 6 core along with the corresponding values of Cycle 3 (Maine Yankee's reference cycle) and Cycle 5 are given in Section 4.4 and Table 4.1 of Reference 2.

The BOC-6 and EOC-6 HFP MTC's are $(-0.83 \pm 0.50) \times 10^{-4}$ $\Delta\rho/°F$. This value appears to be more negative than the value given for the reference cycle because of the change in the operating conditions of the Cycle 6 core. The hot zero power (HZP) and HFP fuel temperature coefficients at both BOC-6 and EOC-6 are about the same as their corresponding values for the reference cycle. The calculational uncertainty for the fuel temperature coefficient assumed in the analyses is 25 percent. The magnitudes of the kinetics parameters (total delayed neutron fraction and prompt neutron generation time) at BOC-6 and EOC-6 lie in the overlapping band of the Cycle 6 and reference cycle ranges established by the 10 percent uncertainties. The slight differences in these values are attributed to the differences in operating conditions and core average exposures.

Since the above data have been obtained using approved methods and practices and since the differences in the results are within the respective calculational uncertainties, we find these data to be acceptable.

2.5.4 Control Requirements

The value of the required shutdown margin is determined from the steam line break (SLB) analysis. As discussed in Section 3.2 the steam line break analysis will be reanalysed later, i.e. prior to an irradiation of 4,000 MWD/MTU in Cycle 6. The licensee has provided an acceptably conservative scoping analysis to permit startup and operations to an irradiation of 4,000 MWD/MTU. By letter (Ref. 25) and during conversations with the licensee , it was determined that the available shutdown margin with one CEA stuck should not be less that 3.57%. In addition the licensee has agreed that this value may be revised upon completion of their steam line break analysis. We find this value acceptable pending the steam line break reanalysis.

2.5.5 Safety Characteristics

Safety related characteristics such as scram reactivity, CEA ejection, CEA drop, insertion limits and augmentation factors are presented in Section 4.6 of Reference 2. The results of our review are presented below and in the Safety Analysis, Section 3.0.

2.5.5.1 Scram Reactivity

Table 4.3 of Reference 2 gives the available scram reactivity calculated for both HFP and HZP conditions at BOC-6 and EOC-6. In addition to uncertainties, these results include allowances for the worst stuck CEA and the permitted CEA insertion. These results, derived by methods already approved, are found acceptable.

2.5.5.2 CEA Insertion Limits

The power dependent insertion limit (PDIL) is given in Figure 4.9 of Reference 2. The PDIL also appears in Figure 3.10.1 of the TS. The curve gives the power level as a function of CEA insertion by group for three and two-loop operation. The data are the same as for the previous cycle and thus are found to be acceptable.

2.5.5.3 Augmentation Factors

The augmentation factors shown in Table 4.7 of Reference 2 are incorporated as a power spike penalty in the calculation of the core power to incipient fuel centerline melt. The set of augmentation factors represents the most restrictive values expected in the Cycle 6 core. These factors have been derived by using approved methods. The factors are compared to the Cycle 3 (reference cycle) augmentation factors. The comparison shows that the Cycle 3 data are slightly more restrictive. The differences are attributed to the presence in Cycle 3 of the replacement fuel (Core 1 design fuel) and its different densification characteristics. Based on the favorable comparison of the Cycle 6 with the Cycle 3 data, we conclude that these results are acceptable.

3.0 Safety Analysis

3.1 General

The description of the core change for Maine Yankee Cycle 6 is given in Reference 2. In summary, 73 assemblies will be discharged, 72 new (ENC) subassemblies and one burned type E assembly from core 2 will be loaded. Descriptions of the ENC fuel and the CE fuel (some of which remains in core) are given in Reference 4. These references describe the small differences in the mechanical and hydraulic characteristics of the two fuels. The fuel changes have produced small changes in power distribution and coefficients, and have necessitated a reexamination of anticipated operational occurrences and accidents.

The licensee has reviewed all relevant anticipated operational occurrences (AOOs) and postulated accidents, except the main steam line rupture, for which only a bounding calculation has been provided at this time. All other AOO's and postulated accidents fall within bounds established in the FSAR or analyses presented for other previously approved loadings. The codes and models used are appropriate and their use has been approved for previous loadings. The results of the calculations for Cycle 6 fall within previous limits as to shutdown margin, DNB, temperature and pressure.

The staff finds that, with the exception of the steam line rupture accident discussed in Section 3.3.1, the AOO's and postulated accidents have been analyzed using approved models and methods, and the results fall within previously accepted limits. These analyses are therefore acceptable. Specific discussion of some items, however, is included.

3.2 Anticipated Operational Occurrences

3.2.1 Control Element Assembly Withdrawal

For Cycle 6, the FSAR design power distribution is more severe than any power distribution predicted within the allowed operating band on symmetric offset at any power level. Therefore, the CEA withdrawal incident results for the reference cycle envelope the conditions for Cycle 6 and are, acceptable.

3.2.2 Boron Dilution

The licensee reanalyzed the boron dilution event of the Cycle 6 reload using the same assumptions made in previous analyses (Ref. 3, 4 and 5). The licensee's calculations show that indications and alarms in the control room will provide adequate time for the plant operators to take appropriate action. The following indications and alarms from the chemical and volume control system will be available to alert the plant operator of a boron dilution event in progress:

a. Boronometer concentration indication;
b. Volume control tank level indication and high and low alarms;
c. Makeup controller flow alarms;
d. Letdown flow temperature indication at outlet of regenerative heat exchanger.

The boron dilution transients assumed beginning of life core conditions since this has the maximum reactivity and the minimum calculated times to loss of shutdown margin. In addition, parametric studies were conducted regarding the effects of: (1) initial and final boron concentration; (2) reactor coolant system volume, and (3) dilution rate.

3.2.2.1 Dilution During Refueling

Boron dilution during refueling assumed:

1. The two most reactive control rods are withdrawn from the core;
2. Initial boron concentration is the minimum allowed by plant procedures during refueling operations;
3. Minimum reactor coolant volume;
4. Maximum dilution rate.

Under these conditions the licensee calculates that it would take more than 30 minutes before the shutdown margin is decreased to zero by dilution.

3.2.2.2 Dilution During Startup

Boron dilution during startup assumed:

1. Primary coolant system completely filled;
2. Maximum dilution rate;
3. Minimum initial boron concentration;
4. All control rods withdrawn from the core.

The licensee calculated that it would take approximately 1.5 hours to dilute the primary system to the critical boron concentration.

3.2.2.3 Dilution at Hot Standby and at Power

Boron dilution at hot standby and at power assumed:

1. Maximum dilution rate;
2. Initial boron concentration corresponds to the maximum expected for any critical condition, including uncertainties. (The concern is the rate at which reactivity can be added along with the resulting power excursion and rise in reactor coolant temperature.)

The licensee calculates that it would take approximately 17 minutes of continuous dilution at the maximum charging rate to completely absorb a 1% shutdown margin.

3.2.2.4 Dilution at Hot and Cold Shutdown

The staff has discussed the boron dilution events during hot and cold shutdown conditions with the licensee particularly during the Mode 5 (cold shutdown) coincident with a reduced reactor coolant system volume. Maine Yankee has revised their Technical Specifications so that the minimum shutdown margin is now increased from 2% Δk/k to %5 Δk/k. The licensee states when the reactor coolant system is drained down during Mode 5 operation, adminstrative procedures are used to ensure that additional boration beyond that required by the Technical Specifications has occurred.

The licensee has committed to document calculations showing that inadvertent boron dilution during hot and cold shutdown modes will still result with sufficient time (i.e. at least 15 minutes) for operator action before the shutdown margin is lost. Based on a comparison with a similar facility, we concur with the licensee that sufficient time for operator action should be available.

3.2.2.5 Conclusion

We have reviewed the licensee's boron dilution analysis for the Cycle 6 reload. The assumptions and methodology used by the licensee are consistent with those made in analyses that have been previously reviewed and approved by the staff. Indications and alarms are present in the control room so that plant operators will be properly alerted to an inadvertent boron dilution of the primary system.

Based on our review we conclude that the licensee has adequately addressed the boron dilution event for the Cycle 6 reload and has provided adequate protection to prevent return to critically and is, therefore, acceptable.

3.3 Dropped CEA

Using the same analysis assumptions as those for Cycles 4 and 5, CEA drop calculations were performed for Cycle 6. Table 4.5 of Reference 2 shows the calculated worths of the most limiting dropped CEA's during Cycles 5 and 6 with the resulting maximum 1-pin radial powers evaluated at BOC-6. The corresponding EOC-6 data are shown in Table 4.6 of the same reference. These results are presented as an envelop of the maximum percent increase in peaking versus reactivity worth of the dropped CEA in Figure 4.8 of Reference 2. This figure shows that the BOC-6 increases in the peaking at the maximum power pin is about the same as that evaluated for BOC-5. The EOC-6 maximum power pin increase in the peaking as a function of the dropped CEA worth is higher than the corresponding EOC-5 increase by amounts ranging from 0 to 1.7 percentage points. The EOC increase in the power peaking with the CEA worth is enveloped by the BOC-5 data and is therefore acceptable.

3.4 Postulated Accidents

3.4.1 Steam Line Break Accidents

The introduction of the steam driven main feedwater pump in Cycle 6 is noted in the licensee's submittal describing the Safety Analysis (Ref. 2 Sections 5.4.1 and 5.6.1). The characteristics of a steam line break accident will be reanalyzed to take the new installation into account. The revisions to the steam system require a code with greater capabilities than the previous FLASH analysis, and the licensee is therefore adapting RETRAN to this application. The RETRAN analysis is not expected to be completed at the time for cycle 6 startup; however, the licensee has submitted a scoping analysis to justify, in a conservative fashion, that the temperature transient and consequent reactivity insertion accompanying a steam line rupture event could not possibly exceed the plant shutdown capability during the early part of the cycle (Ref. 25). The concern in this instance is that a temperature transient imposed on the primary system resulting from the steamline break might exceed the shutdown margin and cause a return to criticality. The scoping analysis assumes an extremely conservative primary system cooldown rate i.e., an instantaneous cooldown to 212°F, and shows that the reactivity insertion associated with this temperature change does not exceed the available control element assembly worth. Conservatisms in the scoping analysis include a 10% uncertainty on control assembly worth and a highest worth stuck rod. Reactivity coefficients considered include Doppler feedback with 25% uncertainty and moderator temperature with 15% uncertainty. The step temperature change from the limiting condition of hot zero power to 212°F still leaves a 0.21% margin of subcriticality up to an average irradiation in cycle 6 of 4000 MWD/MTU.

The staff finds this analysis to be acceptably conservative to permit startup and operations to an irradiation of 4000 MWD/MTU in Cycle 6. Approval for the continuation of operations in Cycle 6 beyond 4000 MWD/MTU will require that this analysis be appropriately supplemented.

3.4.2 CEA Ejection

Calculated worths and maximum 1-pin powers resulting from the worst ejected rods are shown in Table 4.4 of Reference 2 for hot, full and zero power at BOC-6 and EOC-6. The maximum ejected worth for this cycle is smaller than its reference cycle (Cycle 3) counterpart. Correspondingly, the 1-pin radial peaks at BOC-6 and EOC-6 appear to be less pronounced when compared with the reference cycle values. Since these results have been obtained with approved methods and since the values are bounded by the reference cycle values we find them to be acceptable.

The licensee's analysis shows that the CEA ejection accident results for HFP are bounded by Cycle 5 and/or reference cycle results. For HZP the licensee provided the results of analyses which show clad damage occurs and none of the fuel experiences incipient centerline melting. The CEA ejection accident results for Cycle 6 are, therefore, acceptable.

We have reviewed the changes made to the CHIC-KIN program used by the licensee in analyses of the CEA Ejection and Siezed Rotor Accidents. The modifications properly allow the thermal properties of the fuel and clad to vary as a function of temperature.

3.4.3 Loss-of-Coolant Accident

Three fuel-related items were addressed by the licensee for the postulated loss-of-coolant accident (LOCA). These were (1) Cycle 5-Cycle 6 design differences, (2) clad swelling and rupture as described in NUREG-0630 (Ref. 15), and (3) enhanced fission gas release as described in NUREG-0418 (Ref. 5). Because the design of the fresh fuel introduced into the Cycle 6 core is identical to that previously used at Maine Yankee, the Cycle 5-Cycle 6 differences are largely due to the burnup distribution in the core.

To evaluate the impact of these changes on the LOCA analyses, the licensee performed (1) a break spectrum analysis, (2) a burnup sensitivity study and (3) a cosine axial distribution study. This partial, rather than complete, LOCA re-analysis was performed because (1) Cycle 5-Cycle 6 design differences were not expected to significantly change the results, (2) previous analyses with NUREG-0630 models indicated that the reference LOCA analysis remains bounding, and (3) enhanced fission gas release effects do not occur for burnups below 20 GWd/MtU. For the burnup sensitivity study, the rod heatup calculations were performed for exposure below 20 GWd/MtU and core heatup calculations were performed for exposure above 20 GWd/MtU. The results show that the Maine Yankee Cycle 6 core continues to satisfy regulatory requirements for LOCA analysis and is, therefore, acceptable.

4.0 Other Matters

4.1 Auxiliary Feedwater System and Main Feedwater Isolation System

The licensee plans to revise the logic functions associated with the delivery of auxiliary feedwater at the start of Cycle 6. At the same time the main feedwater isolation system will be upgraded to safety grade components and an additional trip function will be incorporated. These changes are described in outline form in Reference 2, Sections 5.6.2 and 5.6.3, but are described more fully in previous correspondence (Ref. 26 and 27), are functionally in conformance with NRC Bulletin 80-04, and are therefore satisfactory. They are, however subject to post-implementation review with regard to their instrumentation and control details.

5.0 Technical Specification Changes In Support of Cycle 6

5.1 Thermal Margin/Low Pressure Trip

The licensee proposes to modify coefficients for thermal margin low pressure trip from A = 2004.3 to A = 2060.7 in TS 2.1.1(b), and revise Figures 2.1-1a and 2.1-1b. The pages affected are 2.1-1, 2.1-4, and 2.1-5.

These changes are acceptable because they reflect Cycle 6 power descriptions and use approved methodology (Ref. 4) to generate the appropriate values.

5.2 CEA Group, Power Distribution, Moderate Temperature Coefficient Limits and Coolant Conditions

Revise Technical Specification 3.10.A.3 by changing the available shutdown margin with one CEA stuck out from 3.2% to 3.57%. This change has been discussed with the licensee by telephone communication and by Maine Yankee letter Reference 28. This new shutdown margin has been reviewed and found acceptable in Section 2.5.4 of this SE.

The page affected is page 3.10-1.

5.3 Power Distribution Limits

The licensee proposes to modify the power distribution limit wording only to be consistent with the Cycle 6 loading. The proposed modifications change subheadings for limits: change Type J to read Fresh Fuel and change Types E, G, H & I to read Exposed Fuel. The page affected is 3.10-2.

This change is acceptable because it reflects the Cycle 6 evaluation in Section 5.4.5 of Reference 2.

5.4 Flux Peaking Augmentation Factors

The licensee proposed to delete Figure 3.10-2 of page 3.10-10.

This change is acceptable because the affected figure is no longer applicable.

5.5 Incore Detector Alarm

The licensee proposes to replace Figure 3.10-3 with revised Figures 3.10-2 and 3.10-3. The page affected is 3.10-11.

This change is acceptable because it reflects Cycle 6 power distributions. Approval of the Cycle 6 power distribution calculation is discussed in Section 2.5.2 of this SE.

5.6 Power Distribution Limits

The licensee proposed to replace Figure 3.10-4 with a revised Figure 3.10-4. The page affected is 3.10-12.

This change is acceptable because it specifies the maximum unrodded radial peaking factors used as input in the safety analyses for Cycle 6. This is discussed in Section 2.5.2 of this SE.

5.7 CEA Insertion Limits

The licensee proposes to replace Figure 3.10-5 with a revised Figure 3.10-5. The page affected is 3.10-13.

This change is acceptable because it reflects the assumptions used in the generation of the Cycle 6 set points.

5.8 Definitions

5.8.1 Hot Shutdown Condition

The licensee proposes to modify the definition from: When the reactor is subcritical by an amount greater than or equal to the margin specified in Technical Specification 3.10 (paragraph A.3) and T_{AVG} is greater than 500°F, to: When the reactor is subcritical by 5% $\Delta k/k$ and T_{AVG} is greater that 500°F.

The page affected is page 1 of Definitions.

The changes in definitions reflect current operating practice and are, therefore, acceptable.

5.8.2 Cold Shutdown Boron Concentration

The licensee proposes to modify the definition from: Boron concentraton sufficient to provide $k_{eff} \leq 0.98$ with all control rods in the core and the highest worth control rod fully withdrawn, to: The boron concentration shall be sufficient to maintain the reactor at least 5% Δk/k subcritical with all control rods in the core.

The pages affected are pages 2 and 3 of Definitions.

The changes in definitions reflect current operating practice and are, therefore, acceptable.

5.9 Other Technical Specification Changes

5.9.1 Combined Heatup, Cooldown and Pressure Temperature Limitations

5.9.1.1 Introduction

By letter dated March 25, 1981, the Maine Yankee Atomic Power Company proposed to modify the Technical Specifications by replacing Figures 3.4-2 and 3.4-3 with revised versions of Figures 3.4-2 and 3.4-3, attached to the referenced letter. Figure 3.4-2 contains curves plotting the fluence against burnups at the 1/4 T and 3/4 T reactor vessel beltline positions and Figure 3.4-3 contains a curve extracted from Regulatory Guide 1.99 showing the shift in RT_{NDT} against fluence for the limiting material in the Maine Yankee reactor vessel.

Appendix G of 10 CFR Part 50, "Fracture Toughness Requirements", requires that pressure-temperature limits be established for the reactor coolant system heatup and cooldown operation, inservice leak and hydrostatic testing, and for reactor core operation. Pressure-temperature limits are required to ensure that the stresses in the reactor vessel do not exceed ASME Code acceptable values, which provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The purpose of Appendix H of 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements", is to monitor the change in fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from the neutron irradiation and thermal environment. The magnitude of the shift in RT_{NDT} is proportional to the neutron fluence received by the materials in the pressure vessel. The shift in RT_{NDT} is predicted by Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials". Appendix G of the ASME Boiler and Pressure Code, 1971 Edition, including Summer 1972 Addenda, "Protection Against Nonductile Failure", presents the procedure for obtaining allowable loadings on Class 1 components and conservatively relating the stress intensity factor, K_{IR}, to the reference nil-ductility temperature, RT_{NDT}.

In order to check the validity of the predicted shift in RT_{NDT} in compliance with the requirements of Appendix H, surveillance specimens are periodically removed from the reactor and tested. The test results are compared to the predicted shift in RT_{NDT} and the pressure-temperature limits for reactor operation are accordingly revised.

Surveillance capsule 263 was removed from Maine Yankee during Cycle 4/5 refueling outage after 4.58 equivalent full power years (EFPY) operation. The neutron fluence as determined from dosimetry in capsule was 6.9×10^{18} n/cm^2 (E>1MeV). The actual fluence was 2.3 times greater than the predicted fluence as stated in the surveillance program.

5.9.1.2 Evaluation

The irradiation induced changes in the mechanical properties of the pressure vessel materials were determined. The limiting material in the beltline region of the reactor vessel was the weld metal, containing 0.36% copper and 0.15% phosphorus. The adjusted shift in RT_{NDT} was 205°F and the upper shelf decreases to 58 ft lbs for the test material. A comparison of the predicted and measured shift in RT_{NDT} at the 1/4 T position at the end of Cycle 4 was 104°F compared to 164°F. The end-of-life fluence at the reactor inside surface was 1.6×10^{19} compared to $2.7 \times 10^{19} n/cm^2$ (E>1MeV). The difference between the predicted values and measured results was attributed to calculational error in the original prediction, according to the licensee.

The Battelle report, "Maine Yankee Nuclear Plant Reactor Vessel Surveillance Program: Capsule 263", BCL-585-21, provides the revised curves for predicting fluence versus reactor power history. The curves in conjunction with the fluence versus RT_{NDT} shift of Regulatory Guide 1.99 were used in the revision of the pressure-temperature limits for heatup, cooldown and hydrotest of the reactor coolant system. Although test data were presented to indicate that the recommendation of Regulatory Guide 1.99 was conservative for the Maine Yankee reactor vessel, the licensee elected to incorporate the additional conservatism in the generation of operating limitations. The proposed modification to the TS reflect the revision in fluence resulting from the reactor vessel surveillance program.

5.9.1.3 Conclusion

We conclude that the proposed modifications to the TS for the Maine Yankee plant are acceptable and will ensure that the reactor vessel is operated under pressure-temperature limits in compliance with Appendix G of 10 CFR Part 50. The revisions in the heatup and cooldown limitations during operation, testing, maintenance and postulated accident conditions constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 31 of Appendix A of 10 CFR Part 50. The pressure-temperature limits were calculated to the recommendation of Appendix G of the ASME Code and Regulatory Guide 1.99.

5.9.1.4 Recommendation

We have reviewed the proposed capsule removal schedule in Table 4.5-2 of the FSAR. Although the schedule complies with the intent of Appendix H of 10 CFR Part 50, we recommend that the next capsule be removed and tested prior to 12.7 EFPY operation. After 12.7 EFPY operation, the fluence at the 1/4 T position in the reactor vessel will equal or exceed $6.9 \times 10^{18} n/cm^2$ (E>1MeV), the fluence received by surveillance capsule 263.

5.9.2 Safety Injection System

5.9.2.1

The licensee proposes to replace page 3.19-1 of Technical Specification 3.19 with a revised page 3.19-1. By letter dated February 13, 1981 Maine Yankee proposed to modify the technical specifications requirement to disable the Safety Injection Tank and reactor coolant system loop isolation valve power operators in the open position when the reactor is critical.

Currently, Technical Specification 3.19.b.1 requires the Safety Injection Tank isolation valve power operators to be "racked out". Since these breakers were not designed to be racked out, the licensee met the intent of the specification by disconnecting and taping the power leads and locking out the breakers. In place of the disconnected leads the licensee proposes to install a disconnect switch in series with the valves power supply breakers to be locked in the open position while the reactor is critical. The breakers will also be locked open. We find the locked open disconnect switch a suitable alternative to disconnected leads and therefore, acceptable.

Technical Specification 3.19.b.2.b. requires that for the reactor coolant system loop isolation valves: "The breaker thermal overload links shall be physically removed from the breakers." The licensee states that these links were not intended to be used for that purpose. Consequently, the licensee proposes to install disconnect switches in series with each valve power supply breaker to be locked in the open position while the reactor is critical in lieu of removing the breaker thermal overload links. We find installing disconnect switches locked in the open position a suitable alternative to removing the thermal links and, therefore, acceptaible.

The affected page is 3.19-1.

5.9.2.2

By letter dated June 3, 1981 as supplemented June 30, 1981 and July 6, 1981, the licensee proposes to revise Technical Specification 3.19.b.3, 4.6.A.1.b and 4.6.A.2.f to add surveillance requirements to two check valves isolating the high pressure reactor coolant loop from the low pressure safety injection system. This change is intended to verify the integrity of the check valves. The change was submitted in compliance with our confirmatory order dated April 20, 1981. The pages affected are 3.19-1, 4.6-1, 4.6-2, 4.6-3 and 4.6-4. Pages 3.19-2 and 4.6-5 have been added. We have reviewed the proposed changes and find the surveillance applied, as required by the proposed change, acceptable.

6.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of 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.

Date: July 10, 1981

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2. "Maine Yankee Cycle 6 Core Performance Analysis," Yankee Atomic Electric Company Report YAEC-1259, undated, Attachment B to Reference 1 above.

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28. Robert H. Groce (MYAPC) to Mr. Robert A. Clark (NRC), July 2, 1981.