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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-3

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

DOCKET NO. 50-29

Introduction

During the refueling and maintenance outage which began on October 21, 1978, Yankee Atomic Electric Company (the licensee) has completed refueling of Yankee-Rowe for Core XIV operation and other tasks. The licensee has also installed four containment isolation valves as part of a modification to enhance the safe shutdown capability of Yankee-Rowe and has completed an acceptable steam generator inspection program. This Safety Evaluation documents our review of these matters in support of a proposed license amendment, which would authorize Core XIV operation, with appropriate changes to the facility Technical Specifications.

A. Core XIV Reload

Discussion

By letter dated September 8, 1978 (Reference 1) as supplemented by two (2) submittals dated November 21, 1978 (References 2 and 6) Yankee Atomic Electric Company (the licensee) requested changes to the Technical Specifications to permit operation of the reloaded Core XIV.

The proposed reload for Core XIV consists of replacing 40 Core XIII fuel assemblies with 40 fresh fuel assemblies which are placed in the peripherial region and reshuffling once burned fuel assemblies from Core XIII into the inner region of the core. The reloaded plant would be operated at the same conditions as Core XIII, i.e., a power level of up to 600 Mwt with 2000 psia system pressure, 515°F core inlet temperature and 4.40 Kw/ft core average linear heat generation rate.

7812210308

7812210 8

The analyses performed for the Core XIV reload core design were based on the following assumptions:

(1) Core average burnup for the beginning-of-life Core XIV is 5,268 MWD/MTU.

(2) Full power lifetime for Core XIV is 14,100 MWD/MTU.

(3) Core XIV operation to be within the plant operating limitations given in the Technical Specifications including the changes proposed in the licensee's application.

The licensee has proposed the following changes to the Technical Specifications:

(1) relocate the channel #2 low pressure sensor in the engineered safeguard system to monitor main coolant pressure instead of pressurizer pressure.

(2) reset the low pressure safety injection accumulator timer setpoints to increase accumulator rundown time from 4.0 ± 0.75 sec to 11.85 ± 0.23 sec. This change is required because of the relocation of the channel #2 low pressure sensor, and

(3) modify the following curves specifying operational limits for the core:

Fig. 3.2.1 Allowable Peak Rod Liner Heat Generation Rate;

Fig. 3.2.2 Factor F;

Fig. 3.2.3 Xenon Multiplier Redistribution;

Fig. 3.2.4 Multiplier for Reduced Power.

Evaluation

Fuel Design

The fresh fuel used in the Core XIV reload was manufactured by the Exxon Nuclear Corporation (ENC). Mechanical and thermal design of this fuel is identical to the fuel used in Core XIII (References 3 and 4). This design has been reviewed and approved by the NRC (Reference 5). The mechanical and thermal conditions in which the core will operate are also identical to those in which Core XIII operated. Therefore, we conclude that the Core XIV fuel design is acceptable.

Nuclear Design

The Core XIV loading consists of two zone patterns used in the preceding core with 36 once burned fuel assemblies (average burnup: 11,120 MWD/MTU) located in the inner region and 40 fresh fuel assemblies located at the periphery of the core. Core XIV will have a higher boron concentration at the beginning of the cycle (BOC) than Core XIII. This results in less negative moderator temperature, void and pressure coefficients of reactivity. The effect of these changes in reactivity coefficients have been accounted for in the reevaluation of anticipated occurrences and postulated accidents for Core XIV discussed in the "Safety Analysis" section below. Other nuclear characteristics of Core XIV are similar to those of Core XIII.

Control rod configuration in Core XIV remains the same as in Core XIII. Control is accomplished with rod group C which has a slightly higher reactivity worth in Core XIV than in Core XIII. The licensee has compared total control rod worths in Core XIV with Core XIII and has shown that the excess shutdown margin is higher in Core XIV. Startup tests will provide additional verifications that sufficient margin is available during the Core XIV operation. The control rod insertion curve is identical for both core cycles.

The maximum power spikes for different axial positions were calculated using an approved method (Reference 7). Power spikes were higher in Core XIV, but the licensee has acceptably shown that this would not have any significant impact on plant operation or accident analyses (Reference 7).

The factors accounting for the maldistribution of Xenon, which were introduced in Core XIII (Reference 8), were recalculated for Core XIV. Both factors (1) the Multiplier for Reduced Power and (2) the Multiplier for Xenon Redistribution, exhibited only a relatively small change from their values in Cycle XIII. This does not significantly change the margin of safety.

The methods used for calculating nuclear parameters were generally similar to those used for Core XIII (Reference 4). The modifications made in the present reload calculations are discussed below:

(1) Modification was made to the nodal neutronic coupling model chosen for use in the SIMULATE model analysis. The TRILUX

formulation for the neutron transport between nodes was replaced by Borresen's PRESTO formulation (Reference 9) which represented a more exact analytical solution.

(2) The FOG program for calculating reactivity parameters, such as moderator and fuel temperature coefficients, boron worth and critical boron concentration was replaced by the SIMULATE program. The licensee has shown that this change produced improved results.

(3) Modifications were introduced to the PDQ code.

(a) PDQ mesh structure was improved with resulting increase in the number of physical regions of the core for which exact cross sections were calculated. This change improved predictive capability of the code.

(b) A "soft" spectrum cross section formalism was replaced by a "hard" spectrum cross section formalism in the water region around the cruciform control rods and rod followers. A comparison of the computed results with measured data (Reference 13) has indicated that the use of "hard" spectrum cross sections was more exact.

(c) Burnup dependent buckling was introduced into the 2D-PDQ Nuclear Design Model. Making buckling power dependent removed the inaccuracies resulting in under-prediction of power in the center of the core and overpredicting it on its periphery as cycle burnup increased.

The first two modifications of the calculational methods were previously reviewed in connection with the Maine Yankee reload. The modifications to the PDQ code were specifically reviewed for this reload. All the modifications were found to be acceptable for Yankee-Rowe.

Based on the information provided by the licensee for this reload and other information available to us in connection with other licensing actions we concluded that the nuclear design for Core XIV meets all the required safety criteria.

Thermal-Hydraulic Design

The methods of analysis employed by the licensee in evaluating the thermal hydraulic characteristics of Core XIV are virtually identical to the methods used in the evaluation of Core XIII (Reference 4). Only minor adjustments were made in code input data to reflect the changes in the reload cycle power distributions. The results of this analysis indicate some variations in the hot channel parameters relative to those calculated for Core XIII. Maximum linear rod power increased by about 7 percent, the center-line pellet temperature by 570°F and DNB ratio decreased by 6 percent. However, all parameters remain within the acceptable design limits. The hot channel factors also exhibited small changes. The total heat flux factor increased by $\Delta(F_Q) = 0.17$ and total enthalpy rise factor decreased by $\Delta(F_{\Delta H}) = 0.06$. They remained, however, below the acceptable design values of $F_Q = 2.76$ and $F_{\Delta H} = 1.81$. Maximum linear rod power was further restricted by the LOCA considerations and hence an additional safety margin was imposed for this parameter.

The change of parameters for the reloaded core did not affect the safety limit curves. The same curves as specified for Core XIII apply for the Core XIV reload (Reference 4).

We have reviewed the thermal hydraulic design of Core XIV and conclude that in most cases the difference in the thermal hydraulic parameters between Core XIII and Core XIV is insignificant. In addition, since none of these parameters exceed the limit set by the design criteria, the thermal hydraulic design of Core XIV was found to be acceptable.

Safety Analysis

The licensee has reviewed the anticipated occurrences and the postulated accidents which were reported in Reference 10. Each transient was considered and compared with the analyses presented in the above reference which was previously approved by the NRC (Reference 11). In most cases it was found that the effect of the postulated incidents in Core XIV could be accommodated within the conservatism of the initial assumptions used in previous applicable safety analyses. For those incidents which were not bounded by the previously approved analyses, new safety analyses were provided which demonstrated that the applicable design basis limits were not exceeded.

The licensee has shown that the following incidents are bounded by the reference cycle (Core XI) analyses: control rod withdrawal, boron dilution at power, failure to borate during cooldown, control rod drop, isolated loop startup, loss of load, loss of feedwater flow, steam generator tube rupture and the small break LOCA.

We have reviewed the justifications given by the licensee for not requiring to reanalyze these incidents and find that for all these cases basic parameters in the reload core fall within the limits set by the reference cycle. The corresponding analyses are therefore bounding and need not be repeated.

The licensee has reanalyzed the remaining incidents which were shown not to be bounded by the reference cycle analyses. The results of these analyses are discussed below:

(1) Boron dilution during refueling and at hot standby.

These incidents were reanalyzed with new input parameters for Core XIV and new values for the minimum time to reach criticality were obtained. Although for boron dilution at hot standby this time is shorter than in the reference cycle analysis, the licensee has demonstrated (Reference 2) that sufficient time is provided for the operator to take appropriate action to prevent the reactor from becoming critical.

(2) Loss of Coolant Flow

The licensee has reassessed the effect of a less negative moderator temperature coefficient existing in the reloaded core. This was done by reanalyzing the most limiting loss of flow incident which could occur for the Core XIV operating condition. The analysis showed that the more favorable values of other Core XIV parameters more than compensated for the effect of a less negative moderator coefficient and that the limits set by the reference analysis were not exceeded.

(3) Control rod ejection accidents at full and zero power

The control rod ejection accident at full power was not bounded by the reference analysis because the value of the moderator temperature coefficient was less negative for the reload core. A reanalysis was therefore performed with a conservative assumption of zero moderator temperature coefficient. The analysis has verified that no clad damage or fuel melting would occur as a result of the accident.

The control rod ejection accident at zero power could not be bounded by the reference analysis due to higher ejected rod worth for the reload core. The new analysis performed with conservative values for input parameters provided an adequate proof that no clad damage or fuel melting could be caused by rod ejection in Core XIV.

(4) Steamline rupture accident

For this accident a new analysis had to be performed in order to calculate the increase in core reactivity caused by the moderator cooldown resulting from the postulated steamline rupture in Core XIV. This value was subsequently used in determining the Power Dependent Rod Insertion Limits (PDIL) which would provide sufficient shutdown reactivity margin. The analysis has indicated that the addition of reactivity resulting from the cooldown could be conservatively accommodated within the presently existing shutdown margin with no need for changing PDILs.

(5) Large break LOCA

Since Cores XIII and XIV have identical hydraulic designs and since the physics parameters used in Core XIII calculations bound the Core XIV values, the limiting break remained unchanged for Core XIII and Core XIV. This break was used in the sensitivity study which was performed to determine the Allowable Rod Linear Heat Generation Rates (ARLHGR) for the reload core. This sensitivity study was carried out for fresh fuel, the most highly exposed, recycled fuel and the highest power, recycled fuel. The licensee has demonstrated that the values of ARLHGR for the fresh and the highest power, recycled fuel are limiting in Core XIV (References 1, 2 and 6). These values are presented in the curve included in the proposed Technical Specifications (FIG. 3.2.1). We have reviewed the sensitivity study and agree with the licensee's findings.

The licensee found that the concentration of boric acid during the hydrostatic and low power physics tests, performed as part of the reload operation, was slightly higher than the concentration used in the post-LOCA long term cooling analysis. This caused some concern that boric acid might precipitate if a LOCA were to occur during these tests. The licensee has addressed this problem by providing a long term coolings analysis assuming increased boric acid concentrations (Reference 14). The analysis has indicated that with the concentration existing during the tests boric acid precipitation could not occur after a postulated LOCA. Our own calculations confirmed the licensee's conclusion.

Relocation of Primary Coolant Pressure Sensor

The licensee has proposed to relocate the channel #2 primary coolant pressure sensor from its present location in the pressurizer to the new location in the main coolant loop. This sensor transmits the signal required to operate the Safety Injection Actuation System. Its relocation was required in order to eliminate delay in sensing coolant pressure during the transient following a LOCA. The relocation of the sensor required resetting the accumulator timer setpoints so that the accumulator pressurization delay time remained unchanged. The licensee has determined that the setpoints have to be changed from its present value (Core XIII) of 4.0 $\pm$ 0.75 to the new value of 11.85 $\pm$ 0.23 sec for Core XIV. In Reference 2 the licensee discussed the quantitative basis for this change. We have reviewed the arguments presented by the licensee and find that the proposed modification would improve the performance of the Safety Injection System. The modification is therefore acceptable.

Startup Testing

The licensee has described the physics startup test program for Core XIV (References 1 and 2). This program includes the test acceptance criteria and the actions to be taken if the acceptance criteria are not met.

The program includes critical boron concentration measurements, control rod operability verifications, rod group reactivity worth measurements, ejected and dropped control rod worth measurements, isothermal temperature coefficient measurements and low power core map. Power ascension tests will consist of core maps at full power. The licensee will submit a startup test report to the NRC in the required period of time. The physics startup test program has been reviewed by us and found to be acceptable.

Summary of Findings

From our review of the material submitted by the licensee on the Core XIV reload we find:

(1) The mechanical and thermal design of the new fuel is identical to Core XIII hence new analyses for fuel performance are not required. The exposure time at which clad flattening is predicted to occur is longer than the maximum fuel exposure in Core XIV. An adequate margin of safety will therefore be available.

(2) The nuclear design of the reload Core XIV is acceptable. Most of the nuclear characteristics are bounded by the previous analyses and it was shown that sufficient shutdown margin will exist for operation with Core XIV. The modifications introduced to the analytical procedures were reviewed and found acceptable.

(3) In most cases considered in the accident analysis the effect of a postulated incident could be accommodated within the conservatism of the reference cycle analyses. Those incidents which could not be bounded by these analyses were reanalyzed. It was found that none of them exceeded safety limits. Introduction of fresh fuel required establishing new limits for the maximum allowable LHGR. These limits were determined by the large break LOCA considerations.

(4) The relocation of the primary coolant pressure sensor to its new position in the main coolant loop and the resulting change of the accumulator timer setpoints were found to improve the performance of the system by providing simultaneous signals from both channel #1 and channel #2 sensors.

(5) The proposed Technical Specifications provide the necessary requirements for safe Core XIV operation and are therefore acceptable.

B. <u>Steam Generator Inspections and Proposed Technical Specifications</u>

<u>Discussion</u>

During the Core XIV refueling and maintenance outage the licensee performed a steam generator inspection program. The purpose of this program was to determine the condition of the steam generators and to make necessary repairs based on the inspection results. The licensee included a summary of the results of this inspection in the November 27, 1978 supplement. To provide requirements in the Technical Specifications for subsequent steam generator inspections we have reviewed the licensee's application (Proposed Change No. 119, Supplement 1) dated November 30, 1976, as supplemented by letter dated November 27, 1978. The licensee's proposal was to add a new

Section 3/4.4.10 "Steam Generators" in the Technical Specifications, that would set forth steam generator inspection requirements in conformance with Regulatory Guide 1.83 and the Westinghouse Standard Plant Technical Specifications (STS).

Evaluation

Revision 1 of Regulatory Guide 1.83 and the Westinghouse Standard Plant Technical Specifications (STS) delineate Pressurized Water Reactor (PWR) steam generator inspection programs that are currently acceptable to the NRC staff. The licensee has accepted the STS with two exceptions: The first exception is with regard to references to an Operating Basis Earthquake. Because Yankee-Rowe is a relatively old plant (commercial operation commenced in July 1961), there was no requirement to define the operating basis earthquake (OBE) during the original plant design. Therefore, references to the Operating Basis Earthquake have been deleted from the technical specifications. The requirement to perform additional, unscheduled inspections following a seismic occurrence greater than the OBE and the definition of an unserviceable tube as one containing a defect large enough to affect its structural integrity in the event of an OBE have been deleted. Although the reference to the OBE has been deleted from the definition of an unserviceable tube, the proposed plugging limit of 40% is consistent with that set for other plants designed by Westinghouse and provides adequate allowance for seismic events. The reference to the OBE will be included in the future when it will be established as part of the ongoing Systematic Evaluation Program (SEP) for Yankee-Rowe.

The second exception to the STS is in the definition of tube inspection. As defined in the STS a tube inspection means an inspection from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg side. The early model steam generators at Yankee-Rowe are small generators (1620 tubes/generator) and the inner row tubes have small bend radii which will not allow passage of a standard eddy current testing probe. Therefore, the definition has been changed to require inspection of tubes through the U-bend where practicable. This definition is expanded in the basis where it is explained that the first seven rows of tubes from the inside of the bundle will not allow passage of the standard eddy current testing probe. The main reasons for inspecting the U-bend portions of the tubes is the concern for possible stress corrosion cracking resulting from denting related hour-glassing of the flow slots in the upper tube support plate. This could pull the legs of the U-bends together and result in overstressing the tubes.

The licensee has completed inspections of 100% of the tubes in steam generators No. 1 and No. 4 during Core XIV refueling outage. One hundred percent inspections of steam generators 2 and 3 were performed in July 1977. Results of both these inspection programs revealed no indications of tube denting in any of the Yankee-Rowe steam generators. Approximately 40 tubes which had wastage type indications just above the tube sheet were plugged. Since Yankee-Rowe has not experienced tube denting or tube support plate hourglassing and since the type of degradation observed has been limited to wastage type defects near the tube-sheet, the concern over U-bend cracking is alleviated. However, in the event that denting is ever observed in the Yankee-Rowe steam generators, special procedures for inspecting the small U-bend tubes will have to be developed at that time.

Based on the above considerations we have concluded that the proposed technical specifications adequately conform to the STS and are therefore acceptable.

C. Modification to Enhance the Shutdown Capability

Discussion

During a site visit of Yankee-Rowe in September 1978, by our fire protection review team, a concern was identified about the lack of ability to safely shutdown the plant independent of damage in certain areas inside the turbine building, due to fires.

To alleviate our immediate concern, the licensee has completed a modification during the Core XIV refueling outage to provide a shutdown capability independent of the turbine building. As described in the licensee's October 30, 1978 application (Proposed Change No. 164), supplemented by letter dated November 21, 1978, the modification would consist of a piping interconnection between the new hot leg injection line (approved by us in Amendment 52, issued November 14, 1978) and each of the four steam generator blowdown lines. The new piping would join the steam generator blowdown lines upstream of the existing automatic containment isolation valves. This interconnection would provide a means of feeding the steam generators from the safety injection pumps or a charging pump, independent of equipment inside the turbine building normally used for safe plant shutdown.

To provide assurance against interference with the normal functions of either the Emergency Core Cooling System or the charging system multiple locked manual valves have been installed. The four manual valves, one in each of the new steam generator feed lines would serve as containment isolation valves. The licensee has proposed to include these four valves in the listing of containment isolation valves in Table 3.6-1 in the Technical Specifications.

Evaluation

The manual containment isolation valves proposed to be installed on the new steam generator feed lines would be locked-close during normal plant operation and would satisfy the requirement of General Design Criterion 57, Appendix A to 10 CFR 50 for closed system isolation valves. The piping system design and installation would be in accordance with requirements in the quality assurance program previously approved by us, and would meet or exceed the design requirements of the original piping codes.

The proposed containment isolation valves would be subject to containment leak rate test and the proposed technical specification changes to include these valves under Type C testing in accordance with Appendix J, 10 CFR 50, are therefore acceptable.

We have also reviewed the piping interconnections and valves to determine potential interaction with the Emergency Core Cooling System (ECCS) through the new hot leg injection piping, previously approved by us in Amendment 52. We have concluded that because of the installation of multiple locked-close valves, the modification will not adversely affect the performance of the ECCS.

Based on the above considerations we have concluded that the modifications described in Proposed Change No. 164 enhances the safe shutdown capability in the event of damage in certain areas inside the turbine building, due to potential fires and is therefore acceptable.

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.

Conclusions

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: December 6, 1978

REFERENCES

1. Yankee Atomic Electric Company letter dated September 8, 1978, transmitting Proposed Change No. 163 and Core XIV Performance Analysis.

2. Yankee Atomic Electric Company letter dated November 21, 1978, transmitting additional information.

3. Proposed Change No. 125, "Core XII Refueling," submitted on July 14, 1975 and Supplement No. 3 to Proposed Change No. 125, submitted November 7, 1975.

4. Proposed Change No. 145, Supplement 1, submitted April 13, 1977.

5. Amendment No. 43, Letter from A. Schwencer, DOR/NRC, Attention: Robert H. Groce, August 25, 1977, transmitting Reload Safety Evaluation For Core XIII.

6. Yankee Atomic Electric Company letter dated November 21, 1978, transmitting Proposed Change No. 163, Supplement 1.

7. YAEC-1140, "Yankee Power Spike Model," November 1977.

8. YAEC-1098, "A Study of Xenon Transients in Yankee Rowe Core XII," dated March 1976.

9. S. Borrensen, "A Simplified, Coarse Mesh, Three-Dimensional Diffusion Scheme for Calculating the Gross Power Distribution in a Boiling Water Reactor," Nucl. Sci. and Eng., Vol. 44, Page 37-43 (1971).

10. Proposed Change No. 115, "Core XI Refueling," submitted DOL/AEC on March 29, 1974.

11. Amendment No. 9, Letter from K. R. Goller, DOL/AEC to YAEC, Attention: G. C. Andognini, July 30, 1974.

12. Yankee Atomic Electric Company letter (W. P. Johnson) to NRC, dated October 7, 1977, transmitting Yankee Rowe Core XIII LOCA Core Inlet Temperature and Accumulator Delay Sensitivity Analysis.

13. WMY 78-43 Maine Yankee letter to NRC, dated April 28, 1978.

14. Yankee Atomic Electric Company letter to NRC, dated November 10, 1978.