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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. DPR-3

(CHANGE NO. 123 TO THE TECHNICAL SPECIFICATIONS)

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

DOCKET NO. 50-29

Introduction

By letters dated July 14, 1975 [as supplemented October 10, October 28, November 7 (Proprietary Information appended), November 21, and November 26, 1975]; July 8, 1975 (as supplemented November 24, and November 26, 1975); and September 23, 1975, Yankee Atomic Electric Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-3 for the Yankee Nuclear Power Station (Yankee-Rowe). The purpose of the requests is to change the operating limits and to add requirements relating to the Emergency Core Cooling System (ECCS) in the Technical Specifications. The revisions to the Technical Specifications are based on an acceptable evaluation model that conforms to the requirements of 10 CFR Part 50, §50.46, to permit operation of Yankee-Rowe with Core XII reloaded with new Exxon Nuclear Company (ENC) fuel assemblies and recycled Gulf United Nuclear Fuels Company (GUNF) fuel assemblies irradiated in the preceding Core XI.

Discussion

The Yankee-Rowe reactor core consists of 76 fuel assemblies, each having a 16x16 array of fuel rods. The Core XII reloaded core utilizes a two-regional configuration with the 40 ENC fuel assemblies located at the periphery of the core and the one-cycle exposure GUNF fuel assemblies occupying the interior of the core.

The licensee provided the needed technical information for our review, including a general description of the reload core, detailed mechanical design data on the reload fuel, the results of the nuclear and thermal-hydraulic evaluation, accident and transient analysis in support of the Core XII reload application. Since this is the first application for a Yankee-Rowe reload with ENC fuel assemblies (the second ENC reload application to PWRs following H. B. Robinson, Unit 2), ENC has provided documentation on the ENC, ECCS cooling performance analysis models and computer codes.

We have examined the methods employed by the licensee and ENC and concluded that their application to the design and analyses of the Yankee-Rowe reconstituted Core XII is acceptable. Further, from our review of the available reload information, we conclude that it is acceptable for the licensee to proceed with Core XII operation in the manner proposed. Our review and evaluation of the licensee's Core XII reload submittal is discussed in the following sections.

Evaluation

A. Fuel and Mechanical Design

The proposed reload core (Core XII) consists of 40 fresh Exxon Nuclear fuel assemblies and 36 one-cycle exposure GUNF fuel assemblies. A comparison of the mechanical designs for the ENC and GUNF fuel assemblies indicates the following differences:

1. The ENC fuel assembly has an open lattice design which uses stainless steel guide bars for structural support. The GUNF design has a stainless steel shroud around the assembly. The fuel rod pitch in the ENC design has been increased from 0.468 inch to 0.472 inch to compensate for the removal of the shroud.

2. The upper nozzle of the ENC fuel assembly design permits removal and reinsertion of fuel rods.

3. The ENC spacer grid is a stainless steel and Inconel bimetallic assembly representative of the manufacturer's standard product line.

4. The lower nozzle of the ENC design is mechanically attached to the guide bars.

5. The ENC fuel pellet design for Yankee-Rowe is the same as all other ENC pellet designs but shorter than the GUNF Core XI pellet design.

The licensee has performed proof tests to assess the structural integrity of the new fuel assembly design. These tests examined the strength of the locking system between the upper nozzle assembly and the guide bars and the strengths of the spacer assembly, lower nozzle, and upper tie plate. Tests were also conducted to determine the support stiffness of the spacer dimples used in design calculations and the functional loading between the spacer grid contact points and the fuel rod cladding. In addition, ENC performed a short duration combined fretting wear and pressure drop test on a prototypical Yankee-Rowe fuel assembly.

Several analyses using generically approved codes confirmed that the ENC fuel assembly design limits were met. These analyses included strain limit calculations, cladding collapse calculations, and fuel temperature calculations (including the effects of fuel densification). We found that the ENC engineering methods, design limits, and results were satisfactory.

Nine one-cycle exposure Core XI GUNF fuel assemblies to be reinserted in Core XII were visually inspected and no apparent rod bow was observed. Examination of similarly designed GUNF BWR and PWR fuel, irradiated between approximately 5000 to 15,000 MWD/MTU, also indicated no apparent rod bow. In addition, the ENC and GUNF fuel designs have a larger thickness to diameter ratio than the standard Westinghouse 15x15 fuel design (0.0066 compared to 0.0057). The distance between the spacer grids of the ENC and GUNF fuel is considerably smaller than that of Westinghouse fuel (14.7 compared to 26.2 inches). These differences should result in decreased rod bow compared to that measured in the Westinghouse fuel assemblies.

Yankee-Rowe is one of the first plants to use ENC, PWR type reload assemblies. ENC's only other operating experience to date on PWR fuel has been with two lead assemblies in the Rochester Gas and Electric Corporation's R. E. Ginna reactor which were inspected after one cycle and had no leakers. Responding to our question regarding surveillance of ENC fuel assemblies, the licensee has committed to an inspection program of at least 5 of the 40 Core XII ENC reload assemblies during each of the next two refueling outages. Specific components to be examined will include top and bottom nozzles, guide bars, spacer grids, and fuel tubes. If irradiation of the Core XII fuel will be continued beyond the next two cycles, the surveillance program will also be continued through the highest burnup achieved.

From our review of the information provided by the licensee on the Core XII reload submittal, we have determined.

1. The ENC fuel rod mechanical design is compatible with the previously approved GUNF Core XI fuel design and provides acceptable engineering safety margins.

2. The analysis performed acceptably accounts for the effects of fuel densification, cladding collapse, and cladding strain.

3. The results of the out-of-pile proof tests reported verify the adequacy of the fuel mechanical design.

4. The surveillance program for the ENC Core XII fuel assemblies is acceptable for evaluation of the fuel performance.

We, therefore, conclude that, from a mechanical design standpoint, reloading of the core with the previously approved GUNF fuel assemblies in Core Region A and with the new ENC fuel assemblies in Core Region B and operation of Yankee-Rowe with the reloaded Core XII are acceptable.

B. Nuclear Design

The Core XII reload configuration departs from the three-zone pattern used in preceding Yankee-Rowe cores. Core XII includes substantially more than the usual amount of fresh fuel, and therefore has a less negative BOL (Beginning-of-Life) moderator coefficient due to a higher boron concentration for a lower average burnup. Core XII also has a larger effective delayed neutron fraction than would an equilibrium reload core because there is less plutonium present. This occurs because of the absence of two-cycle exposure assemblies in Core XII and because there are fewer burned assemblies. The values used in the accident analysis are chosen in a conservative manner for each analysis. The range of data calculated for Core XII, therefore, lie within that used for the accident analysis for the evaluation of the preceding Core XI which we have approved previously.

The licensee's calculation of control group worths for Core XII indicates that there is a substantial excess margin over a 1%Δρ design shutdown margin allowance throughout the cycle life. A 7.5% uncertainty allowance has been included in the calculation of rod worth. Comparison with Core XI calculations of measured worths indicates that this uncertainty allowance is conservative. Startup measurements will provide additional verification that the shutdown margin will be maintained throughout Cycle XII operation.

The nuclear calculations for Cycle XII were performed by the licensee using the same calculational methods employed for Core XI which we have previously found acceptable.

Peak linear heat generation rates (LHGR) are restricted as a function of Cycle XII exposure as indicated in the section on ECCS analysis. The licensee has proposed changes to the Technical Specifications which limit the reactor power as follows:

$$\text{Allowable Fraction of Power} = \frac{\text{LOCA limit LHGR}}{\text{Full power LHGR}}$$

The full power LHGR is the product of:

1. the core average LHGR at full power (4.34 kW/ft),

2. the measured total nuclear peaking factor, F_Q^N, (determined every 1000 equivalent full power hours),

3. a factor which allows for the increase in peaking factor if the rods are inserted to their insertion limit (1.0 for unrodded measurement of F_Q^N),

4. a factor which allows for the maximum increase in peaking factor which could occur from zenon redistribution (1.10),

5. a flux peaking augmentation factor which allows for axial gaps in the fuel from densification (axially variable),

6. a measurement uncertainty allowance (1.05),

7. an allowance for power level uncertainty (1.03),

8. an allowance for engineering uncertainty (1.04), and

9. an allowance for the pellet stack height shortening from fuel densification (1.009).

Items 3 through 9 impose a total uncertainty penalty in excess of 30%. This is conservative because it is extremely unlikely that all of these factors would have their combined maximum adverse effect at any given time. The proposed Technical Specification D.2.C will effectively limit the reactor power to a level consistent with the LHGR used in the LOCA analysis and it will do so in a very conservative manner. We, therefore, find it acceptable.

C. Thermal and Hydraulic Design

The features in the design of the new ENC fuel assemblies that might affect the thermal and hydraulic performance of Core XII involve the change of the structural supports and a slight increase in the clearance between fuel rods. The GUNF fuel has perforated shrouds that provide structural supports for the fuel rod assemblies. The new ENC fuel is of an open lattice design that utilizes eight non-fueled tie rods in lieu of shrouds for providing the structural supports for the fuel rod assemblies. The new ENC fuel assemblies and the one-cycle exposure GUNF fuel assemblies in Core XII have the same fuel rod dimensions. However, the clearance between fuel rods is slightly larger in the new ENC fuel to compensate for the absence of the shrouds.

The hydraulic resistance of the new ENC fuel assemblies is less than that of the one-cycle exposure GUNF assemblies in Core XII because of the absence of shrouds (resulting in decreased wetted surface) and the reduced contraction/expansion losses across the spacers. As a result, the average flow in the new ENC assemblies is higher than in the GUNF assemblies. In our review, we have also considered potential flow diversion between the different fuel types at the spacer elevations because of the differences in hydraulic resistance. We find that the mismatch in hydraulic characteristics in Core XII is less pronounced (therefore, less flow diversion) than existed in Core X which contained stainless steel and zircaloy clad fuel assemblies of distinctly different designs.

The licensee's previous thermal-hydraulic analysis for Core X was based on Cat II calculations. We found the results of those conservative calculations acceptable to support operation with Core X (containing zircaloy clad fuel with perforated shrouds) at a peak linear heat generation rate of 12.2 kW/ft with a hot channel enthalpy rise factor of 2.24. For Core XI which consisted of 72 zircaloy clad assemblies and four stainless steel clad assemblies, the licensee's thermal and hydraulic calculations were based on Cobra-3C. Those calculations showed that the DNB ratios predicted for a range of abnormal operating transients in Core XI all exceeded 2.0 for design hot channel conditions of 12.9 kW/ft and $F_{\Delta H}$ of 1.81.

The licensee's LOCA analysis for Core XII shows that the limiting hot channel conditions will be lower than the design conditions used for the thermal and hydraulic design of Core X or Core XI. Because the operating limits for Core XII are more restrictive (due to the restrictions imposed by the ECCS core cooling performance analysis) than those previously approved by us for Core X and Core XI, we find that the thermal and hydraulic design of Core XII to be acceptable.

D. Accident and Transient Analysis

1. ECCS Cooling Performance (LOCA) Analysis

a. Evaluation Model

The licensee has evaluated through ENC the Yankee-Rowe ECCS cooling performance using a calculational model that conforms to the requirements of 10 CFR Part 50, §50.46.

The calculational model used by ENC for Yankee-Rowe is similar to the approved H. B. Robinson ECCS performance evaluation model addressed in the staff's Safety Evaluation of September 11,

1975, and its supplement. Time step and nodalization studies have been submitted in support of the model. We have reviewed the use of the H. B. Robinson model for the Yankee-Rowe ECCS performance evaluation with respect to the differences in plant design, particularly the shorter core, the thinner fuel rods, and the different accumulator arrangement in the Yankee-Rowe facility. We have determined that the H. B. Robinson model conservatively accommodates those differences and therefore conclude that its application to the Yankee-Rowe plant is acceptable.

b. Break Spectrum

Using the acceptable evaluation model described in the preceding section, the licensee provided in the November 26, 1975 submittal the results of the analysis of a limited break spectrum [the largest double-ended cold leg(DECL)] guillotine break and an equivalent DECL split break. This analysis identified the DECL split break at the pump discharge is to be the most limiting break, with a calculated peak clad temperature of 1883°F, well below the acceptable limit of 2200°F as specified in 10 CFR Part 50, §50.46(b). In addition, the maximum local metal/water reaction of less than 2.2% and the total core wide metal/water reaction of less than 1% were within the allowable limits of 17% and 1%, respectively. Based on this analysis the licensee proposed to limit the peak linear heat generation rate (LHGR) to 9 kW/ft for implementing the analysis results.

We have compared the results of the breaks analyzed for Yankee-Rowe with the results of analyses that we have previously reviewed and approved for various other facilities, particularly that which was performed with the approved ENC model for the H. B. Robinson facility. Based on this comparison, we have concluded that the licensee has provided acceptable bounding calculations, has identified the most limiting break, and has proposed acceptable Technical Specification limits for the LHGR which assure operation of Yankee-Rowe with Core XII in compliance with 10 CFR Part 50, §50.46.

To improve the effectiveness of the Yankee-Rowe Technical Specifications, we have also included in Section D.2.e(4) of Appendix A a requirement to maintain the accumulator water level at a minimum of 700 ft^3, as proposed in the licensee's September 23, 1975 submittal. The licensee has used this value as an input in the ECCS cooling performance evaluation for Core XII which we find acceptable.

c. ECCS Containment Pressure Evaluation

Appendix K to 10 CFR Part 50 requires that the effect of operation of all the installed pressure reducing systems and processes be included in the ECCS evaluation. For the evaluation it is conservative to minimize the containment pressure since this will increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core. Following a loss-of-coolant accident (LOCA), the pressure in the containment will be increased by the addition of steam and water from the primary reactor system into the containment atmosphere. After initial blowdown, heat flow to the ECCS water from the core, primary metal structures, and steam generators will produce additional steam. This steam together with any ECCS water spilled from the primary system will flow through the postulated break into the containment. This energy will be released to the containment during both the blowdown and later ECCS operation phases; i.e., reflood and post-reflood phases.

Energy removal occurs within the containment by several means. Steam condensation on the containment shell and internal structures serves as a passive energy heat sink that becomes effective early in the blowdown transient. Yankee-Rowe has a bare steel containment shell capable of transferring heat generated during a LOCA from inside the containment to the atmosphere outside the containment. No other systems are needed for heat removal. When the energy removal rate exceeds the rate of energy addition from the primary system, the containment pressure will decrease from its maximum value.

The ECCS containment pressure calculations for Yankee-Rowe were done using the ENC ECCS evaluation model. We have reviewed the ENC ECCS evaluation model and discussed it in our Safety Evaluation dated September 11, 1975. We concluded that ENC's containment pressure model was acceptable for the ECCS cooling performance evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

The licensee submitted justification for the containment input data for Yankee-Rowe by letter dated November 26, 1974. The licensee has reevaluated the containment net free volume

and the passive heat sinks with regard to the conservatism for the ECCS performance analysis. This reevaluation was based on measurements within the as-built containment to which additional margin was added. There are no active heat removal systems installed in the Yankee-Rowe containment.

ENC has used a constant value of 11.8 psig in the core heatup calculations. We have compared this value with our independent calculation of the containment pressure as a function of time based on ENC's calculated mass and energy release data and the containment input data for the Yankee-Rowe plant. This analysis demonstrates the value of 11.8 psia used by ENC to be conservative to a time of 115 seconds after the accident. (Peak cladding temperature is predicted to occur before 115 seconds.)

We have concluded that the containmen[illegible]ure analysis for Yankee-Rowe is reasonably conserv[illegible]refore, conforms with Appendix K to 10 CFR Par[illegible]

d. Single Failure Criterion

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operational shall be those available after the most damaging single failure of ECCS equipment has occurred. The worst single failure of ECCS equipment which would minimize the ECCSavailable to cool the core was identified by the licensee as the failure of one of the three emergency diesels to start.

A review of the Yankee-Rowe piping and instrumentation diagrams indicated that the inadvertent actuation of specific motor-operated valves could affect the appropriate single failure assumptions. We had identified the following motor-operated valves which did not satisfy the single failure criterion.

MOV#	Component Function	Failure Mode
SI-MOV-1	Accumulator isolation	Closure would cause loss of accumulator injection to RCS
SI-MOV-22 SI-MOV-23 SI-MOV-24 SI-MOV-25	Isolation valves from SI header to cold leg injection lines	Closure would cause reduction of ECC flow to RCS
SI-MOV-4	Crossover from LPSI pump discharge to HPSI pump suction	Closure of the valve would prevent boosting of SI flow to HPSI pumps for small break
SI-MOV-46	Flow control from HPSI pumps	Closure of the valve would eliminate HPSI flow to RCS
SI-MOV-49	HPSI test/recirc line valve (mini-flow)	Fail closed-overheat pumps
CH-MOV-522	Isolation valve between charging pump discharge and LPSI discharge header	Valve should remain closed
CH-MOV-523 CH-MOV-524	Isolation valve from charging pump discharge header to hot leg of loop 4 (provide normal charging function and hot leg injection for long-term recirculation).	Closure of either valve during long term recirculation would prevent hot leg injection
CS-MOV-536 CS-MOV-537 CS-MOV-538 CS-MOV-539	Isolation valves from SI header to cold leg injection lines	Closure would reduce ECC flow to RCS
CS-MOV-532	LPSI test/recirculation line valve (full flow)	Opening of valve would reduce LPSI flow to RCS
CS-MOV-533 CS-MOV-535	LPSI pump discharge isolation valve	Closure would prevent LPSI flow to RCS

MOV#	Component Function	Failure Mode
CS-MOV-301 CS-MOV-302 CS-MOV-309 CS-MOV-310 CS-MOV-318 CS-MOV-319 CS-MOV-325 CS-MOV-326	RCS loop isolation valves	Closure would isolate a RCS loop
SI-SV-56 SI-SV-57	Accumulator nitrogen pilot relief valves	Premature nitrogen relief would reduce accumulator injection

The licensee has reviewed the existing Emergency Operating Procedures and the consequences of the identified single failures and has proposed the following.

(1) The present Emergency Operating Procedures will be modified to eliminate the ECCS "cutback" mode of operation.

(2) During power operation, a.c. power will be removed from the following motor-operated valves with the valves in their normally opened position by removal of the circuit breaker from the motor control center: SI-MOV-1, SI-MOV-4, SI-MOV-22, SI-MOV-23, SI-MOV-24, SI-MOV-25, SI-MOV-46, and SI-MOV-49.

(3) Valves CH-MOV-522, CH-MOV-523, and CH-MOV-524 will be rewired to a common circuit breaker and a second series breaker will be installed. During power operation, a.c. power will be removed from CH-MOV-522, CH-MOV-523, and CH-MOV-524 with CH-MOV-522 in its normally closed position and CH-MOV-523 and CH-MOV-524 in their normally opened position by opening both of the series breakers.

(4) Using EICSB Branch Technical Position 18 as guidance, position indication for valves CH-MOV-522, CH-MOV-523, and CH-MOV-524 will be provided by rewiring, making the position indication features independent of breaker position. In addition, valve position indication for CH-MOV-522 will be provided by operator monitoring of flow through flow indicator FI-2 and valves CH-MOV-523 and CH-MOV-524 will be placed under periodic operator surveillance to verify their proper alignment.

(5) During power operation, a.c. power will be removed from the following motor-operated valves with the valves in their normally opened position by disconnecting the power cables as they leave the motor starters: CS-MOV-533, CS-MOV-535, CS-MOV-536, CS-MOV-537, CS-MOV-538, CS-MOV-539, MC-MOV-301, MC-MOV-302, MC-MOV-309, MC-MOV-310, MC-MOV-318, MC-MOV-319, MC-MOV-325, MC-MOV-326.

(6) During power operation, a.c. power will be removed from motor-operated valve CS-MOV-532 with the valve in its normally closed position by disconnecting the power cables as they leave the motor starter. In addition, this valve will be permitted to be opened to provide the capability for mixing of the coolant in the Safety Injection tank on a quarterly basis for a period of approximately thirty minutes.

(7) The control circuitry to each of the accumulator pilot operated relief valves, SI-SV-56 and SI-SV-57, will be modified to include level switch contacts in each of the conductors to and from the valve solenoid operators. This will preclude a single failure from opening the accumulator relief valves before termination of the injection mode of operation.

(8) As noted in item 3 above, CH-MOV-523 and CH-MOV-524 will be deactivated in their normally opened position. This will assure that normal charging flow and long term hot leg recirculation will be provided. Since hot leg injection during the injection mode of operation has not been justified as an acceptable procedure, the licensee has proposed to prevent hot leg injection during injection mode operation by tripping the charging pumps upon receipt of a safety injection actuation signal. Safety injection actuation is comprised of systems "A" and "B" which together provide redundant actuation signals to all safety injection equipment. In order to provide the required tripping function and meet the single failure criterion, the licensee has proposed to provide two series contactors in the power supply circuitry to each charging pump. The "A" safety injection signal initiates a trip to one contactor while the "B" safety injection signal initiates a trip to the other contactor. This modification will assure that charging pump flow will be provided to the hot legs only when required for normal charging and long term recirculation, and prevent hot leg injection during the injection phase of operation.

(9) The onsite emergency power system consists of three diesel generator busses, each powering a high pressure and a low pressure safety injection pump. The ECCS analysis demonstrates that the loss of one diesel and its associated equipment can be tolerated without exceeding Appendix K requirements. In our review, however, we have determined that two of the three diesel generator busses are not independent. Busses 1 and 3 are each a normal and an alternate source of power for a swing bus. Yankee-Rowe has two swing busses which power redundant ECCS valve trains and are each connected to busses 1 and 3 through an automatic transfer switch. The automatic transfer capability is not required to meet ECCS acceptance criteria. The flexibility afforded by these swing busses is far outweighed by the reality that a single failure can compromise two of the three safety trains. We have required that this aspect of the onsite emergency power system design meet the single failure criterion and conform to the recommendations of Regulatory Guide 1.6. To this end, the licensee has proposed to rack out and lock the alternate supply breakers and within 60 days from the date of startup with Core XII propose and provide mechanical interlocks so that the normal and alternate supply breakers cannot be closed simultaneously. Also, we have required that the automatic transfer switch feature be removed and that the switch be operated manually. We find that these modifications will bring the onsite emergency power system into conformance with the single failure criterion and are therefore acceptable.

Conclusions

The staff has reviewed the licensee's proposals for satisfying the single failure criterion and has found them to be acceptable. With the modifications, as discussed above, the plant will satisfy the requirements of Appendix K to 10 CFR Part 50 of the Commission's regulations.

e. <u>Long Term Boron Concentration Buildup</u>

We have reviewed the licensee's proposed emergency operating procedures and the systems designed for preventing excessive boric acid buildup in the reactor vessel during the post-LOCA long term cooling period. The licensee has proposed for the first 20 to 24 hours after a LOCA to inject borated water

from the containment sump into the RCS cold legs by means of the purification pumps. After 20 to 24 hours, the ECCS will be realigned and the borated solution will be injected simultaneously into the hot and cold legs. This will be accomplished by diverting a portion of the purification pump output into the suction of the charging pumps which will deliver the flow to the hot legs. A hot leg injection flow of 25.2 gpm is required to control the boric acid concentration in the core.

We have reviewed the procedure and concluded that it will satisfactorily maintain the concentration of boric acid in the core below the solubility limit provided that:

(1) Power is disconnected to motor-operated valves CH-MOV-523 and CH-MOV-524 with the valves in their normally opened position to assure the delivery of the required hot leg injectant (See Section D.1.d above), and

(2) The licensee utilizes existing flow metering instrumentation for measuring and controlling the hot leg recirculation flow to assure that the hot leg recirculation flow required to control the boric acid concentration in the core is provided.

f. <u>Submerged Valves</u>

The licensee has submitted an analysis on July 8, 1975, which reviewed the Yankee-Rowe equipment arrangement. We have concluded that no valve motors within containment required for ECCS operation or long-term core cooling will become submerged following a LOCA.

Conclusions

Based on our review, we conclude that:

1) The ECCS cooling performance (LOCA) analysis submitted by the licensee is in conformance with the requirements of Appendix K to 10 CFR Part 50. Additional analyses will be submitted by the licensee which will confirm that the trends predicted for the H. B. Robinson plant are appropriate for reference by Yankee-Rowe, and that the double-ended cold leg split break is the most limiting break size for Yankee-Rowe.

2) The ECCS cooling performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR Part 50, §50.46(b).

3) ECCS cooling performance will be adequate despite any postulated failure of a single active component.

4) Adequate systems and procedures exist to provide long term cooling to the reactor vessel.

2. Control Rod Ejection Accident

For the reference cycle (Cycle XI) a rod ejection analysis was performed using the most limiting parameters during the core life. The input parameters are more favorable for the reload cycle (Core XII) except for the delayed neutron fraction, which is slightly lower. However, using the delayed neutron fraction to compute the ejected rod worth for the limiting case, the value for the reference cycle is 0.86 dollars while for the reload cycle the value is 0.46 dollars. Therefore, the results for the reload cycle will be bounded by those for the reference cycle and are acceptable.

3. Control Rod Drop Incident

The licensee's bounding analysis of the control rod drop incident indicates that damage would not result from this incident even if no drop in core power were assumed. We find that the licensee's analysis and results are acceptable.

4. Control Rod Withdrawal Incident, Boron Dilution Incident, Isolated Loop Startup Incident, Loss of Load Incident, Loss of Feedwater Flow Incident, Loss of Coolant Flow Incident, Steam Line Rupture Accident, and Steam Generator Tube Rupture Accident

Transient and accident analyses were performed for the reference cycle (Core XI) using the most limiting parameters during the core life. For the reload cycle, the input parameters are more favorable than for the reference cycle. Therefore, the results for the reload cycle will be bounded by those for the reference cycle. We find this acceptable.

5. Other Accidents and Transients

The remaining accidents and transients in the licensee's FSAR are not affected by the proposed core design changes and therefore the previous acceptable results still apply.

Summary of Findings

From our review of the material submitted by the licensee on the Core XII reload, including the ECCS cooling performance evaluation, we find:

1. The mechanical design of the new ENC fuel, the nuclear and thermal-hydraulic analyses, and the analyses of accidents and transients are acceptable.

2. The ECCS cooling performance for Core XII has been calculated with an approved evaluation model in conformity with Appendix K and meets the acceptance criteria in 10 CFR Part 50, §50.46(b).

3. The modifications to the ECCS to preclude single failures and to prevent boron precipitation during the long term core cooling phase following a LOCA are acceptable. The modifications as described in this Safety Evaluation must be completed before proceeding with power ascension following completion of the power physics testing.

4. The proposed Technical Specifications, implementing the ECCS cooling performance evaluation results, provide acceptable limits (these limits are more severe than the restrictions in the Commission's December 27, 1974 Order for Modification of License, which they supersede) for the safe operation of Yankee-Rowe with Core XII.

Conclusions

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: DEC 4 1975

REFERENCES

"Order for Modification of License", letter sent to Yankee Atomic Electric Company (YAEC) from Robert A. Purple dated December 27, 1974.

Letter from D. E. Vandenburgh to Robert A. Purple dated July 8, 1975, submitting YAEC's Proposed Change No. 117, Supplement No. 3.

Letter from D. E. Vandenburgh to Robert A. Purple dated November 24, 1975, submitting YAEC's Proposed Change No. 117, Supplement No. 5.

Letter from D. E. Vandenburgh to Robert A. Purple dated November 26, 1975, submitting Proposed Change No. 117, Supplement No. 6.

Letter from W. P. Johnson to Robert A. Purple dated July 14, 1975, submitting Proposed Change No. 125.

Letter from D. E. Vandenburgh to USNRC dated October 10, 1975, submitting Proposed Change No. 125, Supplement No. 1.

Letter from D. E. Vandenburgh to the Office of Nuclear Reactor Regulation dated October 28, 1975, submitting Proposed Change No. 125, Supplement No. 2.

Letter from D. E. Vandenburgh to the Office of Nuclear Reactor Regulation dated November 7, 1975, submitting Proposed Change No. 125, Supplement No. 3 (Proprietary Information appended).

Letter from D. E. Vandenburgh to the Office of Nuclear Reactor Regulation dated November 21, 1975, submitting Proposed Change No. 125, Supplement No. 4, Revision 1.

Letter from D. E. Vandenburgh to the Office of Nuclear Reactor Regulation dated November 26, 1975, submitting Proposed Change No. 125, Supplement No. 5.

Letter from D. E. Vandenburgh to Robert A. Purple dated September 23, 1975, submitting Proposed Change No. 132.

Safety Evaluation Report Regarding Review of the Exxon Nuclear Company PWR ECCS Codes and the H. B. Robinson Reactor ECCS Evaluation Model for Conformance to All Requirements of Appendix K to 10 CFR Part 50 by the Office of Nuclear Reactor Regulation, USNRC, September 1[illegible], 1975, and Supplement No. 1 dated Novembe[illegible] 1975.