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

YANKEE ATOMIC ELECTRIC COMPANY

DOCKET NO. 50-029

YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 128
License No. DPR-3

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by Yankee Atomic Electric Company (the licensee) dated September 14, 1989 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-3 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 128, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 128

FACILITY OPERATING LICENSE NO. DPR-3

DOCKET NO. 50-029

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
1-6	1-6
3/4 1-24	3/4 1-24
3/4 1-28	3/4 1-28
3/4 1-29	3/4 1-29
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-4	3/4 2-4
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
B 3/4 1-1	B 3/4 1-1
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
6-16	6-16
6-17	6-17

PURGE - PURGING

1.32 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

MEMBER(S) OF THE PUBLIC

1.33 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the production of electricity.

SITE BOUNDARY

1.34 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee. Any area within the site boundary used for residential quarters or recreational purposes shall be considered to be beyond the site boundary for purposes of meeting gaseous effluent dose specifications. (Realistic occupancy factors shall be applied at these locations for the purposes of dose calculations.)

SOLIDIFICATION

1.35 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

CORE OPERATING LIMITS REPORT

1.36 CORE OPERATING LIMITS REPORT - This report is the unit-specific document that provides the core operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.4. Plant operation within these operating limits is addressed in individual specifications.

LIMITING CONDITION FOR OPERATION (Continued)

b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1.1 is determined at least one per 12 hours, and

c. A power distribution map is obtained from the incore detection system and F_Q and $F^N \Delta H$ are verified to be within their limits within 72 hours.

d. The THERMAL POWER level is reduced to $\leq$75% of RATED THERMAL POWER within one hour and within the next 4 hours the Power Range and Intermediate Power Range Neutron Flux high trip setpoint is reduced to $\leq$108% of the 75% of allowable THERMAL POWER, or

e. The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$8 inches of the inoperable rod within one hour while maintaining the rod sequence and insertion limits as specified in the CORE OPERATING LIMITS REPORT. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each control rod shall be determined to be within the limit by verifying the individual rod positions at least once per 4 hours.

4.1.3.1.2 Each control rod not fully inserted shall be determined to be OPERABLE by movement of at least 4 inches in any one direction at least once per 31 days.

4.1.3.1.3. The maximum reactivity insertion rate due to withdrawal of the highest worth control rod group shall be determined not to exceed 1.5 x 10^{-4} $\Delta k/k$ per second at least once per 18 months.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control groups shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control groups inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control groups to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control group shall be determined to be within the insertion limits at least once per 4 hours.

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*See Special Test Exceptions 3.10.2 and 3.10.4.

#With $K_{eff} \geq 1.0$.

Figure 3.1-2

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3/4.2 POWER DISTRIBUTION LIMITS

PEAK LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The peak linear heat generation rate (LHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT during steady-state operation.

APPLICABILITY: MODE 1.

ACTION:

With the peak LHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

a. Within 15 minutes reduce THERMAL POWER to not more than that fraction of the RATED THERMAL POWER as expressed below:

$$\text{Fraction of RATED THERMAL POWER} = \frac{\text{Limiting LHGR}}{\text{Peak Full Power LHGR}}$$

b. Within 4 hours reduce the Power Range and Intermediate Power Range Neutron Flux high trip setpoint to $\leq$108% of the fraction of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The peak LHGR shall be determined to be within the limits specified in the CORE OPERATING LIMITS REPORT using the incore detection system to obtain a power distribution map:

a. Prior to initial operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 1,000 EFPH,

c. The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The below factors shall be included in the calculation of peak full power LHGR:

a. Heat flux power peaking factor, F^N_q, measured using the incore detection system at a power $\geq$10%.

b. The multiplier for xenon redistribution is a function of core lifetime specified in the CORE OPERATING LIMITS REPORT. In addition, if Control Rod Group C is inserted outside the operating band for 100% allowable power, allowable power may not be regained until power has been at or below a reduced level defined below for at least twenty-four hours with Control Rod Group C within the operating band for 100% allowable power.

Reduced Power = Allowable fraction of full power times multiplier specified in the CORE OPERATING LIMITS REPORT

Exceptions:

1. If the rods are inserted outside the operating band for 100% allowable power and power does not go below the reduced power calculated above, hold at the lowest attained power level for at least twenty-four hours with Control Rod Group C within the operating band for 100% allowable power before returning to allowable power.

2. If the rods are inserted outside the operating band for 100% allowable power and zero power is held for more than forty-eight hours, no reduced power level need be held on the way to the allowable fraction of full power.

c. Shortened stack height factor, 1.009.

d. Measurement uncertainty:*

1. 1.05, when at least 17 incore detection system neutron detector thimbles are OPERABLE, or

2. 1.068, when less than 17, and greater than or equal to 12, incore detection system neutron detector thimbles are OPERABLE.

Figure 3.2-1

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Figure 3.2-3

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Figure 3.2-4

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3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements are a function of the plant operating status. For critical conditions, minimum shutdown margins are limited by the Power Dependent Insertion Limits (PDIL) as specified in the CORE OPERATING LIMITS REPORT. For 470°F $\leq T_{avg}$, the requirement for a SHUTDOWN MARGIN is established by postulated steam line break considerations with ECCS and NRVs available and covers the requirements to preclude inadvertent criticality. For 330°F $\leq T_{avg} <$ 470°F, the requirement for a SHUTDOWN MARGIN is sufficient to preclude inadvertent criticality and covers the requirements of steam line breaks with automatic initiation of ECCS and NRVs blocked. With $T_{avg} <$ 330°F, the reactivity transients resulting from a steam line break cooldown are minimal. 5% $\Delta k/k$ SHUTDOWN MARGIN (with all rods inserted) provides adequate protection to preclude criticality for all postulated accidents with the reactor vessel head in place.

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. Normally, when full power is reached after each refueling, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted steady-state curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction and the reactivity status of the core can be continuously evaluated, and any deviation would be thoroughly investigated and evaluated.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Conditions I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria.

3/4.2.1 PEAK LINEAR HEAT GENERATION RATE

Limiting the peak Linear Heat Generator Rate (LHGR) during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

When operating at constant power, all rods out, with equilibrium xenon, power peaking in the Yankee Rowe core decreases monotonically as a function of cycle burnup. This has been verified by both calculation and measurement on Yankee cores and is in accord with the expected behavior in a core that does not contain burnable poison. The all-rods-out power peaking measured prior to exceeding 75% of RATED THERMAL POWER after each fuel loading thus provides an upper bound on all-rods-out power peaking for the remainder of that cycle. Thereafter the measured power peaking shall be checked every 1,000 equivalent full power hours and the latest measured value shall be used in the computation. The only effects which can increase peaking beyond this value would be control rod insertion and xenon transients and these are accounted for in calculating peak LHGR.

The core is stable with respect to xenon, and any xenon transients which may be excited are rapidly damped.

The xenon multiplier specified in the CORE OPERATING LIMITS REPORT was selected to conservatively account for transients which can result from control rod motion at full power.

The multiplier is defined as the ratio of the maximum value of F_z due to xenon induced top peaked power redistribution and the F_z of the nominal operating axial shape. This is consistent with the methodology used to derive the LHGR limits, which were generated based on the worst top-peaked axial power distribution. The minimum value of the multiplier is unity.

BASES (Continued)

The limits on power level and control rod position following control rod insertion were selected to prevent exceeding the maximum allowable linear heat generation rate limits specified in the CORE OPERATING LIMITS REPORT within the first few hours following return to power after the insertion. With Yankee's highly damped core, the 24 hour hold allows sufficient time for the initial xenon maldistribution to accommodate itself to the new power distribution. The restriction on control rod location during these 24 hours assures that the return to allowable fraction of full power will not cause additional redistribution due to rod motion.

After 48 hours at zero power, the average xenon concentration has decayed to about 20% of the full power concentration. Since the xenon concentrations are so low, an increase in power directly to maximum allowable power creates transient peaking well below the value imposed by the xenon redistribution multiplier. Thus, any increase in power peaking due to this operation is below the value accounted for in the calculation of the LHGR.

These conclusions are based on plant tests and on calculations performed with the SIMULATE three dimensional nodal code used in the analysis of Core XI (reference cycle) described in Proposed Change No. 115, dated March 29, 1974.

The Factors d, e, and f in Specification 4.2.1.2 will be combined statistically as the "root-sum-square" of the individual parameters. This method for combining parameter uncertainties is valid due to the independence of the parameters involved. Factor d accounts for uncertainty in the power distribution measurement with the incore detection system. Factor e accounts for uncertainty in the calorimetric measurement for determining core power level. Factor f accounts for uncertainty in engineering and fabrication tolerances of the fuel. Together, these factors, when combined statistically, yield an uncertainty of 8.5% for less than 17 and greater than or equal to 12 operating incore thimbles, and 7.1% for greater than or equal to 17 operating thimbles. This factor and Factors a, b, c, and g will be combined multiplicatively to obtain peak LHGR values.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux and enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded, and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors is measurable but will normally only be determined periodically as specified in Specifications 4.2.2.1 and 4.2.3.1. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

6.9.4 CORE OPERATING LIMITS REPORT

6.9.4.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Control rod insertion limits for Specification 3.1.3.5.

b. Peak linear heat generation rate for Specifications 3.2.1 and 4.2.1.1.

c. The xenon redistribution multiplier for Specification 4.2.1.2.

d. The reduced power multiplier for Specification 4.2.1.2.

6.9.4.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC:

a. XN-75-41, Volumes I, II, III and Supplements 1 through 7, "WREM-Based Generic PWR-ECCS Evaluation Model," Exxon Nuclear Corporation, as amended/supplemented by:

1. YAEC-1071, "Yankee Rowe Core XI Decay Heat Redistribution Factor During Shutdown Conditions," June 1974.

2. Proposed Change to Technical Specifications No. 125.

3. Proposed Change to Technical Specifications No. 142.

4. XN-76-44, "Revised Nucleate Boiling Lockout for ENC-WREM-Based ECCS Evaluation Model," Exxon Nuclear Corporation September 1976.

5. YAEC-1125, "Method for Calculating End-of-Bypass Time for Yankee Rowe LOCA Analysis," March 1977.

6. YAEC-1131, "Method for Calculating Low Flow Film Boiling Coefficients for Yankee WREM-Based Generic PWR ECCS Evaluation Model," June 1977.

7. YAEC-1133, "Core Flood Rate Stabilization for Yankee WREM-Based Generic PWR ECCS Evaluation Model," July 1977.

8. Letter, "Yankee Rowe Core XIII LOCA Core Inlet Temperature and Accumulator Delay Sensitivity Analysis," and Errata, October 7 and October 11, 1977.

9. XN-76-27, "WREM-Based Generic PWR ECCS Evaluation Model Update ENC-WREM-II," Exxon Nuclear Corporation, July 1976.

10. YAEC-1231, Revision 1, "Application of a Lower Plenum Phase Separation Model to Yankee Rowe Large Break LOCA Analysis," March 1981.

11. Proposed Change to Technical Specification No. 178.

12. Letter, "LOCA Injection ΔP Penalty," dated August 16, 1985.

13. Proposed Change to Technical Specifications No. 188.

14. Letter, "LOCA Reflood Heat Transfer Models," dated January 5, 1988.

15. Letter, "YAEC Response to NRC Review of Revised Reflood Heat Transfer Model for YNPS LOCA Analysis," dated May 2, 1989.

b. Reactor physics methods as described in Proposed Change to Technical Specification No. 115, as amended/supplemented by:

1. Proposed Change to Technical Specification No. 125.
2. Proposed Change to Technical Specification No. 145.
3. Proposed Change to Technical Specification No. 163.
4. Proposed Change to Technical Specification No. 178.

c. Transient analysis methods as described in Proposed Change to Technical Specification No. 115, as amended/supplemented by:

1. YAEC-1361, "YNPS Main Steam Line Break Analysis," May 1983.

2. YAEC-1398, "YNPS Main Steam Line Break Analysis, Addition of Boron Transport Model," February 1984.

6.9.4.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.4.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.