Docket No.: 50-271

September 18, 1987

Mr. R. W. Capstick
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Capstick:

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications in response to your application dated June 24, 1987, as clarified by letter dated August 11, 1987.

The amendment changes the Technical Specifications to: (1) incorporate additional Maximum Average Planar Linear Heat Generation Limits (MAPLHGR) for the new fuel; (2) revise the Minimum Critical Power Ratio (MCPR) limits by eliminating the fuel type dependence; and (3) update the bases section references associated with certain cycle dependent limits.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/s/

Vernon L. Rooney, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 100 to License No. DPR-28
2. Safety Evaluation

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

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AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE DPR-28
VERMONT YANKEE NUCLEAR POWER STATION

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UNITED STATES
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WASHINGTON, D. C. 20555

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 100
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated June 24, 1987, 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;

   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-28 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications, contained in Appendix A, as revised through Amendment No. 100 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses

Victor Nerses, Acting Director
Project Directorate I-3
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 18, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A Technical Specifications by removing the page identified below and inserting the enclosed page. The revised page is identified by the captioned amendment number and contains marginal lines indicating the area of change.

REMOVE	INSERT
180-a	180-a
180-d	180-d
180-f	180-f
180-g	180-g
----	180-n6*
180-01	180-01

*Denotes new page

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to ensure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting values shown in Tables 3.11-1A through H. For single recirculation loop operation, the limiting values shall be the values from Tables 3.11-1B through E and Table 3.11-1G through H listed under the heading "Single Loop Operation." These values are obtained by multiplying the values for two loop operation by 0.83. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥25% rated thermal power.

Bases:

3.11 Fuel Rods

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

Refer to the appropriate section of the General Electric Company Licensing Topical Report, "United States Supplement, General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-8-US.

(Note: All exposure increments in this Technical Specification section are expressed in terms of megawatt-days per short ton.)

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1.

The MAPLHGR reduction factor of 0.83 for single recirculation loop operation is based on the assumption that the coastdown flow from the unbroken recirculation loop would not be available during a postulated large break in the active recirculation loop, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation", February 1983.

VYNPS

Table 1

SIGNIFICANT INPUT PARAMETERS TO THE
LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core Thermal Power	1664 MWt, which corresponds to 105% of rated steam flow
Vessel Steam Output	6.75×10^6 lbm/h, which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area for Large Breaks - Discharge	2.26 ft^2 (DBA)
- Suction	4.14 ft^2
Number of Drilled Bundles	220

Fuel Parameters:

	Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio*
A.	7D230	7 x 7	18.5	1.4	1.2
B.	8D219	8 x 8	13.4	1.4	1.2
C.	8D274L	8 x 8	13.4	1.4	1.2
D.	8D274H	8 x 8	13.4	1.4	1.2
E.	8D274 (High Gd)	8 x 8	13.4	1.4	1.2
F.	LTA	8 x 8	13.4	1.4	1.2
G.	8DPB289 & P8DPB289	8 x 8	13.4	1.4	1.2
H.	BP8DRB299	8 x 8	13.4	1.4	1.2

* To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

Bases:

3.11A Linear Heat Generation Rate (LHGR)

Refer to the appropriate section of the General Electric Company Licensing Topical Report, "United States Supplement, General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-8-US.

VYNPS

Table 3.11-1H

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee

Fuel Type: BP8DRB299

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft) Two Loop Operation	*Single Loop Operation	PCT (°F)	Oxidation Fraction
200.0	10.7	8.8	2030.	0.019
1,000.0	10.8	8.9	2037.	0.019
5,000.0	11.4	9.4	2093.	0.023
10,000.0	12.2	10.1	2178.	0.029
15,000.0	12.3	10.2	2198.	0.031
20,000.0	12.2	10.1	2193.	0.031
25,000.0	11.7	9.7	2139.	0.026
35,000.0	10.6	8.8	1972.	0.028
41,900.0	9.4	7.8	1800.	0.012

Source: NEDO-21697, August 1977 (revised)

* MAPLHGR for single loop operation is obtained by multiplying MAPLHGR for two loop operation by 0.83.

TABLE 3.11.2

VERMONT YANKEE NUCLEAR POWER STATION
TECHNICAL SPECIFICATION MCPR OPERATING LIMITS

Value of "N" in RBM Equation (1)	Average Control Rod Scram Time	Cycle Exposure Range	MCPR Operating Limits (2&3)
42%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.29
		EOC-2 GWD/T to EOC-1 GWD/T	1.29
		EOC-1 GWD/T to EOC	1.30
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.29
		EOC-2 GWD/T to EOC-1 GWD/T	1.31
		EOC-1 GWD/T to EOC	1.35
41%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.25
		EOC-1 GWD/T to EOC	1.30
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.31
		EOC-1 GWD/T to EOC	1.35
≤40%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.25
		EOC-1 GWD/T to EOC	1.30
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.31
		EOC-1 GWD/T to EOC	1.35

NOTES:

(1) The Rod Block Monitor (RBM) trip setpoints are determined by the equation shown in Table 3.2.5 of the Technical Specifications.

(2) The current analysis for the MCPR Operating Limits does not include the 7x7, 8x8, or 8x8R fuel types. On this basis, if any of these fuel types are to be reinserted, they will be evaluated in accordance with 10CFR50.59 to ensure that the above limits are bounding for these fuel types.

(3) MCPR Operating Limits are increased by 0.01 for single loop operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letters dated June 24 and August 11, 1987, the Vermont Yankee Nuclear Power Corporation, the licensee for the Vermont Yankee Nuclear Power Station, requested amendment of the Vermont Yankee Technical Specifications for the Cycle 13 operation (Ref. 1, 2). The reload includes 136 new assemblies of GE manufacture. The reload design has no unusual features and the proposed Technical Specification changes are related to the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), the Minimum Critical Power Ratio (MCPR) and the updating of the bases and references associated with certain cycle dependent limits. The new fuel is of increased enrichment designed for extended burnup.

2.0 EVALUATION

2.1 Reload Description

The licensee proposes to use the GE fuel type BP8DRB299, which is of slightly higher enrichment than the present fuel type P8DPB289. The cycle loading places the low reactivity (old) assemblies in the periphery. The new assembly type, described in Reference 3, has been reviewed and approved for use in operating BWRs.

2.2 Fuel Design

All new fuel to be inserted into Cycle 13 has been designed by GE. The new fuel pellet diameter is increased, and the pellet is fabricated to a higher nominal density, which then allows less in-reactor densification. The new fuel cladding incorporates a barrier in the internal surface designed to reduce the effects of pellet-cladding interaction. The new fuel's mechanical and chemical compatibility with the reactor environment has been addressed in Reference 3 and found acceptable.

The fuel thermal effects were calculated using the FROSSTEY code (Ref. 4 and 5). The code calculates the pellet-cladding gap conductance and fuel temperatures based on cladding thermal expansion, fission gas release, pellet swelling, pellet densification, pellet cracking and fuel and cladding thermal conductivity. The core average response of gap conductance was estimated as a function of exposure for the peak linear heat generation steady state conditions. These data were used as input for the transient analysis. However, the hot channel calculations were performed for the given fuel bundle type and assembly exposure and were also used in the core transient analysis. The values of the estimated gap conductance for Cycle 13 are higher than the corresponding values of previous cycles, because (a) the radial pellet-clad gap for the new fuel is lower due to increased pellet diameter, (b) the increased pellet density will decrease the reduction of the gap size and (c) the revised surface roughness for the pellet and the cladding improves conductance. The core has also been used to calculate the local linear heat generation rates for fuel centerline incipient melt and 1.0% cladding plastic strain as a function of the (local) exposure.

2.3 Nuclear Design

The first issue in the nuclear design of Cycle 13 is that of the methodology. The CASMO-1/SIMULATE codes have been changed to CASMO-2/SIMULATE due to input limitations of CASMO-1, which did not allow adequate modeling for the new fuel. In addition, the reflecting bundle boundary conditions have been revised in CASMO-2 to handle high flux gradients. The licensee performed a benchmarking exercise comparing CASMO-1/SIMULATE to CASMO-2/SIMULATE for the Vermont Yankee Cycle 9, 10 and 11 and part of Cycle 12 results. The k_{eff} for cold and hot conditions were compared. The differences of the average of the k_{eff} standard deviation and the comparison to the Traversing Incore Probe are almost identical. Given that the differences of the two codes are in format and an improvement in boundary conditions, we find the CASMO-2/SIMULATE system acceptable for the Cycle 13 nuclear design.

The Haling, All Rods Out (ARO) and a rodded core depletion were carried out using SIMULATE (Ref. 6). For the rodded depletion, control rod patterns were developed which produced cycle peaking similar to the Haling power distribution. Beginning of cycle, 1,000 MWD/MT and 2,000 MWD/MT and end of cycle (for full power operating conditions) exposure distributions were calculated to develop reactivity inputs for the core wide transient analyses. The minimum shutdown margin of 1.13%WK occurs at beginning of cycle with the strongest worth rod withdrawn, thus fully meeting the minimum .32%WK requirement. For the standby liquid control system, the cycle was searched to find the most reactive point. At that point, the 800 ppm of boron required by the Vermont Yankee technical specifications would make the core 6.6%WK subcritical, which is more than the required 5.0%WK. From the above, we conclude that the nuclear design was carried out using acceptable methods and that the results fall within the expected range; thus it is acceptable.

2.4 Thermal Hydraulic Design

The steady state thermal-hydraulics analysis was performed with the code FIBWR (Ref. 8 and 9), which calculates the core pressure drop and the total bypass flow for a given total core flow. The detailed core flow paths and power distribution are assumed known. The objective of these calculations is to assure that nucleate boiling is maintained for normal operation and during transients. The GESTAR II lowest allowable MCPR value is 1.07 (Ref. 3) which is derived from the GEXL correlation (Ref. 9). The GEXL correlation has been approved by the NRC. The Vermont Yankee Technical Specifications limit the reload cycle operation to 13.4 kw/ft; this is based on GESTAR II. The MCPR operating limiting value is determined by the limiting transient, which we shall examine in the following paragraph. The thermal-hydraulic design has been performed with approved methods and is also based on the GE fuel analyses in GESTAR II; therefore, it is acceptable.

2.5 Transient and Accident Analyses

2.5.1 Methodology

The BWR transient and accident analysis requires a two tier method. At first a system level simulation is performed to determine the overall plant response to the assumed transient. The characteristics and response of the plant instrumentation determine the characteristics of the transient. This level of simulation is performed with the code RETRAN, which has been approved generically for reloads (Ref. 10). (Note: The YAEC version was approved subject to the approval of the EPRI version listed in Ref. 11. Ref. 12 is the approval of the EPRI version of RETRAN). The second set of calculations is performed to determine the hot channel characteristics using the RETRAN output. This computation is carried out with the TCPYA01 (Ref. 13), which is an approved code. The purpose of all analyses is to determine that the MCPR limit will not be violated.

The initial conditions are chosen, such as to yield conservative transients. For example, 104.5% of power is assumed at a 100% flow. The scram setpoints are set at the technical specification limits and the logic system delays are assumed at the equipment specification limits. The safety and safety/relief valve capacities are based on Technical Specification values, and the set points are based on technical specification upper limits. The response is assumed to be the slowest specified value, and the control rod drive scram speed is based on the technical specification limits.

The reactivity functions, the axial power distributions and the kinetics parameters are generated from the base states established for beginning and end of cycle and two intermediate exposure state points. All state points are characterized by exposure, void history, control rod patterns and core thermal hydraulic conditions. The method, which is approved, is described in detail in Reference 14. Thus, we conclude that the methodology used for the accident and transient analysis of the Vermont Yankee Cycle 13 is based on approved methods and is acceptable.

2.5.2 Analyzed Transients

The following is a summary of the transients analyzed and the estimated results:

- Turbine Trip Without Bypass, Transient (TTWOBP)

  The transient is initiated by a rapid (0.1 sec) closure of the turbine stop valves. The steam bypass valve is assumed to remain closed, and the reactor protection system initiates rod insertion. The scram time is based on the technical specification limit. The transient results in a WCPR of 0.025 and a maximum reactor vessel pressure of 1,282 psia.

- Generator Load Rejection Without Bypass (GLRWOBP)

  This transient is assumed to be initiated by a rapid (.3 sec) closing of the turbine control valves. As in the TTWOBP case, the steam bypass is assumed to remain closed. In this case, the reactor protection system is initiated by the acceleration relay of the turbine control system and is assumed to occur at 0.28 sec. The maximum pressure and MCPR values are the same as in the TTWOBP case.

° Loss of Feedwater Heating, Transient (LOFWH)

This transient is assumed to initiate with the failure (tripped or bypassed) of a group of feedwater heaters which would lower the feedwater temperature by 100°F, thus increasing the reactivity and the reactor power. It is conservatively assumed that the 120% power scram will not be activated. In this transient the WMCPR is 0.18. This transient is not limiting.

° Overpressurization Analysis

To demonstrate compliance with the ASME code limits, the Main Steam Isolation Valves (MSIV) are assumed to close. A 3.0 sec closing time is assumed in accordance with the technical specifications and a reactor scram signal is initiated 0.28 sec after reaching the high flux trip. The results show that the maximum pressure is lower than the allowable limit of 10% above the vessel design pressure, i.e., 1.375 psig.

° Local Rod Withdrawal Error Transient (RWE)

This transient is assumed to occur due to an operator erroneously withdrawing a control rod in the continuous withdrawal mode. To bound the most severe of the postulated rod withdrawal errors, a portion of the MCPR operating limit envelope is specifically defined, such that the cladding limits are not violated. The assumptions for the RWE analysis include: 104.5% power and 100% flow levels; the rod being withdrawn has a high reactivity worth and has a power distribution that places the bundles around the rod near the operating limit. Many rod patterns are tested and the one with the highest WCPR is selected as the bounding case. The rod block monitor system terminates rod movement before the MCPR of 1.07 is reached. The analysis was performed using SIMULATE (Ref. 6). The limiting case is the case with no xenon present at the most reactive point in the cycle. The worst case with equilibrium xenon present is bounded, by a large margin, by the non-xenon case. We conclude that the RWE analysis was performed with approved methods and conservative assumptions, and results in a MCPR that is not less than 1.07; it is therefore acceptable.

- Misloaded Bundle Error Analysis

In this case, two possibilities are analyzed, i.e., the rotated bundle and the mislocated bundle.

Bundle rotation could increase the local pin peaking and the local reactivity if higher enrichment pins are placed adjacent to a wide water gap. The objective of the analysis is to insure that in the worst possible rotation, the LHGR and the MCPR limits are not violated. The CPR response is examined by using SIMULATE to develop the largest possible WCPR for the expected conditions of enrichment, exposure and void history. The same process is repeated for reactivity and the LHGR.

Bundle mislocation could result in a high reactivity assembly being placed into a high importance region. The analysis again is based on SIMULATE to examine the effects of replacing every older interior assembly (in a quarter core) with a new assembly. For every rod sequence and cycle condition, the MCPR for the properly loaded assembly was compared to misloaded locations.

Using the above procedure and approved methods, the rotated assembly was found to be limiting compared to the mislocated assembly. However, the resulting MCPR was 1.07, which is acceptable.

- Control Rod Drop Accident

This transient assumes the uncoupling and subsequent free fall of a rod from a rod bank. Control rod banks are designed and their withdrawal is programmed to minimize the worth of individual rods. However, should a rod be uncoupled, the incremental rod worth should not produce an enthalpy increase of more than 280 cal/gm (see paragraph 15.4.9 of the SRP). The rod worth minimizer prevents inadvertent rod withdrawal out of sequence, and is used to take the plant from an all-rods-in configuration to above 20% power. Above 20% power even multiple operator errors will not create the potential for an enthalpy increase over 280 cal/gm. However, below 20% power all the sequences are examined for their incremental worth using the xenon-free SIMULATE model.

The control rod drop accident was analyzed using an approved method and showed that no rod drop will cause a fuel rod enthalpy increase greater than 280 cal/gm. The analysis is therefore acceptable.

° Loss of Coolant Accident (LOCA) Analysis

Reanalysis of the LOCA at Vermont Yankee was necessary to assure that the MAPLHGR operating limits remained valid with the new fuel type introduced in Cycle 13. An approved evaluation method was used to establish the operating limits (References 15 and 16). The estimated MAPLHGR operating limits as a function of the average planar exposure show that the peak cladding temperature remains below 2,200°F and the maximum cladding oxidation does not exceed 17% of the cladding thickness, as required by 10 CFR 50.46. Therefore, we conclude that the proposed LOCA analysis and the estimated MAPLHGR operating limits are acceptable.

2.6 MCPR Operating Limits

The accident analysis described in the preceding paragraphs has established the new operation MCPR limits (and LCOs) for the proposed new fuel. Our review of the accident analysis showed that it was performed with approved methods and results were acceptable; hence, we find that the proposed MCPR limits are acceptable. The events analyzed are limiting; no other design basis transients would produce more restrictive MCPR operating limits for Cycle 13.

2.7 Technical Specification Changes

1) The changes in TS 3.11, Table 3.11-1H and the bases 3.11A are acceptable because the limits are derived from analyses performed using approved methods.

2) The MCPR operating limits in Table 3.11-2 Section 3.11c are acceptable as discussed in Sections 2.5 and 2.6, and

3) The bases, in Section 3.11A and Section 3.11B, conform with the most recent changes in GESTAR II. These are acceptable.

We have reviewed the proposed license amendment pertaining to Cycle 13 operation of the Vermont Yankee reactor. Based on this review, we conclude that the required information was submitted and that the fuel, the nuclear and thermal hydraulic designs and the transient and accident analyses are acceptable. The proposed Technical Specification changes are therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

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. The issuance of the amendment will not, therefore, be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Lois

Dated: September 18, 1987

5.0 REFERENCES

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8. EPRI-NP-1923, (Project 1754-1 Final Report) "FIBWR, A Steady-State Core Flow Distribution Code for Boiling Water Reactors, Code Verification and Qualification Report" by A. F. Ansari, et al, dated July 1981.

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15. NEDO-21697, "LOCA Analysis Report for the Vermont Yankee Nuclear Power Station," General Electric Company, dated August 1977.

16. Letter from USNRC to Vermont Yankee Nuclear Power Corporation, SER for NEDO-21697, dated November 30, 1987.