August 9, 1995

Mr. Donald A. Reid
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
Ferry Road
Brattleboro, VT 05301

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89201)

Dear Mr. Reid:

The Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated March 31, 1994, as supplemented by letters dated September 9, 1994, and June 22, 1995.

The amendment modifies the requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with the Boiling Water Reactor (BWR) Owners Group long-term solution Option I-D described in the Licensing Topical Report, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960 June 1991" and NEDO-31960, Supplement 1, dated March 1992. NEDO-31960 and NEDO-31960, Supplement 1, were accepted by the NRC staff in a letter to L.A. England (BWR Owners Group) dated July 12, 1993.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> Notice.

Sincerely,

Original signed by:

Daniel H. Dorman, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 146 to DPR-28
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Docket No. 50-271

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2. Safety Evaluation

cc w/encls: See next page

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cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated March 31, 1994, as supplemented by letters dated September 9, 1994, and June 22, 1995, 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 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 146 , 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee

Phillip F. McKee, 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: August 9, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 146

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

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
14	14
122	122
123	123
124	124
125	125
126	126
127	127
128	128
134	134
138	138
144	144
145	145
--	145a
267	267
270	270

BASES:

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses are performed to demonstrate that the APRM flux scram over the range of settings from a maximum of 120% to the minimum flow biased setpoint of 54% provide protection from the fuel safety limit for all abnormal operational transients including those that may result in a thermal hydraulic instability.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. If the scram requires a change due to an abnormal peaking condition, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1.A.1.a, thus assuring a reactor scram at lower than design overpower conditions. For single recirculation loop operation, the APRM flux scram trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR defined in the Core Operating Limits Report.

3.6 LIMITING CONDITIONS FOR OPERATION

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Single Loop Operation

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

4.6 SURVEILLANCE REQUIREMENTS

3. The surveillance requirements of 4.6.F.1 and 4.6.F.2 do not apply to the idle loop and associated jet pumps when in single loop operation.

4. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle. Baseline data for evaluating 4.6.F.2 while in single loop operation shall be updated as soon as practical after entering single loop operation.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION

b. The requirements for avoiding potentially unstable thermal hydraulic conditions defined in Technical Specification 3.6.J are met.

c. The idle loop is isolated by electrically disarming the breaker to the recirculation pump motor generator set drive motor prior to startup or, if disabled during reactor operation, within 24 hours, and until such time as the inactive recirculation loop is to be returned to service.

d. The recirculation system controls will be placed in the manual flow control mode.

4.6 SURVEILLANCE REQUIREMENTS

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

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3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

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3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

H. Recirculation System

1. Operation with one recirculation loop is permitted according to Specification 3.6.G.1.

2. With no Reactor Coolant System recirculation loops in operation, initiate measures such that the unit is in hot shutdown within the next 12 hours.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

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3.6 LIMITING CONDITIONS FOR OPERATION

I. Shock Suppressors (Snubbers)

1. Except as noted in 3.6.I.2 and 3.6.I.3 below, all required safety-related snubbers shall be operable whenever its supported system is required to be operable.

2. With one or more required snubbers inoperable, within 72 hours, replace or restore the snubber to operable status and perform an engineering evaluation per Specification 4.6.I.1b and c, on the supported component. In all cases, the required snubbers shall be made operable or replaced prior to reactor startup.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, the supported system shall be declared inoperable and the appropriate action statement for that system shall be followed.

4.6 SURVEILLANCE REQUIREMENTS

I. Shock Suppressors (Snubbers)

1. Each snubber shall be demonstrated operable by performance of the following inspection program.

   a. Visual Inspections

   Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers Per Inspection Period	Next Required Inspection Intervals
0	18 months ±25%
1	12 months ±25%
2	6 months ±25%
3, 4	124 days ±25%
5, 6, 7	62 days ±25%
8 or more	31 days ±25%

   The snubbers may be categorized into two groups: the accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

3.6 LIMITING CONDITIONS FOR OPERATION

J. Thermal Hydraulic Stability

1. When the reactor mode switch is in RUN:

   a. Under normal operating conditions the reactor shall not intentionally be operated within the power flow exclusion region defined in Core Operating Limits Report (COLR).

   b. If the reactor has entered the power flow exclusion region (COLR), the operator shall immediately insert control rods and/or increase recirculation flow to establish operation outside of the region.

4.6 SURVEILLANCE REQUIREMENTS

J. Thermal Hydraulic Stability

This page has been deleted.

BASES: 3.6 and 4.6 (Cont'd)

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow (except in the case of single loop operation when reverse flow is subtracted from the total jet pump flow). Thus, the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

G. Single Loop Operation

Continuous operation with one recirculation loop was justified in "Vermont Yankee Nuclear Power Station Single Loop Operation", NEDO-30060, February 1983, with the adjustments specified in Technical Specification 3.6.G.1.a.

During single loop operation, the idle recirculation loop is isolated by electrically disarming the recirculation pump motor generator set drive motor, until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Under single loop operation, the flow control is placed in the manual mode to avoid control oscillations which may occur in the recirculation flow control system under these conditions.

H. Recirculation System

Twelve hours is a reasonable period of time to reach hot shutdown conditions. Operation of the reactor may not occur without forced recirculation flow.

BASES: 3.6 and 4.6 (Cont'd)

I. Shock Suppressors (Snubbers)

All snubbers are required operable to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested once each operating cycle. Observed failures of these sample snubbers shall require functional testing of additional units.

BASES: 3.6 and 4.6 (Cont'd)

J. Thermal Hydraulic Stability

The reactor design criteria is such that thermal hydraulic oscillations are prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of an instability, a power/flow exclusion region to be avoided during normal operation is calculated using the approved methodology as stated in Specification 6.7.A.4. Since the exclusion region may change each fuel cycle, the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in this region and to immediately exit upon an entry. Entries into the exclusion region are not part of normal operation. An entry may occur as a result of an abnormal event, such as a single recirculation pump trip. In these events, operation in the exclusion region may be needed to prevent equipment damage, but actual time spent inside the exclusion region is minimized. Though each operator action can prevent the occurrence and protect the reactor from an instability, the APRM flow-biased scram function is designed to suppress global oscillations, the most likely mode of oscillation, prior to exceeding the fuel safety limit. While global oscillations are the most likely mode, protection from out-of-phase oscillations are provided through avoidance of the exclusion region and administrative controls on reactor conditions which are primary factors affecting reactor stability.

The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurement. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Statistical Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the fifteenth of each month following the calendar month covered by the report. These reports shall include a narrative summary of operating experience during the report period which describes the operation of the facility.

4. Core Operating Limits Report

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following: (a) The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a, (b) The K_f core flow adjustment factor for Specification 3.11.C., (c) The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a, (d) The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a, 2.1.B.1, and 3.11.B, and (e) The Power/Flow Exclusion Region for Specifications 3.6.J.1a and 3.6.J.1b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).

Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (Approved by NRC, SER, dated September 15, 1982).

Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (Approved by NRC SER, dated September 15, 1982).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1249P, April 1981 (Approved by NRC SER, dated September 27, 1985).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplemental Information to VYNPC April 19, 1990 Response Regarding FROSSTEY-2 Fuel Performance Code," BVY 90-054, dated May 10, 1990 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Responses to Request for Additional Information on FROSSTEY-2 Fuel Performance Code," BVY 91-024, dated March 6, 1991 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "LOCA-Related Responses to Open Issues on FROSSTEY-2 Fuel Performance Code," BVY 92-39, dated March 27, 1992 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNRC) to USNRC, "FROSSTEY-2 Fuel Performance Code - Vermont Yankee Response to Remaining Concerns," BVY 92-54, dated May 15, 1992 (Approved by NRC SER, dated September 24, 1992).

Report, "Loss-of-Coolant Accident Analysis for Vermont Yankee Nuclear Power Station," NEDO-21697, August 1977, as amended (Approved by NRC SER, dated November 30, 1977).

Report, "General Electric Standard Application for Reactor Fuel (GESTARII)," NEDE-24011-P-A, GE Company Proprietary (the latest NRC-approved version will be listed in the COLR).

Report, General Electric Nuclear Energy, "BWR Owner's Group Long-Term Solutions Licensing Methodology," NEDO-31960, June 1991 (Approved by NRC SER, dated July 12, 1993).

Report, General Electric Nuclear Energy, "BWR Owner's Group Long-Term Solutions Licensing Methodology," NEDO-31960, Supplement 1, March 1992 (Approved by NRC SER, dated July 12, 1993).

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. The COLR, 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.

B. Reportable Occurrences

This section deleted.

C. Unique Reporting Requirements

1. Annual Radioactive Effluent Release Report

a. Within 90 days after January 1 of each year, a report shall be submitted covering the radioactive content of effluents released to unrestricted areas during the previous calendar year of operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 146 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 letter dated March 21, 1994, as supplemented by letters dated September 9, 1994, and June 22, 1995, the Vermont Yankee Nuclear Power Corporation (the licensee) submitted a request for changes to the Vermont Yankee Nuclear Power Station Technical Specifications (TS). The requested changes include: (1) modify the requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with the BWR Owners Group (BWROG) long term solution Option I-D, and (2) add an exclusion region to the Core Operating Limits Report (COLR). The NRC staff reviewed the licensee's proposal to apply solution I-D to the Vermont Yankee Nuclear Power Station and approved such application in a safety evaluation dated March 30, 1995. The licensee has implemented solution I-D for Cycle 18 which began in May 1995. The stability limits are included in the Cycle 18 COLR. The September 9, 1994, and June 22, 1995, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The NRC staff was assisted in this review by its consultant, Oak Ridge National Laboratory (ORNL). The staff reviewed the licensee's submittals and adopted the findings recommended in ORNL's technical evaluation report (Attachment 1).

2.0 EVALUATION

The proposed changes include modifying the requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with BWROG long term solution Option I-D, and adding an exclusion region and its approved supporting methodologies to the COLR. The following specifications are proposed changes.

(1) Bases 2.1.A.1.a - APRM Flux Scram Trip Setting (Run Mode)

The current Basis for the 120% flux scram setpoint is changed from 120% to the range from 120% to 54%. Analysis of the flow biased portion of APRM flux scram over its range from 120% to 54% performed by the licensee indicates protection is provided from all abnormal operational occurrences (AOOs) including those that may result in an instability. Therefore, the staff has no objection to this change.

(2) TS 3.6.G.1.b - Single Loop Operation

This change removes limiting condition for operation (LCO) requirements which allow limited single loop operation within the stability exclusion region. New analysis indicates that such operation is not allowed. A statement is added to this TS requiring avoidance of "potentially unstable thermal hydraulic conditions" as defined in TS 3.6.J. TS 3.6.J requires immediate action, upon entering the stability exclusion region, to establish operation outside of the region. It also states that the power flow exclusion region is defined in the COLR. TS 6.7.A.4 is revised to incorporate the NRC-approved methodology for defining the region in the COLR. Therefore, the staff has concluded that the revised TS incorporates appropriate controls to prevent single loop operation within the exclusion region and is thus acceptable.

(3) Surveillance 4.6.F.3

This change removes the requirement to obtain baseline neutron flux noise data. This information is required for operating within defined exclusion regions. It is not necessary since operation in the exclusion region is not allowed. Thus, the change is acceptable.

(4) TS 3.6.H - Recirculation System

This LCO change eliminates the requirement to monitor APRM and LPRM neutron noise flux level in the exclusion region. It is acceptable since operation in the exclusion region is not allowed.

(5) TS 3.6.J - Thermal Hydraulic Stability

The entire section is revised to include the requirements for operation as follows: (a) normal plant operation is not allowed in the analytically defined exclusion region, and (b) immediate exit is required for any inadvertent region entry. This revision is acceptable since it reflects the solution Option I-D implementation criteria.

(6) Figure 3.6.4 - Stability Exclusion Regions

This figure is relocated to the COLR. The NRC-approved methodology for determining this figure is incorporated in TS 6.7.A.4. The staff has determined that this figure may be modified by the licensee, without affecting nuclear safety, provided that these changes are determined using the NRC-approved methodology incorporated in TS 6.7.A.4 and consistent with all applicable limits of the plant safety analysis that are addressed in the Updated Final Safety Analysis Report. Because plant operation will continue to be limited in accordance with NRC-approved methodologies and consistent with 10 CFR 50.36, this change is acceptable.

(7) Bases 3.6.G and 4.6.G - Single Loop Operation

The Basis for single loop operation is modified to delete reference to thermal hydraulic stability. Analysis supporting the exclusion region boundary and flow biased neutron flux scram for stability are bounding for all modes of operation (Ref. 2). Therefore, explicit reference to the thermal hydraulic stability exclusion region for the single loop mode is not necessary. The staff has no objection to this change.

(8) Bases 3.6.H and 4.6.H - Recirculation System

The Basis for the recirculation system is modified to remove reference to thermal hydraulic stability. The staff has no objection to this change since Bases of stability operating restrictions are identified in Bases 3.6.J.

(9) Bases 3.6.J and 4.6.H - Thermal Hydraulic Stability

The Bases for thermal hydraulic stability are revised to reflect the current approach for avoiding and protecting the fuel from thermal hydraulic instabilities. Therefore, the staff has no objection to this change.

(10) TS 6.7.A.4 - Core Operating Limits Report

The power/flow exclusion region for TS 3.6.J.1a and 3.6.J.1b is proposed as an additional cycle-specific parameter to be removed from the TS to the COLR. Also, the approved Topical Reports (NEDO-31960 and NEDO-31960, Supplement 1), which are the methodologies used to support this proposed TS change, are proposed to be added to the TS. The staff has reviewed the proposed changes and found them acceptable.

The proposed TS changes discussed above include interim implementation of long term solution Option I-D, relocation of the power/flow exclusion region to the COLR, and addition of the approved topical report relating to the Option I-D methodologies. Based on the staff's review in conjunction with ORNL's evaluation (Attachment 1), the staff concluded that the power/flow exclusion region may be modified by the licensee, without affecting nuclear safety, provided that such changes are determined using the specified, NRC-approved methodologies and consistent with all applicable limits of the plant safety analysis that are addressed in the FSAR. Because plant operation will continue to be limited in accordance with NRC-approved methodologies and consistent with 10 CFR 50.36, the staff therefore finds the proposed changes acceptable. Complete implementation of solution Option I-D will require additional plant procedures to cover power distribution controls and exclusion region confirmation analyses for new fuel (Ref. 2).

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The 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 and changes surveillance requirements. The amendment also changes recordkeeping or reporting requirements. The NRC 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 (60 FR 507). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from D. A. Reid (VYNPC) to USNRC, "Proposed Change No. 173, BWR Thermal Hydraulic Stability and Plant-Information Requirements for BWROG Option I-D Long Term Stability Solution", BVY 94-36, March 31, 1994.

2. Letter from D. A. Reid (VYNPC) to USNRC, "Submittal of Vermont Yankee Nuclear Power Station Application of BWROG Thermal Hydraulic Stability Long Term Solution Option I-D ", BVY 93-72, July 7, 1993, transmitting Licensing Topical Report, Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) to the Vermont Yankee Nuclear Power Plant, GENE-637-018-0793, General Electric Nuclear Energy, July 1993.

Principal Contributor: T. Huang

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Attachment: ORNL Technical Evaluation Report

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ORNL/NRC/LTR-95/08

Contract Program:	**Technical Support for the Reactor Systems Branch (L1697/P2)**
Subject of Document:	**Review of Proposed Technical Specification Changes for Interim Implementation of Solution I-D in Vermont Yankee**
Type of Document:	**Technical Evaluation Report**
Author:	**José March-Leuba**
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NRC Monitor:	**T. L. Huang, Office of Nuclear Reactor Regulation**

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Attachment 1

SUMMARY

This technical evaluation report addresses the Technical Specification changes proposed by Vermont Yankee (VY) to implement a Long Term Stability Solution of I-D type, which were submitted to NRC in Ref 1 ("VY proposed change #173", dated March 31, 1994). The main conclusion from this review is that the proposed changes are an adequate interim implementation of Solution I-D. These changes are not, however, a full implementation of Solution I-D because they do not address power distribution controls or reload confirmation procedures. It is our understanding that the licensee has plans to address the remaining implementation issues by the startup of Cycle 19, which is expected in January 1996.

INTRODUCTION

Our initial review of the proposed changes[1] indicated that the I-D implementation was not complete because it did not include provisions for reload confirmation analyses or power distribution controls. The results of our preliminary review were discussed with the licensee in a meeting held on August 10, 1994. The conclusion reached in the August 10 meeting was that NRC would issue a "generic" Solution I-D SER, which would recognize that Solution I-D is an acceptable Long Term Solution. The generic SER would also specify the minimum requirements that a Solution I-D implementation must satisfy in the area of power distribution controls and reload confirmation procedures. A TER addressing the technical issues that will be covered in the generic SER was issued in September 1994.[2]

During a February 8, 1995, meeting at NRC headquarters, the licensee described their proposed power distribution controls and reload confirmation analyses. It appears that Vermont Yankee will complete the implementation of Solution I-D when the SOLOMON Monitor/Predictor software (a GE product based in the ODYSY code) is installed to guarantee power distribution controls. The protection provided by the flow-biased scram for core-wide oscillations will be confirmed when the new BWROG delta-CPR correlation becomes available in mid 1995. The licensee also indicated that they plan to use the LAPUR code to calculate exclusion regions for reload confirmation analyses. During this meeting, the licensee stated that they plan to have a full Solution I-D implementation ready by the startup of Cycle 19, which is expected in January 1996.

PROPOSED TECHNICAL SPECIFICATION CHANGES

There are three major modifications proposed to the Vermont Yankee Technical Specifications:[1]

(1) An exclusion region will be defined in the Core Operating Limits Report (COLR)

(2) The reactor cannot be operated intentionally within the power-flow exclusion region. If the reactor has entered the exclusion region, the operator is instructed to exit the region immediately by either: (a) inserting control rods, or (b) increasing recirculation flow.

(3) Eliminate the current requirements to acquire baseline neutron noise data, and to monitor neutron noise levels while operating in the exclusion region.

Other proposed modifications include:

(1) Revise the basis for the APRM flux scram setting. Previously, this section only addressed the 120% high flux scram; he revised procedures state that the plant is now taking credit for the flow-biased scram.

(2) Single loop operation is allowed but only outside the exclusion region.

(3) Delete the figure showing the stability exclusion regions in the current specifications; the new region has been moved to the COLR.

(4) The bases for thermal hydraulic stability is revised to reflect the current Solution I-D approach. Reference to thermal hydraulic stability is removed from the bases of single loop operation and the recirculation system.

SOLUTION I-D IMPLEMENTATION

The technical bases for Solution I-D are discussed in detail in references 3 and 4 and were reviewed in ref. 2. A solution I-D implementation must satisfy the following criteria:[2]

(1) A exclusion region is defined conservatively and intentional operation is not allowed inside the region. Instabilities are only likely if the reactor is operated inside the exclusion region unintentionally, which reduces significantly the probability of occurrence.

(2) In case an instability occurs in a Solution I-D plant, it is likely to be an core-wide instability because Solution I-D plants have: (a) tight inlet orifices, and (b) small cores. Both of these characteristics make out-of-phase instabilities unlikely when reasonable power distribution controls are in place.

(3) The flow-biased scram provides automatic protection against core-wide instabilities, which is the most likely oscillation mode, and little or no protection for the out-of-phase mode, which is highly unlikely. Thus, the probability that unstable power oscillations will result in fuel design limits violations is low in Solution I-D plants.

The proposed changes to Technical Specifications represent an adequate interim implementation of Solution I-D from the technical point of view because they satisfy criteria numbers (1) and (3) described above. Solution I-D will be completely implemented in Vermont Yankee when criteria number (2) is satisfied by implementing power distribution controls and reload confirmation procedures.

The proposed changes satisfy criterion number (1) because they provide a requirement that a cycle-specific exclusion region be defined in the COLR and also provide administrative controls to avoid the region. Note that the requirement is that the exclusion region must be reviewed and confirmed in a cycle-to-cycle bases, not that it must be changed every cycle.

Criterion number (3) is satisfied by the calculations provided in ref. 5. Those calculations[5] show that the flow-biased scram will provide automatic protection before fuel limits are violated if large amplitude power oscillations develop because of a core-wide mode instability. We must note that the calculations shown in reference 3 are based on an interim delta-CPR correlation based on preliminary analyses of the impact on CPR of power oscillations that were performed by GE. The final delta-CPR correlation is expected in mid 1995, but the conclusions from ref. 5 are not expected to be qualitatively different when the more accurate correlation is used because Solution I-D is only required to show protection for core-wide oscillations.

Criterion number (2) is partially satisfied by Vermont Yankee because it is a small-core plant with tight (i.e., higher friction) bundle inlet orifices. These characteristics make out-of-phase instabilities unlikely as long as reasonable power distribution controls are in place. The criterion is only partially satisfied by the proposed Technical Specification changes because power distribution controls are not addressed. It is our understanding that these controls will be addressed by plant procedures rather than by a Technical Specification change.

CONCLUSIONS

Based on our present review of the Technical Specification changes proposed in reference 1, we conclude that this changes represent an adequate interim implementation of Long Term Stability Solution I-D. Complete implementation of Solution I-D will require additional plant procedures to cover power distribution controls and exclusion region confirmation analyses for new fuel cycles.

REFERENCES

1. Letter, Vermont Yankee Nuclear Power Corporation to U.S. NRC, *Proposed Change No. 173, BWR Thermal Hydraulic Stability and Plant-Information Requirements for BWROG Option 1-D Long Term Stability Solution.* BVY 94-36, March 31, 1994.

2. ORNL/NRC/LTR-93/23, *Review of Technical Issues Related to Long Term Solution I-D "Regional Exclusion with Flow Biased Scram"* September 1994.

3. General Electric Company, *BWR Owners' Group Long-Term Stability Solutions Licensing Methodology*, NEDO-31960, May 1991.

4. General Electric Company, *BWR Owners' Group Long-Term Stability Solutions Licensing Methodology*, NEDO-31960 Supplement 1, March 1992.

5. General Electric Company, *Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) to the Vermont Yankee Nuclear Power Plant*, GENE-637-018-0793, DRF A00-04021, July 1993.