April 15, 1998

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 160 TO FACILITY OPERATING LICENSE NO. DPR-28, VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. M98087)

The Commission has issued the enclosed Amendment No. 160 to Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated September 11, 1996, as supplemented by letter dated December 8, 1997. The information provided on December 8, 1997, did not change the original proposed no significant hazards determination.

The amendment involves a change to the safety and relief valve setpoint tolerance and power operation with an inoperable safety relief valve.

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

Sincerely,

Richard P. Croteau, 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. 160 to License No. DPR-28
2. Safety Evaluation

cc w/encls: See next page

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cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. David R. Lewis
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037-1128

Mr. Richard P. Sedano, Commissioner
Vermont Department of Public Service
120 State Street, 3rd Floor
Montpelier, VT 05602

Public Service Board
State of Vermont
120 State Street
Montpelier, VT 05602

Chairman, Board of Selectmen
Town of Vernon
P.O. Box 116
Vernon, VT 05354-0116

Mr. Richard E. McCullough
Operating Experience Coordinator
Vermont Yankee Nuclear Power Station
P.O. Box 157
Governor Hunt Road
Vernon, VT 05354

G. Dana Bisbee, Esq.
Deputy Attorney General
33 Capitol Street
Concord, NH 03301-6937

Chief, Safety Unit
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Deborah B. Katz
Box 83
Shellburne Falls, MA 01370

Mr. Raymond N. McCandless
Vermont Division of Occupational
and Radiological Health
Administration Building
Montpelier, VT 05602

Mr. Gautam Sen
Licensing Manager
Vermont Yankee Nuclear Power
Corporation
185 Old Ferry Road
Brattleboro, VT 05301

Resident Inspector
Vermont Yankee Nuclear Power Station
U. S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, VT 05354

Mr. Peter LaPorte, Director
ATTN: James Muckerheide
Massachusetts Emergency Management
Agency
400 Worcester Rd.
P.O. Box 1496
Framingham, MA 01701-0317

Jonathan M. Block, Esq.
Main Street
P. O. Box 566
Putney, VT 05346-0566

DATED: April 15, 1998

AMENDMENT NO. 160 TO FACILITY OPERATING LICENSE NO. DPR-28 - VERMONT YANKEE NUCLEAR POWER STATION

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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. 160
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 September 11, 1996, as supplemented by letter dated December 8, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

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:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 160 , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O Thomas, 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: April 15, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 160

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

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

Remove	Insert
18	18
120	120
142	142

1.2 SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings for controlling reactor system pressure.

Objective:

To provide for protective action in the event that the principal process variable approaches a safety limit.

Specification:

A. Reactor coolant high pressure scram shall be less than or equal to 1055 psig.

B. Primary system relief and safety valve settings shall be as specified in Table 2.2.1.

TABLE 2.2.1

Primary System Relief and Safety Valve Settings

Number and Type of Valve(s)	Lift Setting[1]
1 safety relief valve	1080 psig
2 safety relief valves	1090 psig
1 safety relief valve	1100 psig
2 safety valves	1240 psig

Note:

(1) As-left setpoint tolerance ±1%. As-found setpoint tolerance ±3%.

3.6 LIMITING CONDITIONS FOR OPERATION

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves and at least three of the four relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

4.6 SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter.

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage; the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equivalent drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Safety analyses have shown that only three of the four relief valves are required to provide the recommended pressure margin of 25 psi below the safety valve actuation settings as well as compliance with the MCPR safety limit for the limiting anticipated overpressure transient. For the purposes of this limiting condition, a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

The setpoint tolerance value for as-left or refurbished valves is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of set pressure. However, the code allows a larger tolerance value for the as-found condition if the supporting design analyses demonstrate that the applicable acceptance criteria are met. Safety analysis has been performed which shows that with all safety and safety relief valves within ±3% of the specified set pressures in Table 2.2.1 and with one inoperable safety relief valve, the reactor coolant pressure safety limit of 1375 psig and the MCPR safety limit are not exceeded during the limiting overpressure transient.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.160 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 September 11, 1996, as supplemented by letter dated December 8, 1997, the Vermont Yankee Atomic Power Corporation (VY or the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station Technical Specifications (TSs). The proposed amendment would revise the TS sections 2.2.B and 3.6.D.1. The changes would allow the licensee to increase the allowable safety/relief valve (SRV) and safety valve (SV) as-found setpoint tolerance from ±1% to ±3% and would permit operation up to 100% of rated thermal power (RTP) with a single inoperable SRV. The staff provided a request for additional information (RAI) to the licensee regarding the proposed changes. The licensee provided a response to the RAI by letter dated December 8, 1997, which did not change the original proposed no significant hazards determination and did not expand the scope of the original Federal Register notice.

The staff has previously granted approval to individual BWRs to increase the as-found SRV tolerance to three percent. The basis for the approval was a staff safety evaluation report (SER) for a licensing topical report (LTR) evaluating the setpoint tolerance increase (Reference 2). The staff SER included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance. Although VY has not referenced the staff SER, the staff will consider the six conditions provided in the SER as necessary conditions for acceptance of the TS modifications.

2.0 EVALUATION

The safety objective of the SRVs is to prevent overpressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at VY includes four SRVs and two SVs, arranged into four setpoint groupings of one SRV set at 1080 psig, two SRVs at 1090 psig, one SRV at 1100 psig and both SVs at 1240 psig. Existing TS provide a ±1% as-found / ±1% as-left setpoint tolerance. The proposed modifications would provide a ±3% as-found / ±1% as-left setpoint tolerance. The licensee's submittal was evaluated against the conditions provided in the setpoint relaxation SER and additional areas including impact on peak clad temperature (PCT) and surveillance intervals for SRV testing. The staff concluded that the conditions were met by the licensee.

Transient Analysis / Reload Methodology

The licensee considered the impact of the tolerance increase on abnormal operational transients (AOTs). For VY, analysis of AOTs has been conducted utilizing the 3% tolerance and one SRV out of service assumptions. The transients which generate the limiting drop in critical power ratio are the turbine trip without bypass (TTWOBP), generator load reject without bypass (GLRWOBP), and the loss of feedwater heater (LOFWH). The licensee has stated that the event resulting in the greatest change in critical power ratio (limiting ΔCPR) is GLRWOBP. A hot channel analysis was performed with the limiting core conditions (EOFPL, limiting scram times). The results show that the combined effects of a 3% setpoint tolerance increase and an inoperable SRV cause the ΔCPR to increase by 0.02. This is caused by positive moderator feedback from higher pressure in the top part of the core as control rods are inserted. The licensee has stated that continuing compliance with fuel integrity limits is obtained when the revised ΔCPR changes are incorporated into operating MCPR limits identified in the Core Operating Limits Report (COLR). This is acceptable to the staff.

Design Basis Pressurization Event

The licensee has re-evaluated the limiting design basis pressurization transient using the 3% tolerance limit to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) pressure vessel code upset limit. The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10% over design pressure (110% x 1250 psig = 1375 psig). The limiting pressurization AOT analyzed is a Main Steam Isolation Valve (MSIV) closure event occurring at end of full power life without credit for reactor trip on MSIV position sensing. The licensee analyzed the MSIV closure event with the 3% tolerance and one inoperable SRV, and calculated the maximum vessel pressure to be 1316 psig. This is within the 1375 psig ASME limit, and is acceptable to the staff.

TS Operability Statement for SRVs and SVs

The licensee has stated that all plant specific transient analyses have been conducted considering the increased SRV and SV tolerance and assuming one inoperable SRV. This is consistent with TS requirements and is acceptable to the staff.

Re-evaluation of High Pressure Systems Performance

The licensee re-evaluated performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3% tolerance limit. VY has three systems which are required to inject to the vessel at high pressure conditions: High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC) and Standby Liquid Control (SLC). The licensee has stated that for events where HPCI and RCIC are credited, at least two SRVs will operate with the as-found tolerance and allow HPCI or RCIC to supply flow to the reactor vessel. The SLC system is expected to operate for Anticipated Transient Without Scram (ATWS) conditions. For a 3% higher pressure during ATWS conditions, the SLC would deliver flow at the TSs required flow rate. The staff has evaluated the analysis and found it acceptable.

Evaluation of Motor-Operated Valves

In support of the SRV and SV tolerance increase from +/-1% to +/-3%, the licensee reviewed the motor-operated valves (MOVs) whose differential pressure could be affected and are required to operate during plant transients or accidents. The licensee determined that all such MOVs can operate within their performance capability. This is acceptable to the staff.

Alternate Operating Modes

The licensee also evaluated the increased tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) The analyses for the proposed changes included evaluations for the currently approved operating domains. Vermont Yankee has not extended its operating domain as defined by the power flow map in the Core Operating Limits Report (COLR), and has performed the setpoint analysis with bounding power/flow setpoints. This is acceptable to the staff.

Containment Response / Hydrodynamic Loads

The licensee also evaluated the effect of the increased tolerance limit on (1) the containment hydrodynamic loads during loss of coolant accidents, and (2) the hydrodynamic loads on the SRV discharge lines and the suppression chamber.

The licensee examined the potential effects of the proposed amendment on the containment and containment heat removal system. The containment design basis accident is a double-ended break at the suction of a recirculation pump. For this event, the reactor coolant system depressurizes very rapidly and thus, the SRVs are not challenged. Also, the RCS inventory and primary system heat sources that would contribute to the containment mass and energy are not increased. The setpoint tolerance thus has no effect on the capability of the containment to perform its design basis safety function (i.e., the containment peak temperature and pressure loads would not be adversely affected). The staff notes that small break LOCAs also would not lead to increased RCS pressure and subsequent SRV/SV challenges.

An increase in SRV setpoint tolerance involves a potential increase in SRV discharge dynamic and hydrodynamic loads on the SRV discharge piping and the torus. The licensee analyzed the loads and compared the increases to the margins determined in the Mark I Long Term Program. The results demonstrated that the increased torus loads are acceptable for all SRVs and SVs. Similarly, the increases in the loads on the SV piping and main steam lines due to the increased SV setpoint tolerance were evaluated and found to be acceptable. This is acceptable to the staff.

ECCS-LOCA

The licensee addressed the potential safety impact of increasing the SRV opening pressure on emergency core cooling systems (ECCS) performance, including three postulated pipe break scenarios: large break LOCA (LBLOCA), small break LOCA (SBLOCA) and main steamline break (MSLB) outside containment. For the LBLOCA, the licensee states that operation at full power with an inoperable SRV was not addressed since for all LOCA cases only one SRV is challenged; an inoperable SRV would not change the results of LOCA analyses. This is acceptable to the staff. The licensee addressed the impact of the increased setpoint tolerance on LOCA analysis for the SBLOCA and MSLB. The SBLOCA analysis examined break sizes from 0.4 ft^2 to 0.05 ft^2. The resulting impact on PCT varied from -28° F to +32° F, and all cases were below 2200° F and all 10 CFR 50.46 criteria were met. The licensee has also stated that the MSLB is bounded by the LBLOCA results. This is acceptable to the staff.

SRV/SV Surveillance Test Frequency

The licensee stated that all 4 SRVs and both SVs will be tested during each outage similar to past practice except that the new tolerance of +/-3% will be used for the as-found acceptance criteria. The plant TS Sections 4.6.D.1 and 4.6.E require a minimum of 50% of the SRVs and SVs to be tested which also meets the testing requirements of Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. Therefore, the licensee's surveillance test frequency meets both TS and ASME Code requirements.

2.1 Summary

The proposed amendment will allow the licensee to increase the allowable SRV and SV setpoint tolerance from ±1% to ±3% and will allow operation up to 100% of rated thermal power with a single inoperable SRV. In support of the modifications, the licensee has submitted an analysis of the limiting pressurization transient, analysis of anticipated operational transients for impact on ΔCPR and review of LOCA analysis, considering an inoperable SRV and the revised SRV and SV tolerance. The licensee will continue to ensure that acceptance criteria for the limiting pressurization transient, AOTs and design-basis accidents will be observed and remain acceptable. The staff has reviewed the licensee's analyses and concluded that the proper analyses were performed, the results were acceptable, and the changes are therefore, acceptable.

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 NRC staff has determined that the amendment involves no significant increase in 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 (62 FR 17241). 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.

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.

Principal Contributor: G. Golub

Date: April 15, 1998

REFERENCES

1. Letter from J. K. Thayer to U.S. NRC, "Proposed Change No. 185 - Safety and Relief Valve Setpoint Tolerance and Power Operation With an Inoperable SRV", dated September 11, 1996.

2. Letter from A. C. Thadani (NRC) to C. L. Tully (BWROG), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Licensing Topical Report,'" dated March 8, 1993.

3. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990.

4. NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel, GESTAR II," and NEDE-24011-P-A-11-US, "GESTAR II U. S. Supplement."