RAS M-169

DOCKETED
USNRC

August 12, 2008 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

March 2, 2006

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: EXTENDED POWER UPRATE (TAC NO. MC0761)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 229 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS), in response to your application dated September 10, 2003, as supplemented by letters dated October 1, and October 28 (2 letters), 2003; January 31 (2 letters), March 4, May 19, July 2, July 27, July 30, August 12, August 25, September 14, September 15, September 23, September 30 (2 letters), October 5, October 7 (2 letters), December 8, and December 9, 2004; February 24, March 10, March 24, March 31, April 5, April 22, June 2, August 1, August 4, September 10, September 14, September 18, September 28, October 17, October 21 (2 letters), October 26, October 29, November 2, November 22, and December 2, 2005; January 10, and February 22, 2006.

The amendment increases the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWt) to 1912 MWt, which is an increase of approximately 20 percent. The increase in power level is considered an extended power uprate (EPU). The amendment includes revisions to the VYNPS Operating License and Technical Specifications that are necessary to implement the EPU.

The related Safety Evaluation (SE) has been determined to contain proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly-available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed.

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee
Docket No. 50-271-LR Official Exhibit No. 14
OFFERED by: Applicant/Licensee Intervenor ______
NRC Staff Other ______
IDENTIFIED on 7/23/08 Witness/Panel Scarbrough/Hsu/Rowley
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk MAC

Template Secy-027

DS-03

A copy of the "Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration," which is being forwarded to the Office of the Federal Register for publication, is also enclosed.

Sincerely,

/RA/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 229 to License No. DPR-28
2. Non-proprietary SE
3. Proprietary SE
4. Notice

cc w/encls 1, 2, and 4: See next page

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Sincerely,
/RA/
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Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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cc w/encls 1, 2, and 4: See next page

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Accession Nos.:
Package: ML060050024
Cover letter, Amendment, and Notice: ML060050022
License and TS pages: ML060390107
Non-proprietary SE: ML060050028
Proprietary SE: ML060050051

*Cover letter only

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DATE	3/1/06	3/1/06	3/2/06	

The following provided safety evaluation input by memos dated**

OFFICE	EEIB-A/SC	EEIB-B/SC	EMCB-A/SC	EMCB-B/SC	EMCB-C/SC
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DATE**	9/27/04	7/1/04, 12/15/05, 9/29/04	11/24/04	8/9/04	9/9/04
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DATE**	7/31/04	10/3/05	10/4/04, 10/6/04, 9/30/05	9/30/05, 10/11/05	

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Vermont Yankee Nuclear Power Station

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ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 229
License No. DPR-28

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

   A. The application for amendment filed by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensee) on September 10, 2003, as supplemented by letters dated October 1, and October 28 (2 letters), 2003; January 31 (2 letters), March 4, May 19, July 2, July 27, July 30, August 12, August 25, September 14, September 15, September 23, September 30 (2 letters), October 5, October 7 (2 letters), December 8, and December 9, 2004; February 24, March 10, March 24, March 31, April 5, April 22, June 2, August 1, August 4, September 10, September 14, September 18, September 28, October 17, October 21 (2 letters), October 26, October 29, November 2, November 22, and December 2, 2005; January 10, and February 22, 2006, 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:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 229, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended to revise paragraph 3.A of Facility Operating License No. DPR-28 to reflect the new maximum licensed reactor core power level of 1912 megawatts thermal. The licensee is also amended to add new license conditions 3.K, 3.L, and 3.M as follows:

K. Minimum Critical Power Ratio

When operating at thermal power greater than 1593 megawatts thermal, the safety limit minimum critical power ratio (SLMCPR) shall be established by adding 0.02 to the cycle-specific SLMCPR value calculated using the NRC-approved methodologies documented in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," as amended, and documented in the Core Operating Limits Report.

L. Transient Testing

1. During the extended power uprate (EPU) power ascension test program and prior to exceeding 168 hours of plant operation at the nominal full EPU reactor power level, with feedwater and condensate flow rates stabilized at approximately the EPU full power level, Entergy Nuclear Operations, Inc. shall confirm through performance of transient testing that the loss of one condensate pump will not result in a complete loss of reactor feedwater.

2. Within 30 days at nominal full-power operation following successful performance of the test in (1) above, through performance of additional transient testing and/or analysis of the results of the testing conducted in (1) above, confirm that the loss of one reactor feedwater pump will not result in a reactor trip.

M. Potential Adverse Flow Effects

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer).

1. The following requirements are placed on operation of the facility above the original licensed thermal power (OLTP) level of 1593 megawatts thermal (MWt):

a. Entergy Nuclear Operations, Inc. shall monitor hourly the 32 main steam line (MSL) strain gages during power ascension above 1593 MWt for increasing pressure fluctuations in the steam lines.

b. Entergy Nuclear Operations, Inc. shall hold the facility for 24 hours at 105%, 110%, and 115% of OLTP to collect data from the 32 MSL strain gages required by Condition M.1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall provide the evaluation to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission.

c. If any frequency peak from the MSL strain gage data exceeds the limit curve established by Entergy Nuclear Operations, Inc. and submitted to the NRC staff prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall return the facility to a power level at which the limit curve is not exceeded. Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

d. In addition to evaluating the MSL strain gage data, Entergy Nuclear Operations, Inc. shall monitor reactor pressure vessel water level instrumentation or MSL piping accelerometers on an hourly basis during power ascension above OLTP. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data, Entergy Nuclear Operations, Inc. shall stop power ascension, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

e. Following start-up testing, Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis and provide that resolution to the NRC staff by facsimile or electronic transmission to the NRC project manager. If the uncertainties are not resolved within 90 days of issuance of the license amendment authorizing operation at 1912 MWt, Entergy Nuclear Operations, Inc. shall return the facility to OLTP.

2. As described in Entergy Nuclear Operations, Inc. letter BVY 05-084 dated September 14, 2005, Entergy Nuclear Operations, Inc. shall implement the following actions:

   a. Prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall install 32 additional strain gages on the main steam piping and shall enhance the data acquisition system in order to reduce the measurement uncertainty associated with the acoustic circuit model (ACM).

b. In the event that acoustic signals are identified that challenge the limit curve during power ascension above OLTP, Entergy Nuclear Operations, Inc. shall evaluate dryer loads and re-establish the limit curve based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency.

c. After reaching 120% of OLTP, Entergy Nuclear Operations, Inc. shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the steam dryer monitoring plan (SDMP) limit curve with the updated ACM load definition and revised instrument uncertainty, which will be provided to the NRC staff.

d. During power ascension above OLTP, if an engineering evaluation is required in accordance with the SDMP, Entergy Nuclear Operations, Inc. shall perform the structural analysis to address frequency uncertainties up to ±10% and assure that peak responses that fall within this uncertainty band are addressed.

e. Entergy Nuclear Operations, Inc. shall revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with General Electric Services Information Letter 644, Revision 1; and to identify the NRC Project Manager for the facility as the point of contact for providing SDMP information during power ascension.

f. Entergy Nuclear Operations, Inc. shall submit the final extended power uprate (EPU) steam dryer load definition for the facility to the NRC upon completion of the power ascension test program.

g. Entergy Nuclear Operations, Inc. shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC, including methodology for updating the limit curve, prior to initial power ascension above OLTP.

3. Entergy Nuclear Operations, Inc. shall prepare the EPU startup test procedure to include the (a) stress limit curve to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria are not satisfied; and (i) verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above OLTP. Entergy Nuclear Operations, Inc. shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above OLTP.

4. When operating above OLTP, the operating limits, required actions, and surveillances specified in the SDMP shall be met. The following key attributes of the SDMP shall not be made less restrictive without prior NRC approval:

    a. During initial power ascension testing above OLTP, each test plateau increment shall be approximately 80 MWt;

    b. Level 1 performance criteria; and

    c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.

    Changes to other aspects of the SDMP may be made in accordance with the guidance of NEI 99-04.

5. During each of the three scheduled refueling outages (beginning with the spring 2007 refueling outage), a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer, including flaws left "as is" and modifications.

6. The results of the visual inspections of the steam dryer conducted during the three scheduled refueling outages (beginning with the spring 2007 refueling outage) shall be reported to the NRC staff within 60 days following startup from the respective refueling outage. The results of the SDMP shall be submitted to the NRC staff in a report within 60 days following the completion of all EPU power ascension testing.

7. The requirements of paragraph 4 above for meeting the SDMP shall be implemented upon issuance of the EPU license amendment and shall continue until the completion of one full operating cycle at EPU. If an unacceptable structural flaw (due to fatigue) is detected during the subsequent visual inspection of the steam dryer, the requirements of paragraph 4 shall extend another full operating cycle until the visual inspection standard of no new flaws/flaw growth based on visual inspection is satisfied.

8. This license condition shall expire upon satisfaction of the requirements in paragraphs 5, 6, and 7 provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is due to fatigue.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License
and Technical Specifications

Date of Issuance: March 2, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 229

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

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

Facility Operating License

Remove	Insert
3	3
9	9
---	10
---	11
---	12
---	13

Technical Specifications

Remove	Insert
3	3
6	6
7	7
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228	228

7590-01-P

U.S. NUCLEAR REGULATORY COMMISSION

ENTERGY NUCLEAR VERMONT YANKEE, LLC AND

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-271

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND FINAL DETERMINATION OF NO SIGNIFICANT

HAZARDS CONSIDERATION

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 229 to Facility Operating License No. DPR-28, issued to Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensee), which revised the Technical Specifications (TSs) and License for operation of the Vermont Yankee Nuclear Power Station (VYNPS) located in Windham County, Vermont. The amendment was effective as of the date of its issuance.

The amendment increases the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWt) to 1912 MWt, which is an increase of approximately 20 percent. The increase in power level is considered an extended power uprate.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

The Commission published a "Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing" related to this action in the FEDERAL REGISTER on July 1, 2004 (69 FR 39976). This Notice provided 60 days for the public to request a hearing. On August 30, 2004, the Vermont Department of Public Service and the New England Coalition filed requests for hearing in connection with the proposed amendment. By Order dated November 22, 2004, the Atomic Safety and Licensing Board (ASLB) granted those hearing requests and by Order dated December 16, 2004, the ASLB issued its decision to conduct a hearing using the procedures in 10 CFR Part 2, Subpart L, "Informal Hearing Procedures for NRC Adjudications."

The Commission published a "Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination" related to this action in the FEDERAL REGISTER on January 11, 2006 (71 FR 1744). This Notice provided 30 days for public comment. The Commission received comments on the proposed no significant hazards consideration as discussed below.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. Public comments received on the proposed no significant hazards consideration determination were considered in making the final determination. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any hearing will be held after issuance.

The Commission published an Environmental Assessment related to the action in the FEDERAL REGISTER on January 27, 2006 (71 FR 4614). Based on the Environmental Assessment, the Commission concluded that the action will not have a significant effect on the quality of the human environment. Accordingly, the Commission determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to this action, see the application for amendment dated September 10, 2003, as supplemented by letters dated October 1, and October 28 (2 letters), 2003; January 31 (2 letters), March 4, May 19, July 2, July 27, July 30, August 12, August 25, September 14, September 15, September 23, September 30 (2 letters), October 5, October 7 (2 letters), December 8, and December 9, 2004; February 24, March 10, March 24, March 31, April 5, April 22, June 2, August 1, August 4, September 10, September 14, September 18, September 28, October 17, October 21 (2 letters), October 26, October 29, November 2, November 22, and December 2, 2005; January 10, and February 22, 2006, which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 2nd day of March, 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

VY EPU SER SECTION 2.1.6

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-28

ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Proprietary information pursuant to
Title 10 of the *Code of Federal Regulations* Section 2.390
has been redacted from this document.
Redacted information is identified by blank space enclosed within double brackets.

suppression pool turbulence and the form of the failed coating will not change. Since the debris accumulated on the pump suction strainers will not cause increased head loss, the licensee concluded that the proposed EPU will not change the NPSH in the RHR and CS pumps.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coatings systems.

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. The NRC staff has reviewed the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in EPRI Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS™ computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

VYNPS has a procedurally controlled FAC inspection program relying on selective component inspections to provide a measure of confidence in the condition of components susceptible to FAC. The program is based on the guidelines developed by EPRI and the American Society of Mechanical Engineers (ASME). The selection of the components for these inspections and their scope is determined by the susceptibility of these components to FAC as determined by the licensee developed criteria. The components which either indicate damage caused by FAC or show signs that such damage could occur before the next outage, are repaired or replaced. The licensee determines predicted rates of wear by FAC using the CHECWORKS™ program. This program calculates wear rates due to FAC from the operating parameters in the plant. Some of these parameters will be affected by the proposed EPU and their changes will have an impact on FAC wear rates. Increase in velocity of flow of single- or two-phase fluid (which is expected to occur in some lines) will produce higher FAC wear rates. The licensee has determined that an increase in the velocities in the main steam line and feedwater lines will cause proportional increases in FAC wear rates. The proposed EPU will also have an effect on moisture and oxygen content, and on temperature. A change of these parameters will impact FAC in the main steam drains, moisture separator drains, and the turbine cross around system piping and will require the licensee to suitably modify the FAC inspection program for these systems. The piping in the extraction steam system at VYNPS is made from material immune to FAC. In response to an NRC staff RAI, the licensee, in Reference 6, provided information on typical expected changes due to FAC in several plant systems subsequent to EPU. After reviewing this information, the staff concurred with the licensee's assessment that the proposed EPU could cause an increase of FAC in some plant systems. Accordingly, the licensee plans to modify the inputs to the CHECWORKS™ program and introduce some changes to the FAC inspection program to account for the changes due to the EPU.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. The NRC staff further concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The reactor water cleanup system (RWCS) provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCS comprise the RCPB. The NRC staff's review of the RWCS included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant's TSs in these areas under the proposed EPU conditions. The NRC's acceptance criteria for the RWCS are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) draft GDC-51, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

The licensee reviewed functional capability of the RWCS to ensure that after the proposed EPU, it will meet the requirements of draft GDC-9, GDC-51, and GDC-70. The review has indicated that, although some operating system parameters will change slightly after the proposed EPU, the system will be able to perform its functions in a satisfactory manner and will meet the requirements imposed by the draft GDCs. The slight changes in operating system parameters result primarily due to increased feedwater flow. The licensee has determined that the changes due to the EPU will produce no detectable effect on system performance including

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E. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1912 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.

B. Technical Specifications

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

C. Reports

Entergy Nuclear Operations, Inc. shall make reports in accordance with the requirements of the Technical Specifications.

D. This paragraph deleted by Amendment No. 226.

E. Environmental Conditions

Pursuant to the Initial Decision of the presiding Atomic Safety and Licensing Board issued February 27, 1973, the following conditions for the protection of the environment are incorporated herein:

b. Surety

(i) The surety agreement must be in a form acceptable to the NRC and be in accordance with all applicable NRC regulations.

(ii) The surety company providing any surety obtained to comply with the Order approving the transfer shall be one of those listed by the U.S. Department of the Treasury in the most recent edition of Circular 570 and shall have a coverage limit sufficient to cover the amount of the surety.

(iii) Entergy Nuclear Vermont Yankee, LLC shall establish a standby trust to receive funds from the surety, if a surety is obtained, in the event that Entergy Nuclear Vermont Yankee, LLC defaults on its funding obligations for the decommissioning of Vermont Yankee. The standby trust agreement must be in a form acceptable to the NRC, and shall conform with all conditions otherwise applicable to the decommissioning trust agreement.

(iv) The surety agreement must provide that the agreement cannot be amended in any material respect, or terminated, without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

Entergy Nuclear Vermont Yankee, LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of this license to Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. shall take no action to cause Entergy Global Investments, Inc., or Entergy International Holdings Ltd. LLC, or their parent companies to void, cancel, or modify the lines of credit to provide funding for Vermont Yankee as represented in the application without prior written consent of the Director of the Office of Nuclear Reactor Regulation.

K. Minimum Critical Power Ratio

When operating at thermal power greater than 1593 megawatts thermal, the safety limit minimum critical power ratio (SLMCPR) shall be established by adding 0.02 to the cycle-specific SLMCPR value calculated using the NRC-approved methodologies documented in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," as amended, and documented in the Core Operating Limits Report.

L. Transient Testing

1. During the extended power uprate (EPU) power ascension test program and prior to exceeding 168 hours of plant operation at the nominal full EPU reactor power level, with feedwater and condensate flow rates stabilized at approximately the EPU full power level, Entergy Nuclear Operations, Inc. shall confirm through performance of transient testing that the loss of one condensate pump will not result in a complete loss of reactor feedwater.

2. Within 30 days at nominal full-power operation following successful performance of the test in (1) above, through performance of additional transient testing and/or analysis of the results of the testing conducted in (1) above, confirm that the loss of one reactor feedwater pump will not result in a reactor trip.

M. Potential Adverse Flow Effects

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer).

1. The following requirements are placed on operation of the facility above the original licensed thermal power (OLTP) level of 1593 megawatts thermal (MWt):

   a. Entergy Nuclear Operations, Inc. shall monitor hourly the 32 main steam line (MSL) strain gages during power ascension above 1593 MWt for increasing pressure fluctuations in the steam lines.

   b. Entergy Nuclear Operations, Inc. shall hold the facility for 24 hours at 105%, 110%, and 115% of OLTP to collect data from the 32 MSL strain gages required by Condition M.1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall provide the evaluation to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission.

   c. If any frequency peak from the MSL strain gage data exceeds the limit curve established by Entergy Nuclear Operations, Inc. and submitted to the NRC staff prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall return the facility to a power level at which the limit curve is not exceeded. Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

d. In addition to evaluating the MSL strain gage data, Entergy Nuclear Operations, Inc. shall monitor reactor pressure vessel water level instrumentation or MSL piping accelerometers on an hourly basis during power ascension above OLTP. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data, Entergy Nuclear Operations, Inc. shall stop power ascension, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

e. Following start-up testing, Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis and provide that resolution to the NRC staff by facsimile or electronic transmission to the NRC project manager. If the uncertainties are not resolved within 90 days of issuance of the license amendment authorizing operation at 1912 MWt, Entergy Nuclear Operations, Inc. shall return the facility to OLTP.

2. As described in Entergy Nuclear Operations, Inc. letter BVY 05-084 dated September 14, 2005, Entergy Nuclear Operations, Inc. shall implement the following actions:

a. Prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall install 32 additional strain gages on the main steam piping and shall enhance the data acquisition system in order to reduce the measurement uncertainty associated with the acoustic circuit model (ACM).

b. In the event that acoustic signals are identified that challenge the limit curve during power ascension above OLTP, Entergy Nuclear Operations, Inc. shall evaluate dryer loads and re-establish the limit curve based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency.

c. After reaching 120% of OLTP, Entergy Nuclear Operations, Inc. shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the steam dryer monitoring plan (SDMP) limit curve with the updated ACM load definition and revised instrument uncertainty, which will be provided to the NRC staff.

d. During power ascension above OLTP, if an engineering evaluation is required in accordance with the SDMP, Entergy Nuclear Operations, Inc. shall perform the structural analysis to address frequency uncertainties up to ±10% and assure that peak responses that fall within this uncertainty band are addressed.

e. Entergy Nuclear Operations, Inc. shall revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with General Electric Services Information Letter 644, Revision 1; and to identify the NRC Project Manager for the facility as the point of contact for providing SDMP information during power ascension.

f. Entergy Nuclear Operations, Inc. shall submit the final extended power uprate (EPU) steam dryer load definition for the facility to the NRC upon completion of the power ascension test program.

g. Entergy Nuclear Operations, Inc. shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC, including methodology for updating the limit curve, prior to initial power ascension above OLTP.

3. Entergy Nuclear Operations, Inc. shall prepare the EPU startup test procedure to include the (a) stress limit curve to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria are not satisfied; and (i) verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above OLTP. Entergy Nuclear Operations, Inc. shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above OLTP.

4. When operating above OLTP, the operating limits, required actions, and surveillances specified in the SDMP shall be met. The following key attributes of the SDMP shall not be made less restrictive without prior NRC approval:

   a. During initial power ascension testing above OLTP, each test plateau increment shall be approximately 80 MWt;

   b. Level 1 performance criteria; and

   c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.

   Changes to other aspects of the SDMP may be made in accordance with the guidance of NEI 99-04.

5. During each of the three scheduled refueling outages (beginning with the spring 2007 refueling outage), a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer, including flaws left "as is" and modifications.

6. The results of the visual inspections of the steam dryer conducted during the three scheduled refueling outages (beginning with the spring 2007 refueling outage) shall be reported to the NRC staff within 60 days following startup from the respective refueling outage. The results of the SDMP shall be submitted to the NRC staff in a report within 60 days following the completion of all EPU power ascension testing.

7. The requirements of paragraph 4 above for meeting the SDMP shall be implemented upon issuance of the EPU license amendment and shall continue until the completion of one full operating cycle at EPU. If an unacceptable structural flaw (due to fatigue) is detected during the subsequent visual inspection of the steam dryer, the requirements of paragraph 4 shall extend another full operating cycle until the visual inspection standard of no new flaws/flaw growth based on visual inspection is satisfied.

8. This license condition shall expire upon satisfaction of the requirements in paragraphs 5, 6, and 7 provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is due to fatigue.

4. This license is effective as of the date of issuance and shall expire at midnight on March 21, 2012.

A-127
12/17/90

FOR THE ATOMIC ENERGY COMMISSION

Original Signed By
Roger S. Boyd /f/

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosures:
Appendix A Technical Specifications

Date of Issuance:
Feb. 28, 1973

1.0 DEFINITIONS

or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1912 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1912 thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser vacuum trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Deleted

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and the core flow is greater than 10% of rated:

1. A Minimum Critical Power Ratio (MCPR) of less than 1.07 (1.09 for Single Loop Operation) shall constitute violation of the Fuel Cladding Integrity Safety Limit (FCISL).

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip setting of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Allowable Value (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram Allowable Value shall be:

Two loop operation:

$S \leq 0.33W + 50.45\%$ for $0\% < W \leq 30.9\%$

$S \leq 1.07W + 27.23\%$ for $30.9\% < W \leq 66.7\%$

$S \leq 0.55W + 62.34\%$ for $66.7\% < W \leq 99.0\%$

With a maximum of 117.0% power for $W > 99.0\%$

Single loop operation:

$S \leq 0.33W + 48.00\%$ for $0\% < W \leq 39.1\%$

$S \leq 1.07W + 19.01\%$ for $39.1\% < W \leq 61.7\%$

$S \leq 0.55W + 51.22\%$ for $61.7\% < W \leq 119.4\%$

With a maximum of 117.0% power for $W > 119.4\%$

where:

S = setting in percent of rated thermal power (1912 MWt)

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure ≤800 psia or Core Flow ≤10% of Rated)

When the reactor pressure is ≤800 psia or core flow ≤10% of rated, the core thermal power shall not exceed 23% of rated thermal power.

C. Power Transient

To ensure that the safety limit established in Specification 1.1A and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The safety limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

D. Whenever the reactor is shutdown with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the enriched fuel when it is seated in the core.

2.1 LIMITING SAFETY SYSTEM SETTING

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

In the event of operation at > 23% Rated Thermal Power the APRM gain shall be equal to or greater than 1.0.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

D. Reactor low-low water level Emergency Core Cooling System (ECCS) initiation shall be at least 82.5 inches above the top of the enriched fuel.

E. Turbine stop valve scram shall, when operating at greater than 25% of Rated Thermal Power, be less than or equal to 10% valve closure from full open.

F. Turbine control valve fast closure scram shall, when operating at greater than 25% of Rated Thermal Power, trip upon actuation of the turbine control valve fast closure relay.

G. Main steam line isolation valve closure scram shall be less than or equal to 10% valve closure from full open.

H. Main steam line low pressure initiation of main steam line isolation valve closure shall be at least 800 psig.

BASES:

1.1 FUEL CLADDING INTEGRITY

A. Refer to General Electric Company Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (most recent revision).

The fuel cladding integrity Safety Limit (SL) is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations.

The MCPR fuel cladding integrity SL is increased for single loop operation in order to account for increased core flow measurement and TIP reading uncertainties.

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$10% of Rated)

The General Electric critical power correlation (also known as the GEXL critical power correlation) is applicable for operation at pressures greater than or equal to 800 psia and core flows greater than or equal to 10% of rated flow. For operation at lower pressures or core flows, the following basis is used:

At power levels at or below the low pressure, low flow (low power) thermal limit, the minimum core flow occurs for natural circulation, and as the power to flow ratio in natural circulation increases with increasing power, the maximum and most limiting power to flow ratio occurs for natural circulation at the low power thermal limit. This condition is therefore also the condition with the minimum margin to critical power. Analysis of the natural circulation flow rate at the low power thermal limit has shown that the core average mass flux is 0.3-0.4 Mlb/hr-ft^2 and the corresponding core pressure drop is 5-6 psi. For these conditions, full scale ATLAS test data have shown a critical power of 4-5 MWt. Analysis has also shown that a maximum radial peaking factor of 2 is expected at the low power thermal limit condition. Since the low power thermal limit basis corresponds to a maximum average bundle power of 1.2 MWt or less, fuel bundles with radial peaking factor as high as 3 will have margin to critical power. This bounds any radial peaking, and therefore the low power thermal limit is conservative. An average bundle power of 1.2 MWt occurs at 23% rated thermal power. Thus, a limit of 23% rated thermal power for operation with reactor pressure less than or equal to 800 psia is conservative.

BASES: 1.1 (Cont'd)

With no reactor coolant recirculation loops in operation, the plant must be brought to a condition in which the LCO does not apply. Operation of at least one reactor coolant recirculation loop provides core flow greater than natural circulation, so the margin to a critical power condition is significantly greater than this bounding example for all normal operating conditions with power less than the low power thermal limit. Therefore, a low power thermal limit of 23% rated thermal power is conservative.

Additionally, a core thermal power limit of 23% rated thermal power ensures consistency with the threshold for requiring thermal limit monitoring (i.e., average planar linear heat generation rate, linear heat generation rate, and minimum critical power ratio). This assures that for those power levels where thermal limit monitoring is required, the General Electric critical power correlation is applicable.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Vermont Yankee has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the enriched fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the enriched fuel provides adequate margin. This level will be continuously monitored.

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 Allowable Value (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 (1912 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 setting 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. The relationship between recirculation drive flow and reactor core flow is non-linear at low core flows. Due to stability concerns, separate APRM flow biased scram trip setting equations are provided for low core flows.

The APRM flow biased flux scram Allowable Value is the limiting value that the trip setpoint may have when tested periodically, beyond which appropriate action shall be taken. For Vermont Yankee, the periodic testing is defined as the calibration. The actual scram trip is conservatively set in relation to the Allowable Value to ensure operability between periodic testing. 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. The single loop

BASES: 2.1 (Cont'd)

Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 23% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psia.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

BASES: 2.1 (Cont'd)

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram signal may be bypassed at ≤ 25% of reactor Rated Thermal Power.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists. This scram signal may be bypassed at ≤ 25% of reactor Rated Thermal Power.

G. Main Steam Line Isolation Valve Closure Scram

The isolation scram anticipates the pressure and flux transients which occur during an isolation event and the loss of inventory during a pipe break. This action minimizes the effect of this event on the fuel and pressure vessel.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating: Refuel (1)	Startup(12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required ACTIONS When Minimum Conditions For Operation Are Not Satisfied (3)
1.	Mode Switch in Shutdown (5A-S1)		X	X	X	1	A
2.	Manual Scram (5A-S3A/B)		X	X	X	1	A
3.	IRM (7-41(A-F))						
	High Flux	≤120/125	X	X		2	A
	INOP		X	X		2	A
4.	APRM (APRM A-F)						
	High Flux (flow bias)	Two loop operation: (4) S≤ 0.33W+ 50.45% for 0% < W ≤ 30.9% S≤ 1.07W+ 27.23% for 30.9% < W ≤ 66.7% S≤ 0.55W+ 62.34% for 66.7% < W ≤ 99.0% With a maximum of 117.0% power for W > 99.0% Single loop operation: (4) S≤ 0.33W+ 48.00% for 0% < W ≤ 39.1% S≤ 1.07W+ 19.01% for 39.1% < W ≤ 61.7% S≤ 0.55W+ 51.22% for 61.7% < W ≤ 119.4% With a maximum of 117.0% power for W > 119.4%			X	2	A or B
	High Flux (reduced)	≤15%	X	X		2	A
	INOP			X	X	2(5)	A or B
5.	High Reactor Pressure (PT-2-3-55(A-D) (M)	≤1055 psig	X	X	X	2	A

TABLE 3.1.1 NOTES (Cont'd)

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate ACTIONS listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 25% of rated within 8 hours.

4. The specified APRM High Flux scram (flow bias) Trip Setting is an Allowable Value, which is the limiting value that the trip setpoint may have when tested periodically. The actual scram trip setting is conservatively set in relation to the Allowable Value. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Deleted.

8. Deleted.

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at $\leq$ 25% of reactor Rated Thermal Power.

11. Not used.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

       1. Mode switch in shutdown,
       2. Manual scram,
       3. High flux IRM scram
       4. High flux SRM scram in noncoincidence,
       5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be face adjacent or diagonally adjacent.

BASES: 3.1 (Cont'd)

Instrumentation is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The Control Rod Drive Scram System is designed so that all of the water that is discharged from the reactor by the scram can be accommodated in the discharge piping. This discharge piping is divided into two sections. One section services the control rod drives on the north side of the reactor, the other serves the control rod drives of the south side. A part of the piping in each section is an instrument volume which accommodates in excess of 21 gallons of water and is at the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level instrumentation has been provided for the instrument volume which scram the reactor when the volume of water reaches 21 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water, and precludes the situation in which a scram would be required but not be able to perform its function adequately. The present design of the Scram Discharge System is in concert with the BWR Owner's Group criteria, which have previously been endorsed by the NRC in their generic "Safety Evaluation Report (SER) for Scram Discharge Systems", dated December 1, 1980.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass.

Turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure scram signals may be bypassed at ≤ 25% of reactor Rated Thermal Power since, at low thermal power levels, the margins to fuel thermal-hydraulic limits and reactor primary coolant boundary pressure limits are large and an immediate scram is not necessary. This bypass function is normally accomplished automatically by pressure switches sensing turbine first stage pressure. The turbine first stage pressure setpoint controlling the bypass of the scram signals on TCV fast closure and TSV closure is derived from analysis of reactor pressurization transients. Certain operational factors, such as turbine bypass valves open, can influence the relationship between turbine first stage pressure and reactor Rated Thermal Power. However, above 25% of reactor Rated Thermal Power, these scram functions must be enabled.

3.3 *LIMITING CONDITIONS FOR OPERATION*

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 17% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

   (a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

   (b) If all rods, except those that cannot be moved with control rod drive

4.3 *SURVEILLANCE REQUIREMENTS*

positive coupling and the results of each test shall be recorded. The drive and blade shall be coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

   (a) Verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

   (b) The Rod Worth Minimizer diagnostic test shall be performed.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 17% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system provides a scram function in noncoincident configuration. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. Deleted.

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable when the reactor mode switch is in either the "Startup/Hot Standby" or "Run" position, except to allow testing of instrumentation associated with the reactor mode switch interlock functions provided:

1. Reactor coolant temperature is less than or equal to 212° F;

2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and

3. No core alterations are in progress.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at ≥ 1325 psig shall be verified for each pump.

2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

3. Deleted

4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor vessel is available. Both loops shall be tested over the course of two operating cycles.

3.4 LIMITING CONDITIONS FOR OPERATION

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

3. The combination of Standby Liquid Control System pump flow rate, boron concentration, and boron enrichment shall satisfy the following relationship for the Standby Liquid Control System to be considered operable:

$$\frac{Q}{86} \times \frac{M251}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq 1.29$$

where:

C = the concentration of sodium pentaborate solution (weight percent) in the Standby Liquid Control System tank

E = the boron-10 enrichment (atom percent) of the sodium pentaborate solution

$Q \geq$ 35 gpm

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

3. The boron-10 enrichment of the borated solution required by Specification 3.4.C.3 shall be tested and verified once per operating cycle.

BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System (SLCS) is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of natural boron in the reactor core in less than 138 minutes. An 800 ppm natural boron concentration in the reactor core is required to bring the reactor from full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% natural sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

In addition to its original design basis, the Standby Liquid Control System also satisfies the requirements of 10CFR50.62(c)(4) on anticipated transients without scram (ATWS) by using enriched boron. The ATWS rule adds hot shutdown and neutron absorber (i.e., boron-10) injection rate requirements that exceed the original Standby Liquid Control System design basis. However, changes to the Standby Liquid Control System as a result of the ATWS rule have not invalidated the original design basis.

With the reactor mode switch in the "Run" or "Startup/Hot Standby" position, shutdown capability is required. With the mode switch in "Shutdown," control rods are not able to be withdrawn since a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. With the mode switch in "Refuel," only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate shutdown margin by Specification 3.3.A ensures that the reactor will not become critical. Therefore, the Standby Liquid Control System is not required to be operable when only a single control rod can be withdrawn.

Pump operability testing (by recirculating demineralized water to the test tank)in accordance with Specification 4.6.E is adequate to detect if failures have occurred. Flow, circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. The maximum SLCS pump discharge pressure during the limiting ATWS event is 1325 psig. This value is based on a reactor vessel lower plenum pressure of 1292 psia that occurs during the limiting ATWS event at the time of SLCS initiation, i.e., 120 seconds into the event. There is adequate margin to prevent the SLCS relief valve from lifting. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear. In addition, at least once during each operating cycle, one of the standby liquid control loops will be initiated to verify that a flow path from a pump to the reactor vessel is available by pumping demineralized water into the reactor vessel.

BASES: 3.4 & 4.4 (Cont'd)

~~B. Operation With Inoperable Components~~

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with the Requirements of Specification 4.6.E.

C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

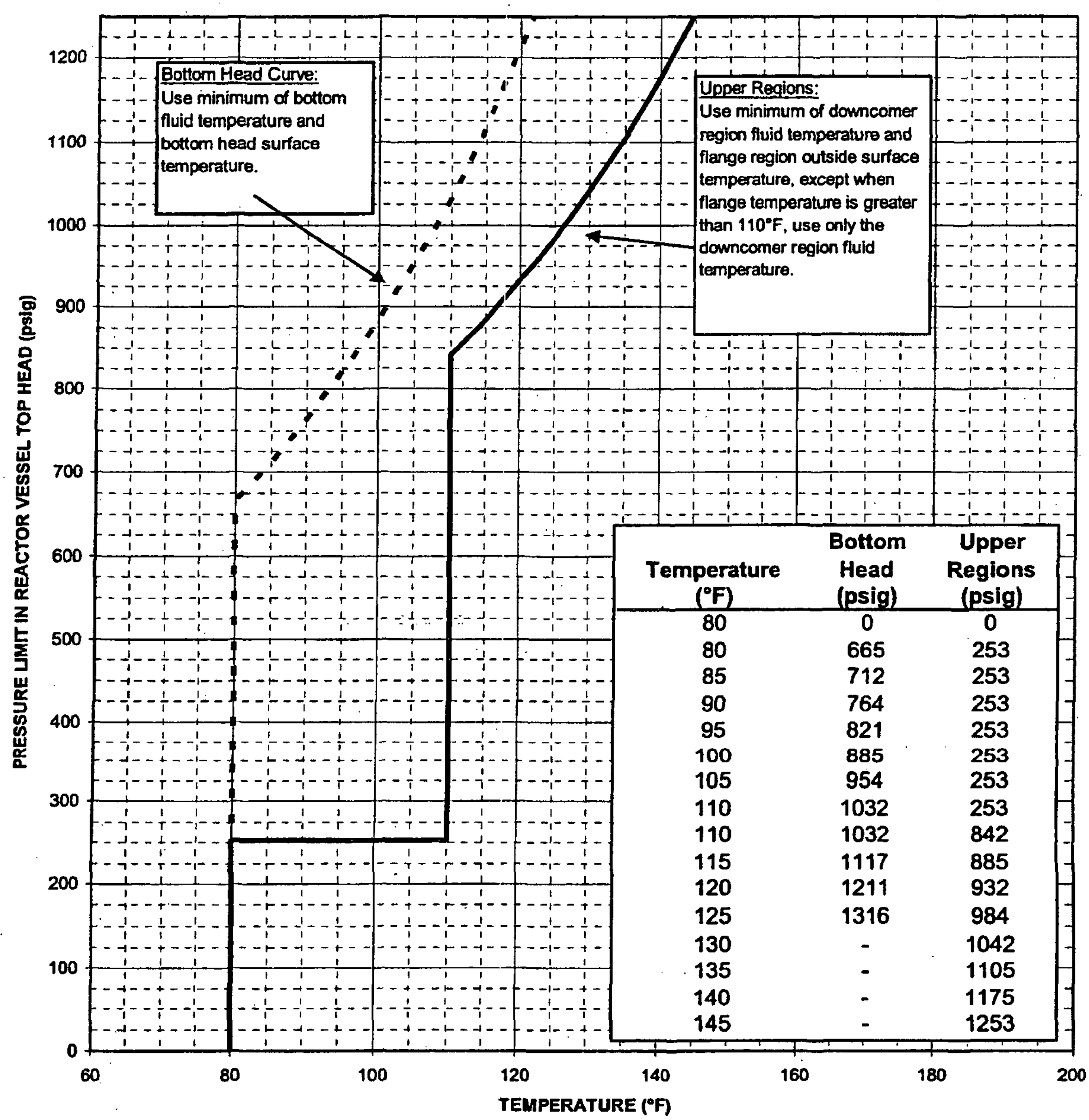
10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule and the plant-specific ATWS analysis, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. If the product of the expression in Specification 3.4.C.3 is equal to or greater than 1.29, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4) and the plant-specific ATWS analysis.

Figure 3.6.1

Reactor Vessel Pressure-Temperature Limitations
Hydrostatic Pressure and Leak Tests, Core Not Critical

40°F/hr Heatup/Cooldown Limit
Valid Through 4.827E8 MWH(t)

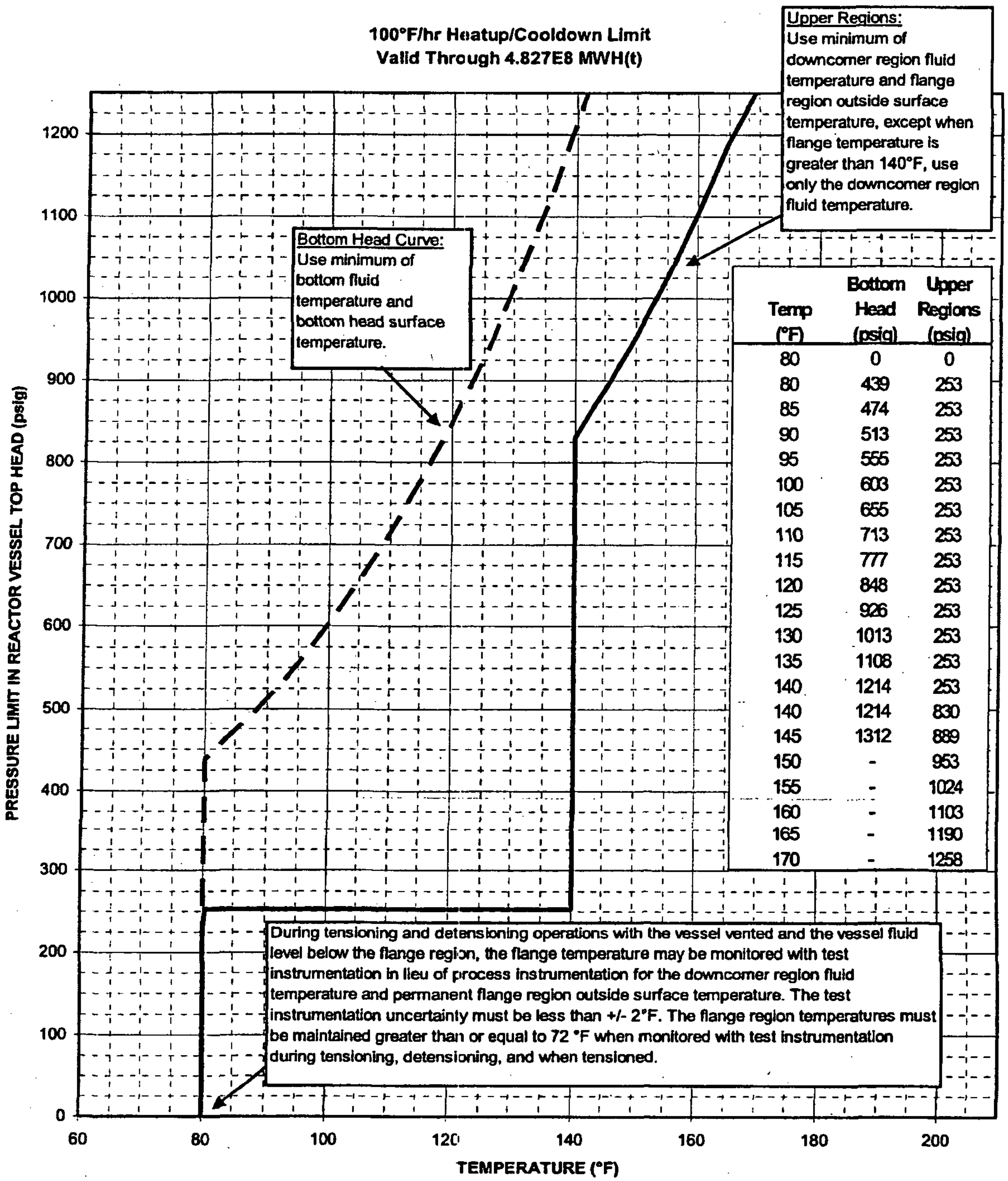


Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	665	253
85	712	253
90	764	253
95	821	253
100	885	253
105	954	253
110	1032	253
110	1032	842
115	1117	885
120	1211	932
125	1316	984
130	-	1042
135	-	1105
140	-	1175
145	-	1253

Figure 3.6.2

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Not Critical

100°F/hr Heatup/Cooldown Limit
Valid Through 4.827E8 MWH(t)

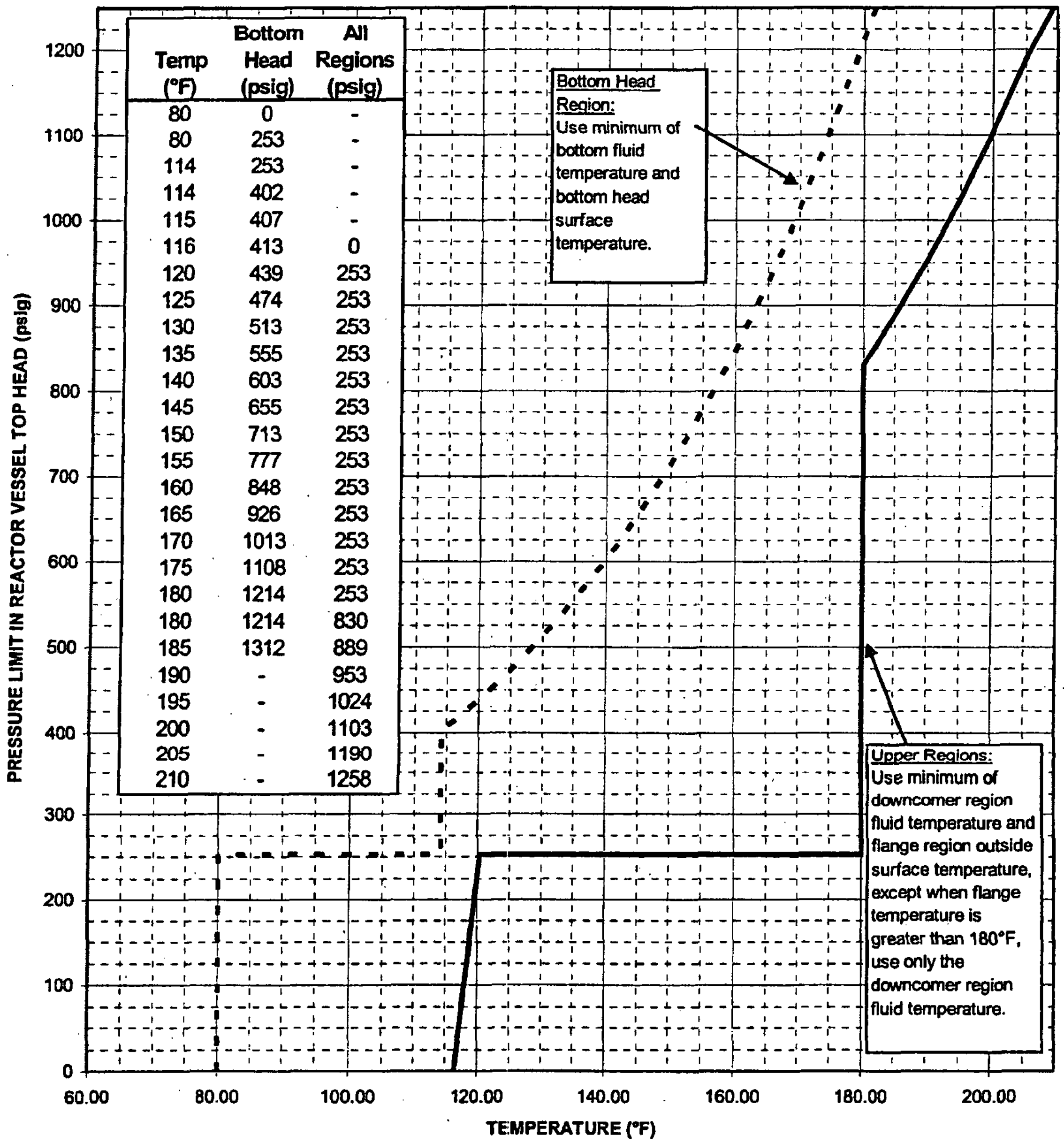


Temp (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

Figure 3.6.3

Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Critical

100°F/hr Heatup/Cooldown Limit
If Pressure < 253 psig, Water Level must be within Normal Range for Power Operation
Valid Through 4.827E8 MWH(t)



Temp (°F)	Bottom Head (psig)	All Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	413	0
120	439	253
125	474	253
130	513	253
135	555	253
140	603	253
145	655	253
150	713	253
155	777	253
160	848	253
165	926	253
170	1013	253
175	1108	253
180	1214	253
180	1214	830
185	1312	889
190	-	953
195	-	1024
200	-	1103
205	-	1190
210	-	1258

BASES:

3.6 and 4.6 REACTOR COOLANT SYSTEM

A. Pressure and Temperature Limitations

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The Pressure/Temperature (P/T) curves included as Figures 3.6.1, 3.6.2, and 3.6.3 were developed using 10CFR50 Appendix G, 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda), and ASME Code Case N-640. These three curves provide P/T limit requirements for Pressure Test, Core Not Critical, and Core Critical. The P/T curves are not derived from Design Basis Accident analysis. They are prescribed to avoid encountering pressure, temperature or temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure boundary, a condition that is unanalyzed.

During heating events, the thermal gradients in the reactor vessel wall produce thermal stresses that vary from compressive at the inner wall to tensile at the outer wall. During cooling events the thermal stresses vary from tensile at the inner wall to compressive at the outer wall. The thermally induced tensile stresses are additive to the pressure induced tensile stresses. In the flange region, bolt preload has a significant affect on stress in the flange and adjacent plates. Therefore heating/cooling events and bolt preload are used in the determination of the pressure-temperature limitations for the vessel.

The guidance of Branch Technical Position - MTEB 5-2, material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RT_{NDT}) for all pressure boundary components. For the plates and welds adjacent to the core, fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fast neutron fluence (3.18×10^{17} n/cm^2 at the reactor vessel inside surface) for a gross power generation of 4.827×10^8 MWH(t), these core region ART_{NDT} values conservatively bound the guidance of Regulatory Guide 1.99, Revision 2.

There were five regions of the reactor pressure vessel (RPV) that were evaluated in the development of the P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These regions will bound all other regions in the vessel with respect to considerations for brittle fracture.

Two lines are shown on each P/T limit figure. The dashed line is the Bottom Head Curve. This is applicable to the bottom head area only and includes the bottom head knuckle plates and dollar plates. Based on bottom head fluid temperature and bottom head surface temperature, the reactor pressure shall be maintained below the dashed line at all times.

BASES: 3.6 and 4.6 (Cont'd)

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 equipment 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 ensure compliance with the MCPR safety limit for the analyzed transients.

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. 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. 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.

3.11 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 assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During operation at ≥ 23% Rated Thermal Power, the APLHGR for each type of fuel as a function of average planar exposure, power, and flow shall not exceed the limiting values provided in the Core Operating Limits Report. For single recirculation loop operation, the limiting values shall be the values provided in the Core Operating Limits Report listed under the heading "Single Loop Operation." If at any time during operation at ≥ 23% Rated Thermal Power it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, APLHGR(s) shall be returned to within prescribed limits within two (2) hours; otherwise, the reactor shall be brought to < 23% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 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, power, and flow shall be determined once within 12 hours after ≥ 23% Rated Thermal Power and daily during operation at ≥ 23% Rated Thermal Power thereafter.

3.11 LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

During operation at ≥ 23% Rated Thermal Power, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR provided in the Core Operating Limits Report.

If at any time during operation at ≥ 23% Rated Thermal Power it is determined by normal surveillance that the limiting value for LHGR is being exceeded, LHGR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor shall be brought to < 23% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked once within 12 hours after ≥ 23% Rated Thermal Power and daily during operation at ≥ 23% Rated Thermal Power thereafter.

3.11 LIMITING CONDITIONS FOR OPERATION

C. ~~Minimum Critical Power Ratio~~ (MCPR)

1. During operation at ≥ 23% Rated Thermal Power the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. If at any time during operation at ≥ 23% Rated Thermal Power it is determined by normal surveillance that the limiting value for MCPR is being exceeded, MCPR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor power shall be brought to < 23% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

C. ~~Minimum Critical Power Ratio~~ (MCPR)

MCPR shall be determined once within 12 hours after ≥ 23% Rated Thermal Power and daily during operation at ≥ 23% Rated Thermal Power thereafter.

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily when operating at ≥ 23% Rated Thermal Power to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 23% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. The 12 hour allowance after thermal power ≥ 23% Rated Thermal Power is achieved is acceptable given the large inherent margin to operating limits at low power levels.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

If a reactor power distribution limit is exceeded, an assumption regarding an initial condition of the DBA analysis, transient analyses, or the fuel design analysis may not be met. Therefore, prompt action should be taken to restore the APLHGR, LHGR or MCPR to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour completion time is sufficient to restore the APLHGR, LHGR, or MCPR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR, LHGR, or MCPR out of specification.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 23%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 23% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 23% rated thermal power is sufficient since power distribution shifts are very slow during normal operation.