Nebraska Public Power District

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NLS2007069
November 19, 2007

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate
Cooper Nuclear Station Docket 50-298, DPR-46

Dear Sir or Madam:

The purpose of this letter is for Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 in accordance with the provisions of 10 CFR 50.40 and 10 CFR 50.90 and to revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). Specifically NPPD is requesting Nuclear Regulatory Commission (NRC) approval of a Measurement Uncertainty Recapture (MUR) power uprate TS change to increase the licensed reactor core power level by 1.62 percent from a Current Licensed Thermal Power of 2381 megawatts thermal (MWt) to 2419 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy Caldon CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. The instrumentation will be installed during refueling outage 24, scheduled to start April 12, 2008.

NPPD has proposed only those license and TS changes that are required to implement the increased power level.

The proposed TS change is consistent with the guidelines in NRC Regulatory Issue Summary, 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." In addition, Requests for Additional Information (RAIs) regarding MUR applications for other nuclear units were reviewed for applicability. Information that addresses many of those RAIs is included in this proposed TS change.

The proposed TS change is described and discussed in attachments and enclosures to this letter. Attachments and enclosures are summarized in the table below:

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NRR

COOPER NUCLEAR STATION
P.O. Box 98 / Brownville, NE 68321-0098
Telephone: *(402) 825-3811 /* ***Fax:*** *(402) 825-5211*
www.nppd.com

DESCRIPTION of ATTACHMENTS and ENCLOSURES	
Attachment 1	License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate
Attachment 2	Operating License And Technical Specifications Pages - Marked Up With Proposed Changes (TRM Pages Included for Information)
Attachment 3	Revised (Clean Copies) Of The Operating License And Technical Specifications Pages
Attachment 4	NRC Regulatory Issue Summary (RIS) 2002-03 Reconciliation
Attachment 5	List of Regulatory Commitments
Enclosure 1	GE-Hitachi Nuclear Energy Safety Analysis Report for Cooper Nuclear Station Thermal Power Optimization, NEDC-33385P (Proprietary Version)
Enclosure 2	Affidavit of Withholding Pursuant to 10 CFR 2.390 GE-Hitachi
Enclosure 3	GE-Hitachi Nuclear Energy Safety Analysis Report for CNS Thermal Power Optimization, NEDO-33385 (Non-Proprietary Version)
Enclosure 4	Caldon ER-592 Rev. 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Cooper NPPD Using the LEFM√+ System" (Proprietary Version)
Enclosure 5	Caldon ER-614 Rev. 1, "LEFM Check√+ Meter Factor Calculation, and Accuracy Assessment for Cooper NPPD " (Proprietary Version)
Enclosure 6	Affidavits of Withholding Pursuant to 10 CFR 2.390 Cameron International Corporation
Enclosure 7	NEDC 06-035 Reactor Core Thermal Power Uncertainty Calculation

Attachment 1 provides a description of the change, a technical and a regulatory analysis, a no significant hazards consideration evaluation pursuant to 10 CFR 50.91(a)(1), and an assessment of environmental impact pursuant to 10 CFR 51.22.

Attachments 2 and 3 provide the revised pages in markup, and in clean typed formats, respectively. TS Bases and Technical Requirements Manual (TRM) pages affected by this amendment request are included for information in Attachment 2.

Attachment 4 provides a reconciliation of RIS 2002-03 requirements with associated Attachments and Enclosures.

Attachment 5 provides a summary of the regulatory commitments associated with the implementation of this request.

Note that Enclosure 1, GE-Hitachi Nuclear Energy (GEH) NEDC-33385P, is proprietary. An affidavit signed by an officer of GEH is provided in Enclosure 2, and is also included in the front of the document. It is requested that this proprietary information be withheld from public disclosure. This request is made pursuant to 10 CFR 2.390. The address of GEH is provided in the cover page of the report included in Enclosure 1. A nonproprietary version for public disclosure is included as Enclosure 3.

Enclosure 4 provides Caldon Engineering Report ER-592 (Proprietary), and Enclosure 5 provides Caldon Engineering Report ER-614 (Proprietary). Non-proprietary versions of these two reports do not exist. Enclosure 6 contains the applications for withholding proprietary information contained in Enclosures 4 and 5 from public disclosure, including affidavits, in conformance with the provisions of 10 CFR 2.390.

Enclosure 7 determines the uncertainty of the reactor thermal power (heat balance) calculation performed by the process computer (GARDEL).

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change does not involve a significant hazards consideration. The bases for these determinations are included in the attachments and enclosures. The proposed change has been reviewed by the necessary safety review committees (Station Operations Review Committee, and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 227 dated September 20, 2007, have been incorporated into this request. NPPD has concluded that the proposed changes satisfy the categorical exclusion criteria of 10 CFR 51.22(c)(9), and preparation of an environmental impact statement or environmental assessment is therefore not required.

This request is submitted under oath pursuant to 10 CFR 50.30(b). By copy of this letter and its attachments and enclosures, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with
10 CFR 50.4(b)(1).

To support the refueling outage currently scheduled to begin April 12, 2008, and to take advantage of the features afforded by the new equipment, NPPD requests NRC approval of the proposed TS change and issuance of the requested license amendment by April 8, 2008, with implementation to be completed within 90 days of issuance.

Should you have any questions concerning this matter, please contact Mr. David Van Der Kamp, at (402) 825-2904.

I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 11/19/07
Date

Sincerely,

Stewart B. Minahan
Vice President Nuclear and
Chief Nuclear Officer

/je

Attachments (5)
Enclosures (7)

cc: Regional Administrator w/ attachments and enclosures
USNRC - Region IV

Cooper Project Manager w/ attachments and enclosures
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/ attachments and enclosures
USNRC - CNS

Nebraska Health and Human Services w/ attachments and enclosures
Department of Regulation and Licensure

NPG Distribution w/o attachments or enclosures

CNS Records w/ attachments and enclosures

ATTACHMENT 1

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS - APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

COOPER NUCLEAR STATION DOCKET 50-298, DPR-46

Revised License and Technical Specifications Pages

License: Page 3; TS Pages: 1.1-4, 3.3-2, 3.3-5, 3.3-6, 3.3-8, 3.4-23, 3.4-24, 3.4-25

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1.0 DESCRIPTION

This is a License Amendment Request (LAR) to amend Facility Operating License DPR-46, and the Technical Specifications for Cooper Nuclear Station (CNS).

Nebraska Public Power District (NPPD) is proposing that the CNS Facility Operating License be amended to reflect an increase in the Rated Thermal Power (RTP) level from a Current Licensed Thermal Power (CLTP) of 2381 megawatt thermal (MWt) to 2419 MWt (1.62% increase). The increase in RTP, evaluated and justified herein, is obtained by installation of a more accurate feedwater flow measuring system. The Leading Edge Flow Meter CheckPlus™ instrumentation system (LEFM CheckPlus System) supplied by Caldon, Inc., will be installed in CNS (References 7.1 and 7.2).

The increased accuracy of the new feedwater flow measuring instrumentation results in increased accuracy of the core thermal power uncertainty calculation ($\leq \pm$ 0.31 percent of core thermal power), versus the previously assumed uncertainty of $\leq \pm$ 2.0 percent of core thermal power. This reduction in uncertainty in the core thermal power calculation (Enclosure 7) allows operation at the proposed increased RTP with no decrease in the confidence level that the actual operating power level is less than the power level required to be assumed in the Emergency Core Cooling Systems (ECCS) accident analyses by 10 CFR 50, Appendix K, "ECCS Evaluation Models."

The improved core thermal power measurement accuracy obviates need for the full 2 percent power margin required to be assumed in the original Appendix K analyses, thereby allowing an increase in thermal power available for electrical generation.

2.0 PROPOSED CHANGE

The proposed license amendment would revise the Cooper Nuclear Station (CNS) Operating License and Technical Specifications (TS) to increase licensed RTP to 2419 MWt, or 1.62% greater than the CLTP of 2381 MWt. The LAR proposes the specific License and TS changes as described below. Technical Requirements Manual (TRM) changes are included for information.

- Paragraph 2.C.(1) in Facility Operating License DPR-46 (page 3) is revised to authorize operation at a steady state reactor core thermal power level not in excess of 2419 MWt.

- The definition of RATED THERMAL POWER (RTP) in TS 1.1, page 1.1-4, is revised to reflect the increase from 2381 MWt to 2419 MWt.

- Reference to "10% RTP" has been scaled down to "9.85% RTP" in the following TS:
    - 3.1.3 CONDITION D (page 3.1-9),
    - 3.1.6 APPLICABILITY (page 3.1-18),
    - SR 3.3.2.1.2 and 3.3.2.1.3 (page 3.3-17),
    - SR 3.3.2.1.6 (page 3.3-18),

- Footnote (f) of Table 3.3.2.1-1 (page 3.3-19).

- Reference to "30% RTP" has been scaled down to "29.5% RTP" in the following TS:
    - 3.3.1.1 Reactor Protection (RPS) Instrumentation REQUIRED ACTION E.1 (page 3.3-2), and related TS SR 3.3.1.1.14 (page 3.3-5);
    - Table 3.3.1.1-1, FUNCTION 8 and 9, APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS (page 3.3-8).

- TS Table 3.3.1.1-1, Average Power Range Monitors ALLOWABLE VALUE of FUNCTION 2.b, Neutron Flux-High (Flow Biased), page 3.3-6, referenced by LCO 3.4.1c (Recirculation Loops Operating), is revised from "$\leq$ 0.66 W + 71.5% RTP [(b)]" to "$\leq$ 0.75 W + 62.0% RTP [(b)]." Footnote (b) is revised from "0.66 W + 71.5% - 0.66" to "0.75W + 62.0% - 0.75."

- ALLOWABLE VALUE on page 3.3-51 for TS Table 3.3.6.1-1, FUNCTION 1.c., Main Steam Line Flow – High, is revised from "$\leq$ 144% rated steam flow" to "$\leq$ 142.7% rated steam flow."

- Validity of the TS Pressure/Temperature Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3, pages 3.4-23, 3.4-24, and 3.4-25, respectively, will be revised from "30 EFPY" to "28 EFPY."

<u>TS BASES, TRM and TRM BASES – for information only</u>

- Reference to "30% RTP" has been scaled down to "29.5% RTP" in the following TS BASES:
    - B 3.3.1.1 APPLICABLE SAFETY ANALYSES (four places on page B 3.3-18, two places on page B 3.3-19),
    - <u>SR 3.3.1.1.14</u> (three places on page B 3.3-31),
    - <u>SR 3.7.7.2</u> (page B 3.7-31).

- Reference to "10% RTP" has been scaled down to "9.85% RTP" in the following TS BASES:
    - B 3.1.3 <u>D.1 and D.2</u> (two places) (page B 3.1-18),
        - <u>E.1</u> (page B 3.1-19);
    - B 3.1.6 BACKGROUND (page B 3.1-34),
        - APPLICABLE SAFETY ANALYSES (page B 3.1-35),
        - APPLICABILITY (two places) and ACTIONS (page B 3.1-36),
        - <u>SR 3.1.6.1</u> (page B 3.1-37);
    - B 3.3.2.1 BACKGROUND (page B 3.3-43),
        - APPLICABLE SAFETY ANALYSES (two places) (page B 3.3-46),
        - <u>SR 3.3.2.1.2 and SR 3.3.2.1.3</u> (pages B3.3-50 and B 3.3-51),
        - <u>SR 3.3.2.1.6</u> (page B 3.3-52);
    - B 3.10.7 APPLICABILITY (two places) (page B 3.10-31).

- In TRM Table T 3.3.1-1, page 3.3-6, ALLOWABLE VALUES for the APRM FUNCTION 3.a. Upscale (Flow Biased) are revised from "≤ (0.66W + 60.5% - 0.66 ΔW)" to "≤ (0.75W + 51.0% - 0.75 ΔW)." Footnote (f) to this table is revised to reflect the increase in rated power from 2381 MWt to 2419 MWt.

- A new TRM T 3.3.5, and associated BASES B 3.3.5, "Feedwater Flow Instrumentation," has been created to address the new LEFM CheckPlus instrumentation and actions required when the system is out of service.

These changes recognize the impact on RTP of installing higher accuracy feedwater flow instrumentation, and incorporate adjustments required by the associated setpoint and plant analyses.

3.0 BACKGROUND

On June 1, 2000, a revision to 10 CFR 50, Appendix K was issued effective on July 31, 2000. The stated objective of this rulemaking was to reduce an unnecessarily burdensome regulatory requirement. Appendix K was originally issued to ensure an adequate performance margin of the Emergency Core Cooling System (ECCS) in the event a design-basis Loss of Coolant Accident (LOCA) was to occur. The margin is provided by conservative features and requirements of the evaluation models and by the ECCS performance criteria. The original regulation did not require that the power measurement uncertainty be demonstrated, but rather mandated a 2% margin. The new rule allows licensees to justify a smaller margin for power measurement uncertainty. Because there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, the final rule by itself did not allow increases in licensed power levels. Because the licensed power level for a plant is a TS limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. This LAR includes a justification of the reduced power measurement uncertainty and the basis for the modified ECCS analysis. These items are addressed in Enclosure 1.

CNS was originally licensed to operate at a maximum power level of 2381 MWt, which includes a 2% margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR 50, Appendix K. This appendix has since been revised as described above to permit licensees to use an assumed power level less than 1.02 times the licensed power level provided the new power level is demonstrated to account for uncertainties due to power level instrument error.

CNS will install a Caldon LEFM CheckPlus System for feedwater flow measurement. This will be in addition to the venturi-based feedwater flow measurement system CNS currently uses to obtain the daily calorimetric heat balance measurements. Use of the LEFM CheckPlus System will reduce the calorimetric core power measurement uncertainty to ≤ ± 0.31%. Based on this, CNS is proposing to reduce power measurement uncertainty, while meeting the requirements of

10 CFR 50, Appendix K, to permit an increase of 1.62% in licensed power level. As discussed below, reduction in power measurement uncertainty does not constitute a significant change to the Emergency Core Cooling system (ECCS) evaluation model as defined in 10 CFR 50.46(a)(3)(i).

Uncertainty in feedwater flow measurement is the most significant contributor to core power measurement uncertainty. Use of the LEFM CheckPlus System provides a more accurate measurement of feedwater flow that supplements accuracy of the venturi-based instrumentation originally installed at CNS. Caldon Engineering Report ER-80P, as supplemented by Caldon Engineering Report ER-157P, References 7.1 and 7.2, documents the theory, design, and operating features of the system and its ability to achieve increased accuracy of flow measurement. In a Safety Evaluation (SE) dated March 8, 1999 (Reference 7.3), the NRC approved ER-80P for referencing in license applications for power uprate. ER-157P, which supplements ER-80P, was provided for NRC review on July 6, 2001 by Entergy (letter number CNRO-200100029). On December 20, 2001, the NRC issued a SE approving ER-157P (Reference 7.4). The NRC again addressed and approved Caldon's ER-80P and ER-157P on July 5, 2006 (Reference 7.9). Additional details regarding the LEFM CheckPlus System and its application at CNS are provided in the following discussion.

4.0 TECHNICAL ANALYSIS

CNS is presently licensed for a RTP limit of 2381 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase licensed core power level by 1.62% to 2419 MWt. NPPD has evaluated the impact of the proposed core power uprate on nuclear steam supply systems (NSSS), balance of plant (BOP) systems, and safety analyses. The results of NPPD's evaluation are summarized in Enclosure 1 of this submittal, and as discussed herein. The results of all analyses and evaluations performed demonstrate that acceptance criteria will continue to be met.

4.1 General Approach For Plant Analyses Using Plant Power Level

Rated thermal power is used as an input to most plant safety, component, and system analyses. Analyses for which a 2% increase was applied to the initial power level to account solely for the power measurement uncertainty do not need to be re-performed for the 1.62% uprate conditions. This is based on the fact that the sum of increased core power level (1.62%) and the decreased power measurement uncertainty ($\leq \pm$ 0.31%) fall within the previously analyzed conditions.

The power calorimetric uncertainty calculation described in Section 4.2.5 below indicates that with the Caldon LEFM CheckPlus System installed, the power measurement uncertainty (based on a 95-percent probability at a 95-percent confidence level) is $\leq \pm 0.31\%$. Thus, these analyses only need to reflect a 0.31% power measurement uncertainty. Accordingly, the existing 2.00% uncertainty can be allocated such that 1.62% is applied to provide sufficient margin to address the uprate to 2419 MWt, and 0.07% is retained in the analysis to still account for the power measurement uncertainty.

Various core and fuel performance analyses described in Enclosure 1 are reanalyzed or reevaluated on a cycle-specific basis. Other analyses performed at a nominal power level have either been evaluated or re-performed for the 1.62% increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.62% uprate conditions.

4.2 LEFM Ultrasonic Flow Measurement

The LEFM CheckPlus System is based upon ultrasonic transit time principles to determine fluid velocity. This flow measurement method yields highly accurate flow readings and has been approved by the NRC for power uprate applications as documented in Caldon Topical Reports ER-80P, Rev. 0, and ER-157P, Rev. 5 (References 7.1 through 7.4).

This instrumentation is not safety-related. It is, however, designed and manufactured in accordance with Caldon's 10 CFR 50 Appendix B Quality Assurance Program, and their Verification and Validation (V&V) Program. The V&V Program fulfills the requirements of ANSI/IEEE-ANS 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," and ASME NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with the guidance for software V&V in EPRI TR-103291s, "Handbook for Verification and Validation of Digital Systems, December 1994." Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM Check System are provided in Topical Report ER-80P, Section 6.4, and Table 6.1. ER-157P (Reference 7.2), Section 2, supplements ER-80P to address the LEFM CheckPlus System, which is essentially two Check Systems that feed one electronics cabinet.

4.2.1 Use Of LEFM CheckPlus System To Determine Calorimetric Power

The LEFM CheckPlus System measures transit times of pulses of ultrasonic energy traveling multiple acoustic paths, both with the flow and against it, which form two orthogonal measurement planes. From these measurements, the system forms multiple path length fluid velocity products, which are numerically integrated to determine volumetric flow. The system also measures sound velocity along the acoustic paths which, along with feedwater pressure inputs, are used to determine fluid temperature and density. The LEFM CheckPlus System then calculates mass flow, and transmits the signals to the Plant Computer for use in thermal power calculations and system monitoring. This power determination will be used directly to calibrate the plant's nuclear power instruments.

The LEFM CheckPlus System which CNS will install is an improvement on the Caldon LEFM Check System. The earlier LEFM Check System had eight transducers mounted at both ends of four measurement paths arranged at different chord lengths across a single plane. The allowance of 0.6% in total power measurement uncertainty when using the LEFM Check System was derived by Caldon in ER-80P. NRC approval of ER-80P to support a 1.0% power uprate was received March 8, 1999 (Reference 7.3). Supplement ER-160P was later issued by Caldon to support a power uprate of 1.4% when using the LEFM Check System. ER-160P was previously reviewed and approved by the NRC in connection with a similar LAR submitted for the Watts

Bar Nuclear plant. The NRC staff approved the report in its January 19, 2001 SE for Watts Bar (ADAMS accession number ML010260074).

The Caldon LEFM CheckPlus System is similar to the LEFM Check System, except that it has 16 transducers on eight acoustic measurement paths grouped into two orthogonal planes with four measurement paths in each plane. The LEFM CheckPlus System essentially combines two LEFM Check Systems. In order to ensure independence, each measurement plane employs its own timing clock in the LEFM CheckPlus System. The LEFM CheckPlus System provides feedwater flow measurement that is more accurate than that provided by a LEFM Check System. It will support a power uprate of up to 1.7%. Superiority in measurement accuracy arises from two distinct advantages in the LEFM CheckPlus System, both of which are described in Caldon Report ER-157P (Reference 7.2). The NRC staff approved the report in its December 20, 2001 SE (Reference 7.4). These advantages are:

- Because of the orthogonal geometry of the two measurement planes, any transverse components of the fluid velocity will be cancelled out when the two companion measurements in each plane are averaged. The average of two numerical integrations of four pairs of axial velocity measurements in orthogonal planes is inherently more accurate than the integration of four measurements in a single plane.

- Because there are twice as many measurements being taken, the total statistical error due to uncertainties in both transit time measurements and path length geometry is reduced. This advantage arises due to the statistical treatment of the uncertainties, the mathematics of which is supported by ANSI/ASME Power Test Code PTC 19.1-1985.

The individual contributions to mass flow measurement uncertainty by the two Caldon systems are tabulated for comparison in Table 1 of ER-157P. This table identifies the differences between the uncertainties associated with the two LEFM systems and provides an association with the two advantages of the LEFM CheckPlus System listed above. This table shows that the accuracy of the LEFM CheckPlus System exceeds the accuracy of the LEFM Check System.

The LEFM CheckPlus System at CNS will consist of a flow element to be installed in each of the two feedwater inlet lines just downstream of the mixing pipe in the Feedwater Pump Room, and an electronics cabinet installed in the Turbine Building basement. The installation of each of the flow elements will conform to the requirements in Caldon Topical Reports ER-80P and ER-157P. The system will utilize continuous calorimetric power determination by direct serial link with the plant computer, and will incorporate self-verification features. These features ensure that performance is consistent with the design basis.
Caldon derived calibration data for each of the LEFM CheckPlus System spool pieces using a site-specific model test at Alden Research Laboratories with calibration standards traceable to National Standards. A copy of the Alden Labs certified calibration report is included in the Caldon Design Basis Uncertainty Analysis for the system. The LEFM CheckPlus System will be installed and commissioned according to Caldon procedures in conformance with ER-80P and

ER-157P, including verification of ultrasonic signal quality and hydraulic velocity profiles, as compared to those tested during site-specific model testing.

4.2.2 LEFM Inoperability

The redundancy inherent in the two measurement planes of an LEFM CheckPlus System makes the system tolerant to component failures. The system features automatic self-checking. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy envelope. The on-line monitoring and diagnostics tests include the acoustic processing unit transmitters, timing circuits, signal quality, path sound velocity, hydraulic profile as represented by path velocities, and active computation as reported by watchdog timers. The system provides display and storage of verification test results. Failure messages are generated and monitored in the control room, if system failure events are detected.

A process will be implemented to use the LEFM CheckPlus System feedwater mass flow and temperature to adjust or calibrate the existing feedwater flow nozzle-based signals. If the LEFM CheckPlus System or a portion of the system becomes inoperable, control room operators are promptly alerted by control room computer indications. Feedwater flow input to the core thermal power calculation would then be provided by the existing flow nozzles, or a combination of flow nozzle(s) and LEFM flow data. Calculations have been performed to support the uncertainty of LEFM and flow nozzle inputs to the core thermal power calculation. In addition, since the flow nozzles are calibrated to the last validated good data from the LEFM CheckPlus System, it will be acceptable to remain at 2419 MWt for up to 72 hours to enact LEFM system repairs. The TRM will be revised prior to implementation of the uprated power to include CheckPlus System out-of-service administrative controls.

4.2.3 Maintenance And Calibration

Calibration and maintenance of the LEFM CheckPlus System will be performed using site procedures developed from the Caldon LEFM CheckPlus System technical manuals. Ultrasonic signal verification and alignment is performed automatically with the LEFM CheckPlus System. Signal verification is possible by review of signal quality measurements performed and displayed by the LEFM CheckPlus System. Routine preventive maintenance procedures include physical inspections, power supply checks, back-up battery replacements, and internal oscillator frequency verification.

Work on the CNS LEFM CheckPlus System will be performed by site I&C personnel qualified per the CNS I&C Training Program, and who will have been formally trained on the CheckPlus System by Caldon. Work will be performed in accordance with site work control procedures.

The CNS LEFM CheckPlus is under Caldon's V&V Program, and procedures are maintained for user notification of important deficiencies.

4.2.4 Procedures, Training And Simulator

Procedures governing normal operation, emergency operation, and off-normal operation, as well as equipment changes that may be affected by the power uprate, will be identified in the design change process and revised prior to implementation of uprated power. Appropriate personnel will receive training on the Caldon LEFM CheckPlus System, as well as on the affected procedures. This training consists of briefings, required reading, classroom sessions, and a simulator demonstration, as needed, and will be conducted prior to operation at the uprated power. Simulator changes and validation for the power uprate will be performed in accordance with ANSI/ANS 3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," prior to implementation of the requested license amendment.

4.2.5 Uncertainty Determination Methodology

NPPD has completed the thermal power uncertainty calculation for CNS, indicating an uncertainty of ≤ 0.31% of rated thermal power for the site-specific installation (Enclosure 7).

The calculations are consistent with the methodology described in Topical Report ER-80P (Reference 7.1), as supplemented by Engineering Report ER-157P (Reference 7.2). The uncertainty calculation supports an overall uncertainty in the reactor power measurement of 0.31%. The uncertainty is at a 95% probability and 95% confidence level. Enclosure 7 provides a discussion for uncertainty in the CNS heat balance using the LEFM CheckPlus System.

LEFM CheckPlus System operating procedures will ensure the assumptions and requirements of the uncertainty calculation remain valid.

4.2.6 Monitoring, Verification and Error Reporting

As discussed in 4.2, above, the LEFM CheckPlus System for this application is non-safety-related. However, the system is designed and manufactured under Caldon's standard quality control program which provides for configuration control, deficiency reporting and correction, and maintenance. System software and laboratory calibration tests are required to meet the requirements of 10 CFR 50, Appendix B.

At CNS the LEFM CheckPlus System will be included in the preventive maintenance program, and the CNS Quality Assurance Program. Conditions that are adverse to quality are documented under the Corrective Action Program. The software falls under the CNS Software Quality Assurance Program. Vendor notifications are controlled in the CNS Operating Experience Program. Those vendor notifications considered applicable are entered into the Corrective Action Program for disposition. The equipment manuals are also included in the CNS vendor manual program.

CNS operating procedures will be revised to ensure that the plant does not intentionally exceed the proposed RTP of 2419 MWt. CNS will continue to maintain current shift power average and power excursion guidelines to maintain RTP within the licensed steady state RTP. This approach is consistent with existing operating procedures.

4.2.7 Hydraulic Modeling

The LEFM CheckPlus System spool pieces were calibrated at Alden Research Laboratory (ARL). This testing included a full-scale model of the CNS hydraulic geometry and tests in straight pipe. The calibration factor for the CNS spool pieces is based on these tests and documented in a Caldon Engineering Report, Enclosure 5. A review of the observed profiles for the various pipe models at ARL and the observed profiles at CNS will be conducted as part of the final commissioning by Caldon, Inc.. During power ascension following refueling outage 24 (RE24), final acceptance of the CNS uncertainty analysis will occur upon completion of the commissioning process and verification of bounding calibration test data. This step will provide final positive confirmation that actual performance in the field meets the uncertainty assumptions for the instrumentation, and that it is consistent with the assumptions of Engineering Reports ER-80P and ER-157P.

4.2.8 RIS 2002-03, Item I.1.D and ER-157P Criteria

In approving ER-80P and ER-157P, the NRC established four criteria to be addressed by each licensee. The four criteria of RIS 2002-03 Item I.1.D, Reference 7.7, and a discussion of how each will be satisfied follows:

Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

Response to Criterion 1 (See also 4.2.2, above)

Implementation of the measurement uncertainty recapture power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM system. As stated in 4.2.3 above, plant maintenance and calibration procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the LEFM CheckPlus System OPERABLE and raising power above 2381 MWt. The incorporation of, and continued adherence to, these requirements will assure that the LEFM CheckPlus System is properly maintained and calibrated.

The proposed allowed outage time for operation at the Thermal Power Optimization (TPO) power level with a LEFM out of service is 72 hours, provided steady state conditions persist during the 72 hours (no power changes in excess of 10% during the period). There are four bases for this proposed time period:

- There is an on-line calibration of a set of alternate plant instruments to be used if the LEFM is out of service for a longer period. These alternate instruments will be calibrated to the last good value provided by the LEFM, and their accuracy will gradually degrade over time associated with nozzle fouling and transmitter drift. Provided steady state conditions persist, the gradual accuracy degradation is likely to be imperceptible for a 72-hour period.

- 72 hours gives plant personnel time to make repairs and to verify normal operation of the LEFM CheckPlus System while it is within its original uncertainty bounds, and while at the same power level and indications as before the failure.

- The plant will be operated based on the calibrated alternate plant instruments when the LEFM CheckPlus System is not available. It is considered prudent to provide time to become accustomed to operation with the alternate plant instruments prior to requiring a reduction in power. A reduction in power could in many cases be avoided altogether, since a repair would be accomplished prior to the expiration of the 72-hour period.

- Since a plant transient may result in calibration changes of the alternate instruments, if the plant experiences a power change of greater than 10% during the 72 hour period, then the maximum permitted power level would be 2381 MWt (see 4.2.2).

Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response to Criterion 2

This Criterion is not applicable to CNS. CNS currently uses a venturi-based feedwater flow measurement system to obtain the daily calorimetric heat balance measurements. CNS is installing a new LEFM CheckPlus System as the basis for the requested uprate. It will be installed during RE24.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

NPPD uses a core thermal power uncertainty calculation approach consistent with ASME PTC-19.1 (1985), Measurement Uncertainty; ISA 67.04.02-2000, Methodologies for the Determination of Set Points for Nuclear Safety-Related Instrumentation; and Caldon's Topical

Report ER-80P, as supplemented by ER-157P. The combination of errors within instrument loops is accomplished in accordance with plant and NRC-approved GE Setpoint Methodology as described in Reference 7.6.

Criterion 4

For plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

Criterion 4 does not apply to CNS. The calibration factor for the CNS spool pieces was established by tests of these spools at Alden Research Laboratory in August of 2007. These included tests of a full-scale model of the CNS hydraulic geometry and tests in a straight pipe. An Alden data report for these tests and a Caldon engineering report (Enclosure 5) evaluating the test data are on file. The calibration factor used for the LEFM CheckPlus System at CNS is based on these reports. The uncertainty in the calibration factor for the spools is based on the Caldon engineering report. The site-specific uncertainty analysis, Enclosure 7, documents these analyses.

Final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process which is expected to be completed in May of 2008. The commissioning process will verify bounding calibration test data (See Appendix F of Reference 7.1). This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation.

4.2.9 Total Power Measurement Uncertainty

Refer to Table I of Enclosure 4, and the tables in Section 3.10 of Enclosure 7 for a summary of the core thermal power measurement uncertainty at CNS. These tables detail plant-specific calculations that identify parameters and their individual contributions to power uncertainty.

4.2.10 Startup Testing

Core power from the Average Power Range Monitors (APRMs) will be rescaled to the uprated power level prior to exceeding the CLTP level. Any necessary adjustments of the APRM alarm and trip settings will be made.

Demonstration of an acceptable fuel thermal margin will be performed prior to and during power ascension at each of the following steady-state heat balance power levels: 95% and 100% of the CLTP level, and 100% of the uprated power level. Fuel thermal margin will be projected to the uprated RTP point after the measurements at 95% and 100% of CLTP level are taken and the estimated margin is determined. The demonstration of core and fuel conditions will be performed using current CNS methods.

In preparation for operation at the uprated power level, routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95% and 100% of the CLTP level and at 100% of the uprated power level.

The operational aspect of the uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing during power ascension testing. Reactor pressure control system testing, consistent with the guidelines of NEDC-33385P, Safety Analysis Report for Cooper Nuclear Station THERMAL POWER OPTIMIZATION, Enclosure 1, will be performed during power ascension testing. During these tests, a water level change of ± 3 inches and pressure setpoint change of ± 3 psi will be used. If necessary, the controllers and actuator elements will be adjusted.

- The performance of the feedwater level control system will be recorded at 95% and 100% of the CLTP level, and confirmed at the uprated power level during power ascension.

- The turbine pressure controller setpoint will be readjusted at 95% and 100% CLTP level and held constant. Adjusting the pressure setpoint prior to recording the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor, and the turbine control valves.

At TPO test conditions, samples will be taken and measurements made to determine that gaseous release and the chemical and radiochemical quality of reactor water and feedwater remain within acceptable limits.

Radiation conditions will be monitored at TPO conditions to ensure that personnel exposures are maintained as low as reasonably achievable, radiation survey maps are accurate, and radiation zones are properly posted.

A Startup Test Report will be submitted within 90 days following resumption of power operation following RE24.

4.2.11 Adverse Flow Effects

CNS is committed to examining the Steam Dryer in accordance with a Boiling Water Reactor Vessel Internals Project (BWRVIP-139), Reference 7.8. BWRVIP is reviewing plant experience reports to determine what a reasonable re-inspection frequency should be. Currently CNS has instituted a 4 cycle re-inspection frequency in the CNS Vessel Internals Project Program, making the next full BWRVIP inspection occur in RE27.

Regulatory Guide 1.20, Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing, describes a methodology that the NRC staff considers acceptable for use in a vibration assessment program. Even though not required by this guideline, prior to exceeding CLTP and ascension to TPO, CNS will ensure compliance with the methodology contained in Reg. Guide 1.20.

4.2.12 Miscellaneous

The feedwater flow setpoint which limits recirculation pump speed to 22% (22% Speed Limiter) is used to prevent recirculation cavitation in the lower part of the power-to-flow map at low feedwater temperatures. It is expressed in percent of rated flow (20% feedwater flow). Although the setpoint in absolute flow remains the same, it will appear lower due to the increase in TPO feedwater flow. The boundary of the lower part of the power-to-flow map, "Do Not Operate Region," is established to prevent recirculation and jet pump cavitation. The region is based on the absolute values of flow and temperature and independent of power. The boundary will move downward due to the re-scaled power-to-flow map.

The low power setpoint in which rod patterns are enforced by the Rod Worth Minimizer will remain the same in terms of MWt, but will be lowered with respect to percent power.

5.0 REGULATORY SAFETY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

Nebraska Public Power District (NPPD) is proposing that the Cooper Nuclear Station (CNS) Operating License be amended to reflect an increase in the licensed reactor power level from 2381 megawatts thermal (MWt) to 2419 MWt. These changes result from increased accuracy of the feedwater flow measurements to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation, the Caldon Leading Edge Flow Meter CheckPlus System. The basis for this change is consistent with the July 31, 2000 revision to 10 CFR 50 Appendix K, which was made to allow operating reactor licensees to use an uncertainty factor of ≤ 2% of rated reactor thermal power in analyses of postulated design basis loss-of-coolant accidents.

NPPD has evaluated the proposed changes at CNS and has determined that applicable regulations and requirements continue to be met. The spectrum of hypothetical accidents and transients has been investigated and were shown to meet the plant's currently licensed regulatory criteria. Emergency Core Cooling System performance was evaluated at the proposed power uprate and was shown to meet the criteria of 10 CFR 50.46 and 10 CFR 50 Appendix K. Challenges to the containment under postulated accident conditions have been evaluated, and the containment and its associated cooling systems continue to meet the intent of 10 CFR 50 Appendix A, Criterion 38, Long Term Cooling, and Criterion 50, Containment.

NPPD has determined that the proposed changes do not require any exemptions or relief from any regulatory requirements, other than the Technical Specifications (see Attachments 2 and 3), and do not affect conformance with any General Design Criteria as currently described in the Updated Safety Analysis Report (USAR).

5.2 No Significant Hazards Consideration

NPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, Issuance of Amendment, as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
   Response: No.

   The comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of components and systems that could be affected by this change. Evaluation of accident analyses confirmed the effects of the proposed uprate are bounded by the current dose analyses. All systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable.

   The primary loop components (reactor vessel, reactor internals, control rod drive housings, piping and supports, recirculation pumps, etc.) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

   All of the Nuclear Steam Supply Systems (NSSS) will still perform their intended design functions during normal and accident conditions. The balance of plant (BOP) systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level.

   Because the integrity of the plant will not be affected by operation at the uprated condition, NPPD has concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the core thermal power uncertainty measurement allows a majority of the current safety analyses to be used, with small changes to the core operating limits, to support operation at a core power of 2419 MWt. Other analyses performed at a nominal power level have either been evaluated or re-performed for the 1.62% increased power level. The results demonstrate that acceptance criteria of the applicable analyses continues to be met at the 1.62% uprate conditions. As such, all CNS USAR Chapter 14 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 1.62% uprated condition.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier, and from the standpoint of compliance with the required acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed or approved by the Nuclear Regulatory Commission, or that are in compliance with regulatory review guidance and standards. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NPPD concludes that the proposed amendment presents no significant hazards consideration under the three standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment will not:

(i) involve a significant hazards consideration,

(ii) involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or

(iii) involve a significant increase in individual or cumulative occupational radiation exposure.

NPPD has determined the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, no environmental impact statement or environmental assessment is required in connection with the proposed amendment. The basis for this determination is as follows:

(i) Section 5.2, above, provides the justification as to why this proposed change to the licensed power level does not involve a significant hazards consideration.

(ii) The proposed increase to the new maximum licensed power level of 2419 MWt has been reviewed with respect to offsite releases, with the conclusion that 10 CFR 20 limits will continue to be met, and that the processing of liquid and gaseous radwaste limits will not be adversely affected. Consequently, the proposed change does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

(iii) Operating CNS at the new maximum licensed power level of 2419 MWt will not result in a significant increase to typical occupational exposures. Therefore, a significant increase in individual or cumulative occupational radiation exposure will not occur.

6.0 PRECEDENCE

Similar amendment requests have been approved for:

FACILITY	AMMENDMENT(S)	APPROVAL DATE	ACCESSION #
Grand Gulf	156	10/10/2002	ML022890295
River Bend	129	1/31/2003	ML030350194
Hatch	238, 180	9/23/2003	ML032691360
Peach Bottom	247, 250	11/22/2002	ML031010365
Seabrook (PWR)	110	5/22/2006	ML061360034

NPPD chose the first four plants to review based on similarity of equipment installed, and as suggested on the NRC web page, Information for Boiling-Water Reactor Measurement Uncertainty Recapture Power Uprates. NPPD chose the Seabrook LAR because of the clear way it was organized, and because it was a recent submittal.

7.0 REFERENCES

7.1 Caldon, Inc., Engineering Report - 80P, "TOPICAL REPORT Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

7.2 Caldon Inc., Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System," Revision 5, October 2001.

7.3 Letter from US NRC to C. L. Terry (Texas Utilities Electric), "Comanche Peak Steam Electric Station, Units 1 & 2 -Review of Caldon Engineering Topical Report ER 80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM√™ System," March 8, 1999.

7.4 Letter from US NRC to M. A. Krupa (Entergy Operations, Inc.), 'Waterford Steam Electric Station, Unit 3; River Bend Station and Grand Gulf Nuclear Station -Review of Caldon, Inc., Engineering Report ER-157P," December 20, 2001 (ML01350256).

7.5 GE-Hitachi Nuclear Energy NEDC-32938P-A, Revision 2, Class III (Proprietary), "Licensing Topical Report Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," May 2003.

7.6 NEDC-31336P, "General Electric Instrument Setpoint Methodology" September 1996.

7.7 Nuclear Regulatory Commission Regulatory Issue Summary, 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

7.8 BWRVIP-139: BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines (EPRI Proprietary Final Report, April 2005)

7.9 Letter from US NRC to Ernest M. Hauser, President, Caldon, Inc., "Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters (UFMs) (TAC No. MC6424)," July 5, 2006 (ML061700222)

ATTACHMENT 2

OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES - MARKED UP WITH PROPOSED CHANGES

(TRM PAGES INCLUDED FOR INFORMATION)

APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE COOPER NUCLEAR STATION DOCKET 50-298, DPR-46

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility. /018 /018 /018

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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 or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2381~~ 2419 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through /107
Amendment No. ~~227~~; are hereby incorporated in the license. The /~~227~~
licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are
entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter /*
dated May 17, 2006. /*

(4) <u>Fire Protection</u> /199,**

The licensee shall implement and maintain in effect all provisions of the /199,**
approved fire protection program as described in the Cooper Nuclear Station /199,**
(CNS) Updated Safety Analysis Report and as approved in the Safety /199,**
Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; /199,**
April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, /199,**
1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, /199,**
1989; August 15, 1995; and July 31, 1998, subject to the following provision: /199,**
/199,**

The licensee may make changes to the approved fire protection program /199,**
without prior approval of the Commission only if those changes would not /199,**
adversely affect the ability to achieve and maintain safe shutdown in the /199,**
event of a fire. /199,**

*Revised by letter dated March 5, 2007
** Revised by letter dated August 9, 2007

1.1 Definitions

LOGIC SYSTEM FUNCTIONAL TEST (continued)	from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2381~~ 2419 MWt. 1
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time segment from the time the sensor contacts actuate to the time the scram solenoid valves deenergize.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free;

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. ---------NOTE--------- Not applicable when THERMAL POWER > ~~10~~% RTP. 9.85 ---------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1	Restore compliance with BPWS.	4 hours
	<u>OR</u>		
	D.2	Restore control rod to OPERABLE status.	4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> Nine or more control rods inoperable.	E.1	Be in MODE 3.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

9.85

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ ~~10~~% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1 --------NOTE--------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ---------------------	
	Move associated control rod(s) to correct position.	8 hours
	<u>OR</u>	
	A.2 Declare associated control rod(s) inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~30%~~ 29.5 RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.12	-------------------NOTES------------------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.13	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.14	Verify Turbine Stop Valve — Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~ 29.5% RTP.	18 months
SR 3.3.1.1.15	-------------------NOTE------------------- Neutron detectors are excluded. -- Verify the RPS RESPONSE TIME is within limits.	18 months

1

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux — High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
	5(a)	3	H	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux — High (Startup)	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 14.5% RTP
b. Neutron Flux-High (Flow Biased)	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 0.75 ~~≤ 0.66~~ W ~~+ 71.5%~~ + 62.0 RTP(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~[0.66 W + 71.5% - 0.66 ΔW]~~ 0.75 W + 62.0% − 0.75ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level —High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
b. Level Switch	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
8. Turbine Stop Valve —Closure	≥ ~~30~~% RTP ≥29.5	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed 1
9. Turbine Control Valve Fast Closure, DEH Trip Oil Pressure —Low	≥ ~~30~~% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1018 psig 1
10. Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5(a)	1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.13	NA
	5(a)	1	H	SR 3.3.1.1.9 SR 3.3.1.1.13	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.2	------------NOTE------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ ~~10~~ 9.85% RTP in MODE 2. ---------------------------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------NOTE------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ ~~10~~ 9.85% RTP in MODE 1. ---------------------------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------NOTE------------ Neutron detectors are excluded. ---------------------------- Verify the RBM: a. Low Power Range — Upscale Function is not bypassed when THERMAL POWER is ≥ 27.5% and < 62.5% RTP and a peripheral control rod is not selected. b. Intermediate Power Range — Upscale Function is not bypassed when THERMAL POWER is ≥ 62.5% and < 82.5% RTP and a peripheral control rod is not selected. c. High Power Range — Upscale Function is not bypassed when THERMAL POWER is ≥ 82.5% RTP and a peripheral control rod is not selected.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.5	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is ≤ ~~10~~ 9.85% RTP.	18 months
SR 3.3.2.1.7	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range — Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	(h)
b. Intermediate Power Range — Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	(h)
c. High Power Range — Upscale	(c),(d)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	(h)
d. Inop	(d),(e)	2	SR 3.3.2.1.1	NA
e. Downscale	(d),(e)	2	SR 3.3.2.1.1 SR 3.3.2.1.5	≥ 92/125 divisions of full scale
2. Rod Worth Minimizer	$1^{(f)}, 2^{(f)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8	NA
3. Reactor Mode Switch — Shutdown Position	(g)	2	SR 3.3.2.1.7	NA

(a) THERMAL POWER ≥ 27.5% and < 62.5% RTP and MCPR < 1.70 and no peripheral control rod selected.

(b) THERMAL POWER ≥ 62.5% and < 82.5% RTP and MCPR < 1.70 and no peripheral control rod selected.

(c) THERMAL POWER ≥ 82.5% and < 90% RTP and MCPR < 1.70 and no peripheral control rod selected.

(d) THERMAL POWER ≥ 90% RTP and MCPR < 1.40 and no peripheral control rod selected.

(e) THERMAL POWER ≥ 27.5% and < 90% RTP and MCPR < 1.70 and no peripheral control rod selected.

(f) With THERMAL POWER ≤ ~~10~~ 9.85% RTP.

(g) Reactor mode switch in the shutdown position.

(h) Less than or equal to the Allowable Value specified in the COLR.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low (Level 1)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 835 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ ~~144~~ 142.7% rated steam flow
d. Condenser Vacuum - Low	1, $2^{(a)}$, $3^{(a)}$	2	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 8 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2 per location	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low (Level 3)	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 3 inches
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.84 psig
c. Reactor Building Ventilation Exhaust Plenum Radiation- High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 49 mR/hr
d. Main Steam Line Radiation - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3 times full power background
e. Reactor Vessel Water Level -Low Low Low (Level 1)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches

(continued)

(a) With any turbine stop valve not closed.

Cooper Heatup/Cooldown, Core Not Critical Curve (Curve B), ~~30~~ 28 EFPY

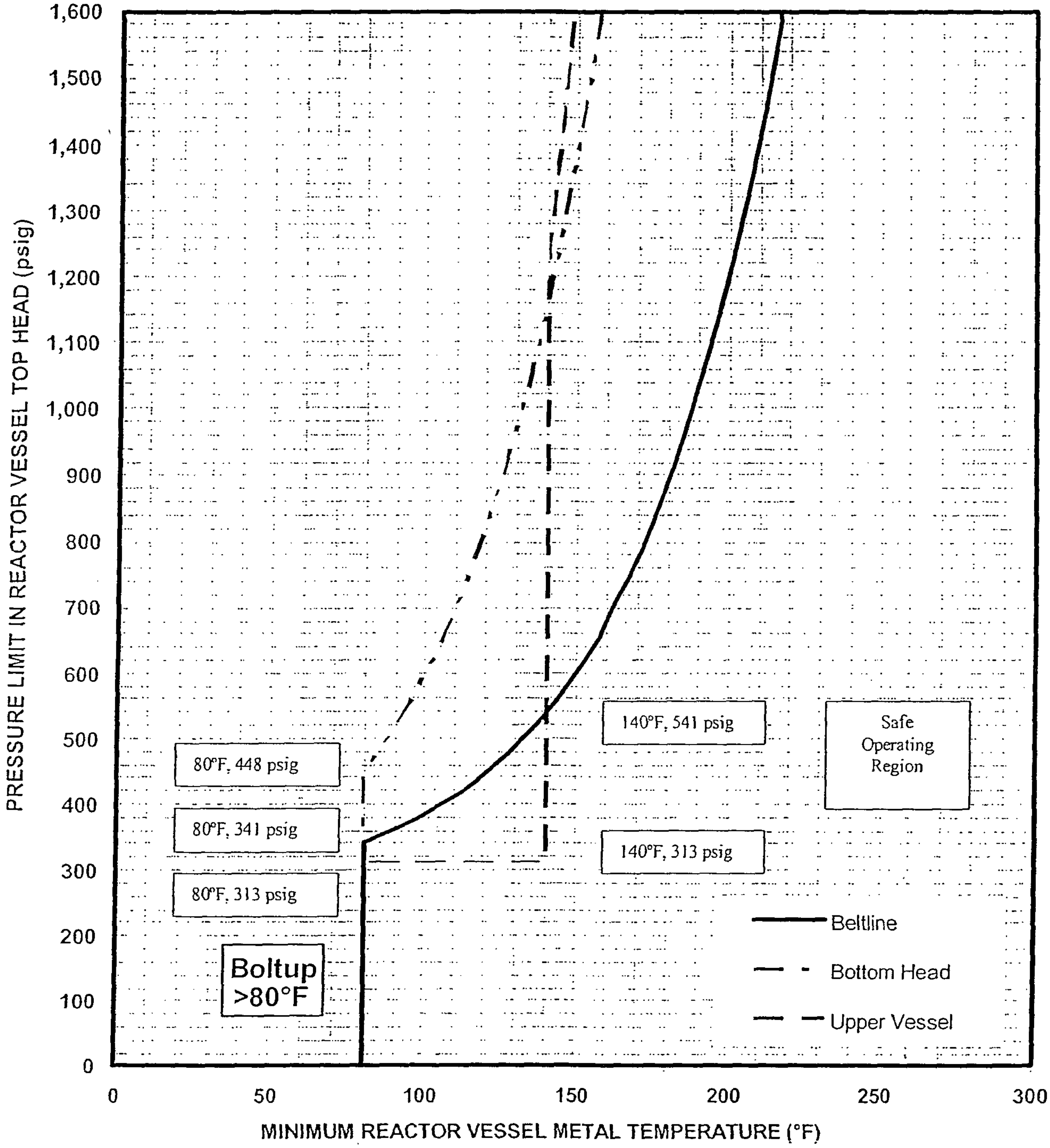

Figure 3.4.9-1 (page 1 of 1)
Pressure/Temperature Limits for
Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown
Valid Through ~~30~~ 28 EFPY

28

Cooper Pressure Test Curve (Curve A), ~~30~~ EFPY

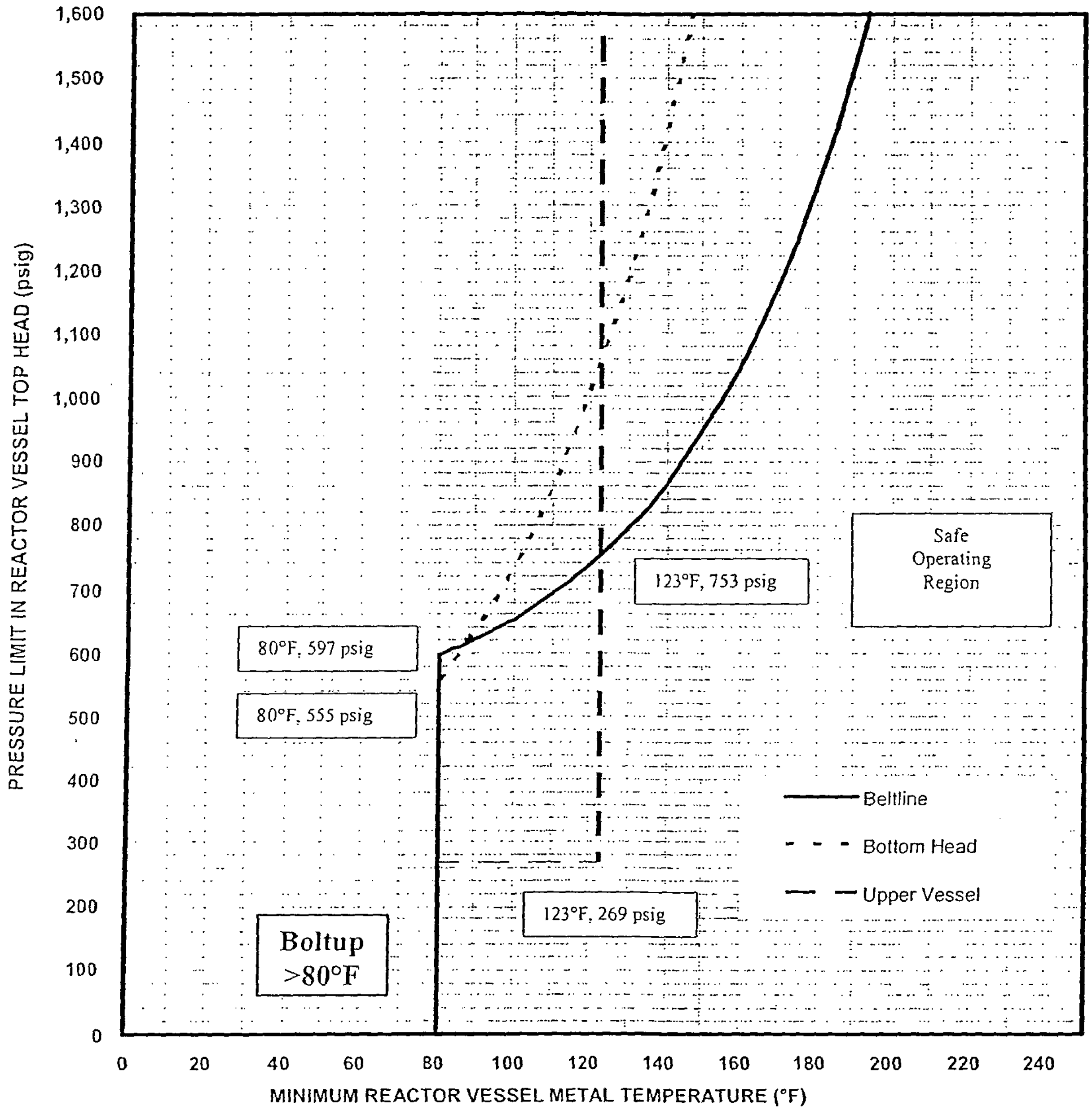

Figure 3.4.9-2 (page 1 of 1)
Pressure/Temperature Limits for
Inservice Hydrostatic and Inservice Leakage Tests
Valid Through ~~30~~ EFPY

28

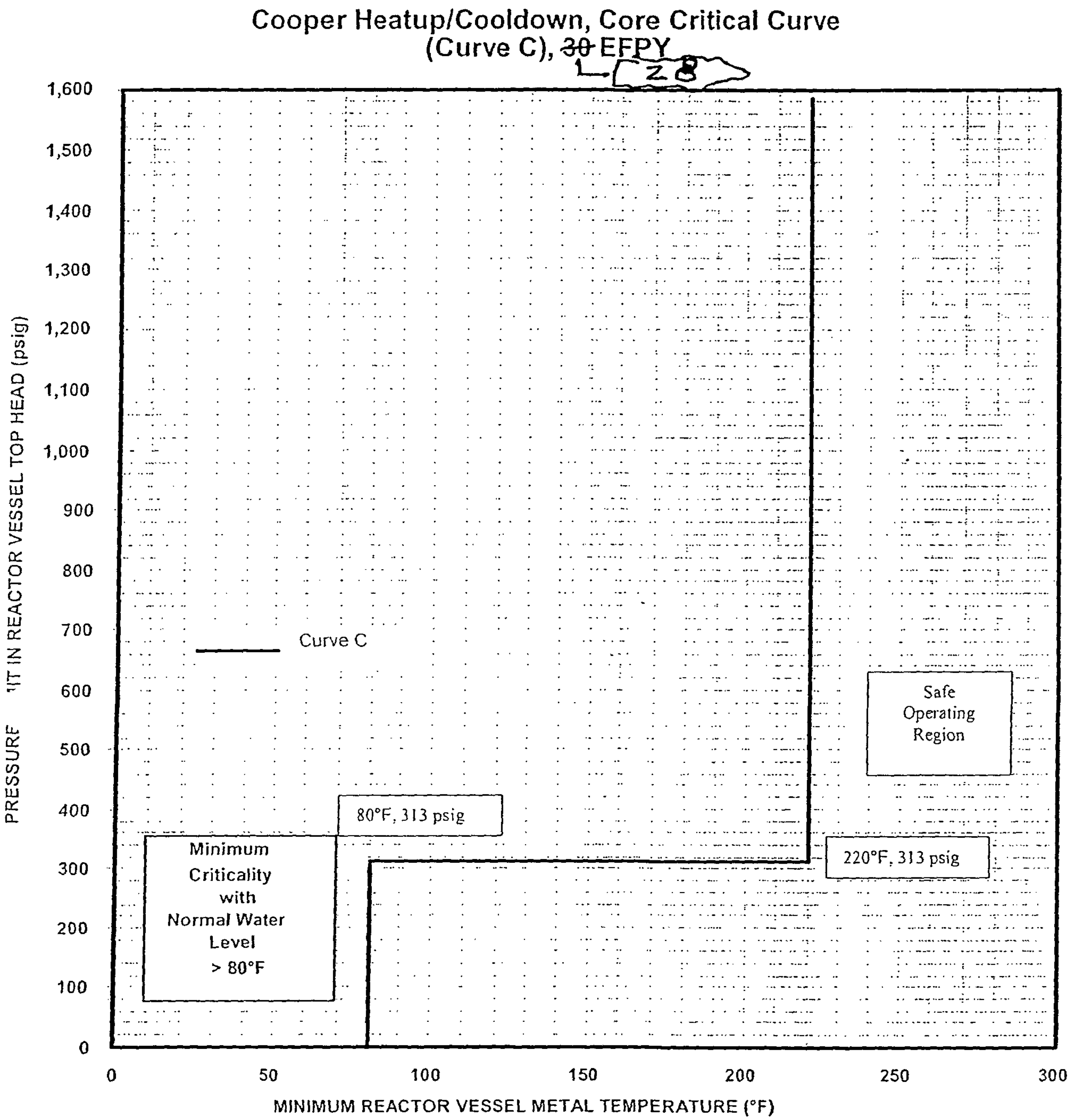

Figure 3.4.9-3 (page 1 of 1)
Pressure/Temperature Limits for Criticality
Valid Through ~~30~~ EFPY
28

BASES

ACTIONS

C.1 and C.2 (continued)

electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ ~~10~~ 9.85% RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 6) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > ~~10~~ 9.85% RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met, or there are nine or more inoperable control rods, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the

(continued)

BASES

ACTIONS

E.1 (continued)

active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above ~~10~~ 9.85% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the Banked Position Withdrawal Sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to ~~10~~ 9.85% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

(continued)

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to ~~10~~ 9.85% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The generic BPWS analysis (Ref. 8) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 10) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled. When using the Reference 10 control rod sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under LCO 3.3.2.1, Condition D controls.

In order to use the Reference 10 BPWS shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no Single Operator Error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are provided in Reference 10. If the requirements for use of the BPWS control rod insertion process contained in Reference 10 are followed, the plant is considered to be in compliance with BPWS requirements, as required by LCO 3.1.6.

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 9).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

BASES (continued)

9.85

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ ~~10~~% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > ~~10~~% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

9.85

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ ~~10~~% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence.

9.85

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod

BASES

ACTIONS

B.1 and B.2 (continued)

worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq$ ~~10~~% RTP.
9.85

REFERENCES

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1 (Revision specified in the COLR).
2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.
3. NUREG-0979, Section 4.2.1.3.2, April 1983.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
5. 10 CFR 100.

BASES

APPLICABLE, SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

29.5 closure will produce a half scram. This Function must be enabled at THERMAL POWER ≥ ~~30~~% RTP as measured by turbine first stage pressure. This is accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

The Turbine Stop Valve-Closure Allowable Value is selected to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Four channels of Turbine Stop Valve-Closure Function, with two channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if both TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~30~~% RTP. This 29.5 Function is not required when THERMAL POWER is < ~~30~~% RTP since the Reactor Vessel Pressure-High and the Average Power Range Monitor Neutron Flux-High (Fixed) Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, DEH Trip Oil Pressure-Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, DEH Trip Oil Pressure-Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 2. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, DEH Trip Oil Pressure-Low signals are initiated by low digital-electrohydraulic control (DEHC) fluid pressure in the emergency trip header for the control valves. There are four pressure switches which sense off the common header, with one pressure switch assigned to each separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ ~~30~~% RTP as 29.5 measured by turbine first stage pressure. This is accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

BASES

APPLICABLE, SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Turbine Control Valve Fast Closure, DEH Trip Oil Pressure-Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, DEH Trip Oil Pressure-Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ ~~30~~ 29.5% RTP. This Function is not required when THERMAL POWER is < ~~30~~ 29.5% RTP, since the Reactor Vessel Pressure-High and the Average Power Range Monitor Neutron Flux-High (Fixed) Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch-Shutdown Position

The Reactor Mode Switch-Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a keylock four-position, four-bank switch. The reactor mode switch will scram the reactor if it is placed in the shutdown position. Scram signals from the reactor mode switch are input into each of the two RPS manual scram logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two channels of Reactor Mode Switch-Shutdown Position Function, with one channel in each manual scram trip system, are available and required to be OPERABLE. The Reactor Mode Switch-Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.14

This SR ensures that scrams initiated from the Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~30~~% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during an in-service calibration at THERMAL POWER ≥ ~~30~~% RTP to ensure that the calibration is valid. 1

29.5

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~30~~% RTP, then the affected Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions are considered inoperable. Open main turbine bypass valve(s) can also affect these two functions. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE. 1 1

The Frequency of 18 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.15

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 12.

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

The 18 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

BASES

BACKGROUND (continued)

RBM averaging but remain in the display and LPRM alarm logic. Assignment of power range detector assemblies to be used in RBM averaging is controlled by the selection of control rods. The minimum number of LPRM inputs required to each RBM channel to prevent an instrument inoperative alarm is four when using eight LPRM assemblies, three when using six LPRM assemblies, and two when using four LPRM assemblies. The RBM is automatically bypassed and the output set to zero if a peripheral control rod is selected since the RBM function is not required for these rods. In addition, any one of the two RBM channels can be manually bypassed. If any LPRM detector assigned to a RBM is bypassed, the computed average signal is adjusted automatically to compensate for the number of LPRM input signals to average. When a control rod is selected, the signal conditioner gain is automatically adjusted so that the output level of the signal conditioner always corresponds to a constant level (relative to the initialization reference signal of 100/125 of full scale). The gain set will be held constant during the movement of that rod, thus providing an indication of the change in the relative local power level. Whenever the reactor power level is below the lowest RBM operating range, the RBM is zeroed and RBM outputs are bypassed. If the indicated power increases above the preset limit, a rod block will occur. In addition, to preclude rod movement with an inoperable RBM, a downscale trip and an inoperative trip are provided. A rod block signal is generated if an RBM downscale trip or an inoperable trip occurs, since this could indicate a problem with the RBM channel. The downscale trip will occur if the RBM channel signal decreases below the downscale trip setpoint after the RBM channel signal has been normalized. The inoperable trip will occur during the nulling (normalization) sequence, if the RBM channel fails to null, too few LPRM inputs are available, a module is not plugged in, or the function switch is moved to any position other than "Operate."

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to ~~10~~% RTP. 9.85
The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 5 and 6. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 7) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 7 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under the controls in Condition D.

The RWM Function satisfies Criterion 3 of Reference 4.

Since the RWM is a system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ ~~10~~% RTP. 9.85
When THERMAL POWER is > ~~10~~% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 5). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the channel will perform the intended function. It includes the Reactor Manual Control System input. It also includes the local alarm lights representing upscale and downscale trips, but no rod block will be produced at this time. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 10).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the system will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the RWM includes performing the RWM computer on line diagnostic test satisfactorily, attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. For SR 3.3.2.1.2, the CHANNEL FUNCTIONAL TEST also includes attempting to select a control rod not in compliance with the prescribed sequence and verifying a selection error occurs. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ ~~10~~ 9.85% RTP in MODE 1. This allows

BASES

SURVEILLANCE REQUIREMENTS (continued)

entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ ~~10~~ 9.85% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 10).

SR 3.3.2.1.4

The RBM power range setpoints control the enforcement of the appropriate upscale trips over the proper core thermal power range of the Applicability Notes (a), (b), (c), (d), and (e) of ITS Table 3.3.2.1-1. The RBM Upscale Trip Function setpoints are automatically varied as a function of power. Three Allowable Values are specified in the COLR as denoted in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the reference APRM signal's input to each RBM channel. Below the minimum power setpoint of 27.5% RTP or when a peripheral control rod is selected, the RBM is automatically bypassed. These power Allowable Values must be verified periodically by determining that the power level setpoints are less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 184 day Frequency is based on the actual trip setpoint methodology utilized for these channels.

SR 3.3.2.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8.

The Frequency is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.6

9.85

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The setpoint where the automatic bypass feature is unbypassed must be verified periodically to be > ~~10~~% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

SR 3.3.2.1.7

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch — Shutdown Position Function to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch — Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.2 (continued)

29.5

Cycling open a bypass valve at slightly above ~~30%~~ RTP may affect the RPS Turbine Stop and Control Valve functions.

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analyses. The response time limits are specified in the COLR. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section VII-11.3.
2. Amendment 25 to the FSAR.
3. NEDC 96-006, "Estimate of Steam Tunnel's HELB," March 3, 1996.
4. USAR, Section XIV-5.8.1.
5. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36 (c)(2)(ii) (Ref. 3) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Control rod testing may be performed in compliance with the prescribed sequences of LCO 3.1.6, and during these tests, no exceptions to the requirements of LCO 3.1.6 are necessary. For testing performed with a sequence not in compliance with LCO 3.1.6, the requirements of LCO 3.1.6 may be suspended, provided additional administrative controls are placed on the test to ensure that the assumptions of the special safety analysis for the test sequence are satisfied. Assurances that the test sequence is followed can be provided by either programming the test sequence into the RWM, with conformance verified as specified in SR 3.3.2.1.8 and allowing the RWM to monitor control rod withdrawal and provide appropriate control rod blocks if necessary, or by verifying conformance to the approved test sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. These controls are consistent with those normally applied to operation in the startup range as defined in the SRs and ACTIONS of LCO 3.3.2.1, "Control Rod Block Instrumentation."

APPLICABILITY

9.85

Control rod testing, while in MODES 1 and 2, with THERMAL POWER greater than ~~10%~~ RTP, is adequately controlled by the existing LCOs on power distribution limits and control rod block instrumentation. Control rod movement during these conditions is not restricted to prescribed sequences and can be performed within the constraints of LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.3.2.1. With THERMAL POWER less than or equal to ~~10%~~ RTP, the provisions of this Special Operations LCO are necessary to

(continued)

Table T3.3.1-1 (Page 2 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUES
2. IRM				
a. Detector Not Full In	2	6	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.8	NA
b. Upscale	2	6	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≤ 108/125 of Full Scale
c. Inoperative	2	6	TSR 3.3.1.3 TSR 3.3.1.4	NA
d. Downscale	2[d]	6	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≥ 2.5/125 of Full Scale
3. APRM				
a. Upscale (Flow Biased)	1	4	TSR 3.3.1.1 TSR 3.3.1.2 TSR 3.3.1.5 TSR 3.3.1.7[e]	≤ (~~0.66W + 60.5% − 0.66 ΔW~~) [f][g] 0.75W+51.0% −0.75ΔW
b. Upscale (Startup)	2	4	TSR 3.3.1.1 TSR 3.3.1.4 TSR 3.3.1.7	≤ 11.5%
c. Inoperative	1, 2	4	TSR 3.3.1.1 TSR 3.3.1.5	NA
d. Downscale	1	4	TSR 3.3.1.1 TSR 3.3.1.5 TSR 3.3.1.7	≥ 3%
e. Upscale (Fixed)	1	4	TSR 3.3.1.1 TSR 3.3.1.2 TSR 3.3.1.5 TSR 3.3.1.7	≤ 109.5%

(continued)

(d) With IRMs on Range 2 or above.

(e) Calibration of the recirculation loop flow transmitters is only required once every 18 months.

(f) W is the two-loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power (~~2381~~ 2419 MWt = 100% RTP).

(g) ΔW is the difference between two-loop and single-loop effective drive flow and is used for single recirculation loop operation. ΔW = 0 for two loop recirculation loop operation.

T 3.3 INSTRUMENTATION

T 3.3.5 Feedwater Flow Instrumentation

TLCO 3.3.5 Both Leading Edge Flow Meter CheckPlus instrumentation systems shall be OPERABLE.

APPLICABILITY: MODE 1 and THERMAL POWER > 2381 MWt

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more systems inoperable.	A.1 Verify no reduction in power $\geq$ 10% occurs during the 72 hour COMPLETION TIME of REQUIRED ACTION A.2.	Immediately
	AND	
	A.2 Restore required instruments to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of CONDITION A not met.	B.1 Initiate an orderly power reduction to $\leq$ 2381 MWt.	Immediately
	OR	
	B.2 Verify power is no greater than 2381 MWt.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---

Not required to be performed until 12 hours after THERMAL POWER > 2381 MWt.

SURVEILLANCE		FREQUENCY
TSR 3.3.5.1	Perform CHANNEL CHECK.	12 hours

B 3.3 INSTRUMENTATION

B 3.3.5 Feedwater Flow Instrumentation

BASES

The highly accurate Leading Edge Flow Meter CheckPlus Instrumentation allowed an increase in Licensed Thermal Power from 2381 MWt to 2419 MWt by reducing instrument uncertainty. When one or both channels of this instrumentation is out of service, operation at 2419 MWt is allowed for up to 72 hours following discovery of an INOPERABLE channel, providing no downward power change in excess of 10% occurs during the 72 hours (this could result in calibration changes of the alternate instruments). If the instrumentation cannot be repaired within 72 hours, then power must be reduced to and maintained no higher than 2381 MWt until the instrumentation is repaired. If a decrease in power in excess of 10% occurs during the 72 hour period, then power must be maintained no higher than 2381 MWt (the original Licensed Thermal Power) until the instrumentation is repaired.

ATTACHMENT 3

REVISED (CLEAN COPIES) OF THE OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES

APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE COOPER NUCLEAR STATION DOCKET 50-298, DPR-46

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility. /018 /018 /018

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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 or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal). /

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications. /107 /

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006. /* /*

(4) Fire Protection | /199,**

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision: | /199,** | /199,** | /199,** | /199,** | /199,** | /199,** | /199,** | /199,**

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. | /199,** | /199,** | /199,** | /199,**

*Revised by letter dated March 5, 2007
** Revised by letter dated August 9, 2007

1.1 Definitions

LOGIC SYSTEM FUNCTIONAL TEST (continued)	from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2419 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time segment from the time the sensor contacts actuate to the time the scram solenoid valves deenergize.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

(continued)

Actions (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -------------NOTE------------- Not applicable when THERMAL POWER > 9.85% RTP. --------------------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. OR D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 9.85% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1 -------------NOTE------------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ---------------------------------	
	Move associated control rod(s) to correct position.	8 hours
	OR	
	A.2 Declare associated control rod(s) inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 29.5% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.12	---------------------------NOTES--------------------------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.13	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.14	Verify Turbine Stop Valve — Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is ≥ 29.5% RTP.	18 months
SR 3.3.1.1.15	---------------------------NOTE--------------------------- Neutron detectors are excluded. --- Verify the RPS RESPONSE TIME is within limits.	18 months

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux — High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
	5(a)	3	H	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux — High (Startup)	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 14.5% RTP
b. Neutron Flux-High (Flow Biased)	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 0.75 W + 62.0% RTP(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) [0.75 W + 62.0% - 0.75 ΔW] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level - High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
b. Level Switch	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
8. Turbine Stop Valve — Closure	≥ 29.5% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, DEH Trip Oil Pressure — Low	≥ 29.5% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1018 psig
10. Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5[(a)]	1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.13	NA
	5[(a)]	1	H	SR 3.3.1.1.9 SR 3.3.1.1.13	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.2	-----NOTE----- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 9.85% RTP in MODE 2. ----- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	-----NOTE----- Not required to be performed until 1 hour after THERMAL POWER is ≤ 9.85% RTP in MODE 1. ----- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	-----NOTE----- Neutron detectors are excluded. ----- Verify the RBM: a. Low Power Range — Upscale Function is not bypassed when THERMAL POWER is ≥ 27.5% and < 62.5% RTP and a peripheral control rod is not selected. b. Intermediate Power Range — Upscale Function is not bypassed when THERMAL POWER is ≥ 62.5% and < 82.5% RTP and a peripheral control rod is not selected. c. High Power Range — Upscale Function is not bypassed when THERMAL POWER is ≥ 82.5% RTP and a peripheral control rod is not selected.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.5	-----NOTE----- Neutron detectors are excluded. ---------- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is $\leq$ 9.85% RTP.	18 months
SR 3.3.2.1.7	-----NOTE----- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ---------- Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range — Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	(h)
b. Intermediate Power Range — Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	(h)
c. High Power Range — Upscale	(c),(d)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	(h)
d. Inop	(d),(e)	2	SR 3.3.2.1.1	NA
e. Downscale	(d),(e)	2	SR 3.3.2.1.1 SR 3.3.2.1.5	$\geq$ 92/125 divisions of full scale
2. Rod Worth Minimizer	1(f),2(f)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8	NA
3. Reactor Mode Switch — Shutdown Position	(g)	2	SR 3.3.2.1.7	NA

(a) THERMAL POWER $\geq$ 27.5% and < 62.5% RTP and MCPR < 1.70 and no peripheral control rod selected.

(b) THERMAL POWER $\geq$ 62.5% and < 82.5% RTP and MCPR < 1.70 and no peripheral control rod selected.

(c) THERMAL POWER $\geq$ 82.5% and < 90% RTP and MCPR < 1.70 and no peripheral control rod selected.

(d) THERMAL POWER $\geq$ 90% RTP and MCPR < 1.40 and no peripheral control rod selected.

(e) THERMAL POWER $\geq$ 27.5% and < 90% RTP and MCPR < 1.70 and no peripheral control rod selected.

(f) With THERMAL POWER $\leq$ 9.85% RTP.

(g) Reactor mode switch in the shutdown position.

(h) Less than or equal to the Allowable Value specified in the COLR.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low (Level 1)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 835 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 142.7% rated steam flow
d. Condenser Vacuum - Low	1, 2(a), 3(a)	2	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 8 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2 per location	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low (Level 3)	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 3 inches
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.84 psig
c. Reactor Building Ventilation Exhaust Plenum Radiation- High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 49 mR/hr
d. Main Steam Line Radiation - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3 times full power background
e. Reactor Vessel Water Level -Low Low Low (Level 1)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches

(continued)

(a) With any turbine stop valve not closed.

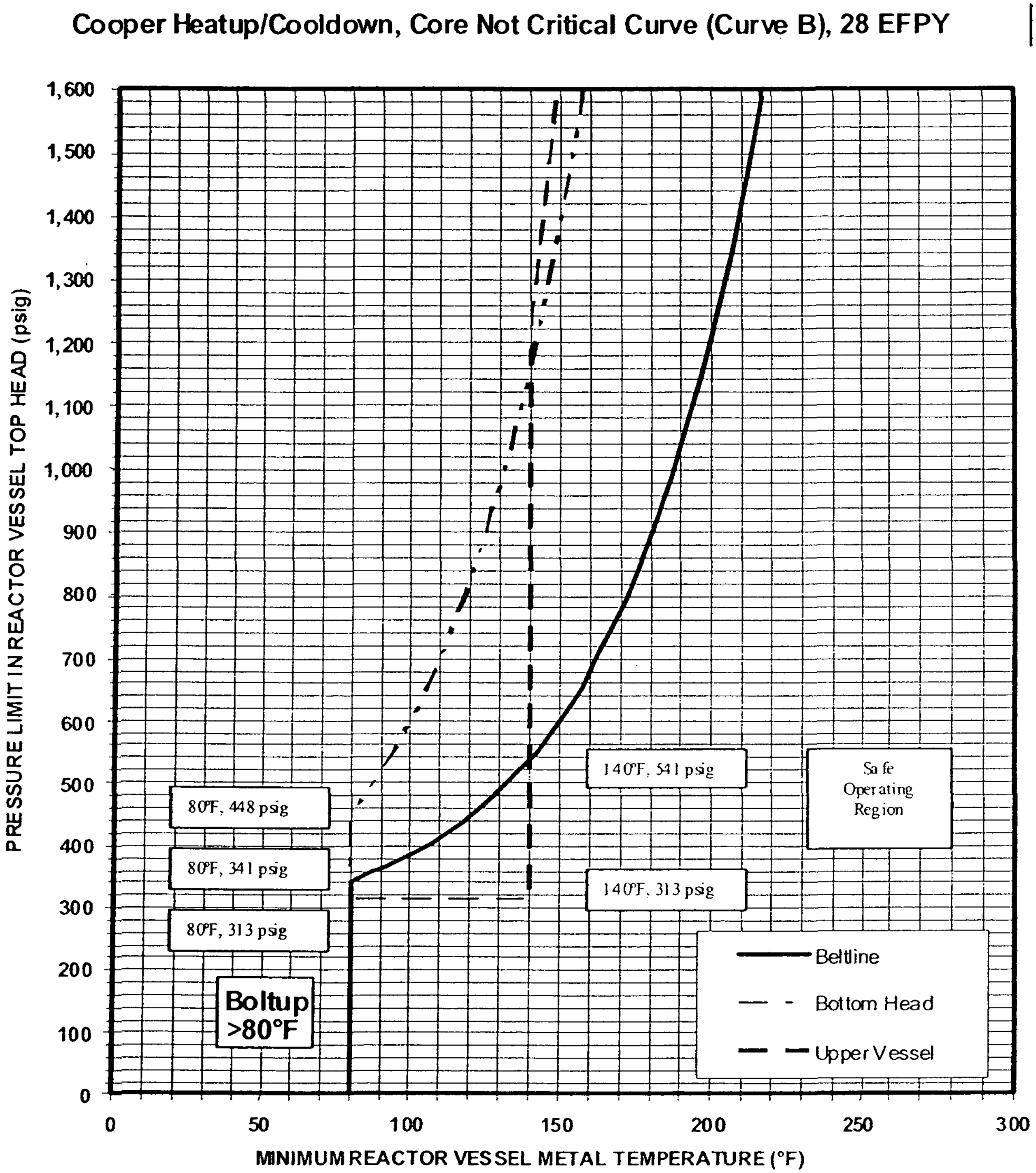

Figure 3.4.9-1 (page 1 of 1)
Pressure/Temperature Limits for
Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown
Valid Through 28 EFPY

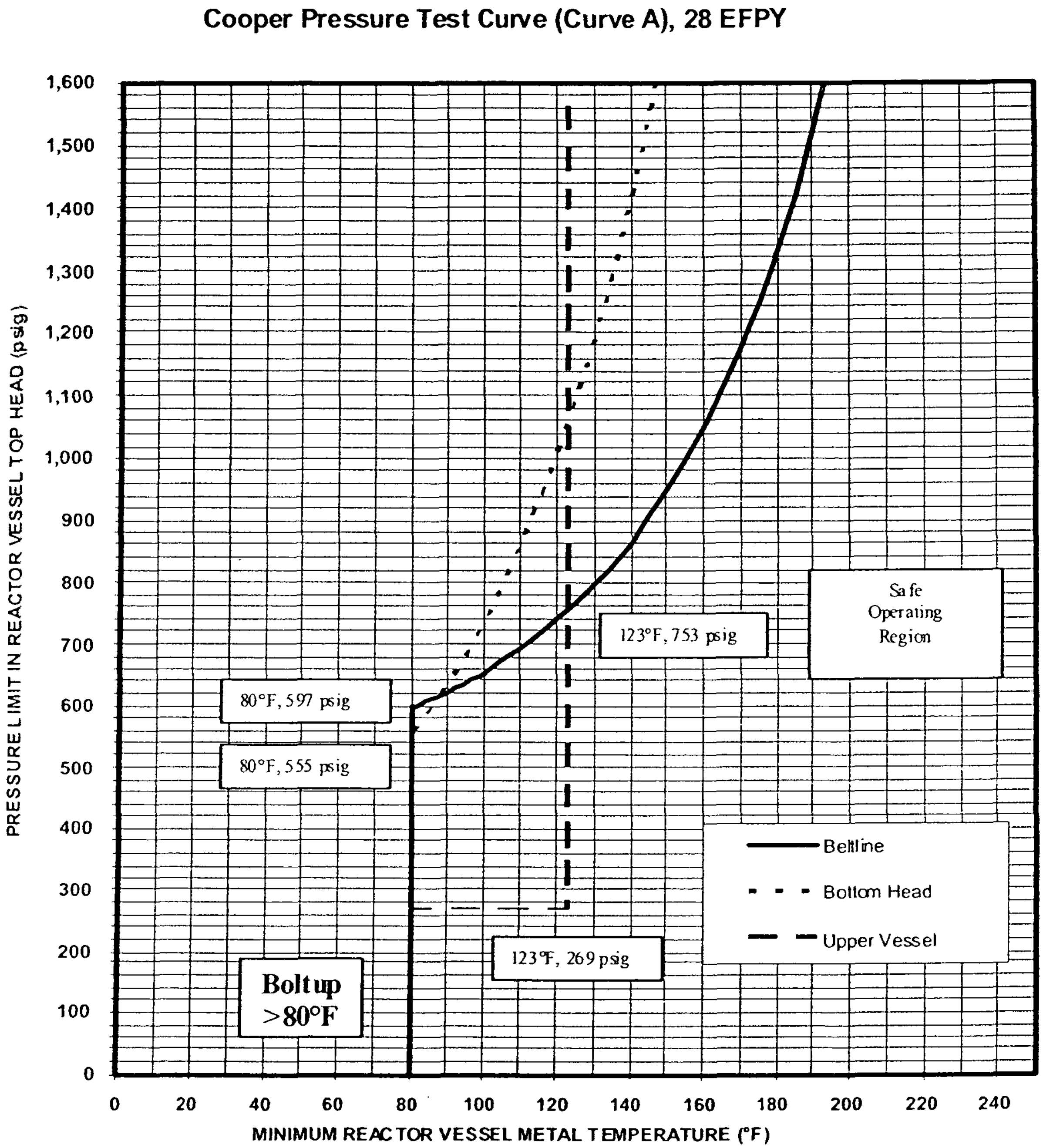

Figure 3.4.9-2 (page 1 of 1)
Pressure/Temperature Limits for
Inservice Hydrostatic and Inservice Leakage Tests
Valid Through 28 EFPY

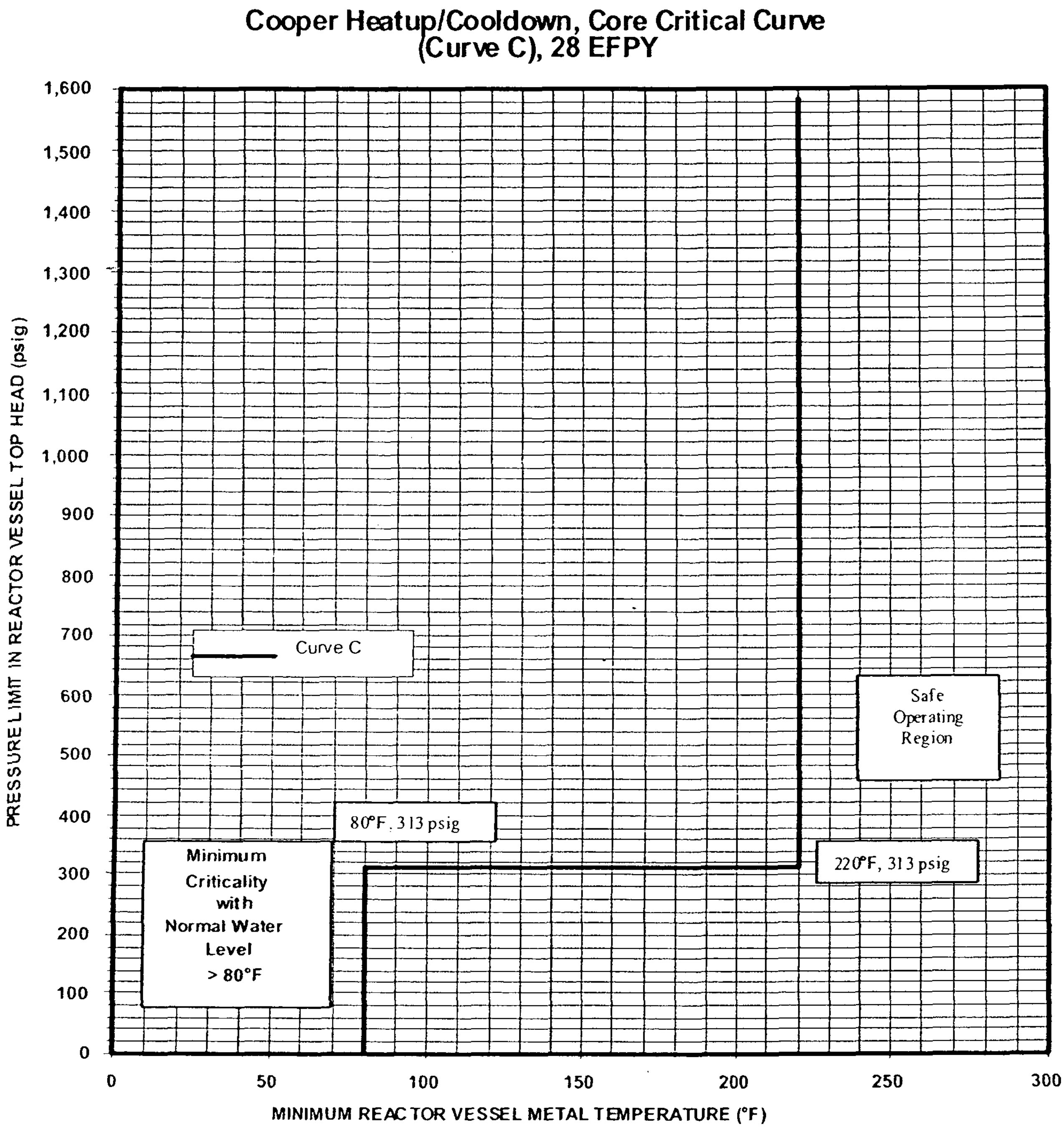

Figure 3.4.9-3 (page 1 of 1)
Pressure/Temperature Limits for Criticality
Valid Through 28 EFPY

ATTACHMENT 4

NRC REGULATORY ISSUE SUMMARY 2002-03 RECONCILIATION

APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE COOPER NUCLEAR STATION DOCKET 50-298, DPR-46

On January 31, 2002, the Nuclear Regulatory Commission issued "NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" (RIS 2002-03), which provides guidance on the content of measurement uncertainty recapture power uprate projects. Specifically, Attachment 1 of RIS 2002-03 provides a detailed breakdown of the specific subject matter that each licensee should address when applying for this type of power uprate.

To ensure technical issues were appropriately addressed, CNS performed the evaluations based upon the guidance of NEDC-32938P-A (Reference 7.5 of Attachment 1). The results of the evaluations performed are fully documented in NEDC-33385P (Enclosure 1).

RIS 2002-03 states the need to identify the bounded and unbounded accident and transient analyses. Enclosure 1 presents the analyses performed or confirmed for each subject area. Within the section for each subject area, the existing analysis is identified as bounding or nonbounding. Where generic analyses from NEDC-32938P-A are used, the generic analysis is verified as being applicable to CNS within Enclosure 1. Where necessary, the specific sections present the results of the new analysis. Necessary accidents and transients applicable to CNS, which encompass the transients and accidents listed under the items in RIS 2002-03, are evaluated in Enclosure 1.

Table 4-1, beginning on the next page, is a cross-reference of the guidance provided in RIS 2002-03 to the documents that provide the discussions of and/or justifications for the proposed Technical Specifications changes addressed herein.

TABLE 4-1

RIS 2002-03 CROSS REFERENCE TO LOCATION IN CNS LAR

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
I.	***Feedwater Flow Measurement Technique and Power Measurement Uncertainty***			
I.1	Detailed description of plant-specific implementation of feedwater flow measurement technique and power increase gained as result of implementing technique	A1	1.0 3.0 4.0	1.0 Description 3.0 Background 4.0 Technical Analysis
I.1.A.	NRC approval of topical report on flow measurement technique	A1	3.0 4.2	3.0 Background 4.2 LEFM Ultrasonic Flow Measurement
I.1.B.	Reference to NRC's approval of proposed measure-ment technique	A1	3.0 4.2	3.0 Background 4.2 LEFM Ultrasonic Flow Measurement
I.1.C.	Plant implementation	A1	3.0 4.2	3.0 Background 4.2.1 Use Of LEFM CheckPlus System To Determine Calorimetric Power
I.1.D.	Disposition of NRC criteria	A1	4.2	4.2.8 RIS 2002-03, Item I.1.D and ER-157P Criteria
I.1.E.	Total power measurement uncertainty calc. for the plant	A1 E7	4.2 All	4.2.9 Total Power Measurement Uncertainty NEDC 06-035 Reactor Core Thermal Power Uncertainty Calculation

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
I.1.F.	Calibration and maintenance	A1	4.2	4.2 Monitoring, Verification and Error Reporting 4.2.6 Hydraulic Modeling 4.2.7 RIS 2002-03, Item I.1.D and ER-157P Criteria (Criterion 1)
I.1.G.	Proposed outage time for LEFM and basis for selected time	A1	4.2	4.2.2 LEFM Inoperability 4.2.8 RIS 2002-03, Item I.1.D and ER-157P Criteria
I.1.H.	Proposed actions if outage time is exceeded, and basis for actions	A1	4.2	4.2.2 LEFM Inoperability 4.2.8 RIS 2002-03, Item I.1.D and ER-157P Criteria
II.	***Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level***			
II.1.	Matrix for bounded accidents and transients	E1	All 9.0	GE-Hitachi (GEH) Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P. 9.0 Reactor Safety Performance Evaluations
III.	***Accidents and Transients for Which the Existing Analyses of Record Do Not Bound Plant Operation at the Proposed Uprated Power Level***			
III.1, 2, 3	Matrix for unbounded accidents and transients	E1	All 9.0	GE-Hitachi (GEH) Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P. 9.0 Reactor Safety Performance Evaluations (No unbounded accidents or transients involved)

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.	***Mechanical/Structural/Material Component Integrity and Design***			
IV.1.A.i	Reactor vessel, nozzles, supports	E1	3.2	3.2 Reactor Vessel 3.2.1 Fracture Toughness 3.2.2 Reactor Vessel Structural Evaluation 3.2.2.1 Design Conditions 3.2.2.2 Normal and Upset Conditions 3.2.2.3 Emergency and Faulted Conditions
IV.1.A.ii	Reactor core support structures and vessel internals	E1	3.3 3.4	3.3 Reactor Internals 3.3.1 Reactor Internal Pressure Difference 3.3.2 Reactor Internals Structural Evaluation 3.3.3 Steam Separator and Dryer Performance 3.4 Flow-Induced Vibration
IV.1.A.iii	Control rod drive mechanisms	E1	2.5	2.5 Reactivity Control

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.A.iv	Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles	E1	3.4	3.4 Flow-Induced Vibration
			3.5	3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping
			3.6	3.6 Reactor Recirculation System
			3.7	3.7 Main Steam Line Flow Restrictors
			3.8	3.8 Main Steam Isolation Valves
			3.9	3.9 Reactor Core Isolation Cooling
			3.10	3.10 Residual Heat Removal System
			3.11	3.11 Reactor Water Cleanup System

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.A.v	Balance of plant (BOP) plping (NSSS interface systems, safety-related cooling water systems, and containment systems)	E1	3.5	3.5. Piping Evaluation 3.5.2. Balance-of-Plant Piping Evaluation
			3.9	3.9 Reactor Core Isolation Cooling
			3.10	3.10 Residial Heat Removal System
			3.11	3.11 Reactor Water Cleanup System
			4.2	4.2.1 High Pressure Coolant Injection 4.2.3 Core Spray 4.2.4 Low Pressure Coolant Injection
			6.4	6.4.1 Service Water Systems 6.4.3 Component Cooling Water System
IV.1.A.vi	Steam generator tubes, secondary side internal support structures, shell and nozzles	NA	NA	NA
IV.1.A. vii	Reactor coolant pumps	NA	NA	NA
IV.1.A. viii	Pressurizer shell, nozzles, and surge lines	NA	NA	NA

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.A.ix	Safety-related valves	E1	3.1	3.1. Nuclear System Pressure Relief/Overpressure Protection
			3.8	3.8. Main Steam Isolatlon Valves
			4.1	4.1. Containment System Performance
				4.1.1 Generic Letter 89-10 Program
				4.1.2. Generic Letter 95-07 Program
			6.5	6.5. Standby Liquid Control System
IV.1.B.i	Stresses	E1	3.2	3.2. Reactor Vessel
				3.2.2. Reactor Vessel Structural Evaluation
			3.4	3.4. Flow-Induced Vibration
			3.5	3.5. Piping Evaluation
				3.5.1 Reactor Coolant Pressure Boundary Piping
				3.5.2 Balance-of-Plant Piping Evaluation
			7.1	7.1 Turbine-Generator
IV.1.B.ii	Cumulative usage factors	E1	3.2	3.2. Reactor Vessel
				3.2.2 Reactor Vessel Structural Evaluation

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.B.iii	Flow induced vibration	E1	3.4	3.4 Flow-Induced Vibration
IV.1.B.iv	Changes in temperature (pre- and post-uprate)	E1	1.3	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features
			Table 1-2	Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.v	Changes in pressure (pre- and post-uprate)	E1	1.3	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features
			Table 1-2	Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vi	Changes In flow rate (pre- and post-uprate)	E1	1.3	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features
			Table 1-2	Thermal-Hydraulic Parameters at TPO Uprate Conditions

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.B.vii	High-energy line break locations	E1	10.1	10.1 High Energy Line Break 10.1.1 Steam Line Breaks 10.1.2 Liquid Line Breaks
IV.1.B. viii	Jet impingement and thrust forces	E1	10.1 10.2	10.1 High Energy Line Break 10.1.1 Steam Line Breaks 10.1.2 Liquid Line Breaks 10.1.2.7 Pipe Whip and Jet Impingement 10.2 Moderate Energy Line Break
IV.1.C.i	Reactor vessel pressurized thermal shock calculations	E1	3.1	3.1 Nuclear System Pressure Relief/Overpressure Protection
IV.1.C.ii	Reactor vessel fluence evaluation	E1	3.2	3.2. Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.iii	Reactor vessel heatup and cooldown pressure-temperature limit curves	E1	3.2	3.2. Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.iv	Reactor vessel low temperature overpressure protection	E1	3.2	3.2. Reactor Vessel 3.2.1 Fracture Toughness

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.C.v.	Reactor vessel upper shelf energy	E1	3.2	3.2. Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.vi	Reactor vessel surveillance capsule withdrawal schedule	E1	3.2	3.2. Reactor Vessel 3.2.1 Fracture Toughness
IV.1.D	Code of record	E1	3.2 3.5	3.2. Reactor Vessel 3.2.2 Reactor Vessel Structural Evaluation 3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piplng 3.5.2 Balance-of-Plant Piplng Evaluation
IV.1.E	Component inspection/testing programs and erosion/corrosion programs	E1	3.5 10.6	3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping (including flow-accelerated corrosion and piping inspection programs) 3.5.2 Balance-of-Plant Piping Evaluation (including flow-accelerated corrosion and piping inspection programs) 10.6 Plant Life

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
IV.1.F	NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"	NA	NA	NA
V.	***Electrical Equipment Design***			
V.l.A	Emergency diesel generators	E1	6.1	6.1 AC Power 6.1.2 On-Site Power
V.l.B	Station blackout equipment	E1	9.3	9.3.2 Station Blackout
V.l.C	Environmental qualification of electrical equipment	E1	10.3	10.3 Environmental Qualification 10.3.1 Electrical Equipment
V.l.D	Grid stability	E1	6.1	6.1 AC Power 6.1.1 Off-Site Power

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VI.	***System Design***			
VI.1.A	NSSS Interface Systems for BWRs (e.g., suppression pool cooling, as applicable)	E1	3.4	3.4 Flow-Induced Vibration
			3.5	3.5 Piping Evaluation
				3.5.1 Reactor Coolant Pressure Boundary Piping
				3.5.2 Balance-of-Plant Piping Evaluation
			3.6	3.6 Reactor Recirculation System
			3.7	3.7 Main Steam Line Flow Restrictors
			3.8	3.8 Main Steam Isolation Valves
			3.9	3.9 Reactor Core Isolation Cooling
			3.10	3.10 Residual Heat Removal System
			3.11	3.11 Reactor Water Cleanup System
VI.1.B	Containment Systems	E1	4.1	4.1 Containment System Performance
				4.1.1 Generic Letter 89-10 Program
				4.1.2 Generic Letter 95-07 Program
				4.1.3 Generic Letter 96-06

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VI.1.C	Safety-related cooling water systems	E1	6.4	6.4 Water Systems 6.4.1 Service Water Systems 6.4.5 Ultimate Heat Sink
VI.1.D	Spent fuel pool storage and cooling systems	E1	6.3	6.3 Fuel Pool 6.3.1 Fuel Pool Cooling 6.3.2 Crud Activity and Corrosion Products 6.3.3 Radiation Levels 6.3.4 Fuel Racks

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VI.1.E	Radioactive waste systems	E1	4.5	4.5 Standby Gas Treatment System
			8.1 through 8.6	8.1 Liquid and Solid Waste Management 8.2 Gaseous Waste Management 8.3 Radiation Sources in the Reactor Core 8.4 Radiation Sources in Reactor Coolant 8.4.1 Coolant Activation Products 8.4.2 Activated Corrosion Products 8.4.3 Fission Products 8.5 Radiation Levels 8.6 Normal Operation Off-Site Doses
VI.1.F	Engineered safety features (ESF) heating, ventilation, and air conditioning systems	E1	4.4	4.4 Main Control Room Atmosphere Control System
			4.7	4.7 Post-LOCA Combustible Gas Control System
			6.6	6.6 Power Dependent Heating, Ventilation and Air Conditioning

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VII. *Other*				
VII.1	Operator actions and effects on time available	E1	4.1	4.1 Containment System Performance
			6.7	6.7 Fire Protection
			9.3	9.3 Special Events
				9.3.2 Station Blackout
			10.5	10.5 Operator Training and Human Factors
VII.2.A	Emergency and abnormal operating procedures	A1	4.2	4.2.4 Procedures, Training, and Simulator
VII.2.B	Control room controls, displays (including the safety parameter display system) and alarms	A1	4.2	4.2.2 LEFM Inoperability
				4.2.3 Maintenance and Calibration
				4.2.4 Procedures, Training, and Simulator
VII.2.C	The control room plant reference simulator	A1	4.2	4.2.4 Procedures, Training, and Simulator
		A5		List of Regulatory Commitments
VII.2.D	The operator training program	A1	4.2	4.2.4 Procedures, Training, and Simulator
		E1	10.5	10.5 Operator Training and Human Factors

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VII.3	Licensee intent to complete modifications identified in item 2 above (including training of operators), prior to implementation of the power uprate	A1 A5	4.2	4.2.4 Procedures, Training, and Simulator 4.2.10 Startup Testing 4.2.12 Miscellaneous List of Regulatory Commitments
VII.4	Licensee intent to revise plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level (reduce from 2%)	A1 A5	4.2	4.2.2 LEFM Inoperability 4.2.4 Procedures, Training and Simulator List of Regulatory Commitments

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VII.5.A	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on types and amounts of effluents released offsite, and whether bounded by final environmental statement and previous Environmental Assessments for the plant	A1 E1	5.3 8.0	5.3 Environmental Considerations 8.0 Radwaste and Radiation Sources
VII.5.B	10 CFR51.22, Exclusion of Environmental Review. including discussion of effect of the power uprate on individual and cumulative occupational radiation exposure	A1 E1	5.3 8.5	5.3 Environmental Considerations 8.5 Radiation Levels

NRC RIS 2002-03		DOCUMENT		DOCUMENT DESCRIPTION
Item No.	DESCRIPTION	#	SECTION	
VIII.	***Changes to Technical Specifications, Protection System Settings, Emergency System Settings***			
VIII.1	A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate	A1 A2	2.0 All	2.0 Proposed Change To Operating License And Technical Specifications Operating License And Technical Specifications Pages Marked Up With Proposed Changes (TRM Pages Included for Information)
VIII.1.A	Description of the change	A1 A2	2.0 All	2.0 Proposed Change To Operating License And Technical Specifications Operating License And Technical Specifications Pages Marked Up With Proposed Changes (TRM Pages Included for Information)
VIII.1.B	Description of the analyses affected by and/or supporting the change	E1	All	GEH Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P
VIII.1.C	Justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by change	E1	All	GEH Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P (No unbounded accidents and transients are involved)

ATTACHMENT 5

LIST OF REGULATORY COMMITMENTS

APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE COOPER NUCLEAR STATION DOCKET 50-298, DPR-46

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station (CNS) of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
Final acceptance of the CNS uncertainty analysis and verification of bounding calibration test data will occur.	NLS2007069-01	During power ascension and commissioning process following RE24
Procedure changes governing normal operation, emergency operation, and off-normal operation, as well as equipment changes that may be affected by power uprate, will be made.	NLS2007069-02	Prior to implementation of uprated power
Technical Requirements Manual will be revised to include CheckPlus System out-of-service administrative controls.	NLS2007069-03	Prior to implementation of uprated power
Core power from Average Power Range Monitors (APRMs) will be rescaled to the uprated power level and any necessary adjustments of APRM alarm and trip settings will be made.	NLS2007069-04	Prior to exceeding Current Licensed Thermal Power (CLTP) level
Demonstration of an acceptable fuel thermal margin will be performed at each of the following steady-state heat balance power levels: 95% and 100% of CLTP, and 100% of uprated power level.	NLS2007069-05	Prior to and during power ascension to 100% of uprated power level
Routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95% and 100% of CLTP, and at 100% of uprated power level.	NLS2007069-06	Prior to and during power ascension to 100% of uprated power level

Operational aspect of the uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing, and reactor pressure control system testing. During this testing, a water level change of ± 3 inches, and pressure setpoint changes of ± 3 psi will be used. If necessary, controllers and actuator elements will be adjusted. • Performance of feedwater level control system will be recorded at 95% and 100% of CLTP, and confirmed at uprated power level. • Turbine pressure controller setpoint will be readjusted at 95% and 100% CLTP level and held constant prior to recording baseline power ascension data.	NLS2007069-07	Prior to and during power ascension to 100% of uprated power level
Ensure compliance with the methodology contained in Reg. Guide 1.20 for vibration assessment.	NLS2007069-08	Prior to exceeding CLTP and ascension to uprated power level
Appropriate personnel will receive training on Caldon LEFM CheckPlus System, and on affected procedures.	NLS2007069-09	Prior to operation at uprated power.
Simulator changes and validation for power uprate will be completed in accordance with ANSI/ANS 3.5-1985.	NLS2007069-10	Prior to implementation of the requested license amendment
A Startup Test Report will be submitted.	NLS2007069-11	Within 90 days following resumption of power operation following RE24
A process will be implemented to use the LEFM CheckPlus System feedwater mass flow and temperature to adjust or calibrate the existing feedwater flow nozzle-based signals.	NLS2007069-12	Following power ascension to 100% of uprated power level.

ENCLOSURE 2

AFFIDAVIT OF WITHHOLDING PURSUANT TO 10 CFR 2.390

GE-HITACHI NUCLEAR ENERGY

APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE COOPER NUCLEAR STATION DOCKET NO. 50-298, DPR-46

GE Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, James F. Harrison, state as follows:

(1) I am Vice President Fuel Licensing, Regulatory Affairs, GE Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH Safety Analysis Report NEDC-33385P, Revision 0, "Cooper Nuclear Station Thermal Power Optimization" Class III (GEH Proprietary Information), dated November 2007. The proprietary information is delineated by a [[dotted underline inside double square brackets{3}]] In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information, which fit into the definition of proprietary information, are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information, which reveals aspects of past, present, or future GEH customer-funded development, plans and programs, resulting in potential products to GEH;

d. Information, which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b above.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains detailed results and conclusions regarding supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for the Thermal Power Optimization for a GEH BWR. The evaluations utilized analytical models and methods, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GEH Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GEH, on the order of several million dollars.

The methodology for the evaluation process was developed by GEH, approved by the US NRC, and is documented in Licensing Topical Report NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization", Revision 2, May 2003. The development of the evaluation process along with the interpretation and application of the analytical results constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 8th day of November 2007.

James F. Harrison

James F. Harrison
Vice President Fuel Licensing
Regulatory Affairs
GE Hitachi Nuclear Energy