Nebraska Public Power District

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NLS2015073
August 6, 2015

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

Subject: License Amendment Request to Revise Technical Specifications to Relocate Pressure and Temperature Limit Curves to a Pressure and Temperature Limits Report
Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 in accordance with the provisions of 10 CFR 50.4 and 10 CFR 50.90 to revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). The proposed amendment will modify TS Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," by replacing the existing reactor vessel heatup and cooldown rate limits and the P/T limit curves with references to the Pressure and Temperature Limits Report (PTLR). A definition for the PTLR will be added to TS Section 1.1, "Definitions," and a section addressing administrative requirements for the PTLR will be added to TS Section 5.6, "Reporting Requirements." The existing CNS Nuclear Regulatory Commission (NRC) approved P/T limit curves for 32 Effective Full-Power Years are not being revised as a part of this relocation. In addition, editorial corrections are being made to the Table of Contents.

NPPD requests approval of the proposed amendment by August 6, 2016 and requests that Enclosure 1, which contains proprietary information, is withheld from public disclosure. This allows one year for NRC review. Once approved, the amendment will be implemented within 60 days.

Attachment 1 provides a description of the proposed TS changes, the basis for the amendment, the no significant hazards consideration evaluation pursuant to 10 CFR 50.91(a)(1), and the environmental consideration pursuant to 10 CFR 51.22. Attachment 2 provides a copy of the marked-up TS pages showing the proposed changes. Attachment 3 provides the final typed format TS pages to be issued with the amendment. Attachment 4 provides conforming changes to the TS Bases for NRC information. Enclosure 1 provides the proprietary version of the PTLR.

Note: Enclosure 1 to this letter contains Proprietary Information. Upon separation from Enclosure 1, the remainder of this document is decontrolled.

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www.nppd.com

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NRR

Enclosure 2 provides the non-proprietary version of the PTLR. Enclosure 3 provides the Affidavit for the proprietary information.

No formal licensee commitments are being made for this proposed amendment.

The proposed TS 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 251, issued July 14, 2015, have been incorporated into this request.

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies are also provided to the NRC Region IV office and the Senior Resident Inspector in accordance with 10 CFR 50.4(b)(1).

Should you have any questions concerning this matter, please contact Jim Shaw, Licensing Manager, at (402) 825-2788.

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

Executed on 08/06/15
(Date)

Sincerely,

Oscar A. Limpias
Vice President - Nuclear and
Chief Nuclear Officer

/dv

Attachments:
1. License Amendment Request to Revise Technical Specifications to Relocate Pressure and Temperature Limit Curves to a Pressure Temperature Limits Report
2. Proposed Technical Specification Revision (Markup)
3. Proposed Technical Specification Revision (Final Typed Format)
4. Proposed Technical Specification Bases Revision (Information Only)

Enclosures:
1. Pressure and Temperature Limits Report (PTLR) for 32 Effective Full-Power Years (EFPY) (Proprietary)
2. Pressure and Temperature Limits Report (PTLR) for 32 Effective Full-Power Years (EFPY) (Non-Proprietary)
3. Affidavit for Proprietary Information Contained in the Pressure and Temperature Limits Report (PTLR) for 32 Effective Full-Power Years (EFPY)

Note: Enclosure 1 to this letter contains Proprietary Information. Upon separation from Enclosure 1, the remainder of this document is decontrolled.

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

Senior Resident Inspector w/attachments and w/o enclosures
USNRC - CNS

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

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

NPG Distribution w/o attachments and w/o enclosures

CNS Records w/attachments and enclosures

Attachment 1

License Amendment Request to Revise Technical Specifications to Relocate Pressure and Temperature Limit Curves to a Pressure and Temperature Limits Report

Cooper Nuclear Station, Docket No. 50-298, DPR-46

Revised Pages
Table of Contents pages i-iv, 1.1-4, 3.4-19, 3.4-20, 3.4-21, 3.4-22, 5.0-21

Deleted Pages
3.4-23, 3.4-24, 3.4-25

New Page
5.0-22

Renumbered Pages
Original 3.4-26 to new 3.4-23
Original 5.0-22 to new 5.0-23

1.0 SUMMARY DESCRIPTION

This is a request to amend the Cooper Nuclear Station (CNS) Facility Operating License DPR-46 Technical Specifications (TS). The proposed amendment would modify TS to relocate pressure and temperature (P/T) limit curves to a Pressure and Temperature Limits Report (PTLR).

Nebraska Public Power District requests approval of the proposed amendment by August 6, 2016, which allows one year for Nuclear Regulatory Commission (NRC) review. Upon receipt of the approved amendment, CNS will implement the change within 60 days.

2.0 DETAILED DESCRIPTION

2.1 Proposed Change

The proposed amendment includes the following TS revisions:

1. TS Section 1.1, "Definitions" - A new definition, "Pressure and Temperature Limits Report (PTLR)," is added. The wording for this definition is consistent with TSTF-419-A (Reference 1).

2. TS Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits" - The current specified heatup and cooldown rates, temperature limits, and associated figures are replaced with a reference to the PTLR.

3. TS Section 5.6, "Reporting Requirements" - A new Section 5.6.7, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," has been added. The format and content of new Section 5.6.7 is consistent with TSTF-419-A. This new section: (1) identifies the individual TS that address reactor coolant system P/T limits; (2) references the NRC-approved topical report that documents PTLR methodologies; and (3) requires the PTLR and any revision or supplement thereto be submitted to the NRC.

4. Table of Contents - Editorial corrections.

2.2 Bases Changes

TS Bases that reflect the relocation of the P/T limit curves are provided in Attachment 4 for the NRC's information. These Bases revisions will be made as an implementing action pursuant to TS 5.5.10, "Technical Specifications (TS) Bases Control Program," following issuance of the approved amendment.

3.0 TECHNICAL EVALUATION

3.1 Technical Justification of Proposed Changes

10 CFR 50, Appendix G, requires the establishment of P/T limits for material fracture toughness requirements of the Reactor Coolant Pressure Boundary materials. 10 CFR 50, Appendix G, requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system hydrostatic tests.

Historically, the P/T limit curves for Boiling Water Reactors (BWR) have been contained in the TS, necessitating the submittal of license amendment requests to update the curves. This caused both the NRC and licensees to expend resources that could otherwise be devoted to other activities.

Generic Letter (GL) 96-03 (Reference 2) allows plants to relocate their P/T limit curves and associated numerical limits (such as heatup and cooldown rates) from the plant TS to a PTLR, which is a licensee-controlled document. As stated in GL 96-03, during the development of the improved Standard Technical Specifications, a change was proposed to relocate the P/T limits contained in the plant TS to a PTLR. As one of the improvements to the Standard Technical Specifications, the NRC staff agreed with the industry that the curves may be relocated outside the plant TS to a PTLR so that the licensee could efficiently maintain these limits. One of the prerequisites for having the PTLR option is that the P/T curves and limits be derived using methodologies approved by the NRC, and that the associated licensing topical reports describing the approved methodologies be referenced in the plant TS.

In order to implement the PTLR, the analytical methods used to develop the P/T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the plant TS. The current CNS P/T curves have been developed in accordance with the SIR-05-044-A, Revision 0 (Reference 3), methodology and include the effects of instrument nozzles located within the beltline.

Revision 1-A of SIR-05-044 provides a current methodology for developing reactor coolant system P/T limit curves and other associated numerical limits for BWRs. Revision 1-A was developed to incorporate the additional requirements related to the effects of nozzles within the beltline region. Because the methodology approved in the NRC SER for the current CNS P/T curves was based on Revision 0 and includes the effects of instrument nozzles, the approved methodology is equivalent to the requirements of Revision 1-A. The PTLR has been prepared in accordance with the requirements of Licensing Topical Reports SIR-05-044, Revision 1-A contained within BWROG-TP-11-022-A, Revision 1 (Reference 8) and 0900876.401, Revision 0-A, contained within BWROG-TP-11-023-A, Revision 0 (Reference 9). NPPD is asking that SIR-05-044-A, Revision 1-A and 0900876.401, Revision 0-A be approved as the analytical method for CNS.

SIR-05-044, Revision 0-A and 1-A, do not include development or licensing of vessel fluence methods. The fluence calculations use the Radiation Analysis Modeling Application (RAMA) code which adheres to the guidance in Regulatory Guide (RG) 1.190. Use of the RAMA code was approved for CNS in the NRC safety evaluation for TS Amendment 219 dated April 27, 2006 (Reference 4).

The P/T curves, currently in TS to be relocated to the PTLR, have been developed to present steam dome pressure versus minimum reactor vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline region. Complete P/T curves were developed for 32 Effective Full-Power Years and were previously approved for CNS in NRC Safety Evaluation for TS Amendment 245 dated February 22, 2013 (Reference 5). This Safety Evaluation concluded that the calculated P/T limits for all regions of the reactor pressure vessel (RPV) meet the criteria of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G, are in compliance with the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60, and are acceptable for incorporation into the CNS TS.

4.0 REGULATORY SAFETY ANALYSIS

4.1 Applicable Regulatory Requirements

As discussed in the Safety Evaluation for SIR-05-044-A, Revision 1, the NRC has established requirements in 10 CFR 50, Appendix G, in order to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The regulation at 10 CFR 50, Appendix G, requires that the P/T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code were used to generate the P/T limits. The regulation at 10 CFR 50, Appendix G, also requires that applicable surveillance data from reactor pressure vessel material surveillance programs be incorporated into the calculations of plant-specific P/T limits, and that the P/T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials. NRC regulatory guidance related to P/T limit curves is found in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," and Standard Review Plan (NUREG-0800) Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock."

Adoption of the NRC-approved methodology described in the SIR-05-044-A, Revision 1, for the preparation of the P/T limit curves ensures that the requirements of 10 CFR 50, Appendix G, will be satisfied. 10 CFR 50, Appendix H, provides criteria for the design and implementation of reactor pressure vessel material surveillance programs for operating light-water reactors. CNS demonstrates its compliance with the requirements of 10 CFR 50, Appendix H, through participation in the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program (Reference 6).

GL 96-03 provides regulatory guidance regarding relocation of P/T curves and associated numerical limits (such as heatup and cooldown rates) from plant TS to a PTLR (a licensee-controlled document). As stated in GL 96-03, a licensee requesting such a change must satisfy the following three criteria:

1) Reference NRC-approved methodologies in the TS.
2) Develop a PTLR to contain the P/T limit curves, associated numerical limits, and any necessary explanation, and
3) Modify applicable sections of the TS accordingly.

The CNS PTLR has been prepared in accordance with the NRC-approved methodology of SIR-05-044-A, Revision 1, as contained within BWROG-TP-11-022-A, Revision 1 (Reference 8) and 0900876.401, Revision 0-A contained within BWROG-TP-11-023-A, Revision 0 (Reference 9).

As discussed in Section 5.0, "Conclusion," of the BWROG-TP-11-022-A, Revision 1, NRC Safety Evaluation:

> "Therefore, the NRC staff concludes that LTR [Licensing Topical Report] BWROG-TP-11-022, Revision 1 satisfies the criteria in Attachment 1 to GL 96-03 and provides adequate methodology for BWR licensees to calculate P-T limit curves.
>
> By using this methodology and following the PTLR guidance in GL 96-03, as amended by NRC TSTF-419, BWR licensees will be able to relocate the P-T limit curves and the associated heatup/cooldown rates from the TS to a PTLR, a licensee-controlled document."

As discussed in Section 5.0, "Conclusion," of the BWROG-TP-11-023-A, Revision 0, NRC Safety Evaluation:

> "Based on the evaluation, the NRC staff concludes that LTR BWROG-TP-11-023, Revision 0 provides acceptable methodology for BWR licensees to obtain plant-specific stress intensity factors for an internal pressure load case and a 100 °F/hr thermal ramp load case for use in developing plant-specific P-T limit curves for RPV WLI [Water Level Instrument] nozzles."

As discussed previously, the PTLR incorporates a fluence calculated in accordance with the RAMA code, which has been approved by the NRC for use at CNS (Reference 4), and is in compliance with RG 1.190.

Proposed revisions to applicable sections of the TS have been prepared and are provided in Attachment 2 to this submittal.

In summary, these proposed TS changes associated with the relocation of the current P/T limits to a PTLR, are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419-A, and the guidance contained in the August 4, 2011, NRC letter (Reference 7) which requires the full methodology citation in TS Section 5.6, "Reporting Requirements."

4.2 Precedent

The NRC has approved similar license amendment requests to relocate P/T limits to a PTLR. Examples for BWRs include:

- Nine Mile Point Nuclear Station, Unit 2 (License Amendment No. 145 issued by NRC letter dated May 29, 2014 - ADAMS Accession No. ML14057A554)

- Peach Bottom Atomic Power Station, Units 2 and 3 (License Amendments Nos. 286 and 289 issued by NRC letter dated April 1, 2013 - ADAMS Accession No. ML13079A219)

- Oyster Creek Nuclear Generating Station (License Amendment No. 269 issued by NRC letter dated September 30, 2008 - ADAMS Accession No. ML082390685)

- James A. Fitzpatrick Nuclear Power Plant (License Amendment No. 292 issued by NRC letter dated October 3, 2008 - ADAMS Accession No. ML082630385)

4.3 No Significant Hazards Consideration

10 CFR 50.91(a)(1) requires that licensee requests for operating license amendments be accompanied by an evaluation of no significant hazard posed by issuance of the amendment. Nebraska Public Power District (NPPD) has evaluated this proposed amendment with respect to the criteria given in 10 CFR 50.92(c). The following is the evaluation required by 10 CFR 50.91(a)(1).

NPPD is requesting an amendment of the Operating License for Cooper Nuclear Station to revise the Technical Specification (TS) to relocate the Pressure and Temperature (P/T) Limit Curves to a Pressure and Temperature Limits Report (PTLR).

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment revises the TS by replacing references to existing reactor vessel heatup and cooldown rate limits and P/T limit curves with references to the PTLR. In 10 CFR 50, Appendix G, requirements are established to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants.

Continued use of an Nuclear Regulatory Commission (NRC) approved methodology for calculating P/T limit curves and relocating those curves to a PTLR provide an equivalent level of assurance that RCPB integrity will be maintained, as specified in 10 CFR 50, Appendix G.

The proposed amendment does not adversely affect accident initiators or precursors, and does not alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety functions is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

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

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

   Response: No.

   The relocation of P/T limits to the PTLR is administrative in nature and does not alter or involve any design basis accident initiators. RCPB integrity will continue to be maintained in accordance with 10 CFR 50, Appendix G, and the accident performance of plant structures, systems, and components will not be affected. These changes do not involve any physical alteration of the plant, and installed equipment is not being operated in a new or different manner. Thus, no new failure modes are introduced.

   Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The proposed amendment is administrative in nature and does not affect the function of the RCPB or its response during plant transients. Continuing to calculate the P/T limits using NRC-approved methodology ensures adequate margins of safety relating to RCPB integrity are maintained. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined, there are no changes to set points at which protective actions are initiated, and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22 provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment or environmental impact statement. 10 CFR 51.22(c)(9) identifies an amendment to an operating license for a reactor which changes an inspection or a surveillance requirement as a categorical exclusion provided that operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amount of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

CNS review has determined that the proposed amendment, which would change a TS, does not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluent that might be released offsite, or (3) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1) Technical Specification Task Force Traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," dated August 4, 2003

2) NRC GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996

3) SIA Topical Report No. SIR-05-044-A, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated April 12, 2007

4) Letter from B. Benney (U.S. Nuclear Regulatory Commission) to R. Edington (Nebraska Public Power District), "Cooper Nuclear Station - Issuance of Amendment RE: Revised Pressure Vessel Fluence and Pressure Temperature Curve Applicability to 30 Effective Full-Power Years of Operation (TAC No. MC8728)," dated April 27, 2006

5) Letter from L. Wilkins (U.S. Nuclear Regulatory Commission) to O. Limpias (Nebraska Public Power District), "Cooper Nuclear Station - Issuance of Amendment RE: Revisions to Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits,' for 32 Effective Full Power Years (TAC No. ME7324)," dated February 22, 2013

6) Letter from M. Honcharik (U.S. Nuclear Regulatory Commission) to C. Warren (Nebraska Public Power District), "Cooper Nuclear Station - Issuance of Amendment to Implement the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program (TAC No. MB7209)," dated October 31, 2003

7) Letter from J. Jolicoeur (U.S. Nuclear Regulatory Commission) to Technical Specifications Task Force, "Implementation of Travelers TSTF-363, Revision 0, 'Revise Topical Report References in ITS 5.6.5, COLR (Core Operating Limits Report),' TSTF-408, Revision 1, 'Relocation of LTOP (Low-Temperature Overpressure Protection) Enable Temperature and PORV (Power-Operated Relief Valve) Lift Setting to the PTLR (Pressure-Temperature Limits Report),' and TSTF-419, Revision 0, 'Revise PTLR Definition and References in ISTS (Improved Standard Technical Specification) 5.6.6, RCS (Reactor Coolant System) PTLR'," dated August 4, 2011

8) Licensing Topical Report BWROG-TP-11-022-A, Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013

9) Licensing Topical Report BWROG-TP-11-023-A, Revision 0, "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," dated May 2013

Attachment 2

Proposed Technical Specification Revision
(Markup)

Cooper Nuclear Station, Docket No. 50-298, DPR-46

Revised Pages
Table of Contents pages i-iv, 1.1-4, 3.4-19, 3.4-20, 3.4-21, 3.4-22, 5.0-21

Deleted Pages
3.4-23, 3.4-24, 3.4-25

Renumbered Pages
Original 3.4-26 to new 3.4-23
Original 5.0-22 to new 5.0-23

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

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).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
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: a. The reactor is xenon free;

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits.

specified in the PTLR

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify: a. RCS pressure and RCS temperature are within the applicable limits specified in ~~Figures 3.4.9-1 and 3.4.9-2~~ the curves in the PTLR.; and b. RCS heatup and cooldown rates are ~~≤ 100°F when averaged over a one hour period.~~ within limits specified in the PTLR.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

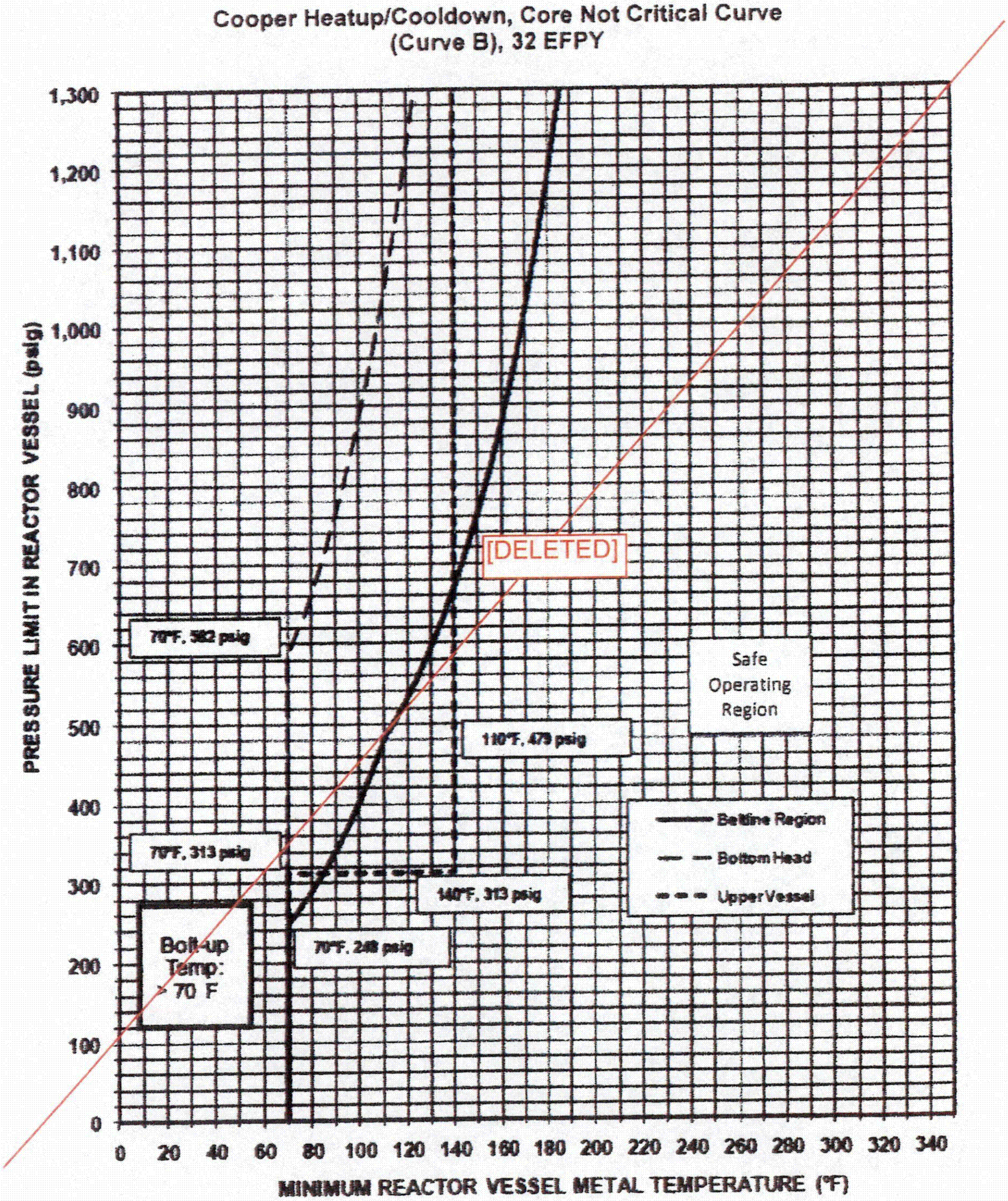
SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~Figure 3.4.9-3.~~ the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	--------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~≤ 145°F.~~ within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	--------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~≤ 50°F.~~ within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

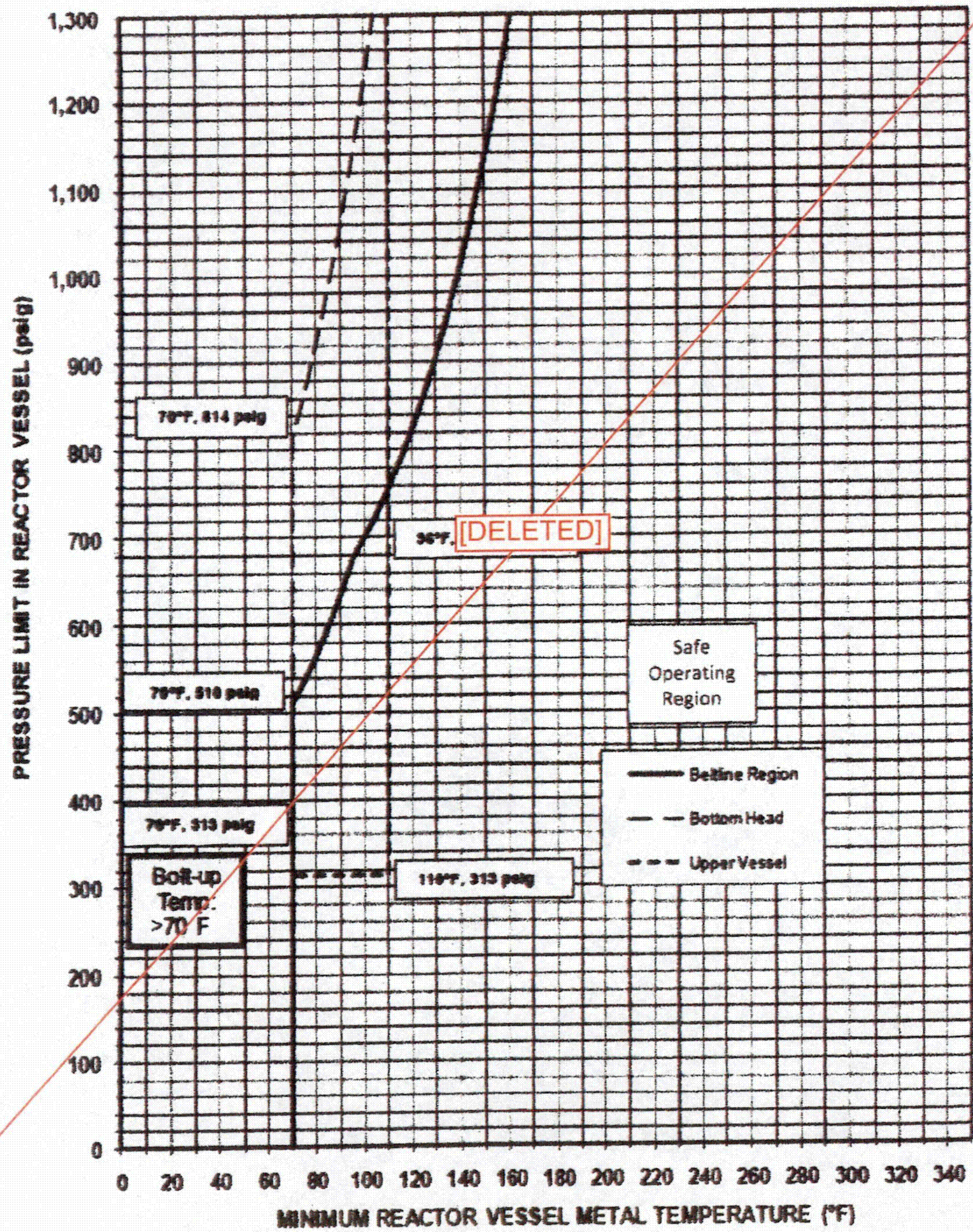
SURVEILLANCE		FREQUENCY
SR 3.4.9.5	-------------------------NOTE------------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F~~. within the limits specified in the PTLR.	30 minutes
SR 3.4.9.6	-------------------------NOTE------------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F~~. within the limits specified in the PTLR.	30 minutes
SR 3.4.9.7	-------------------------NOTE------------------------- Not required to be performed until 12 hours after RCS temperature ≤ 90°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F~~. within the limits specified in the PTLR.	12 hours

Cooper Heatup/Cooldown, Core Not Critical Curve
(Curve B), 32 EFPY

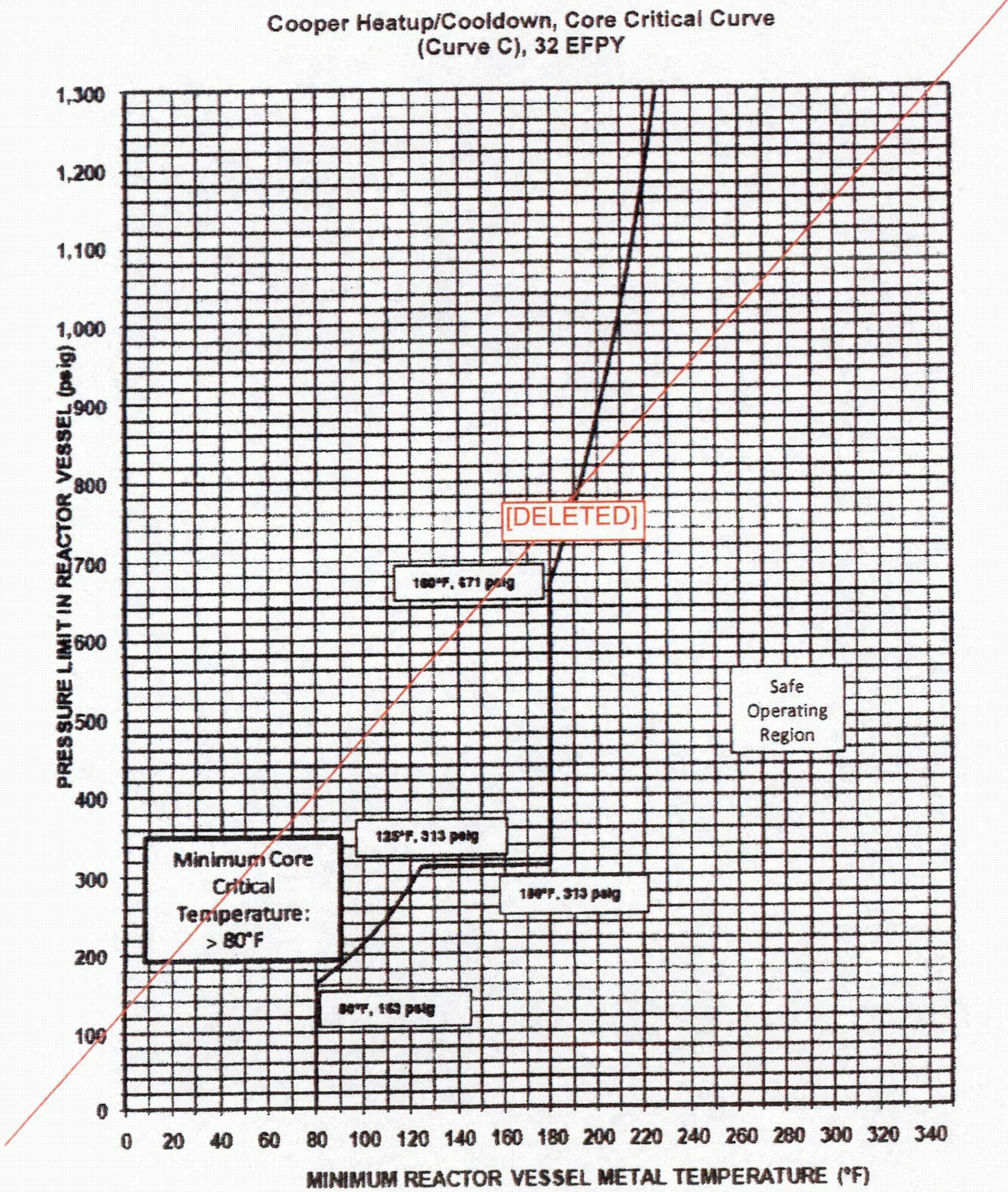


~~Figure 3.4.9-1~~
~~Pressure/Temperature Limits for~~
~~Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown~~
~~Valid Through 32 EFPY~~

Cooper Pressure Test Curve (Curve A), 32 EFPY



~~Figure 3.4.9-2~~
~~Pressure/Temperature Limits for~~
~~Inservice Hydrostatic and Inservice Leakage Tests~~
~~Valid Through 32 EFPY~~



~~Figure 3.4.9-3~~
~~Pressure/Temperature Limits for Criticality~~
~~Valid Through 32 EFPY~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1020 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify reactor steam dome pressure is ≤ 1020 psig.	12 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. NEDE-23785-1-P-A, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident", Volume III, Revision 1, October 1984.

3. NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology" (the approved Revision at the time the reload analysis is performed).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. Reactor pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
   1. Limiting Conditions for Operations Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."
   2. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."
b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
   1. BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.
   2. BWROG-TP-11-023-A, Revision 0 (0900876.401, Revision 0-A), "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," dated May 2013.
c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the deep dose equivalent in excess of 100 mrem but less than 1000 mrem in one hour (measurement made at 12 inches from source of radiation) shall be barricaded (barricade will impede physical movement across the entrance or access to the high radiation area; i.e., doors, yellow and magenta rope, turnstile) and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the SWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A monitoring device which continuously indicates the radiation dose rate in the area.

b. A monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures), with a dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic dose rate monitoring at the frequency specified by Health Physics supervision.

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with dose rates such that a major portion of the body could receive in 1 hour a deep dose equivalent in excess of 1000 mrem (measurement made at 12 inches from source of radiation) shall be provided with locked doors to prevent unauthorized entry. Doors shall remain locked except during periods of access by personnel under an approved SWP which shall specify the dose rates in the immediate work area. For individual high radiation areas accessible to personnel that are located within large areas, such as the containment, or areas where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, then that area shall be barricaded and conspicuously posted. Area radiation monitors that have been set to alarm if radiation levels increase,

(continued)

Attachment 3

Proposed Technical Specification Revision
(Final Typed Format)

Cooper Nuclear Station, Docket No. 50-298, DPR-46

Revised Pages
Table of Contents pages i-iii, 1.1-4, 3.4-19, 3.4-20, 3.4-21, 3.4-22, 5.0-21

New Page
5.0-22

Renumbered Pages
Original 3.4-26 to new 3.4-23
Original 5.0-22 to new 5.0-23

1.1 Definitions

LOGIC SYTEM 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).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
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: a. The reactor is xenon free;

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------------NOTE-------------- Required Action A.2 shall be completed if this Condition is entered. ------------------------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 <u>AND</u> A.2	Restore parameter(s) to within limits. Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 <u>AND</u> B.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------------NOTE-------------- Required Action C.2 shall be completed if this Condition is entered. -------------------------------------	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-----------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --	
	Verify:	30 minutes
	a. RCS pressure and RCS temperature are within the applicable limits specified in the curves in the PTLR; and	
	b. RCS heatup and cooldown rates are within limits specified in the PTLR.	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. --- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	-----------------------------NOTE----------------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. --- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	30 minutes
SR 3.4.9.6	-----------------------------NOTE----------------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	30 minutes
SR 3.4.9.7	-----------------------------NOTE----------------------------- Not required to be performed until 12 hours after RCS temperature ≤ 90°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1020 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1	Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1020 psig.	12 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. NEDE-23785-1-P-A, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident", Volume III, Revision 1, October 1984.

3. NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology" (the approved Revision at the time the reload analysis is performed).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. Reactor pressure and temperature limit for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Limiting Conditions for Operations Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

2. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(continued)

5.6 Reporting Requirements

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT PTLR) (continued)

1. BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.

2. BWROG-TP-11-023-A, Revision 0 (0900876.401, Revision 0-A), "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," dated May 2013.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the deep dose equivalent in excess of 100 mrem but less than 1000 mrem in one hour (measurement made at 12 inches from source of radiation) shall be barricaded (barricade will impede physical movement across the entrance or access to the high radiation area; i.e., doors, yellow and magenta rope, turnstile) and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the SWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A monitoring device which continuously indicates the radiation dose rate in the area.

b. A monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures), with a dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic dose rate monitoring at the frequency specified by Health Physics supervision.

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with dose rates such that a major portion of the body could receive in 1 hour a deep dose equivalent in excess of 1000 mrem (measurement made at 12 inches from source of radiation) shall be provided with locked doors to prevent unauthorized entry. Doors shall remain locked except during periods of access by personnel under an approved SWP which shall specify the dose rates in the immediate work area. For individual high radiation areas accessible to personnel that are located within large areas, such as the containment, or areas where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, then that area shall be barricaded and conspicuously posted. Area radiation monitors that have been set to alarm if radiation levels increase, provide both a visual and an audible signal to alert personnel in the area of the increase. These monitors may be used to meet Specification 5.7.1.a provided that the dose rates and alarms have been established by radiation protection personnel. Stay times or continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide additional positive exposure control over the activities within the area.

Attachment 4

Proposed Technical Specification Bases Revision
(Information Only)

Cooper Nuclear Station Docket No. 50-298, DPR-46

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

~~This Specification~~ contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, criticality, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The P/T limit curves apply to the reactor pressure vessel, since the vessel is the component most subject to brittle failure, and is bounding over other SSCs that comprise the reactor coolant pressure boundary. The fluid temperatures of an idle recirculation loop are not representative of reactor vessel conditions and therefore the P/T limit curves do not apply to an idle recirculation loop.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2). The NRC has also approved the use of alternate fracture toughness curves for establishing these limits (Ref. 10).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

The elements of this LCO are:

within limits specified in the PTLR

within limits specified in the PTLR

the PTLR

within limits specified in the PTLR

a. RCS pressure and temperature (Beltline, Bottom Head, and Upper Vessel) are ~~within the applicable limits of Figure 3.4.9-1 and Figure 3.4.9-2~~, and heatup or cooldown rates are ~~≤ 100°F when averaged over a one hour period~~ during RCS heatup, cooldown, and inservice leak and hydrostatic testing (The Adjusted Reference Temperature (ART) beltline region must be determined from ~~Figure 3.4.9-2~~. During RCS heatup and cooldown operation (i.e., not critical and not performing inservice leak or hydrostatic testing) verify RCS pressure and temperature are ~~within the applicable limits specified in Figure 3.4.9-1~~. During RCS inservice leak and hydrostatic testing verify RCS pressure and temperature are ~~within the applicable limits specified in Figure 3.4.9-2;~~

within limits specified in the PTLR

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~≤ 145°F~~ during recirculation pump startup;

within limits specified in the PTLR

within limits specified in the PTLR

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ~~≤ 50°F~~ during recirculation pump startup;

the PTLR

d. RCS pressure and temperature are within the criticality limits specified ~~in Figure 3.4.9-3~~, prior to achieving criticality; and

within limits specified in the PTLR

e. The reactor vessel flange and the head flange temperatures are ~~≥ 70°F~~ when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the reactor pressure vessel integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

the PTLR limits

Verification that operation is within ~~RCS pressure, RCS temperature, and RCS heatup and cooldown rate limits by monitoring the bottom head drain, recirculation loop temperatures, and RPV metal temperatures (Beltline, Bottom Head, and Upper Vessel)~~ is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the difference between any two readings taken over a 45 minute period is less than 50°F.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing. During RCS heatup and cooldown operation (i.e., not critical and not performing inservice leak or hydrostatic testing) verify RCS pressure and temperature are ~~within the applicable limits specified in Figure 3.4.9-1~~ within limits specified in the PTLR. During RCS inservice leak and hydrostatic testing verify RCS pressure and temperature are ~~within the applicable limits specified in Figure 3.4.9-2~~ within limits specified in the PTLR.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the specified limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 9) are satisfied.