February 6, 2008

Mr. Stewart B. Minahan
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - REQUEST FOR RELIEF NO. RI-29 FOR FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL REGARDING VOLUMETRIC EXAMINATION OF REACTOR PRESSURE VESSEL CIRCUMFERENTIAL SHELL WELDS (TAC NO. MD5260)

Dear Mr. Minahan:

By letter dated April 18, 2007, as superseded by letter dated October 15, 2007, Nebraska Public Power District (the licensee) submitted Request for Relief No. RI-29, from certain requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), for the fourth 10-year inservice inspection (ISI) program interval at Cooper Nuclear Station (CNS). Specifically, the licensee requested permanent relief from the requirements to perform a volumetric examination of reactor pressure vessel (RPV) circumferential shell welds during the fourth 10-year ISI interval, which commenced on March 1, 2006, and applies to the remaining portion of the operating license. The applicable ASME Code for this interval at CNS is the 2001 Edition through the 2003 Addenda.

The proposed alternative eliminates the required examination of RPV circumferential shell welds, and retains the requirement for examination of longitudinal (axial) welds in the RPV shell. Volumetric examination of the axial RPV shell welds (ASME Code, Section XI, IWB-2500, Examination Category B-A, Item No. B1.12) shall be performed for 100 percent of the accessible welds. Examination of the axial welds shall also include those portions of the circumferential welds that intersect the axial welds.

The alternative was proposed pursuant to the provisions of paragraph 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), and is consistent with the staff's Safety Evaluation Report of the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05 report issued July 28, 1998, and the guidance provided by U.S. Nuclear Regulatory Commission (NRC) Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998. Based on the information you provided, the NRC staff concludes that the alternative you have proposed will provide an acceptable level of quality and safety for the remaining term of the CNS operating license. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The detailed results of the NRC staff's review are provided in the enclosed safety evaluation. If you have any questions concerning this matter, please call Mr. F. Lyon of my staff at (301) 415-2296.

Sincerely,

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure: Safety Evaluation

cc w/encl: See next page

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

REQUEST FOR RELIEF NO. RI-29

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated April 18, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071130013), as superseded by letter dated October 15, 2007 (ADAMS Accession No. ML072910269), Nebraska Public Power District (the licensee) submitted Request for Relief No. RI-29 for Cooper Nuclear Station (CNS). Specifically, the licensee requested relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, requirements related to examination of reactor pressure vessel (RPV) circumferential shell welds at CNS. The licensee requested authorization to use a proposed alternative in accordance with technical report Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05, “Boiling Water Reactor Vessel and Internals Project, BWR RPV Shell Weld Inspection Recommendations” (Reference 1), and U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 98-05, “Boiling Water Reactor Licensees Use of The Boiling Water Reactor Vessel and Internals Project-05 Report to Request Relief from Augmented Examinations Requirements on Reactor Pressure Vessel Circumferential Shell Welds” (Reference 2), to the RPV circumferential shell welds examination requirements of the ASME Code, Section XI for the fourth 10-year inservice inspection (ISI) program interval. The fourth 10-year ISI interval at CNS commenced on March 1, 2006, and applies to the remaining portion of the current operating license.

2.0 REGULATORY EVALUATION

ISI of ASME Code Class 1, 2, and 3 components is performed in accordance with the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," and applicable Addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that proposed alternatives to the requirements of paragraph (g) may be used, when authorized by the staff, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), components (including supports) which are classified as ASME Code Class 1, 2 and 3 shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 120-month interval and subsequent intervals comply with the requirements in the latest Edition and Addenda of ASME Code, Section XI incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

Paragraph 50.55a(g)(6)(ii)(A)(2) of 10 CFR requires that all licensees augment their RPV examination by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for RPV shell welds specified in Item No. B1.10, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," of Table IWB-2500-1 to ASME Code, Section XI. Additionally, 10 CFR 50.55a(g)(6)(ii)(A)(2) requires that the examinations cover essentially 100 percent of the RPV shell welds.

Paragraph 50.55a(g)(6)(ii)(A)(2) of 10 CFR defines an "essentially 100 percent" examination as covering 90 percent or more of the examination volume of each weld. The schedule for implementation of the augmented inspection is dependent upon the number of months remaining in the 10-year ISI interval that was in effect on September 8, 1992.

The applicable Section XI ASME Code of record for CNS is the 2001 Edition with the 2003 Addenda.

2.1 Regulatory Background

2.1.1 BWRVIP-05 Report

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998, the BWRVIP submitted the BWRVIP-05 report to the NRC for staff review. This report evaluated the current inspection requirements for RPV shell welds in BWRs, formulated recommendations for alternative inspection requirements, and provided a technical basis for these recommended requirements. As modified, the BWRVIP-05 report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of 100 percent of the axial (i.e., longitudinal) welds and essentially zero percent of the circumferential RPV shell welds, except for the intersections of the axial and circumferential shell welds. In addition, the report included proposals to provide alternatives to the ASME Code, Section XI requirements for successive and additional examinations of circumferential shell welds, provided in paragraphs IWB-2420 and IWB-2430, respectively, of the ASME Code, Section XI.

On July 28, 1998, the staff issued a Safety Evaluation Report (SER) for the BWRVIP-05 report[1] (Reference 3). This evaluation concluded that the failure frequency of RPV circumferential shell welds in BWRs was sufficiently low to justify elimination of ISI of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential shell welds were acceptable. The evaluation indicated that examination of the circumferential shell welds shall be performed if axial shell weld examinations reveal an active degradation mechanism.

In the BWRVIP-05 report, the BWRVIP committee concluded that the conditional probabilities of failure for BWR RPV circumferential shell welds are orders of magnitude lower than that of the axial shell welds. As a part of its review of the report, the staff conducted an independent probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The staff's assessment conservatively calculated the conditional probability of failure from RPV axial and circumferential shell welds during the initial (current) 40-year license period and at conditions approximating an 80-year RPV lifetime for a BWR nuclear plant, as indicated respectively in Tables 2.6-4 and 2.6-5 of the staff's July 28, 1998, SER. The failure frequency for an RPV is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The staff determined the conditional probability of failure for axial and circumferential shell welds in BWR RPVs fabricated by Chicago Bridge and Iron, Combustion Engineering (CE), and Babcock and Wilcox. The analysis identified a low temperature overpressurization (LTOP) event that occurred in a foreign reactor as the limiting event for BWR RPVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics (PFM) calculations. The staff estimated that the probability for the occurrence of the LTOP transient was 1×10^{-3} per reactor-year. For each of the RPV fabricators, Table 2.6-4 of the staff's SER identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) after the initial 40-year license period.

2.1.2 Generic Letter 98-05

On November 10, 1998, the NRC staff issued Generic Letter (GL) 98-05, which states that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RPV shell welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Circumferential Shell Welds") by demonstrating that:

(1) At the expiration of the license, the circumferential shell welds will continue to satisfy the limiting conditional failure probability for circumferential shell welds in the staff's July 28, 1998, SER, and

[1] The staff has identified that in some instances the staff SER is referenced as dated on July 28, 1998, others on July 30, 1998. For clarification purposes the staff notes that this SER is a letter addressed to Carl Terry, BWRVIP Chairman, dated July 28, 1998, and titled "Final SER of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)."

(2) Licensees have implemented operator training and established procedures that limit the frequency of LTOP events to the amount specified in the staff's July 28, 1998, SER.

Licensees will still need to perform the required inspections of “essentially 100 percent” of all axial shell welds.

3.0 TECHNICAL EVALUATION

3.1 ASME Code and Regulatory Requirements for Which Relief is Requested

The licensee requested relief from the following requirements of the ASME Code, Section XI, 2001 Edition with the 2003 Addenda 1992 Edition through portions of 1993 Addenda:

> Subarticle IWB-2500, Table IWB 2500-1, Examination Category B-A, Item No. B1.11, “Pressure Retaining Welds in Reactor Vessel.”

In addition to the ASME Code, Section XI volumetric examinations, licensees are required by 10 CFR 50.55a(g)(6)(ii)(A) to perform a one-time volumetric examination (augmented examination) of the RPV shell welds. This augmented examination of the RPV shell welds is required to ensure structural reliability of the RPV shell welds. The licensee requested relief from the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A) for the CNS RPV circumferential shell welds.

This relief is requested for the following components:

> ISI Class 1, Examination Category B-A, Item No. B1.11, “Circumferential Shell Welds.” This relief is applicable to circumferential RPV shell welds VCB-BB-1, VCB-BA-2, VCB-BB-3, and VCB-BB-4.

3.2 Licensee’s Proposed Alternative to the ASME Code (as stated)

Pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), and consistent with the guidance provided in NRC Generic Letter (GL) 98-05, Nebraska Public Power District requests NRC approval for relief from the examination of RPV circumferential shell welds required by ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11. The proposed alternative consists of permanent relief from the requirement to perform a volumetric examination of the RPV circumferential shell welds, listed above, for the Fourth Ten-year ISI interval at CNS, which applies to the remainder of the current operating license.

As required by Table IWB-2500-1 of ASME Section XI, examination of the longitudinal (axial) RPV shell welds (Examination Category B-A, Item No. B1.12) will be performed for 100% of the accessible welds. Axial weld examination includes that portion of the circumferential welds that intersect each axial weld, or approximately 2% to 3% of the total length of the circumferential welds.

The procedure and personnel employed for these examinations will meet the requirements of ASME Section XI, Appendix VIII, as required by ASME Section XI, 2001 Edition, 2003 Addenda, subject to the limitations set forth by 10 CFR 50.55a(b)(2)(xxiv).

3.3 Licensee's Basis for Alternative (as stated)

GL 98-05 discusses that BWR licensees may request permanent relief (i.e., for the remaining term of operation under the initial license) from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV circumferential shell welds (ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11, Circumferential Shell Welds) by demonstrating that: (1) at the end of their license the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds specified in the NRC staff's July 28, 1998, safety evaluation, and (2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's July 28, 1998, SER. Licensees must continue to perform the required inspections of "essentially 100 percent" of all axial welds.

The basis for this request for relief is documented in the BWRVIP-05 report that was transmitted to the NRC on September 28, 1995, and supplemented by letters dated June 24, and October 29, 1996, May 16, June 4, June 13 and December 18, 1997, and January 13, 1998. The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR RPV circumferential shell welds. The BWRVIP-05 report concludes that the probability of failure of the BWR RPV circumferential shell welds is orders of magnitude lower than that of the axial shell welds. The NRC staff has conducted an independent assessment of the analysis contained in BWRVIP-05 as documented in the final SER of the BWRVIP-05 report. This assessment concluded that the probability of failure of the BWR RPV circumferential shell welds is orders of magnitude lower than that of the axial shell welds, and that the added risk caused by not inspecting the circumferential shell welds is negligible. Additionally, the NRC assessment demonstrated that inspection of BWR RPV circumferential shell welds does not measurably affect the probability of failure. Therefore, the NRC evaluation supports the conclusions of the BWRVIP-05 report.

This NRC independent assessment (Reference 3) utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are: (1) the neutron fluence was the estimated end-of-license mean fluence, (2) the chemistry values are mean values based on vessel types, and (3) the potential for beyond design basis events is considered. Although BWRVIP-05 and subsequent transmittals provide the technical basis supporting the relief request, the following information is provided to show the conservatisms of the NRC analysis relative to the CNS RPV. Results of BWRVIP analyses applicable to CNS are also provided in the discussion below.

Generic Letter 98-05, Criterion 1

Demonstrate at the end of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998, safety evaluation.

For plants with RPVs fabricated by Combustion Engineering (CE), such as CNS, the peak neutron fluence used in the NRC PFM analysis was 2.0×10^{18} n/cm^2 (E>1.0 MeV). However, at CNS, the peak neutron fluence for the RPV is located at the vessel interior wall, and is anticipated to be 1.57×10^{18} n/cm^2 at the end of the current license period. This value is also used as the basis of the current CNS Pressure-Temperature (P-T) curves before being adjusted to one-quarter vessel wall thickness. CNS recalculated the neutron fluence values using the NRC approved RAMA fluence methodology in accordance with the guidelines presented in NRC Regulatory Guide 1.190 (Reference 4) and got a slightly higher estimated fluence value of 1.67×10^{18} n/cm^2 (E>1.0 MeV) at an assumed end-of-license of 32 Effective Full Power Years (EFPY).

The CNS data used for the evaluation based on the BWRVIP-05 methodology are shown in Table 1. This table illustrates that CNS specific information is comparable to that used in Reference 3. The information and results for CNS are provided using the following two approaches from Regulatory Guide 1.99, Revision 2: 1) Position C.1, surveillance data not available, and 2) Position C.2, surveillance data available. As can be seen from Table 1, using Position C.1, the mean adjusted reference temperature (ART) of 37.65 °F [degrees Fahrenheit] is considerably lower than the NRC safety assessment acceptance criteria of 86.4 °F. Using Position C.2 results in a mean ART of 87.61 °F, which is in closer agreement to the NRC safety assessment acceptance criteria of 86.4 °F. Note that the Position C.2 method applies a 57% correction to the ΔRTNDT Position C.1 results.

Table 1

Parameter Description	**CNS Comparative Parameters at End of License (30 EFPY for Bounding Circumferential Weld VCB-BA-2 (4)) Using BWRVIP Methodology (3)**		**NRC Staff Assessment (2) Safety Assessment "VIP"**
Fluence, n/cm^2	1.57×10^{18}		2.0×10^{18}
Initial RTNDT, °F	-50		0
Cu %	0.20		0.13
Ni %	0.69		0.71
Chemistry Factor	175.30		151.7
RG 1.99, Regulatory Position	C.1, Surveillance Data Not Available	C.2, Surveillance Data Available	
Δ RTNDT (°F)	87.65	137.61	86.4
Mean ART (°F) (1)	37.65	87.61	86.4

Footnotes:

(1) Margin term in RG [Regulatory Guide] 1.99 neglected for consistency with NRC assessment therefore Mean ART = Initial RTNDT + ΔRTNDT.

(2) Values based on BWRVIP-05 chemistry information.

(3) Values derived from CNS specific information.

(4) Values correspond to Part No. 1-240, Heat No. 21935, the lower to lower-intermediate girth weld.

The NRC used the data presented in Table 1 to perform probabilistic fracture mechanics calculations using the FAVOR code. Results of this evaluation showed that the conditional probability of failure was 2.81×10^{-5} at 32 EFPY for vessels fabricated by CE (7.03×10^{-7} on a per calendar year basis). The conditional probability of failure was less than 1.0×10^{-6} (no failures in the indicated number of vessel simulations) at 32 EFPY using the BWRVIP-05 methodology and the limiting plant. The NRC evaluation used a frequency of 1×10^{-3}/yr for an overpressure event. This results in a total probability of failure of 7.03×10^{-10}/yr. As presented in the final safety evaluation, NUREG-1560, Vol. 1, core damage frequencies (CDF) for BWR plants were reported to be approximately 10^{-7}/yr to 10^{-4}/yr. In addition, Regulatory Guide (RG) 1.174, Revision 1, indicates that continued operation is acceptable if the plant-specific analyses predict the mean frequency for through wall crack penetration for pressurized thermal shock events is less than $1x10^{-6}$/yr. Since the failure frequency for the CE-fabricated plants due to elimination of circumferential weld examinations contributes less than the allowable amount of change of large early release frequency and CDF, as discussed in RG 1.174, the failure frequency for RPV circumferential welds is sufficiently low to justify elimination of inservice inspection of the subject welds.

A comparison of the mean ART values in Table 1 shows that the CNS value, using Position C.2, is essentially the same as the value derived by the NRC with regards to probabilistic fracture mechanics consideration, for CE units, with margin terms from Regulatory Guide 1.99 neglected for consistency. The probability of failure for CNS and the NRC SER are similar because the mean ART is essentially the same and probability for failure is extremely small ($7.03x10^{-10}$/yr). Therefore, the conclusions stated in the SER are applicable to CNS. Both analyses conclude that the failure probability associated with circumferential welds is extremely small, and that it is orders of magnitude less than that for axial welds. At the expiration of the CNS license, the circumferential welds are expected to satisfy the limiting conditional failure probability for circumferential welds discussed in the NRC staff's July 28, 1998, safety evaluation.

Generic Letter 98-05, Criterion 2

Demonstrate licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's July 28, 1998, safety evaluation.

The NRC staff indicated that the potential for, and consequences of, non-design basis events not addressed in the BWRVIP-05 report should be considered. In particular, the NRC staff stated that non-design basis, cold overpressure transients should be considered. It is highly unlikely that a BWR would experience a cold overpressure transient. The NRC staff described several types of events that could be precursors to BWR RPV cold, over-pressure transients. These were identified as precursors because no cold overpressure event has occurred at a domestic BWR. Also, the NRC staff identified one actual cold overpressure event that occurred during shutdown at a foreign BWR. This event apparently involved several operational errors that resulted in a maximum RPV pressure of 1150 psi [pounds per square inch] with a temperature of 88 °F. The BWRVIP responded with the conclusion that condensate and Control Rod Drive (CRD) pumps could cause conditions that could lead to cold overpressure events. This is summarized in the Final SER for BWRVIP-05.

CNS has in place procedures which monitor and control reactor pressure, temperature, and water inventory during cold shutdown which would minimize the likelihood of a Low Temperature Over-Pressurization (LTOP) event.

The pumps in the High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) systems, as well as the Reactor Feed Pumps of the Feedwater (FW) system, are steam turbine driven. During reactor cold shutdown conditions, only auxiliary steam is available for operation of these systems. Auxiliary steam is isolated from the steam supply lines for these turbines by isolation valves and spool pieces that are normally removed. The HPCI and RCIC steam spool pieces are only installed to support HPCI and RCIC overspeed testing. When HPCI or RCIC overspeed testing is performed with auxiliary steam, the respective pump is uncoupled during the test and the auxiliary steam spool piece is removed upon completion of the test. Therefore, it is unlikely for these systems to contribute to an over-pressurization event while the unit is in cold shutdown.

The Core Spray (CS) system is a low-pressure water source to the RPV and only injects when the reactor pressure is below 400 psig [pounds per square inch gauge]. The CS pumps have a discharge pressure of about 400 psig. An inadvertent initiation of one or both CS subsystem(s) has the potential to pressurize the RPV above the limits of Technical Specification Heatup/Cooldown, Core Not Critical Curve. This curve permits pressure (upper vessel) up to about 313 psig at temperatures up to 80 °F. Between 80 °F and 140 °F, the pressure permitted by Technical Specifications remains constant at about 313 psig. During refueling outages, there is typically only a very short period of time during RPV head de-tensioning and following head re-tensioning in which an overpressurization event could occur. Forced outages are also typically of a short duration. Procedural controls and the short period of time when the RPV coolant temperatures are low and the head is not de-tensioned limits the probability for an overpressurization due to an inadvertent actuation of one or both CS subsystems.

During cold shutdown conditions, the condensate booster pumps of the Condensate system are shutdown. It would require direct operator action to start a Main Condensate Booster pump and inject into the vessel.

During Mode 4 conditions, RPV level and pressure are normally controlled with the CRD and Reactor Water Cleanup (RWCU) or Residual Heat Removal blowdown when in shutdown cooling systems using a “feed and bleed” process. The RPV is not taken water solid during these times, and plant procedures require opening of the head vent valves after the reactor has been cooled to less than 212 °F. If either of these systems were to fail, the operator would adjust the remaining system(s) to control level. Under these conditions, the CRD system typically injects water into the reactor at the rate of approximately 30 to 50 gpm [gallons per minute]. This slow injection rate allows the operator sufficient time to react to unanticipated level changes and, thus, significantly reduces the possibility of an event that would result in a violation of the pressure-temperature limits.

The Standby Liquid Control (SLC) system is another high-pressure water source to the RPV. However, there are no automatic starts associated with the system. SLC injection requires an operator to manually start the system from the control room or from the local test station. Additionally, the injection rate of the SLC pump is approximately 86 gpm, which would give the operator ample time to control reactor pressure in the event of an inadvertent injection.

Pressure testing of the RPV is classified as an “Infrequently Performed Test or Evolution.” This ensures that these tests receive special management oversight and procedural controls to maintain the plant’s level of safety within acceptable limits. The pressure test is conducted so that the required temperature bands for the pressure increases are achieved and maintained prior to increasing pressure. During performance of an RPV pressure test, level and pressure are controlled using the CRD and RWCU systems in a “feed and bleed” process. An alternate reactor water rejection path using main steam line drains can be established for pressure reduction capability in event that RWCU becomes unavailable. Pressure increase is limited to less than 50 psig per minute. This practice minimizes the likelihood of exceeding the pressure-temperature limits during performance of the test.

Procedural Controls/Operator Training to Prevent RPV Cold Over-Pressurization

Operating procedural restrictions, operator training, and work control processes at CNS provide appropriate controls to minimize the potential for RPV cold overpressurization events.

During cold shutdown conditions, reactor water level, pressure, and temperature are maintained within established bands in accordance with procedures. The operations procedure governing control room activities requires that operators frequently monitor for indications and alarms to detect abnormalities as early as possible. This procedure also requires that the Shift Manager or Control Room Supervisor be notified immediately of any changes or abnormalities in indications. Therefore, any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected.

Procedural controls for reactor temperature, level, and pressure are an integral part of operator training. Specifically, operators are trained in methods of controlling water level within specified limits, as well as responding to abnormal water level conditions outside the established limits. Additionally, operators receive training on brittle fracture limits and compliance with the Technical Specification pressure-temperature limits curves. Plant-specific procedures provide guidance to the operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

At CNS, outage work items are scheduled by the outage scheduler. Senior Reactor Operators (SROs) are assigned to provide oversight of outage schedule development to avoid conditions which could adversely impact reactor water level, pressure, or temperature. From the outage schedule, a daily schedule is developed listing the work activities to be performed. These daily schedules are reviewed and approved by SROs and a copy is maintained in the control room.

During outages, work is coordinated through the Work Control Center and Outage Control Center. This provides an additional level of operations oversight. The control room operators are required to provide positive control of reactor water level and pressure within the specified band, including restoration actions being taken. Pre-job briefings are conducted for complex work activities, such as RPV pressure tests or hydrostatic testing that have the potential to cause an overpressurization event. Pre-job briefings are attended by the cognizant individuals involved in the work activity.

Based on the above discussion, the frequency of cold overpressure events is limited to the amount specified in the NRC staff's July 28, 1998, safety evaluation.

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Licensees will also need to perform the required inspection of "essentially 100 percent" of all axial welds.

NPPD [the licensee] examined the accessible regions of the axial welds during the Cycle 18 Refueling Outage in the Third ISI Ten-year Interval. The inability to inspect 100% of the axial welds was due to the presence of obstructions such as the guide rods (see Third Interval Relief Request RI-06, Revision 2, approved by NRC, TAC No. MB2003). In addition to the axial weld coverage, the accessible portions of the circumferential welds were inspected and achieved considerably more than the 2-3% stated in the NRC Final SER for BWRVIP-05 (TAC

No. M93925). As stated in proposed alternative, CNS will examine 100% of the reactor vessel axial welds and the associated 2% to 3% of the intersecting circumferential accessible welds, as required by ASME Section XI, Table IWB-2500-1.

4.0 STAFF EVALUATION

The staff's review focused on confirming that the licensee has adequately documented that the conditions for relief outlined in the SER to the BWRVIP-05 report and GL 98-05 are satisfied.

GL 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," describes the criteria for circumferential weld inspection relief using the methodology in BWRVIP-05.

GL 98-05 allows BWR licensees to request permanent (i.e., for the remaining term of operation under the existing, initial license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, “Circumferential Shell Welds”) by demonstrating that:

(1) At the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff’s July 30, 1998 safety evaluation, and

(2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff’s July 30, 1998, safety evaluation.

The failure probability of the circumferential welds increases with neutron irradiation, and therefore, the fluence used in the estimation of the failure probability is that estimated for the end of the current license. The BWRVIP-05 provides the technical basis for eliminating inspection of the shell circumferential welds. In its evaluation of BWRVIP-05, the NRC staff conducted an independent risk-informed assessment of the circumferential weld failure probability documented in the “Final Safety Evaluation of the BWR Vessel Internals Project BWRVIP-05 Report.” In this evaluation, the staff calculated bounding fluence values for all BWR plants for which the risk is acceptable. The fluence value corresponding to this risk value is also the bounding value for acceptable risk of a specific plant.

The purpose of the staff’s review in sections 4.1 and 4.2 is to: (1) establish that the fluence estimated for the end of the current operating license for CNS is acceptable, i.e., is bounded by the staff calculated value and (2) that operator training and operating procedures have been established to minimize the probability of cold overpressurization and/or prevent such events and address the probable causes of cold overpressurization.

4.1 GL 98-05, Criterion 1

The end of the current license peak vessel fluence for CNS has been calculated using the RAMA code that has been approved by the staff and adheres to the guidance in RG 1.190 (Reference 5). The peak end of license value was calculated to be 1.67×10^{18} n/cm^2. The staff’s plant-specific limiting analysis fluence value to the end of the current license is $2.0x10^{18}$ n/cm^2. Comparison of the above results indicates that the plant fluence value is significantly lower than the staff estimated limiting value and, therefore, the risk for circumferential weld rupture is acceptable.

4.2 GL 98-05, Criterion 2

To satisfy the second condition of GL 98-05 regarding a cold overpressure event, the licensee provided analysis of the potential high-pressure injection sources, administrative controls to prevent such events, and associated operator training.

The licensee noted that for a cold overpressurization event to occur, a series of operator errors are required. This conclusion comes from the two cold overpressure events ever recorded (both occurred in foreign plants). The licensee proposes to use operating procedures and operator training as barriers to cold overpressurization.

4.2.1 Procedural Controls to Prevent Low Temperature Overpressure Events

CNS has in place procedures that monitor reactor pressure, temperature and vessel water inventory during cold shutdowns. Monitoring allows reactor operators to control and minimize the likelihood of cold overpressurization.

4.2.2 High-Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC)

The pumps in both systems are auxiliary steam driven. During cold shutdown, auxiliary steam is isolated from the steam supply lines by isolation valves and spool pieces that are normally removed. Therefore, it is highly unlikely that the HPCI and the RCIC will contribute to cold overpressurization.

4.2.3 The Core Spray System (CSS)

The CSS provides low-pressure water to the reactor vessel and only injects at pressures below 400 psig. At cold shutdown and at temperatures below 140 °F, the maximum allowable pressure is 313 psig. Therefore, the potential for overpressurization exists, but only for a very short period of time during tensioning and detensioning of the vessel head. However, because the time periods are very short and with the operational procedures in place, such overpressurization is very unlikely.

4.2.4 Condensate Booster Pumps (CBPs)

During cold shutdown conditions, the CBPs are not operating. It would require specific operator action to activate the CBPs to inject into the pressure vessel. Operating procedures make such an inadvertent activation very unlikely.

4.2.5 The Control Rod Drive (CRD) Cooling and the Reactor Water Cleanup (RWCU) System

During Mode 4, vessel water level and pressure are controlled with the CRD and the RWCU systems in the feed-and-bleed mode. The vessel is not taken water solid at this time and plant procedures require that the head valves be open at temperatures at or less than 212 °F. Should either of the systems fail, the operator would adjust the remaining system to control level. The CRD injects at a low rate of 30 to 50 gpm giving the operator sufficient time to respond. This minimizes the possibility of a cold overpressurization.

4.2.6 The Standby Liquid Control (SLC) System

The SLC system is a high-pressure system able to inject into the pressure vessel. However, the SLC system requires operator manual action from the control room to inject. In addition, the maximum injection rate is 86 gpm, thus affording the operator sufficient time to prevent overpressurization. Therefore, it is highly unlikely that the SLC would cause a cold overpressurization.

4.2.7 Reactor Pressure Vessel (RPV) Pressure Testing

RPV pressure testing is an infrequent testing procedure. Therefore, it is performed under special care and management oversight. Procedures require that the acceptable temperature range is achieved and established before the pressure is increased. During the test, pressure and level are controlled and achieved using the CRD and the RWCU in the feed-and-bleed mode. Pressure increases are limited to 50 psig per minute. This low rate of increase, the operating procedures, and management oversight make cold overpressurization during a vessel pressure testing highly unlikely. This is the only instance that the vessel is taken water solid.

4.2.8 Procedural Controls and Operator Training

During cold shutdown conditions, reactor water level, water temperature, and vessel pressure are maintained within established and closely monitored limits. Control room procedures require that operators frequently monitor for indications and alarms to detect abnormal conditions as early as possible. Shift managers and supervisors are required to be notified immediately. Thus, any deviations in reactor vessel level, temperature or pressure would be identified and corrected promptly.

Operators receive training in methods of controlling water level pressure and temperature within established limits. In addition, operators receive training on brittle fracture, pressure-temperature limit curves, and technical specification limits.

At CNS, outage work is scheduled by the outage scheduler and senior reactor operators are assigned to provide oversight to assure that the vessel water level, temperature, and pressure conditions are maintained within prescribed limits.

During outages, work is coordinated through the Work Control Center and Outage Control Center which provides additional oversight. In addition, pre-job briefings are conducted to minimize the potential for a cold overpressurization.

Based on the above, the staff finds that the licensee has satisfied the provisions of Criterion 2 of GL 98-05.

4.3 Circumferential Weld Conditional Failure Probability

The NRC staff verified the licensee's calculated mean ART value for the limiting beltline weld metal. By comparing this information with that submitted in the proposed relief request, the staff confirms that the mean RT_{NDT} of the limiting circumferential shell weld at CNS is projected to be 37.6 °F at the end of the current license period of operation (32 EFPY). In this evaluation, the chemistry factor, ΔRT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. The staff finds the licensee's analysis to be acceptable because it meets the requirements specified in the staff's SER for the BWRVIP-05 report, which provides a limiting conditional failure probability of 7.03 x 10^{-7} per reactor-year for a limiting plant-specific mean RT_{NDT} of 86.4 °F for CE-fabricated RPVs. The calculated value of mean RT_{NDT} for the circumferential welds at CNS is lower than that for the limiting plant-specific case for CE-fabricated RPVs, indicating that the conditional failure probability of the CNS circumferential welds is less than 7.03 x 10^{-7} per reactor year. The staff also verified that the licensee's values for the mean ART values shown in Table 1, using Position C.2 in RG 1.99, Revision 2, are essentially the same as the value derived by the NRC with regards to PFM consideration, for CE units. The probability of failure for CNS and the NRC SER are similar because the mean ART is essentially the same and probability for failure is extremely small (7.03 x 10^{-10}/yr). Therefore, the conclusions stated in the SER are applicable to CNS. Both analyses conclude that the failure probability associated with circumferential welds is extremely small, and that it is orders of magnitude less than that for axial welds. At the expiration of the CNS license, the circumferential welds are expected to satisfy the limiting conditional failure probability for circumferential welds discussed in the NRC staff's July 28, 1998, SER. A comparison of the data used in the CNS calculation and the NRC staff assessment is shown in Table 1 of this evaluation.

5.0 CONCLUSION

The staff has reviewed the licensee's submittal and finds that the licensee has acceptably demonstrated that the appropriate criteria in GL 98-05 and in the staff's evaluation of the BWRVIP-05 report have been satisfied regarding permanent relief (i.e., for the remaining portion of the current 40-year license period) from ISI requirements of the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11 for the volumetric examination of the RPV circumferential shell welds.

The NRC staff concludes that authorization of the licensee's alternative examination provides an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination for CNS is authorized.

Additional requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third-party reviews by the Authorized Nuclear Inservice Inspector.

6.0 REFERENCES

1. Electric Power Research Institute, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," TR-105697, Palo Alto, California, September 1995.

2. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

3. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation of the BWR Vessels and Internals Project BWRVIP-05," July 30, 1998.

4. U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide (RG) 1.190, March, 2001.

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