Nebraska Public Power District

Always there when you need us

10 CFR 50.55a

NLS2007019
April 18, 2007

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

Subject: 10 CFR 50.55a Request Number RI-29, Revision 0
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The purpose of this letter is to request that the Nuclear Regulatory Commission (NRC) authorize the Nebraska Public Power District (NPPD) to use an alternative to certain inservice inspection (ISI) code requirements for the Cooper Nuclear Station (CNS) pursuant to 10 CFR 50.55a(a)(3)(i). This request involves ultrasonic examination requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

10 CFR 50.55a Request Number RI-29, Revision 0, is a request for an alternative to allow permanent relief from the requirements to perform a volumetric examination of Reactor Pressure Vessel (RPV) circumferential shell welds during the Fourth Ten-year ISI interval at CNS. The applicable ASME Code for this interval is the 2001 Edition through the 2003 Addenda.

This alternative is in lieu of Section IWB-2500, Table IWB-2500-1, which requires the RPV circumferential shell welds and associated base material to be volumetrically examined once per interval.

RI-29, Revision 0, including the basis and details of the request, is provided in the attachment. This request is applicable to the Fourth Ten-year ISI interval, which commenced on March 1, 2006, and applies to the remaining portion of the current operating license.

NPPD requests approval of this request by January 9, 2008, in order to support planning for Refueling Outage 24 (tentatively scheduled to commence April 2008).

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

A047

Should you have any questions concerning this matter, please contact Paul Fleming, Licensing Manager, at (402) 825-2774.

Sincerely,

Stewart B. Minahan
Vice President - Nuclear and
Chief Nuclear Officer

/dm

Attachment

cc: U.S. Nuclear Regulatory Commission w/attachment
Regional Office - Region IV

Senior Project Manager w/attachment
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachment
USNRC - CNS

NPG Distribution w/o attachment

CNS Records w/attachment

10CFR 50.55a Request Number RI-29, Revision 0
Cooper Nuclear Station
Docket No. 50-298, DPR-46

Proposed Alternative
in Accordance with 10CFR 50.55a(a)(3)(i)

--Alternate Provides Acceptable Level of Quality and Safety--

ASME Code Components Affected

Code Class:	1
References:	IWB-2500-1
Examination Category:	B-A
Item Numbers:	B1.11
Description:	Reactor Pressure Vessel Circumferential Shell Welds
Component Numbers:	VCB-BB-1 VCB-BA-2 VCB-BB-3 VCB-BB-4

Applicable Code Edition and Addenda

American Society of Mechanical Engineers (ASME) Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Cooper Nuclear Station (CNS) is currently in the Fourth Ten-year Inservice Inspection Interval, which commenced March 2006. ASME Section XI, 2001 Edition, 2003 Addenda, IWB-2500, Table IWB-2500-1 requires the reactor pressure vessel (RPV) circumferential shell welds and associated base material to be volumetrically examined once per interval. Deferral of the examinations until the end of interval is permissible; however, IWB-2420(a) requires the sequence of examinations established in the first interval to be repeated during subsequent intervals to the extent practical.

Reason for Request

Boiling Water Reactor Vessel Internals Project (BWRVIP) Report No. BWRVIP-05 (Reference 1) provides the technical basis for permanent relief from the inservice inspection (ISI) requirements of ASME Section XI for the volumetric examinations of RPV circumferential shell welds in boiling water reactors (BWRs). In the report, the BWRVIP concluded that the probabilities of failure for BWR RPV circumferential shell welds are orders of magnitude lower than that of the longitudinal welds. As documented in Reference 3, "Final Safety Evaluation of the BWR Vessel Internals Project BWRVIP-05 Report," the

Nuclear Regulatory Commission (NRC) conducted an independent, risk-informed, probabilistic fracture mechanics assessment (PFMA) of the analysis presented in Reference 1. Based upon the information presented in References 1 and 3, along with the additional information provided in this request, CNS has determined that the proposed alternative presented below provides an acceptable level of quality and safety and satisfies the requirements of 10 CFR 50.55a(a)(3)(i).

Proposed Alternative and Basis for Relief

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 (Reference 2), Nebraska Public Power District (NPPD) 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 welds. Axial weld examination includes that portion of the circumferential welds that intersects each axial weld, or approximately 2% to 3% of the intersecting 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).

The information below provides the basis and details supporting the proposed alternative.

Background

GL 98-05 (Reference 2) discusses that BWR licensees may request permanent (i.e., for the remaining term of operation under the initial license) relief 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 in the NRC 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 NRC staff's July 28, 1998, safety evaluation (Reference 3). 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 report "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)", 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 (Reference 1) 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 risk-informed assessment of the analysis contained in BWRVIP-05 as documented in the final safety evaluation of the BWRVIP-05 report (Reference 3). This assessment also 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 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 PFMA to estimate RPV failure probabilities. Three key assumptions in the PFMA 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.

Discussion

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 end-of-license neutron fluence used in the NRC PFMA was 2.0×10^{18} n/cm^2. However, at CNS, the highest fluence anticipated at the end of the current license period (32 EFPY) is 1.57×10^{18} n/cm^2. Thus, expected embrittlement due to fluence effects is much lower, and the NRC analysis is conservative for CNS in this regard. Therefore, there is conservatism in the already low circumferential weld failure probabilities as related to CNS.

Table 1 illustrates that CNS has additional conservatism in comparison to the NRC's Final Evaluation of BWRVIP-05 Limiting Plant Specific Analysis and Independent Assessment Fracture Analysis limiting case. The chemistry factor, ΔRT_{NDT}, margin term, mean Adjusted Reference Temperature (ART), and upper bound ART are calculated consistent with the guidelines of Regulatory Guide 1.99, Rev. 2. The CNS data used for the evaluation based on the BWRVIP-05 methodology are also shown in Table 1.

Table 1

Parameter Description	CNS Parameters at 32 EFPY (Bounding Circumferential Weld) Using BWRVIP Methodology(1) (2)	NRC Final Safety Evaluation of BWRVIP-05 (Reference 3)
Fluence, n/cm^2	$1.57x10^{18}$	$2.0x10^{18}$
Initial RT_{NDT}, °F	-50	0
Chemistry Factor	175.3	151.7
Cu %	0.2	0.13
Ni %	0.69	0.71
Δ RT_{NDT} (°F)	Monte Carlo Predicted	86.4
Mean ART (°F)	71.6	86.4

Notes associated with Table 1

(1) Values derived from Reference 3.
(2) Values correspond to Part No. 1-240, Heat No. 21935, the lower to lower-intermediate girth weld.

As shown in Table 1, the impact of irradiation results in a lower mean ART for CNS as compared to the results contained in Reference 3. Comparison of the CNS specific data and the data used in the NRC Final Safety Evaluation indicates that the difference between the Mean ART values is the result of the fluence at the end of 32 EFPY, the initial RT_{NDT}, and the chemistry factor.

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 $2.81x10^{-5}$ at 32 EFPY for vessels fabricated by CE ($7.03x10^{-7}$ on a per calendar year basis). The conditional probability of failure was less than $1.0x10^{-6}$ (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 $1x10^{-3}$/yr for an over pressure event. This results in a total probability of failure of $7.03x10^{-10}$/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 $1x\ 10^{-6}$/yr. Since the failure frequency for the CE-fabricated plants due to elimination of circumferential weld examinations contributes less than the amount of change of large early release frequency (LERF) 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 of 71.6°F is bounded by the 86.4°F value used for CE RPVs, with margin terms neglected for consistency. Therefore, the probability of failure stated in BWRVIP-05 and the NRC SER are bounding, and 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 weld will satisfy the limiting conditional failure probability for circumferential welds discussed in the NRC staff's July 28, 1998, safety evaluation (Reference 3).

Generic Letter 98-05, Criterion 2

Demonstrate 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, safety evaluation.

<u>Review of Potential High Pressure Injection Sources</u>

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, over-pressure transients should be considered. It is highly unlikely that a BWR would experience a cold, over-pressure 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, over-pressure event has occurred at a domestic BWR. Also, the NRC staff identified one actual cold, over-pressure 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 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 over-pressure events. This is summarized in the Final Safety Evaluation of BWRVIP-05 (Reference 3).

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 from occurring.

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. If 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. 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 (Figure 3.4.9-1). 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 vessel 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 (RHR) 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. 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 over pressurization 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 over-pressure events is limited to the amount specified in the NRC staff's July 28, 1998, safety evaluation (Reference 3).

Generic Letter 98-05

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

NPPD 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 Safety Evaluation of 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 welds, as required by ASME Section XI, Table IWB-2500-1.

Conclusion

Based on BWRVIP-05, the risk-informed independent assessment performed by the NRC staff, and the above discussion, relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of the circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds), is justified.

In addition to demonstrating consistency with the BWRVIP-05 methodology and conclusions, it was confirmed that CNS has taken steps to reduce the potential for LTOP events through procedural controls and personnel training. An evaluation to identify the sources of increased pressure was also performed, and concluded that the probability of a cold over pressure transient is less than or equal to that used in the NRC evaluation.

In effect, this report demonstrates that the criteria in RG 1.174 regarding defense-in-depth and safety margins are maintained and the USNRC safety goals are not exceeded.

Duration of Proposed Alternative

Relief is requested for the Fourth Ten-year Interval, which is the remainder of the initial CNS operating license.

Precedents

The NRC has previously approved similar relief for several nuclear power plants. CNS RI-29 is similar in content and detail to the request for relief submittals provided by Grand Gulf and River Bend, resulting in NRC approvals per References 4 and 5. These submittals focused on satisfying the conditions established in Generic Letter 98-05 (Reference 2).

Plant	Docket No.	Reference
Grand Gulf Nuclear Station, Unit 1	50-416	4
River Bend Station	50-458	5

References

1. BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 28, 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", date November 10, 1998

3. Letter from G.C. Lainas (U.S. Nuclear Regulatory Commission) to C. Terry (BWRVIP), "Final Safety Evaluation of the BWR Vessel Internals Project BWRVIP-05 Report (TAC No. 93925)," date July 28, 1998.

4. Letter from Robert A. Gramm, U.S. Nuclear Regulatory Commission, to William A. Eaton (Grand Gulf Nuclear Station, Unit 1), "Request for Alternate to Section 50.55a of Title 10 of Code of Federal Regulations (10 CFR) for Examination Requirements of Category B1.11 Reactor Vessel Circumferential Welds (TAC No. MA9787)," dated April 11, 2001

5. Letter from David Terao, U.S. Nuclear Regulatory Commission, Paul D. Hinnenkamp (River Bend Station), "Request for Alternate to 10 CFR 50.55a Examination Requirements of Category B1.11 Reactor Pressure Welds (TAC No. MC8201)," dated 07/14/2006

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2007019

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

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None	N/A	N/A