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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Reactor Pressure Vessel Shell Welds Examination

Ladies and Gentlemen:

On November 10, 1998, Generic Letter (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," was issued which provides guidance for licensees seeking relief from the requirement for examination of BWR RPV circumferential shell welds as recommended in the BWRVIP-05 report. GL 98-05 states that licensees may request relief from the inservice inspection requirements of 10CFR50.55a(g) for volumetric examination of the RPV circumferential shell welds by demonstrating: (1) at the expiration of their 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 GL further states that the licensees will still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Southern Nuclear Operating Company (SNC) has scheduled examination of the RPV axial shell welds during the Spring 1999 Maintenance/Refueling Outage and addresses the two criteria required by Generic Letter 98-05 in the enclosure.

SNC requests approval of an alternative reactor vessel weld examination for the Edwin I. Hatch Nuclear Plant (HNP), Unit No. 1. Approval of this alternative examination is requested in accordance with the provisions of 10 CFR 50.55a(g)(6)(ii)(A)(5) for permanently excluding the examination of RPV circumferential shell welds. The alternative is consistent with the guidance contained in Generic Letter 98-05 and will provide an acceptable level of quality and safety.

In order to plan for the inspections required during the Spring 1999 outage, NRC approval is requested by February 12, 1999. If approval for the permanent deferral cannot be completed by this date, SNC requests NRC approval to delay inspection of the RPV circumferential shell welds for two operating cycles as outlined in NRC Information Notice 97-63, Supplement 1: "Status of NRC Staff's Review of BWRVIP-05", by February 12, 1999 followed by approval of permanent deferral of the examination requirements within a reasonable time period (i.e., within six months).

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SNC's technical justification, as required in Supplement 1 to IN 97-63, is the same as submitted for the permanent request for alternative examination described above.

Should you have any questions in this regard, please contact this office.

Respectfully submitted,

H. L. Sumner, Jr.

IFL/eb

Enclosure: Response to Generic Letter 98-05 Criteria

cc: Southern Nuclear Operating Company
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Enclosure

Southern Nuclear Operating Company's
Response to Generic Letter 98-05 Criteria

GL 98-05 Criterion 1: At the expiration of their 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.

SNC's Response to Criterion 1:

Neutron Fluence/Embrittlement

As published in the August 1992 Federal Register under supplementary information regarding the final rule, the NRC position with regard to augmented examination of reactor vessel shell welds is based on an embrittlement concern (in addition to stress corrosion cracking and service induced cracking) stemming from irradiation material test results which show that certain reactor vessel materials undergo greater radiation damage than previously expected.

The BWR Vessels and Internals Project report (BWRVIP-05), dated September 1995, stated that "Embrittlement issues are addressed in 10CFR50 Appendix G through requirements associated with upper shelf energy (USE) and the reference temperature of nil-ductility transition (RT_{NDT}). In order to account for the effects of embrittlement, adjusted reference temperatures (ARTs), defined as the initial RT_{NDT} plus the irradiation shift for fluence, are determined. It is possible that ARTs may result in pressure-temperature testing criteria that are difficult to meet due to increased temperature requirement. However, due to low BWR fluence, an unacceptable ART will not be reached, even when extended life is planned." The report continues, "In addition to increasing RT_{NDT} the USE of low alloy steel materials decreases with neutron exposure. However, for the relatively low fluence BWR, maintaining a USE above 50 ft-lbs is not a concern. Also... Code margins required by Appendix G are satisfied at USE values as low as 35 ft-lbs and thus is not a safety concern. Based on the above, it can be seen that although irradiation embrittlement of materials can be a significant concern, its effect is minimal for the relatively low fluence environment of a BWR."

Probabilistic Fracture Mechanics (PFM) Analysis

Although BWRVIP-05 provides the technical basis for this relief, an independent NRC risk informed assessment of the analysis contained in the BWRVIP-05 report was conducted. The NRC independent assessment used 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 that estimated to be 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.

The following statement is contained in the "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," dated July 28, 1998. "It should be noted that the failure frequencies for axial welds cited above are relatively high, but that there are known conservatisms in these estimates. For example, these analyses were based on the assumption that the flaws in the axial weld with the limiting material properties and chemistry are located at the inside surface of the BWR RPV and at the location of peak end-of-license (EOL) azimuthal fluence. Since flaws are distributed throughout the weld and the EOL neutron fluence will not occur for many years, the staff has concluded that the present RPV failure frequency is substantially below that reported by the BWRVIP, and independently calculated by the staff, and is not a near-term safety concern".

The following information is provided to show the conservatism of the NRC analysis for Hatch 1. Changes in RT_{NDT} may be used as one of the means for monitoring radiation embrittlement of reactor vessel materials. In the case of Hatch 1, there is a single circumferential weld joint located with the RPV beltline which is the limiting circumferential weld within the vessel, (i.e., relative to RT_{NDT}). For plants with RPVs fabricated by Combustion Engineering (CE), the end-of-license neutron fluence for circumferential welds used in the NRC Staff and BWRVIP Limiting Plant-Specific Analysis (32 EFPY), Table 2.6-4 of the Safety Evaluation for BWRVIP-05, was 0.20E+19 n/cm^2. However the highest surface fluence anticipated for the circumferential welds at the end-of-license for Hatch 1 is 0.132E+19 n/cm^2. Clearly the Hatch 1 specific fluence is bounded by the fluence used in the safety evaluation. Thus, the fluence effect on embrittlement is much lower, and the NRC analysis described in the safety evaluation is conservative for Hatch 1 in this regard. Therefore, there is significant conservatism in the already low circumferential weld failure probabilities as related to Hatch 1.

As shown in Table 1, the largest calculated embrittlement shift in RT_{NDT} (i.e., ΔRT_{NDT}) for the Hatch 1 vessel is 59.8°F at the end of life. The limiting adjusted reference temperature is 76.4°F. By comparison, using the values for fluence (i.e., 0.20E+19 n/cm^2) and weld chemistry (i.e., copper content of 0.13 weight percent, nickel content of 0.71 weight percent) assumed in the CE (VIP) circumferential flaw analysis in Table 2.6-4 of the NRC Safety Evaluation, a ΔRT_{NDT} of 86.4°F would be derived. By further comparison, using the mean values for fluence (i.e., 0.20E+19 n/cm2) and weld chemistry (i.e., copper content of 0.183 weight percent, nickel content of 0.704 weight percent) assumed in the CE (CEOG) circumferential flaw analysis in Table 2.6-4 of the NRC Safety Evaluation, a ΔRT_{NDT} of 98.1°F would be derived. Therefore, the calculated ΔRT_{NDT} value and the adjusted reference temperature for the Hatch 1 vessel are less than the embrittlement shift and the RT_{NDT} estimated in the NRC safety evaluation. Consequently, the Hatch 1 vessel is bounded by the NRC's safety evaluation.

Table 1

Hatch Unit 1
RPV Shell Weld Information

Heat Number	90099	33A277
Neutron surface fluence at the end of life	0.132E+19 n/cm^2	0.132E+19 n/cm^2
Initial (unirradiated) reference temperature (submitted in SNC's response to Generic Letter 92-01, Revision 1, Supplement 1, dated July 28, 1998	-10°F	-50°F
Weld chemistry factor (CF)	91	126
Weld copper content	0.197%	0.258%
Weld nickel content	0.060%	0.165%
Increase in reference temperature due to irradiation (ΔRT_{NDT})	43.2°F	59.8°F
Margin term	43.2°F	56.0°F
Adjusted reference temperature	76.4°F	65.8°F

GL 98-05 Criterion 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.

SNC's Response to Criterion 2:

Consideration of Low Temperature - Over Pressurization Events

Plant Hatch has procedures in place which monitor and control reactor pressure, temperature, and water inventory during all aspects of cold shutdown and refueling operations which would minimize the likelihood of a Low Temperature Over-Pressurization (LTOP) event from occurring. Additionally, these procedures are reinforced through normal, periodic operator training.

The Class 1 System Leakage Pressure Test and Hydrostatic Pressure Test procedures used at Plant Hatch have sufficient procedural guidance to prevent a LTOP event. The Leakage Pressure Test is performed after each refueling outage, while the Hydrostatic Pressure Test is performed once every ten years. The leakage and hydrostatic tests are given special management attention during each performance. Site management provides oversight of the activity and the responsible engineering personnel are required to perform "pre-test briefings" with all essential personnel. These briefings detail the testing evolution with special emphasis on: conservative decision making, anticipated plant conditions, pressure boundary considerations, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications, potential main control room annunciation actions, and the operator actions required should the test

need to be aborted if plant systems responded in an adverse manner. The test procedure also contains specific instruction to ensure that the RPV temperature and pressure are monitored throughout the test duration to ensure compliance with the applicable Technical Specification pressure-temperature curve.

The test is coordinated by engineering department personnel who have training and experience for these particular tests. The test procedures also require the designation of a Responsible Test Engineer on each shift who is a single point of contact accountable and responsible for the coordination of testing from initiation to closure, and for maintaining shift management and line management cognizant of the status of the test.

At the beginning of the test, the procedure requires that the Reactor Pressure Vessel (RPV) water level be stabilized just below the reactor head flange, and the Recirculation System Pump(s) be in operation to heat-up the reactor coolant and RPV shell to comply with the applicable pressure-temperature curve. An air bubble is then injected into the top of the RPV that functions as a snubber. Then the Control Rod Drive (CRD) System pump(s) are utilized to raise RPV level, compress the air bubble, thus increasing pressure, while giving the operator the ability to comfortably control the test pressure. Test pressure is ultimately controlled via a 'feed-and-bleed' technique utilizing the CRD system pump(s) and the Reactor Water Clean-up (RWCU) System dump capability to either the Main Condenser or the Radwaste System. To ensure a controlled, deliberate pressure increase, the rate of increase is procedurally limited throughout the performance of the test and the final test pressure is limited to within a relatively tight band (e.g., 1035 – 1050 psig). If the allowable test conditions are exceeded, the procedure provides direction to trip the CRD system pump(s)), and reduce system pressure utilizing the main steam drain line which discharges to the main condenser. RWCU dump may also be increased, or maximized, as an alternative to using the main steam drain line.

Site engineering and operations personnel have successfully performed the required Class 1 system pressure tests in the described manner since 1987. A minimum of eight such tests have been successfully performed for each Unit at Plant Hatch (16 total tests) with no problems associated with Technical Specification pressure-temperature curve compliance or a LTOP event. The test procedures have been periodically reviewed and continually updated since their creation in 1987 to incorporate engineering and operations personnel comments, concerns, and suggestions, plant modifications, test performance experience, and changes in administrative requirements.

With regard to inadvertent system injection resulting in an LTOP condition, the high pressure make-up systems (High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, as well as the normal feedwater supply (via the Reactor Feedwater Pumps)) are all steam driven. During reactor cold shutdown conditions, no reactor steam is available for the operation of these systems. Therefore, it is not possible for these systems to contribute to an over-pressure event while the unit is in cold shutdown.

In the case of low pressure system initiation, the shutoff head for the Core Spray and Residual Heat Removal Pumps are sufficiently low that the potential for an over-pressurization event which would exceed the Tech Spec pressure-temperature limits, due to an inadvertent actuation of these systems, is very low.

Procedural controls are also in place to respond to an unexpected or unexplained rise in RPV water level which could result from the spurious actuation of an injection system. Actions specified in these procedures include preventing condensate pump injection, securing ECCS system injection, tripping CRD pumps, terminating all other injection sources, and lowering RPV level via the RWCU system.

In addition to procedural controls, periodic Licensed Operator Training further reduces the possibility of the occurrence of LTOP events. Initial Licensed Operator Training and Simulator Training of plant heatup and cooldown events includes performance of surveillance tests and monitoring which ensure pressure-temperature curve compliance. In addition, periodic operator training reinforces management's expectations for strict procedural compliance.

Finally, both Southern Nuclear Operating Company corporate and Hatch site personnel continuously review industry operating plant experiences to ensure that Plant Hatch procedures consider the impact of actual events, including LTOP events. Appropriate changes to procedures and training are then implemented to preclude similar situations from occurring at Plant Hatch.

Based upon the above, the probability of a LTOP event at Plant Hatch is considered to be less than or equal to that used in the USNRC evaluation.