

February 12, 2004

NG-04-0103
10 CFR 50.55a(a)(3)(i)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

DUANE ARNOLD ENERGY CENTER
DOCKET 50-331
LICENSE NO. DPR-49

Request to Implement NDE-R047 Addressing Boiling Water Reactor
Shell Weld Inspection Recommendations of the Boiling Water Reactor
Vessel And Internals Project (BWRVIP) Report BWRVIP-05

References:

1. EPRI Report TR-105697, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), dated September 1995.
2. Letter from NRC to BWRVIP, Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC NO. M93925) dated July 28, 1998.
3. Letter from NRC to BWRVIP, Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC NO. MA3395) dated March 7, 2000.
4. 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.

Pursuant to 10 CFR 50.55a(a)(3)(i), Nuclear Management Company, LLC (NMC) hereby requests NRC approval of proposed Relief Request NDE-R047 for use at the Duane Arnold Energy Center (DAEC). Relief Request NDE-R047 requests permanent relief for the remaining term of the DAEC operating license from the following requirements:

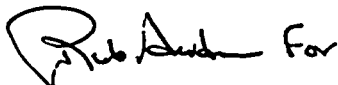
1. Volumetric examination of all reactor pressure vessel (RPV) shell circumferential welds in the RPV in accordance with the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, Examination Category B-A, Item B1.11.
2. Successive inspections for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2420.
3. Additional examinations for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2430.

The technical basis providing justification for relief from the examination requirements of RPV shell circumferential welds is contained in a report submitted from the BWRVIP to the NRC (Reference 1). The NRC evaluated this report and responses to Requests for Additional Information, and issued safety evaluations to the BWRVIP (References 2 and 3). Additionally, the Staff issued NRC Generic Letter (GL) 98-05 (Reference 4) providing guidance to BWR licensees regarding the use of the referenced BWRVIP report when requesting relief; this guidance was used in the development of NDE-R047.

As discussed in the attached request, the basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety. NMC requests approval of the request by February 13, 2005 to support preparations for DAEC's Refueling Outage 19, currently scheduled to begin in March of 2005.

This letter makes no new commitments.

Please contact Steve Catron of the DAEC Regulatory Affairs staff at (319) 851-7234 for questions regarding this submittal.



Mark A. Peifer
Site Vice-President, Duane Arnold Energy Center
Nuclear Management Company, LLC

cc: Regional Administrator, USNRC, Region III
Project Manager, DAEC (NRC-NRR)
NRC Resident Inspector (DAEC)

Attachment

ATTACHMENT
TO NG-04-0103

NUCLEAR MANAGEMENT COMPANY, LLC

DAEC
DOCKET 50-331

10 CFR 50.55a Request Number NDE-R047

11 Pages Follow

10 CFR 50.55a Request Number NDE-R047

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

1. ASME Code Components Affected

Code Class: 1
References: IWB-2500,
Table IWB-2500-1
Examination Categories: B-A
Item Number: B1.11
Description: Relief from Volumetric Examination of Pressure
Retaining Reactor Pressure Vessel (RPV) Shell
Circumferential Welds
Component Numbers: VCB-B001, VCB-A002, VCB-B003, and VCB-B004

2. Applicable Code Edition and Addenda

ASME Section XI 1989 Edition, No Addenda is applicable to the Inservice Inspection (ISI) Program for the Third Ten Year Interval.

3. Applicable Code Requirement

NMC requests relief from the inspection of Reactor Vessel Circumferential (B-A) Welds, Item B1.11, for the third interval inspections and for the remaining term of the current license for the DAEC.

In accordance with the provisions of 10 CFR 50.55a(a)(3)(i), NMC requests permanent relief for the remaining term of the operating license for the DAEC from the following requirements:

- a. Volumetric examination of all Reactor Pressure Vessel shell circumferential welds in the RPV in accordance with the requirements of ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, Examination Category B-A, Item B1.11.
- b. Successive Inspections for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2420.
- c. Additional Examinations for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2430.

4. Reason for Request

NMC requests this relief to reduce inspections and conserve radiological dose, while still maintaining an acceptable level of quality and safety for examination of the affected welds.

5. Proposed Alternative and Basis for Use

I. Alternative Provisions:

Pursuant to 10 CFR 50.55a(a)(3)(i), the DAEC will implement the following alternative provisions for the subject weld examinations. Unless stated otherwise, all references to the ASME code are to ASME Section XI, 1989 Edition, no Addenda.

a. Inservice Inspection Scope

The failure frequency for ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Reactor Pressure Vessel Shell Circumferential Welds, is sufficiently low to justify elimination of inservice inspection, based on the NRC Safety Evaluation (Reference 2).

The ISI examination requirements of ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, Reactor Pressure Vessel Shell Longitudinal, shall be performed, to the extent possible, and shall include inspection of the circumferential welds at the intersection of these welds with the longitudinal welds, or approximately 2 to 3% of the RPV shell circumferential welds.

The procedures for these examinations shall be qualified such that flaws relevant to reactor pressure vessel integrity can be reliably detected and sized, and the personnel implementing these procedures shall be qualified in the use of the procedures.

b. Successive Examination of Flaws

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Reactor Pressure Vessel Shell Circumferential Welds, at intersections with longitudinal welds, successive examinations per IWB-2420, "Successive Inspections," are not required for non-threatening flaws such as embedded flaws from material manufacturing or vessel fabrication which experience negligible or no growth during the design life of the vessel, provided that the following conditions are met:

1. The flaw is characterized as subsurface in accordance with BWR Vessel and Internals Project Report, BWRVIP-05, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (Reference 1).

2. The nondestructive examination (NDE) technique and evaluation that detected and characterized the flaw as originating from material manufacture or vessel fabrication is documented in a flaw evaluation report.

3. The vessel containing the flaw is acceptable for continued service in accordance with ASME Section XI, IWB-3600, "Analytical Evaluation of Flaws," and the flaw is demonstrated acceptable for the intended service life of the vessel.

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, Reactor Pressure Vessel Shell Longitudinal Welds, all flaws shall be reinspected at successive intervals consistent with ASME Code and regulatory requirements.

c. Additional Examinations of Flaws

For ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Reactor Pressure Vessel Shell Circumferential Welds, at the intersection with longitudinal welds, additional requirements per ASME Section XI, IWB-2430, "Additional Examinations," are not required for flaws provided the following conditions are met:

1. If the flaw is characterized as subsurface in accordance with BWRVIP-05, then no additional examinations are required.
2. If the flaw is not characterized as subsurface in accordance with BWRVIP-05, then an engineering evaluation shall be performed, addressing the following as a minimum:
 - A determination of the root cause of the flaw,
 - An evaluation of any potential failure mechanisms,
 - An evaluation of service conditions which could cause subsequent failure,
 - An evaluation per ASME Section XI, IWB-3600 demonstrating that the vessel is acceptable for continued service.
3. If the flaw meets the criteria of ASME Section XI, IWB-3600 for intended service life of the vessel, then additional examinations may be limited to those welds subject to the same root cause conditions and failure mechanisms, up to the number of examinations required by ASME Section XI, IWB-2430(a). If the engineering evaluation determines that there are no additional welds subject to the same root cause conditions or no failure mechanism exists, then no additional examinations are required.

For ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, additional examinations for flaws shall be in accordance with ASME Section XI, IWB-2430, "Additional Examinations." All flaws in RPV shell longitudinal welds shall require additional weld examinations consistent with the ASME Section XI Code and regulatory requirements. Examinations of the circumferential RPV shell welds shall be performed if longitudinal (axial) weld examinations reveal an active, mechanistic mode of degradation.

II. Basis for Relief

Augmented Exam

A September 8, 1992 revision to 10 CFR 50.55a(g)(6)(ii)(A) contains an augmented examination requirement to perform a one time volumetric examination of essentially 100% (>90%) of all circumferential and axial reactor pressure vessel (RPV) shell assembly welds. This rule revoked previously granted relief requests regarding the extent of volumetric examination on ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.10, Circumferential and Longitudinal Reactor Pressure Vessel Shell Welds. 10 CFR 50.55a(g)(6)(ii)(A) required the augmented examinations be performed as specified in the ASME Code Section XI (1995 Edition with the 1996 Addenda).

During refueling outage (RFO) 14 in 1996, the DAEC performed the augmented weld examination of the reactor vessel using the General Electric Reactor Inspection System (GERIS) 2000 ultrasonic examination system. At the DAEC, the volumetric examinations of the reactor pressure vessel shell circumferential welds were performed from the vessel outside diameter using a composite of automated and supplemental manual Ultrasonic (UT) examination techniques; no reportable indications were found.

Complete examination of the subject welds was not obtained due to scanning limitation and access restrictions from various reactor pressure vessel appurtenances and containment structures. For circumferential weld VCB-B001, an examination coverage of 96.5% was obtained, for VCB-A002, an examination coverage of 96.7% was obtained, for VCB-B003, 96.7% was obtained and for VCB-B004, 86.91% was obtained. The examination coverage for VCB-B004 (the Course 3 to Course 4 circumferential weld) was limited due to the presence of vessel stabilizers and an insulation support ring. The insulation support ring is located about 18 inches from the weld. The bottom of the stabilizer brackets are located on the weld. By letter dated October 18, 1999, the NRC granted relief from the requirement to perform an examination of essentially 100% of the weld length for VCB-B004.

GL 98-05

The technical justification for this request for inspection relief is documented in BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), dated September 1995 (Reference 1). The NRC evaluated this report and responses to Requests for Additional Information, and issued Safety Evaluations to the BWRVIP (References 2 and 3).

On November 10, 1998, the NRC issued 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," (Reference 4). This GL stated that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the inservice

inspection 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 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. Licensees will still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Although BWRVIP-05 provides the technical basis supporting the relief request, the following information addresses the GL criteria, and shows the conservatism of the NRC analysis relative to the DAEC reactor pressure vessel.

Criterion 1, Demonstrate that at the expiration of the license, the RPV shell circumferential welds will continue to satisfy the limiting conditional failure probability for RPV shell circumferential welds that is established in the July 30, 1998 Safety Evaluation.

The NRC evaluation of BWRVIP-05 utilized a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis are: (1) the neutron fluence used was the estimated end-of-life 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 table illustrates that the DAEC reactor pressure vessel has additional conservatism in comparison to Table 2.6-4 for the Limiting Plant-Specific Analyses (32 effective full power years (EFPY)) of the NRC's evaluation of BWRVIP-05.

Effects of Irradiation on RPV Circumferential Weld Properties
Duane Arnold Energy Center

Parameter Description	DAEC Comparative Parameters At 32 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 07L669 Lot K004A27A	USNRC Limiting Plant Specific Analysis Parameters at 32 EFPY SER Table 2.6-4***
Copper (Cu), wt%	0.03	0.10
Nickel (Ni), wt%	1.02	0.99
Chemistry Factor (CF)	41	134.9*
End of Life (EOL) Inside Diameter (ID) Fluence, $\times 10^{19}$ n/cm ²	0.355**	0.51
Initial (unirradiated) Reference Temperature $RT_{NDT(U)}$, °F	-50	-65
Increase in Reference Temperature ΔRT_{NDT} , °F	26.4	109.5
Mean (irradiated) Reference Temperature $RT_{NDT(U)} + \Delta RT_{NDT}$, °F	-23.6	44.5

*Revised value from the NRC Safety Evaluation (SE) Supplement (Reference 3).

**By Amendment 253 (Reference 5), the NRC approved revised Reactor Coolant System Pressure-Temperature curves for the DAEC. As discussed in the Safety Evaluation that accompanied the Amendment, the replacement curves were generated using an NRC-approved methodology (General Electric Report NEDC-32983PA, Revision 1, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations," December 2001) for determining the neutron fluence on the Reactor Pressure Vessel (RPV). The methodology used for the RPV fluence calculation is described in GE Report NEDC-32983PA. This methodology follows the guidance in RG 1.190 and has been approved by the NRC staff by letter dated September 14, 2001.

***The DAEC RPV was supplied and erected by the Chicago Bridge and Iron Company.

As shown in the Table, the nickel content for the DAEC bounding weld is slightly higher than the value used in the NRC analysis, however, the values for DAEC copper content and chemistry factor are considerably lower than the values used in the NRC analysis. The unirradiated reference temperature is higher than that used in the NRC analysis. The calculated 32 EFPY fluence for the DAEC is lower than the NRC estimated values. The overall result for the DAEC is a lower calculated mean reference temperature than the NRC analysis mean reference temperature value.

Since the mean (irradiated) reference temperature value for the DAEC RPV shell weld is less than the mean (irradiated) reference temperature value for its corresponding limiting plant reference case study (as shown in the Table), the shell weld is considered to have less embrittlement than the corresponding weld in the case study, and therefore to have a conditional probability of failure less than or equal to that calculated for the reference case study. The RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability, $P(FIE)$, in Table 2.6-4 of the NRC Safety Evaluation through the initial end of license.

This demonstrates that at expiration of the existing license, the circumferential welds of the DAEC RPV will continue to satisfy the limiting conditional failure probability for circumferential welds in the Staff's SE dated July 30, 1998.

Criterion 2, Licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the July 30, 1998 Safety Evaluation.

As discussed below, the NMC has procedures in place for the DAEC that guide operators in controlling and monitoring reactor pressure during all phases of operation. Use of the guidance provided in the operating procedures will prevent a Low Temperature Over-Pressurization (LTOP) event. Also, these procedures are reinforced through operator training, and system design features provide additional insurance against an LTOP event.

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

Both HPCI and RCIC are high-pressure, steam driven systems. These systems use steam-turbine driven pumps to deliver emergency coolant to the RPV. The steam that is used to drive the turbines and actuate the pumps is delivered through the turbine steam supply line, which discharges from the main steam lines of the plant. Since the reactor does not deliver steam to the main steam lines during cold shutdown, the HPCI and RCIC systems are will not cause a cold-overpressurization event while DAEC is in the cold shutdown operating mode.

Feedwater/Condensate System

The feedwater/condensate system is a potential source of high-pressure injection into the reactor vessel. The condensate pumps are a source of water to the reactor feed pumps. The reactor feed pumps provide water to the vessel. A system design feature

of the reactor feed pumps is an automatic trip of all feed pumps on high vessel water level (+211 inches).

With respect to injection by an inadvertent start of a feedwater pump, injection of feedwater with vessel water level greater than +211 inches is controlled by a high water interlock. This interlock prevents operation or starts of the feedwater pumps when water level in the vessel is equal to or greater than +211 inches. Defeating this interlock is procedurally and administratively controlled to prevent inadvertently injecting feedwater into the vessel. The DAEC has high reactor water level and high reactor pressure alarms in the control room. These provide further assurance that an LTOP event will not occur.

The condensate and feed water pumps are used to control vessel level during startup. The startup procedure requires monitoring of reactor vessel temperatures and pressures. The reactor head vents are not closed until the coolant temperature reaches 212°F. This administrative action for head vent closure serves as a mechanism to reduce the likelihood of over pressurization at low temperature. Monitoring of reactor temperature, pressure, and cool down rates, are prescribed in procedures and Technical Specifications.

A low temperature over pressurization event due to injection by the feedwater/condensate system is very unlikely since strict controls on temperature and pressure are imposed by procedures. An unexpected change in reactor water level would allow for operator action. Therefore, this system does not present a significant potential for over pressurization.

Standby Liquid Control (SBLC)

SBLC is another high-pressure water source to the reactor pressure vessel. SBLC is designed with two redundant trains of SBLC piping, each designed with a pump and explosive squib-type discharge valve, delivering flow through a common header to the RPV. No automatic starts are associated with this system; operator action is needed to manually start the system by a key-lock switch; therefore, inadvertent manual initiation of SBLC is an unlikely event.

Procedures have been developed for operation of the SBLC system and operators are trained on the system operation. The injection rate of one SBLC pump is approximately 26.2 gpm; the injection rate of two SBLC pumps is approximately 52.4 gpm. These low flow rates would provide DAEC operators ample time to control reactor pressure in the case of an inadvertent injection of SBLC. Therefore, this system does not present a significant potential for over-pressurization.

Residual Heat Removal (RHR) System, Low Pressure Coolant Injection (LPCI), Core Spray (CS)

The shutoff head for the DAEC Core Spray pumps is about 330 psig, and for the Residual Heat Removal pumps is about 260 psig. An inadvertent injection of LPCI or CS would be detected by operators and the injection would be terminated, based on observation and alarm of reactor vessel level. In addition, during cold shutdown when the reactor head is tensioned, a cold overpressure event is prevented by procedures which require the operator to place the RPV head vent valves in an open position when reactor coolant temperatures are below 212°F. A Core Spray pump may be used for reactor vessel and cavity fill during refueling outages. Under these conditions the reactor vessel head is removed which will prevent over pressurization.

A Condensate, CS, or RHR pump may be used to inject into the RPV in the event of a loss of shutdown cooling. Abnormal Operating Procedure (AOP) 149, Loss of Decay Heat Removal, includes guidance on performing a feed and bleed to the Torus via a safety relief valve (SRV). The handswitch for an SRV is placed in the open position. A Condensate, CS or RHR pump is used to inject water into the RPV until an SRV is open and RPV pressure is about 50 psig above Torus pressure, but as low as practical. Coolant then exits the reactor vessel and flows to the Torus via the SRV discharge line. In this situation, the open SRV prevents an overpressure event.

Control Rod Drive (CRD) and Reactor Water Cleanup (RWCU)

The CRD and RWCU systems are used to control RPV water level and pressure during cold shutdown conditions using a feed and bleed process. The low flow rate of these pumps allows sufficient time for operator action to react to unanticipated level changes and thus pressure changes. Therefore, these systems do not present a significant potential for over pressurization.

The CRD and RWCU systems are also used in the performance of RPV pressure and hydrotests. The DAEC test procedures contain additional requirements to aid in the prevention of a low temperature over-pressurization event. The Class 1 System Leakage Test is typically performed at the conclusion of each refueling outage, while the Hydrostatic Pressure Test (Code Case N498-1) is performed once every ten years. The leakage and hydrotests are considered to be infrequently performed, complex tasks and a requirement is included in them for a briefing with personnel involved in the surveillance. This briefing includes an emphasis on maintaining margins of safety, conservative decision making, review of lessons learned from in-house or industry operating experience, the need for open communications, and the need to abort the test if plant systems respond in an adverse manner. Vessel temperature and pressure are required to be monitored throughout these tests to ensure compliance with the Technical Specification Pressure-Temperature (P-T) limits.

As discussed in the NRC SE of the BWRVIP-05, the risk of cold over pressurization due to CRD injection may be higher if a loss of station power occurs during the pressure test, since the RWCU and CRD pumps would lose their power. If the operator restarts the CRD pumps but does not restore the RWCU, cold CRD flow would accumulate in the lower head region and, without further operator action, the pressure will increase. The beltline region would, nonetheless, stay near the original 200-degree level, maintaining the beltline P-T limits. To preclude this from occurring, special precautions are included in the surveillance test procedure for the DAEC pressure test. One precaution states that in the event of an interruption of offsite power, open CV-2729 (Cleanup System Drain Header Control Valve) and allow the system to depressurize. This will preclude RPV over-pressurization as a result of closure of CV-2729 should control air pressure be lost. Another precaution instructs the operators to immediately trip the CRD pump if RWCU isolates. These actions provide additional protection against an LTOP event.

Reactor Operator Training

Simulator training is conducted on start-up and shut down scenarios in accordance with approved procedures, providing opportunities for the operators to perform RPV pressure and level control. Procedural controls for reactor temperature, water level, and pressure are an integral part of Operator training. Specifically, operators are trained in methods of controlling RPV water level within specified limits, as well as responding to abnormal RPV water level conditions outside the established limits. Plant-specific procedures have been developed to provide guidance to the operators regarding compliance with the Technical Specification requirements on Pressure-Temperature limits.

Work Control Process

During plant outages, work control procedures require that the outage schedule and changes to the schedule receive a risk assessment review commensurate with their safety significance. Senior Operations personnel provide input to the outage schedule to avoid conditions that could adversely impact reactor water level, pressure, or temperature. Schedules are issued listing the work activities to be performed.

During refueling outages, work is coordinated through the Outage Control Center. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor water level or decay heat removal. The Control Room Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Cognizant individuals involved in the work activity attend pre-job briefings. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

Conclusion

In summary, NMC has reviewed the methodology used in BWRVIP-05 (Reference 1), and considered DAEC-specific materials properties and fluence, operational practices, the provisions of the NRC Safety Evaluation Report (Reference 2), and GL 98-05. NMC's operational and procedural controls provide sufficient assurance that it is unlikely that a cold overpressure transient will occur at the DAEC. The probabilistic failure analysis of the circumferential welds in the DAEC RPV, when taken in conjunction with NMC's operational and procedural controls to prevent cold-overpressurization events, provides an acceptable level of quality and safety in lieu of actually performing the volumetric inspections of the circumferential welds as required by ASME Boiler and Pressure Vessel Code, Section XI, Examination Category B-A, Inspection Item B1.11.

6. Duration of Proposed Alternative

Permanent relief is requested for the remaining term of the operating license of the DAEC.

7. Precedent

LaSalle County Station, Units 1 and 2 - Relief Request CR-38, Shell Weld Inspection (TAC Nos. MB9755 and MB9756), from A. Mendiola (NRC) to J. Skolds (Exelon Nuclear) dated January 28, 2004, Docket Nos.: 50-373 and 50-374

8. References:

1. EPRI Report TR-105697, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), dated September 1995.
2. NRC Safety Evaluation Report of Topical Report by the Boiling Water Reactor Vessel and Internals Project: "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations, BWRVIP-5" (TAC No. M93925), July 28, 1998.
3. NRC Safety Evaluation Report of Topical Report by the Boiling Water Reactor Vessel and Internals Project: "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-5 Report (TAC No. MA3395)," March 7, 2000.
4. 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.
5. Amendment 253 to DAEC Technical Specifications Regarding Pressure and Temperature Limit Curves (TAC No. MB8750), by letter dated August 25, 2003, D. Hood (NRC) to M. Peifer (NMC).