AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

March 28, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Ladies and Gentlemen:

ULNRC-05271
10 CFR 50.55a
10 CFR 50.55a(g)(5)(iii)

Ameren UE

DOCKET NUMBER 50-483
UNION ELECTRIC COMPANY
CALLAWAY PLANT
10 CFR 50.55a REQUESTS FOR RELIEF FROM
ASME SECTION XI INSERVICE INSPECTION REQUIREMENTS
FOR THIRD 10-YEAR INSPECTION INTERVAL

Pursuant to 10 CFR 50.55a(a)(3) and/or 10CFR 50.55a(g)(5)(iii), Union Electric Company (AmerenUE) requests NRC approval of the attached three relief requests for the third 10-year inservice inspection interval at Callaway. The Code Edition (and Addenda) applicable to Callaway for its third inspection interval, which began December 19, 2005, is the ASME Boiler Pressure Vessel Code, Section XI, 1998 Edition through 2000 Addenda.

The attached 10CFR50.55a requests, identified as I3R-01, I3R-02, and I3R-04, pertain to examination requirements for applicable plant components and/or piping. I3R-01 is a request to continue the application of an alternative risk-informed methodology for the inservice inspection of Class 1 and 2 piping welds, as previously established at Callaway under a previous 10 CFR 50.55a request that was approved by the NRC during Callaway's second ten-year inspection interval. I3R-02 is a request to utilize Code Case N-700, "Alternative Rules for Selection of Classes 1, 2, and 3 Vessel Welded Attachments for Examination," as an alternative approach for the selection of vessel welded attachments for examination. The third and last of the attached relief requests, I3R-04, is a request to permit a limited visual examination of each of the reactor vessel supports at Callaway, based on the hardship involved in performing a 100% examination of each support. For each of these relief requests, supporting information and essential details, including justification, is provided and contained in each of the attached requests.

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As indicated above, these 10 CFR 50.55a requests support inservice inspection activities for Callaway's third 10-year inspection interval which began December 19, 2005. The next refueling outage (i.e., the first of those that will occur during the third inspection interval) is scheduled for Spring 2007. Since many of the inspection activities affected by the proposed relief requests will be conducted during that outage, and in order to plan and prepare for those activities sufficiently in advance of the outage, AmerenUE respectfully requests NRC review and approval of the attached relief requests by the end of this year (2006).

It may be noted that no new regulatory commitments have been made or identified pursuant to this letter and its attachments. Please contact me at 573-676-8659 or Dave Shafer at 314-554-3104 for any questions you may regarding these relief requests.

Sincerely,

Keith D. Young
Keith D. Young
Manager - Regulatory Affairs

TBE/jdg

Attachments: Relief Request I3R-01
Relief Request I3R-02 (with attachment)
Relief Request I3R-04

cc: U.S. Nuclear Regulatory Commission (Original and 1 copy)
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Mr. Bruce S. Mallett
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Senior Resident Inspector
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Jack N. Donohew (2 copies)
Licensing Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 7E1
Washington, DC 20555-2738

Missouri Public Service Commission
Governor Office Building
200 Madison Street
PO Box 360
Jefferson City, MO 65102-0360

10 CFR 50.55a Request Number I3R-01

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

Alternative Provides Acceptable Level of Quality and Safety

1. **ASME Code Components Affected**

All Code Class 1 and 2 piping welds previously subject to the requirements of ASME Section XI, Table IWB-2500-1 (Examination Categories B-F and B-J) and Table IWC-2500-1 (Examination Categories C-F-1 and C-F-2).

2. **Applicable Code Edition and Addenda**

ASME Boiler and Pressure Vessel Code, Section XI 1998 Edition through 2000 Addenda

3. **Applicable Code Requirement**

ASME Section XI, Tables IWB-2500-1 and IWC-2500-1 for Examination Categories B-F, B-J, C-F-1 and C-F-2 stipulate the selection and examination requirements for Class 1 and 2 piping welds.

4. **Reason for Request**

The above-noted ASME Section XI tables (for Examination Categories B-F, B-J, C-F-1, and C-F-2) specified the original requirements for nondestructive examination of Class 1 and 2 piping welds. In 2001, a risk-informed methodology for the inservice inspection of Class 1 and 2 piping welds was applied at the Callaway Nuclear Power Plant. The risk-informed inservice inspection (RI-ISI) process used in this application is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657, Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." The RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B."

This risk-informed application met the intent and principles of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" and Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping."

The original RI-ISI template, "Risk-Informed Inservice Inspection Program Plan - Callaway Plant (Revision 1)," was submitted to the NRC for approval per 10CFR50.55a(a)(3)(i) via AmerenUE letter ULNRC-4392, dated February 16, 2001. Based upon the information provided in the RI-ISI template, the request to implement the RI-ISI methodology on Class 1 and 2 piping welds was approved by the NRC in a letter dated January 30, 2002 (TAC No. MB1205). The purpose of this current request is for the continued application of the RI-ISI methodology on Class 1 and 2 piping welds during the third ISI interval, based on the RI-ISI alternative providing an acceptable level of quality and safety.

5. **Proposed Alternative and Basis for Use**

The proposed alternative is to continue applying the risk-informed ISI criteria of EPRI TR-112657 during the third ISI interval in lieu of the requirements of ASME Section XI, Table IWB-2500-1 (Examination Categories B-F and B-J) and Table IWC-2500-1 (Examination Categories C-F-1 and C-F-2).

When Callaway submitted their initial RI-ISI application to the NRC for approval, the following standard wording was included in the template submittal:

> "The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant specific feedback."

Most U.S. nuclear power plants have implemented RI-ISI Programs with this standard wording for performing periodic reviews and updates. To address this issue, NEI 04-05, "Living Program Guidance To Maintain Risk-Informed Inservice Inspection Programs For Nuclear Plant Piping Systems," has been developed. As part of the ISI Program Update for the third ISI interval at Callaway, an RI-ISI Living Program Evaluation was performed in accordance with NEI 04-05. The objective of this evaluation was to review plant and industry activities that could impact the bases of the Callaway RI-ISI application as it enters the third ISI interval.

In accordance with NEI 04-05, the following aspects were considered for the evaluation:

- Plant Examination Results
- Piping Failures
  - Plant Specific Failures
  - Industry Failures
- PRA Updates
- Plant Design Changes
  - Physical Changes
  - Programmatic Changes
  - Procedural Changes
- Changes in Postulated Conditions
  - Physical Conditions
  - Programmatic Conditions

The RI-ISI Living Program evaluation resulted in the following seven issues being addressed in the RI-ISI application:

- The plant changes made per Modification Package MP 01-1019 and further documented in Corrective Action Request (CAR) 200200055 were incorporated into the RI-ISI Program. This modification replaced a troublesome safety injection valve with a new one slightly upstream.

- A steam generator replacement modification was completed since the initial RI-ISI application. This modification was primarily a "like-for-like" replacement and as such had minimal impact on the RI-ISI application. Minor updates were made to the RI-ISI documents to account for the replacement, deletion and addition of welds associated with the modification.

- Changes to the plant PRA resulted in changes to the consequence rankings and in subsequent revisions to the RI-ISI Program for three consequence segments. These changes were all on Low Risk segments which remained Low Risk after the changes were incorporated. As such, there was a negligible impact on the overall RI-ISI application.

- The Class 2, 4" NPS (nominal pipe size) auxiliary feedwater lines from the outboard containment isolation valve to the connection to the main feedwater piping in all four trains were added to the RI-ISI Program. This resulted from a change in ASME Section XI Code criteria when updating to the 1998 Edition with 2000 Addenda such that Class 2, 4" NPS and smaller auxiliary feedwater piping is no longer exempt.

- The Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) values for the Class 2, 4" auxiliary feedwater piping addressed above were higher than the upper bound values previously used in the risk impact analysis. New upper bound values of 2.4E-2 and 2.4E-3 were used in the updated risk impact analysis for CCDP and CLERP, respectively.

- Class 2 Borated Refueling Water (BN) system piping greater than 4" NPS shown on Callaway Drawing No. ISI-M-22BN01(Q) was considered to be exempt during Interval 2, but is being considered as non-exempt piping during Interval 3 to be consistent with the Wolf Creek Nuclear Generating Station. This will have minimal impact on the RI-ISI Program as this piping was conservatively evaluated as non-exempt piping during the initial RI-ISI application to remain consistent with the other (Strategic Teaming & Resource Sharing) STARS utilities' RI-ISI submittals. As such, the RI-ISI bases have already been established for the BN system.

- Based on ongoing industry experience with primary water stress corrosion cracking (PWSCC), Assumption No. 7 in the Degradation Mechanism Evaluation was deleted. This assumption was stated as follows:

  "Bi-metallic welds with Inconel buttering are not considered susceptible to the PWSCC degradation mechanism."

For the third interval, Callaway will comply with the criteria of MRP-139 (per the EPRI Materials Reliability Program) for welds that are potentially susceptible to PWSCC. At Callaway, this consists of 14 welds where piping attaches to the reactor pressure vessel and pressurizer. MRP-139 is independent from the RI-ISI Program, yet these 14 welds are in the scope of both programs. From both a technical and administrative standpoint, precedence for the examination of the 14 welds that are potentially susceptible to PWSCC will be taken from how RI-ISI Programs at BWRs comply with the NRC-mandated program for examining welds that are potentially susceptible to intergranular stress corrosion cracking (IGSCC).

During the update of the Callaway ISI Program in preparation for the third ISI interval, other minor corrections were identified (e.g., correction of weld numbers) and were evaluated as part of the RI-ISI Living Program Update. These had no impact on the RI-ISI Program beyond requiring minor editorial corrections to the RI-ISI Program documents.

The RI-ISI Program was reevaluated for the seven issues and other minor corrections using the applicable portions of the same risk-informed process that originally established the risk-informed inspection program. The reevaluation was performed by inserting the new information at the appropriate levels of the analysis. All of the cases that were evaluated in the risk impact analysis during the original RI-ISI application were reevaluated using the new information that was determined for the current application. Results of the risk impact reanalysis were that the overall plant risk, measured as a change in Core Damage Frequency and Large Early Release Frequency, was decreased as a result of the application of the new information. As such, the RI-ISI application on Class 1 and 2 piping welds still maintains an acceptable level of quality and safety.

A summary table of the welds in the RI-ISI Program is provided in Attachment 1. The table reflects changes made as a result of the issues addressed above

6. Duration of Proposed Alternative

This 10CFR50.55a Request is proposed for use during the third inspection interval which began on December 19, 2005.

7. Precedents

The RI-ISI alternative proposed in this 10CFR50.55a Request was previously proposed in a Relief Request that was submitted to the NRC for Callaway's second 10-year inspection interval via AmerenUE Letter ULNRC-4392, dated February 16, 2001. That request to implement the RI-ISI methodology for Class 1 and 2 piping welds, which was based on the information initially provided in accordance with the RI-ISI template, was approved by the NRC by letter dated January 30, 2002 (TAC No. MB1205).

Resubmittal of an RI-ISI application for Class 1 and 2 piping welds was done by the V.C. Summer Nuclear Station for their third ISI interval via South Carolina Electric & Gas Company Letter RC-04-0148, dated September 8, 2004. NRC approval of the requested, continued RI-ISI application per V. C. Summer's 10CFR50.55a Request was subsequently granted by NRC letter dated September 6, 2005 (TAC No. MC4323).

Attachment 1
Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System(1)	Risk		Consequence Rank	Failure Potential		Code Category	1st Approved RI-ISI Interval			New RI-ISI Interval		
	Category	Rank		DMs	Rank		Weld Count	RI-ISI	Other(2)	Weld Count	RI-ISI	Other(2)
AB	6	Low	Medium	None	Low	C-F-2	152	0		157	0	
AE	2	High	High	TASCS	Medium	C-F-2	0(3)	0(3)		9(3)	6(3)	
AE	2	High	High	TT	Medium	C-F-2	0(3)	0(3)		8(3)	2(3)	
AE	4	Medium	High	None	Low	C-F-2	0(3)	0(3)		6(3)	1(3)	
AE	5 (3)	Medium (High)	Medium	TASCS, (FAC)	Medium (High)	C-F-2	17	2		18	2	
AE	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-1	16	0		16	0	
						C-F-2	85	0		90	0	
AL	6	Low	Medium	None	Low	C-F-2	0(3)	0(3)		126(3)	0(3)	
BB	2	High	High	TASCS, TT	Medium	B-J	9	3		9	3	
BB	2	High	High	TASCS	Medium	B-J	24	6		24	6	
BB	2 (2)	High (High)	High	TT, (PWSCC)	Medium (Medium)	B-F	0(4)	0(4)		1(4)	0(4)	
BB	2	High	High	TT	Medium	B-F	1(4)	0(4)		0(4)	0(4)	
						B-J	33	8		33	8	
BB	4 (2)	Medium (High)	High	None, (PWSCC)	Low (Medium)	B-F	0(4)	0(4)		13(4)	5(4)	
BB	4	Medium	High	None	Low	B-F	21(4)	5(4)		8(4)	0(4)	
						B-J	325	29		342	34	
BB	5	Medium	Medium	TT	Medium	B-J	2	1		2	1	
BB	6	Low	Medium	None	Low	B-J	30	0		30	0	
BG	2	High	High	TT	Medium	B-J	3	1		3	1	
BG	4	Medium	High	None	Low	B-J	5	0		5	0	
						C-F-1	84	9		84	9	
BG	5	Medium	Medium	TT	Medium	B-J	4	1		4	1	
BG	6	Low	Medium	None	Low	B-J	1	0		1	0	
						C-F-1	40	0		40	0	
BG	7	Low	Low	None	Low	C-F-1	14	0		14	0	

Attachment 1

Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	1st Approved RI-ISI Interval			New RI-ISI Interval		
	Category	Rank		DMs	Rank		Weld Count	RI-ISI	Other[2]	Weld Count	RI-ISI	Other[2]
BN	4	Medium	High	None	Low	C-F-1	4	0		4	0	
BN	6	Low	Medium	None	Low	C-F-1	5	0		111	0	
EF	5	Medium	Medium	MIC, PIT	Medium	C-F-2	4	1		4	1	
EF	6	Low	Medium	None	Low	C-F-2	46	0		46	0	
EJ	4	Medium	High	None	Low	B-J	19	2		19	2	
						C-F-1	431	44		431	44	
EJ	7	Low	Low	None	Low	C-F-1	35	0		35	0	
EM	5	Medium	Medium	IGSCC	Medium	B-J	20	2		19	2	
EM	6	Low	Medium	None	Low	B-J	110	0		107	0	
						C-F-1	206	0		206	0	
EM	7	Low	Low	None	Low	C-F-1	24	0		24	0	
EN	6	Low	Medium	None	Low	C-F-1	86	0		86	0	
EP	5	Medium	Medium	IGSCC	Medium	B-J	12	2		12	2	
EP	6	Low	Medium	None	Low	B-J	87	0		93	0	

Notes:

1. System designations are as follows:
   - AB – Main Steam System
   - AE – Main Feedwater System
   - AL – Auxiliary Feedwater System
   - BB – Reactor Coolant System
   - BG – Chemical and Volume Control System
   - BN – Borated Refueling Water Storage System
   - EF – Essential Service Water System
   - EJ – Residual Heat Removal System
   - EM – High Pressure Coolant Injection System
   - EN – Containment Spray System
   - EP – Accumulator Safety Injection System

Notes (con't):

2. The column labeled "Other" is generally used to identify augmented inspection program locations that are credited beyond those locations selected per the RI-ISI process, as addressed in Section 3.6.5 of EPRI TR-112657. This option was not applicable for the Callaway Plant RI-ISI application. The "Other" column has been retained in this table solely for uniformity purposes with other RI-ISI application submittals.

3. Due to a change in ASME Section XI Code criteria, 4" NPS Class 2 auxiliary feedwater piping was added to the ISI Program, and therefore the RI-ISI Program, for the first time during the third ISI interval. This consisted of Class 2 piping from the outboard isolation valve to the first check valve (i.e., system "AL") and piping from the first check valve to the branch connection to feedwater (i.e., system "AE") in all four trains. This piping and its associated weldments were outside the scope of the original RI-ISI application.

4. Changes to the information shown for former Code Category B-F welds reflect the implementation of MRP-139 as an augmented examination and mitigation program.

10 CFR 50.55a Request Number I3R-02

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

Alternative Provides Acceptable Level of Quality and Safety

1. <u>ASME Code Components Affected</u>

   All ASME Section XI Code Class 1, 2, and 3 Vessel Welded Attachments

2. <u>Applicable Code Edition and Addenda</u>

   ASME Boiler and Pressure Vessel Code, Section XI 1998 Edition through 2000 Addenda.

3. <u>Applicable Code Requirement</u>

   Relief is requested from the requirements of:

   Table IWB-2500-1, Examination Category B-K, Footnote 4;
   Table IWC-2500-1, Examination Category C-C, Footnote 4; and
   Table IWD-2500-1, Examination Category D-A, Footnote 3.

4. <u>Reason for Request</u>

   This 10CFR50.55a Request addresses two issues. First, ASME Section XI, 1998 Edition through 2000 Addenda, Table IWB-2500-1, Examination Category B-K, Footnote 4; Table IWC-2500-1, Examination Category C-C, Footnote 4; and Table IWD-2500-1, Examination Category D-A, Footnote 3 do not include specific criteria for the selection of welded attachments in situations where a plant has multiple vessels of similar design, function and service. The requirements in these footnotes do specify that "only one welded attachment of only one of the multiple vessels shall be selected for examination," but no additional criteria are provided for the selection of the appropriate welded attachment.

   Second, the identified footnotes do not provide any specific criteria for the selection of welded attachments on single vessels. The current wording under the "Extent of Examination" in Tables IWX-2500-1, Categories B-K, C-C and D-A can be interpreted to require that <u>all</u> welded attachments on a single vessel be examined. This wording has been reconsidered by the ASME Code Committee, and Code Case N-700 (attached) has been published to clarify the requirements for examining welded attachments on both multiple and single vessels.

One of the bases for Code Case N-700 was previously published Code Case N-509, "Alternative Rules for the Selection and Examination of Class 1, 2, and 3 Integrally Welded Attachments," which was incorporated into ASME Section XI in the 1995 Addenda. The technical basis of Code Case N-509 concluded that there have been very few welded attachment failures identified during normal Section XI examinations. Instead, failures have been identified when the connected support member has been found to be deformed due to operational transients or water hammer events. For this reason, Code Case N-509 and Section XI versions that include the 1995 and later Addenda require welded attachments to be examined whenever component support deformation is identified. In addition, a sampling plan for welded attachments was maintained.

Although Code Case N-509 and ASME Section XI beginning with the 1995 Addenda represent an improvement to previously stated Code criteria for the examination of welded attachments, neither addresses the examination of welded attachments on a single vessel nor which welded attachment should be selected for examination. Code Case N-700 provides clarification for the selection of Class 1, 2, and 3 vessel welded attachments for examination, and was developed to address the selection criteria currently not included in Code Case N-509 and ASME Section XI, 1998 Edition through 2000 Addenda.

Code Case N-700 clarifies the requirements for the examination of welded attachment on vessels by stating the following criteria:

1) For multiple vessels of similar design, function and service, only one welded attachment of only one of the multiple vessels shall be selected for examination.

2) For single vessels, only one welded attachment shall be selected for examination.

3) The attachment selected for examination on one of the multiple vessels or the single vessel, as applicable, shall be an attachment under continuous load during normal system operation, or an attachment subject to a potential intermittent load (seismic, water hammer, etc.) during normal system operation if an attachment under continuous load does not exist.

Because the selection criteria provided by Code Case N-700 are supported by the same failure data that forms the basis for Code Case N-509 and since they also address scenarios not specifically or adequately addressed by Section XI, the alternative requirements of N-700 are deemed to be a more complete and detailed set of rules for the selection of welded attachments on vessels. Accordingly, pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative provides an acceptable level of quality and safety

5. **Proposed Alternative and Basis for Use**

In lieu of implementing the requirements of Table IWB-2500-1, Examination Category B-K, Footnote 4; Table IWC-2500-1, Examination Category C-C, Footnote 4; and Table IWD-2500-1, Examination Category D-A, Footnote 3, it is proposed that the alternative requirements of Code Case N-700 be implemented at the Callaway Nuclear Power Plant.

6. **Duration of Proposed Alternative**

Selection of ASME Section XI Code Class 1, 2, and 3 vessel welded attachments would be performed in accordance with Code Case N-700 for the remainder of the third inspection interval (which began on December 19, 2005), or until such time that Code Case N-700 is approved for use by reference in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." Once accepted in Regulatory Guide 1.147, any conditions specified in Regulatory Guide 1.147 will be followed.

7. **Precedents**

By letter dated March 11, 2005, the Tennessee Valley Authority (TVA) requested relief from the ASME Section XI inservice inspection requirements for vessel welded attachments at Browns Ferry Units 2 and 3. In accordance with 10CFR50.55a(a)(3)(i), TVA requested the use of Code Case N-700 as an alternative. In a letter dated July 18, 2005 (TAC Nos. MC6437 and MC6438), the NRC approved TVA's 10CFR50.55a request and authorized it use for the remainder of the ongoing interval or until such time that Code Case N-700 is approved for use in Regulatory Guide 1.147.

Approval Date: November 18, 2003

See Numeric Index for expiration and any reaffirmation dates.

Case N-700
Alternative Rules for Selection of Classes 1, 2, and 3 Vessel Welded Attachments for Examination
Section XI, Division 1

Inquiry: What alternative rules may be used in lieu of those required by Table IWB-2501-1, Table IWC-2500-1, and Table IWD-2500-1, Examination Categories B-K and C-C, footnote 4, and Examination Category D-A, footnote 3, for selection of vessel welded attachments for examination?

Reply: It is the opinion of the Committee that for multiple vessels of similar design, function and service, only one welded attachment of only one of the multiple vessels shall be selected for examination. For single vessels, only one welded attachment shall be selected for examination. The attachment selected for examination on one of the multiple vessels or the single vessel, as applicable, shall be an attachment under continuous load during normal system operation, or an attachment subject to a potential intermittent load (seismic, water hammer, etc.) during normal system operation if an attachment under continuous load does not exist.

10 CFR 50.55a Request Number I3R-04

Relief Request
In Accordance with 10 CFR 50.55a(g)(5)(iii)

Inservice Inspection Impracticality

1. **ASME Code Components Affected**

   Reactor Vessel Supports, Component Numbers 2-RBB01-01, 2-RBB01-02, 2-RBB01-03 and 2-RBB01-04

2. **Applicable Code Edition and Addenda**

   ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition through 2000 Addenda

3. **Applicable Code Requirement**

   Relief is requested from the requirements of ASME Section XI, Table IWF-2500-1, Category F-A, Item Number F1.40 which requires that 100% of Class 1 supports, other than piping supports, be subject to a visual VT-3 examination once every inspection interval.

4. **Impracticality of Compliance**

   Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that compliance with the specified requirements is impractical. Conformance with the applicable inservice inspection requirements would necessitate a design modification to the reactor pressure vessel supports and associated insulation/walkplate to allow 100% visual examination of the subject supports. In addition, limited accessibility and high radiation levels in the area where these supports are located further reduces the percentage of the supports available for visual examination.

   The Callaway reactor vessel is supported by two cold leg nozzles and two hot leg nozzles. There is a support assembly at each of these nozzles that consists of a nozzle weld build-up, shoe plate, air-cooled box, and steel support structure embedded in the primary shield wall. Figures 1 and 2 depict these support assemblies. As shown in these figures, only the nozzle weld build-up and shoe plate are completely accessible for a visual VT-3 examination. Most of the air-cooled box and the entire steel support structure are located beneath a steel walk plate, and only the top of the air-cooled box is directly accessible. An additional 20 to 30 percent of the air-cooled box and a very small percentage of the steel support structure would be made accessible if the steel walk plate and insulation were removed.

The reactor vessel supports are located in a confined space below the refueling pool permanent seal ring. The area can only be accessed through four seal ring hatches. In addition to difficult access, the radiation level in the area is between 1.5 and 2.0 man-rem per hour.

5. **Burden Caused by Compliance**

The large cost of a design modification to the reactor pressure vessel supports and associated insulation/walkplate to allow 100% visual examination of the subject supports is deemed an undue burden. Further, it is estimated that the removal and re-installation of the walk plate and insulation in this confined space, combined with the performance of the visual VT-3 examination, would result in an exposure of approximately 36 man-rem. Removal of the walk plate and insulation, under these conditions, in order to increase the examination coverage of the air-cooled box by approximately 20 to 30 percent and a very small percentage of the steel support structure is considered impractical without a commensurate increase in quality or safety.

6. **Proposed Alternative and Basis for Use**

In lieu of implementing the requirements of Table IWF-2500-1, Category F-A, Item No. F1.40, Callaway proposes to perform a limited VT-3 visual examination, with the walk plate and insulation installed, on the accessible NF portions of the Reactor Vessel support assemblies. If conditions are discovered during this limited VT-3 examination that do not meet the acceptance standards of IWF-3400, the walk plate or insulation will, if necessary, be removed in order to meet the requirements of IWF-3122.2 or IWF-3122.3, as applicable.

7. **Duration of Proposed Alternative**

This 10CFR50.55a request is being proposed for use during the third inspection interval which began on December 19, 2005.

8. **Precedents**

The alternative proposed in this 10CFR50.55a request was proposed in a previous Relief Request (ISI-03) that was submitted to the NRC for Callaway's second 10-year inspection interval, via Union Electric Company letters dated October 12, 1994 and August 18, 1995. Approval of the requested relief was granted by NRC letter from William H. Bateman, USNRC to Donald F. Schnell, dated 12/20/95 (TAC No. M90859).

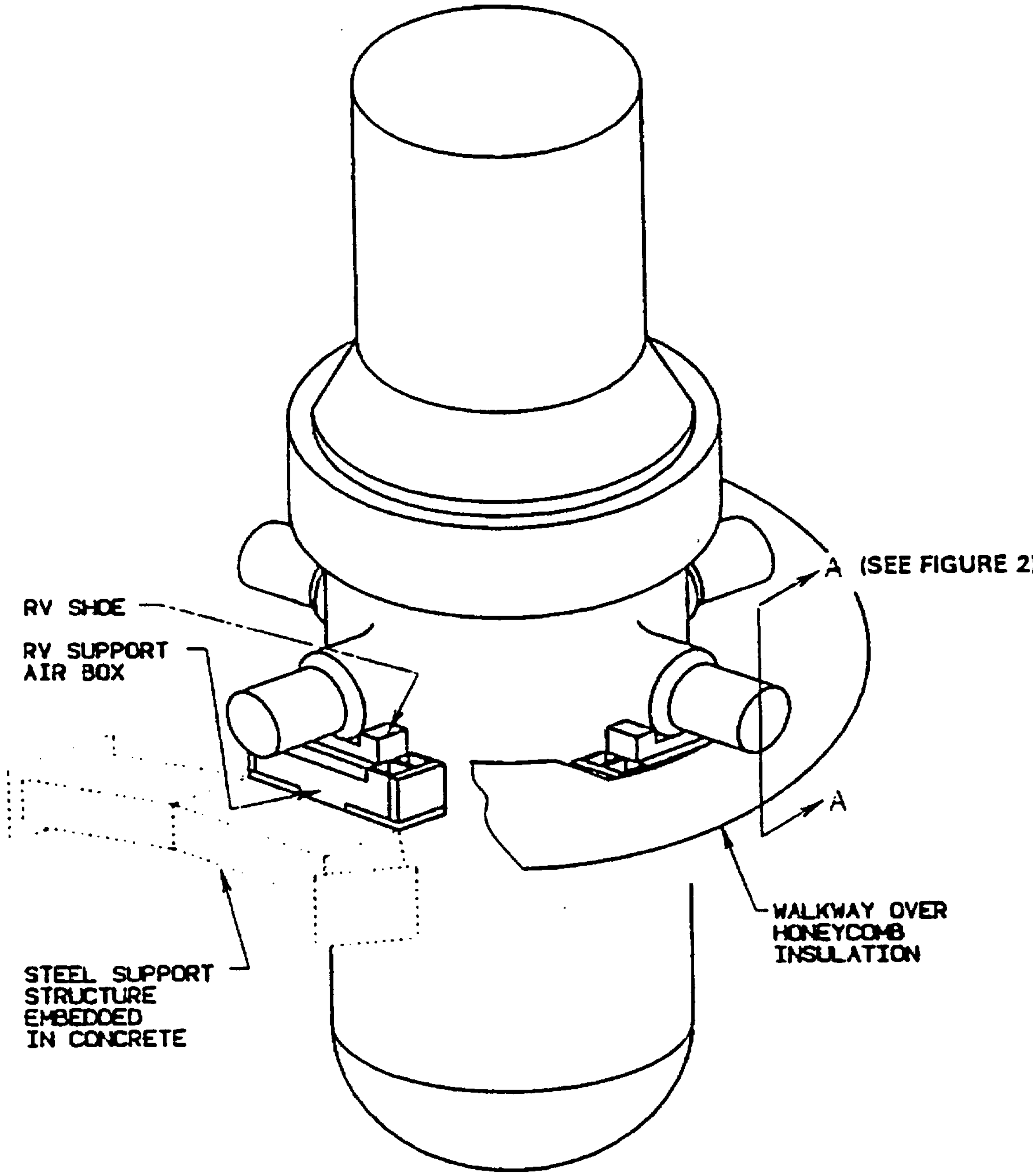
A (SEE FIGURE 2)
RV SHOE
RV SUPPORT
AIR BOX
A
WALKWAY OVER
HONEYCOMB
INSULATION
STEEL SUPPORT
STRUCTURE
EMBEDDED
IN CONCRETE

FIGURE 1

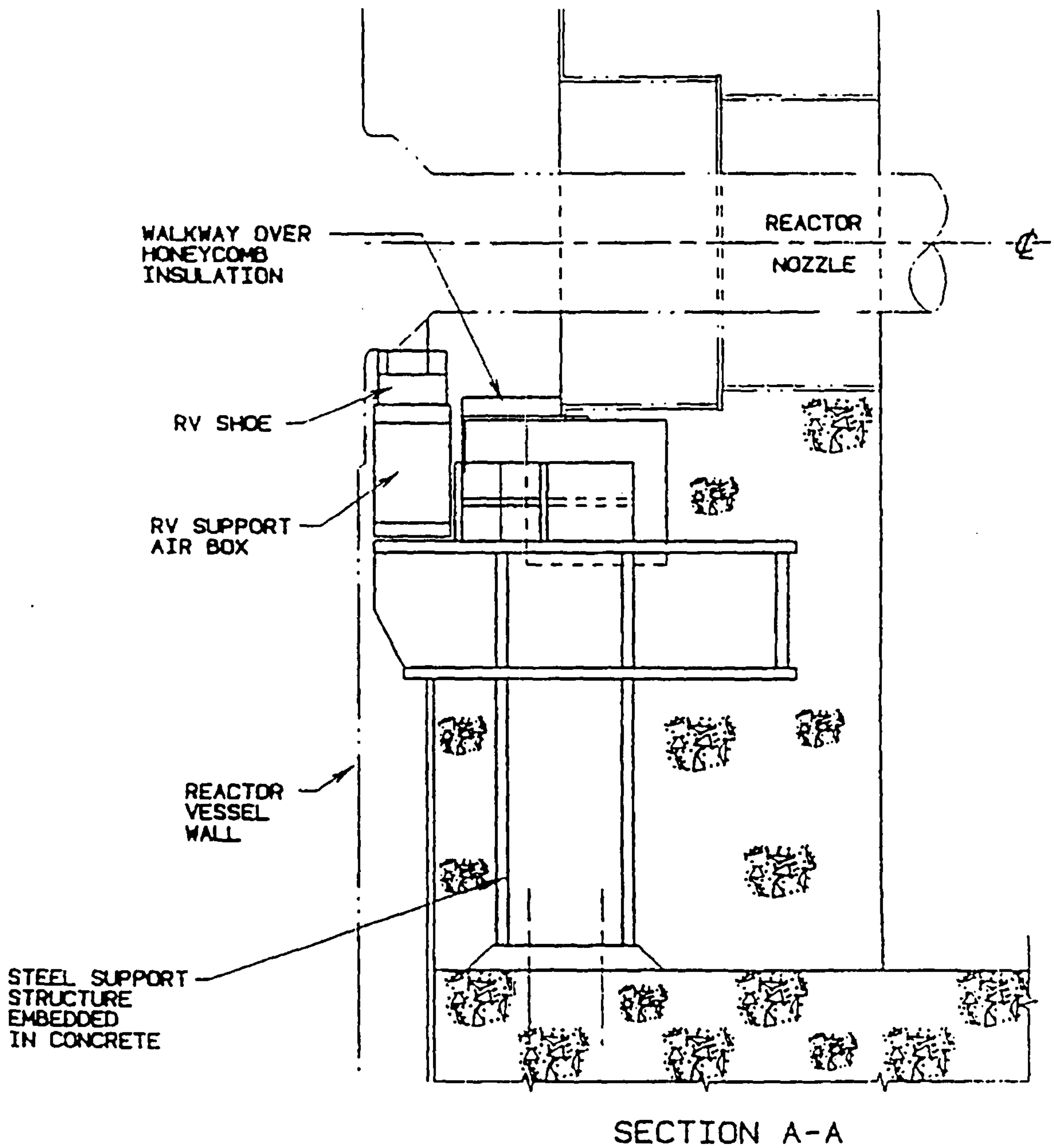
WALKWAY OVER HONEYCOMB INSULATION
REACTOR
NOZZLE
RV SHOE
RV SUPPORT AIR BOX
REACTOR VESSEL WALL
STEEL SUPPORT STRUCTURE EMBEDDED IN CONCRETE
SECTION A-A

FIGURE 2