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FPLEnergy.

Duane Arnold Energy Center

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NG-06-0866
10 CFR 50.55a

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

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Responses to Requests for Additional Information and Revised Relief Requests NDE-R005 and NDE-R007 – Fourth 10-Year Inservice Inspection Program

References
1) Letter from G. Van Middlesworth (FPL Energy) to USNRC, "Fourth 10-Year Inservice Inspection Program," dated June 30, 2006.
2) Letter from Richard B. Ennis (USNRC) to G. Van Middlesworth (FPL Energy) "Request for Additional Information Related to Relief Request NDE-R005 and NDE-R007 (TAC Nos. MD2521 and MD2523)," dated December 8, 2006.

Reference 1 submitted the Fourth 10-Year Inservice Inspection Program including requests for relief to the NRC for approval. Reference 2 requested additional information regarding Relief Requests NDE-R005 and NDE-R007 included in Reference 1. Attachment A provides the responses to the requests of Reference 2 and Attachment B provides revised relief requests based on the responses. Changes to the relief requests previously submitted in Reference 1 are marked in Attachment B with revision lines.

Approval of NDE-R005 is requested by January 30, 2007 and approval of NDE-R007 is requested by October 1, 2007.

This letter makes no new commitments nor changes to any existing commitments. If you have any questions or require additional information, please contact Steve Catron at (319) 851-7234.

Gary Van Middlesworth

Gary Van Middlesworth
Site Vice President, Duane Arnold Energy Center
FPL Energy Duane Arnold, LLC

A047

Attachments: A) Response to Request for Additional Information on Relief Requests NDE-R005 and NDE-R007

B) Revised Relief Requests NDE-R005 and NDE-R007

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Resident Inspector, DAEC, USNRC

ATTACHMENT A

Response to Request for Additional Information on Relief Requests NDE-R005 and NDE-R007

3 Pages to Follow

ATTACHMENT A

Response to Request for Additional Information on Relief Requests NDE-R005 and NDE-R007

Relief Request NDE-R005

1. Page 29 of 45 of the DAEC Fourth Ten Year ISI Plan - Relief Request NDE-R005 states that *"This Table shows the final consequence ranking has not changed for individual line segments, and therefore the change in risk assessment for the new inspection interval as compared to the original RI-ISI [risk-informed ISI] submittal meets the acceptance criteria of the original RI-ISI submittal."* However we identified some inconsistencies below:

1a. Note (16) of this table says that *"The risk rank was increased from low to medium as an effect from the Updated PSA [probabilistic safety assessment]."*

DAEC Response:

Table 1 of NDE-R005 has been updated with the appropriate Risk Ranks. The combination of Failure Potential and Consequence Rank provides a Risk Rank of "Low". Therefore, Note 16 has been deleted and the referenced risk ranks for the affected RCIC and HPCI segments has been changed from "Medium" to "Low." See the revised NDE-R005 in Attachment B of this submittal.

1b. The number of welds selected for the nondestructive examination for the fourth interval are reduced, especially for control rod drive system. Please provide an explanation for this inconsistency and the rationale for the reduction.

DAEC Response:

The original third interval submittal did not commit to examining the CRD segment for RI-ISI. There is a typographical error on what was originally committed to. The original number of required ASME Section XI exams was listed instead of the RI-ISI number of exams for that segment. Therefore, the stated number of welds selected for examination for the control rod drive system in the First Approved Interval column has been changed from 2 to zero. See the revised NDE-R005 in Attachment B of this submittal.

1c. In Table 1, the risk ranking of the reactor core isolation cooling system is increased from low to medium as an effect from the updated PSA, but none of the welds were selected for the inspection, please explain.

DAEC Response:

As stated in the response to 1a, Table 1 was updated with the Risk Rank as a restatement of the Consequence Rank, and not as a combination of the Failure Potential and the Consequence Rank. The combination of Failure Potential and Consequence Rank provides a Risk Rank of "Low," and therefore, none of the welds are selected for inspection.

2. Where your explanations in response to items 1a, 1b, and 1c warrant, please provide results from the cumulative risk impacts analysis demonstrating that the change in risk associated with implementation of the probabilistic risk assessment (PRA) Revision 5B model is consistent with the change in the Electric Power Research Institute TR-112657 Revision B-A risk guidelines.

DAEC Response:

A risk impact analysis was performed for the DAEC in support of the original RI-ISI submittal (Ref: M453-050, "Risk Impact Analysis for the Duane Arnold Energy Center"). The evaluation--which was conducted using the guidance of Section 3.7 of EPRI TR-112657--concluded that unacceptable risk impacts will not occur from implementation of the RI-ISI program and that the acceptance criteria of Regulatory Guide 1.174 are satisfied. Values of conditional core damage probability (CCDP) and conditional large/early release probability (CLERP) used in this evaluation are upper bound values and are not dependent on the version of the PRA model used in deriving the risk consequence of individual line segments in the present relief request. Since the risk ranking of line segments is not changed and the number of exams is not decreased as a result of this relief request, the results and conclusions of the original analysis are unaffected. Therefore, there are no cumulative risk impacts associated with implementation of the PRA Revision 5B model.

3. The RI-ISI consequence results should also be ranked based on the impact on containment performance. Please provide the DAEC large early release frequency (LERF) and PRA model version/date used in support of the RI-ISI relief.

DAEC Response:

Impact of containment performance was considered in determining the consequence of postulated pipe breaks in the subject RI-ISI relief request. The Revision 5B Level II PRA model was used to assess containment performance. This model was completed concurrent with the Revision 5B Level I model, in February 2005. The value of LERF when quantified at a low truncation limit (1E-12 per year) is 1.23E-06 per year.

4. There is no specific mention of the Boiling Water Reactors Owners' Group PRA certification associated with the LERF calculation. Please confirm if this certification has or has not been performed. If the certification has been performed, have the Level A and B Facts and Observations (F&Os) that the peer review team identified regarding LERF been resolved and/or incorporated into the models? If the F&Os have not been incorporated, please state why the unincorporated F&Os are not expected to have an impact on the RI-ISI consequence evaluation.

DAEC Response:

The DAEC Level II PRA model, which is used to calculate LERF, was included in the BWR Owners' Group (BWROG) PRA program peer review and disposition of all Level A and Level B Facts and Observations from this review was completed during the Revision 5 PRA model update project.

Relief Request NDE-R007

1. The system pressure test for nonisolable buried components in the Essential Service Water (ESW) piping requires a test to confirm that flow during operation is not impaired. Relief Request NDE-R007 proposes an alternative to use the results of quarterly pump testing under the inservice test (IST) program for pumps, along with a visual examination of the ground surface, to verify that the flow is not impaired in the buried portion of ESW piping. The NRC staff requests the licensee to clarify how the results of the pump test will be used to confirm adequate flow from a qualitative determination of flow impairment in the buried ESW piping.

DAEC Response –

Relief Request NDE-R007 has been revised to address these issues. See the revised NDE-R007 in Attachment B of this submittal.

ATTACHMENT B

Revised Relief Requests NDE-R005 and NDE-R007

11 Pages to Follow

(5) Relief Request NDE-R005

COMPONENT IDENTIFICATION

Class:	1 and 2
References:	IWB-2500-1
Examination Category:	Class 1 B-F & B-J, and Class 2 C-F-2 Welds
Item Numbers:	B5.10, B5.20, B5.30 B9.11, B9.21, B9.31, B9.32, B9.40, C5.51, C5.81
Component Numbers:	Various

CODE REQUIREMENT

ASME Code Section XI 2001 Edition with 2003 Addenda, IWB-2500-1 requires in part that for each successive 10-Year ISI Interval, 100% of Category B-F welds for the ASME Class 1 piping 4" NPS and greater be selected for volumetric and surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 100% of Category B-F welds for the ASME Class 1 piping less than 4" NPS be selected for surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 100% of Category B-F socket welds for the ASME Class 1 piping be selected for surface examination. IWB-2500-1 requires in part that for each successive 10-Year ISI Interval, 25% of Category B-J welds for the ASME Class 1 piping 4" NPS and greater be selected for volumetric and surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 25% of Category B-J welds for the ASME Class 1 piping less than 4" NPS be selected for surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 25% of Category B-J socket welds for the ASME Class 1 piping be selected for surface examination. IWC-2500-1 requires in part that for each successive 10-Year Interval, 7.5% of C-F-2 welds be examined for ASME Class 2 piping greater than 4" NPS and 3/8" or greater nominal wall thickness for volumetric and surface examination. IWC-2500-1 requires in part that for each successive 10-Year Interval, 7.5% of C-F-2 welds be examined for ASME Class 2 piping 2" NPS or less for surface examination.

REASON FOR RELIEF REQUEST

Section XI, Examination Categories B-F and B-J currently contain the requirements for the non-destructive examination (NDE) of Class 1 piping components. Section XI, Examination Category C-F-2 currently contains the requirements for the NDE of Class 2 piping components. The previously approved Risk Informed Inservice Inspection (RI-ISI) Program (Reference 1) will be substituted for Class 1 and Class 2 piping (Examination Categories B-F, B-J, and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. For example, existing pressure testing requirements remain unchanged.

BASIS FOR RELIEF AND ALTERNATIVE EXAMINATIONS

Pursuant to 10 CFR 50.55a(a)(3)(i), NRC approval of the DAEC RI-ISI as an alternative to the current 2001 Edition through 2003 Addenda, ASME Section XI inspection requirements for Class 1 and Class 2 Code Examination Category B-F, B-J, and C-F-2 piping welds is requested. This request is to extend the relief previously granted to include the Fourth Interval.

The DAEC RI-ISI Program has been developed in accordance with the Electric Power Research Institute (EPRI) methodology contained in EPRI Topical Report TR-112657 Revision B-A, "Risk-Informed Inservice Inspection Evaluation Procedure'" (Reference 2). It was approved for use at DAEC during the 2nd and 3rd Periods of the 3rd Inspection Interval and is requested to be applicable for the 4th Inspection Interval. The DAEC specific RI-ISI program is summarized in Table 1. This Table reflects the recommended approach as provided in the Nuclear Energy Institute (NEI) 04-05 "Living Program Guidance To Maintain Risk-Informed Inservice Inspection Programs For Nuclear Piping Systems" (April 2004) for requesting relief to continue the RI-ISI program into the next inspection interval. This Table shows the final consequence ranking has not changed for individual line segments. Since the final consequence ranking has not changed and since the number of exams is not being decreased, the cumulative risk impact analysis performed in support of the original RI-ISI submittal is unaffected. This evaluation, which used bounding values of conditional core damage probability (CCDP) and conditional large/early release probability (LERP), concludes that unacceptable risk impacts will not occur from implementation of the RI-ISI program and that acceptance criteria of Regulatory Guide 1.174 are satisfied. The RI-ISI program was updated after a rigorous review of inputs and technical elements of the original submittal consistent with the intent of NEI-04-05 (Reference 3) and continues to meet EPRI TR-112657 and Reg. Guide 1.174 risk acceptance criteria. The current Class 1 and 2 piping weld scope is consistent with the submitted scope approved for the 3rd Interval ISI Program as described in Reference 1. The original list DAEC intended to credit for Class 1 or 2 RI-ISI piping weld exams has been substituted on specific occasions with similar welds due to accessibility issues that would have resulted in reduced exam volumes. DAEC chooses welds for examination that are classified within the same risk matrix classification segment, using the same treatment criteria as those originally selected in the first submittal. Socket welds that are chosen by the RI-ISI program for exam will be subjected to VT-2 exams as described by Code Case N-578-1. Welds chosen based on risk consequence alone will be volumetrically examined per ASME Section XI Code 2001Edition through the 2003 Addenda requirements for B-F, B-J, or C-F-2 welds depending on weld type.

The 3rd Interval RI-ISI program required DAEC to complete 38.7 % of the Section XI exams in the 1st Period and the remaining 61.3% of the RI-ISI program welds were to be completed by the end of the 3rd Inspection Interval. This Relief Request is to align the RI-ISI Interval and Code Year with the 4th Interval ISI Program. Therefore, 100% of the RI-ISI Program weld examinations will be completed in the 4th Inspection Interval.

All PRA inputs reported in the RI-ISI relief are derived from the Revision 5B PRA model, which was completed in February of 2005. The base core damage frequency value from this model, excluding internal flooding initiated sequences, is 1.10E-05 per year. The base large early release frequency is 1.23E-06 per year. This same Revision 5B PRA model was used as input to the Mitigating Systems Performance Index (MSPI).

Because of its on-going use as a decision-making tool, the DAEC PRA has been through a peer review as part of the BWR Owners' Group PRA certification program. The peer review team concluded that all of the graded elements are of sufficient detail and quality to support a risk significance determination supported by deterministic insights. The review team also commented on the DAEC's excellent PRA documentation and very consistent level of quality across all elements of the certification.

IMPLEMENTATION SCHEDULE

Relief is requested for extension into the Fourth Ten-Year Interval of the DAEC Inservice Inspection Program.

PRECEDENTS

USNRC previously approved the DAEC RI-ISI program via Reference 1.

ATTACHMENTS

1. Table 1, "Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657, Rev B-A by Risk Category."
2. Table 2, "System Selection and Segment/Element Definition"

REFERENCES

1. USNRC Letter dated January 17, 2003 "Duane Arnold Energy Center – Risk Informed Inservice Inspection Program" (TAC No. MB4751).
2. Revised Risk-Informed Inservice Inspection Evaluation Procedure, EPRI, Palo Alto, CA: 1999. TR-112657, Rev B-A.
3. NEI-04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems," dated April 2004.

Table 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	Weld Count	First Approved Interval		New Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RPV	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-F	2	0		0	
						B-J	2	0		0	
RPV	6	Low	Medium	None	Low	B-F	6	0		0	
						B-J	21	0		0	
RCR	2 (2)	High (High)	High	TT (IGSCC)[15]	Medium (Medium)	B-F	8	2[4]		2[4]	
RCR	2 (2)	High (High)	High	TT (IGSCC)	Medium (Medium)	B-J	69	18[4]		18[4]	
RCR	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	2	0	2[5]	0	2[5]
						B-J	32	4[6]		4[6]	
RCR	5	Medium	Medium	TASCS	Medium	B-J	5	1		1	
RCR	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	26	0		0	
RCR	6	Low	Medium	None	Low	B-J	43	0		0	
RCR	7	Low	Low	None	Low	B-J	4	0		0	
RWCU	4(2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-J	1	1[7]		1[7]	
RWCU	6(5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-F	1	0		0	
						B-J	22	0		0	
RWCU	6	Low	Medium	None	Low	B-F	1	0		0	
						B-J	27	0		0	
RWCU	7	Low	Low	None	Low	B-J	2	0		0	
RCIC	6	Low	Medium	None	Low	B-J	22	0		0	
						C-F-2	7	0		0	
RCIC	6	Low	Medium	None	Low	B-J	5	0		0	
RCIC	7	Low	Low	None	Low	C-F-2	7	0		0	
RHR	2(2)	High (High)	High	TT (IGSCC)	Medium (Medium)	B-F	2	1[8]		1[8]	
						B-J	2	0		0	
RHR	2	High	High	TT	Medium	B-J	8	2		2	
RHR	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	1	1[9]		1[9]	
						B-J	1	0		0	

Table 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	Weld Count	First Approved Interval		New Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RHR	4	Medium	High	None	Low	B-J	7	1		1	
RHR	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	1	0		0	
RHR	6	Low	Medium	None	Low	B-J	31	0		0	
						C-F-2	433	0		0	
CS	2(2)	High (High)	High	(IGSCC)[15]	Low[15](Medium)	B-F	2	1[10]		1[10]	
CS	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	4	0	4[11]	0	4[11]
						B-J	2	1[12]	1[13]	1[12]	1[13]
CS	4	Medium	High	None	Low	B-J	16	2		2	
CS	6	Low	Medium	None	Low	B-J	22	0		0	
						C-F-2	136	0		0	
HPCI	4	Medium	High	None	Low	B-J	3	3		3	
						C-F-2	49	3		3	
HPCI	6	Low	Medium	None	Low	B-J	7	0		0	
						C-F-2	91	0		0	
HPCI	6	Low	Medium	None	Low	B-J	9	0		0	
HPCI	7	Low	Low	None	Low	C-F-2	12	0		0	
MS	4	Medium	High	None	Low	B-J	60	6		6	
MS	6(3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	7	0		0	
MS	6	Low	Medium	None	Low	B-J	38	0		0	
						C-F-2	147[17]	0		0	
FW	2(1)	High (High)	High	TASCS, TT (FAC)	Medium (High)	B-J	8	2		2	
FW	2(1)	High (High)	High	TASCS, CC (FAC)	Medium (High)	B-J	8	2	3[14]	2	3[14]
FW	2(1)	High (High)	High	TASCS (FAC)	Medium (High)	B-J	3	1		1	
FW	4(1)	Medium (High)	High	None (FAC)	Low (High)	B-J	49	5		5	
FW	5(3)	Medium (High)	Medium	TASCS (FAC)	Medium (High)	B-J	4	1		1	
FW	6(3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	5	0		0	
CRD	4	Medium	High	None	Low	B-J	2	1		1	
CRD	6	Low	Medium	None	Low	B-F	2	0		0	
						B-J	31	0		0	
						C-F-2	27	0		0	

Table 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	Weld Count	First Approved Interval		New Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
SLC	4	Medium	High	None	Low	B-J	6	1		1	
SLC	6	Low	Medium	None	Low	B-F	1	0		0	
						B-J	26	0		0	

Table Notes:

1) Systems are described in Table 2.

2) The column labeled "Other" is used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. DAEC added ten welds as examination selections to bring the overall percentage of Class 1 selections to 10%.

3) Not Used.

4) These twenty welds were selected for examination by both the IGSCC Program and the RI-ISI Program. Thermal Transients were identified along with IGSCC, as a potential damage mechanism for these welds. In order to be credited toward both the IGSCC Program and the RI-ISI Program the IGSCC examinations will include the requirements identified in EPRI TR-112657 for thermal transient examinations.

5) These two welds were selected for examination by the IGSCC Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld selections to 10%. Since IGSCC was the only potential damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

6) These four welds were selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

7) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism identified for this weld, the IGSCC examination will be credited toward both programs..

8) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Thermal transients were identified along with IGSCC as a potential damage mechanism for this weld. In order to be credited toward both the IGSCC Program and the RI-ISI Program, the IGSCC examination will include the requirements identified in EPRI TR-112657 for thermal transient examinations.

9) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism for this weld, the IGSCC examination will be credited to both programs.

10) This one weld was selected for examination by both the IGSCC Program and by the RI-ISI Program. For this weld, IGSCC was identified as the potential damage mechanism.

11) These four welds were selected for examination by the IGSCC Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld selections to 10%. Since IGSCC was the only potential damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

12) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism identified for this weld, the IGSCC examination will be credited toward both programs.

13) This one weld was selected for examination by the IGSCC Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld sections to 10%. Since IGSCC was the only potential damage mechanism identified for this weld, the IGSCC examination will be credited toward both programs.

14) These three welds were selected for examination by the NUREG-0619 Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld selections to 10%. For these welds, TASCS and crevice corrosion were identified as potential damage mechanisms. Although, the NUREG-0619 examinations are included in the RI-ISI Program, they are not credited as risk-informed examinations in the risk impact analysis. As such, the NUREG-0619 examinations by themselves could be credited toward both programs. However, to ensure that all potential damage mechanisms are investigated, DAEC has elected to supplement the NUREG-0619 examinations for these three welds with the requirements identified in EPRI TR-112657 for TASCS and crevice corrosion examinations.

15) Recirculation riser safe-end and Core Spray injection safe-end welds are not considered to be subject to crevice corrosion degradation per the "Enhanced Crevice Corrosion Criteria in RI-ISI Evaluations," EPRI Technical Update 1011945, November 2005. The failure potential ranking for Core Spray was moved from medium to low because of the elimination of the degradation mechanism (crevice corrosion).

16) (Reserved)

17) One new weld in Main Steam due to modifications of Main Steam Reheat System adding one weld.

Table 2
System Selection and Segment /Element Definition

System Description	Number of Segments	Number of Elements
RPV-Reactor Pressure Vessel	11	31
RCR-Reactor Coolant Recirculation	56	189
RWCU-Reactor Water Clean-Up	14	54
RCIC-Reactor Core Isolation Cooling	7	41
RHR-Residual Heat Removal	53	486
CS-Core Spray	29	182
HPCI – High Pressure Coolant Injection	21	171
MS- Main Steam	48	252
FW-Feedwater	20	77
CRD-Control Rod Drive	8	62
SLC – Standby Liquid Control	6	33
Totals	273	1578

7) Relief Request NDE-R007

COMPONENT IDENTIFICATION

Code Class:	3
Examination Category:	D-B
Item Number:	D2.10
Component Numbers:	Various

CODE REQUIREMENT

ASME Section XI, 2001 Edition through the 2003 Addenda, IWA-5244 (b) states that for buried components where a VT-2 visual examination cannot be performed, the examination requirement is satisfied by the following:

1) The system pressure test for buried components that are isolable by means of valves shall consist of a test that determines the rate of pressure loss. Alternatively, the test may determine the change in flow between the ends of the buried components. The acceptable rate of pressure loss or flow shall be established by the Owner.
2) The system pressure test for non-isolable buried components shall consist of a test to confirm that flow during operation is not impaired.

Reason for Request

IWA-5244(b)(l) requires either a pressure loss test or a test that determines the change in flow between the ends of the buried components for isolable sections of buried piping. The acceptable rate of pressure loss or flow shall be established by the Owner. Sections of River Water Supply, Emergency Service Water (ESW), and Residual Heat Removal Service Water (RHRSW) System buried piping were not designed with consideration for isolation valves adequate for performing a pressure loss type test or do not contain instrumentation adequate for measuring changes in flow between the ends of the buried piping.

The River Water Supply System contains large diameter buried piping (24 inch diameter) that runs from the River Intake Structure to the Pump House and is greater than 1500 feet in length. The ESW System and the RHRSW System contain large diameter buried piping (16 inch diameter for RHRSW and 8 inch and 6 inch diameter for ESW) that runs from the Pump House to the Turbine Building and is greater than 500 feet in length. The subject piping design for these systems did not provide for isolation valves that are capable of supporting a pressure loss type test considering the volume of the piping and the available capacity of test pumps. The system isolation valves were only intended to provide isolation for maintenance activities with only static system pressure.

River Water Supply and ESW were designed with a single flow element per train located in the Pump House. ESW has some additional flow instrumentation on some downstream components, but not for every branch on a train. RHRSW was designed with a single flow element per train located in the Reactor Building before the Residual Heat Removal System Heat Exchanger. Therefore, the installed instrumentation is inadequate for measuring the flow difference at each end of the buried piping. The use of ultrasonic flow instrumentation was considered, but the piping configurations do not provide for the straight runs of piping required for accurate flow measurement.

Both the River Water Supply and RHRSW systems include four pumps each with two pumps designated to each of two independent trains. The River Water Supply pumps and RHRSW pumps have installed excess capacity. Therefore, each of the independent trains of both the River Water and RHRSW systems can accommodate a leak and still satisfy the accident analysis requirements. ESW has one pump per train. The ESW system supplies various plant heat exchangers, which have flow margin due to heat transfer requirements.

PROPOSED ALTERNATIVE AND BASES FOR USE

In lieu of performing a system pressure test in accordance with the requirements specified in IWA-5244(b)(1), DAEC shall use the provisions of IWA-5244(b)(2) to confirm that flow during operation is not impaired. IWA-5244(b)(2) states that the system pressure test for non-isolable buried components shall consist of a test to confirm that flow during operation is not impaired. The proposed alternative provides an acceptable level of quality and safety.

The integrity of the buried piping for the River Water Supply and RHRSW, will be verified during quarterly pump testing. Using the downstream instruments, flow rate is set at the fixed reference value in accordance with ASME OM Code 2001 Edition through 2003 Addenda and documented in the test record. The pump discharge pressure is then measured and used to determine the head produced by the pump. Head and flow rate are interdependent variables, which together define pump hydraulic performance. As the pump degrades, the developed head will decrease at the reference flow rate. However, since the location of the flow rate instruments are downstream of the buried piping, a decrease in pump head during testing may also indicate through-wall leakage in the buried portion of the RHRSW or River Water Supply piping.

Significant through-wall leakage would be evident because the total flow rate would increase even though the downstream indicated flow rate is set at the reference value. Therefore, a satisfactory quarterly service water pump test also verifies the integrity of the buried system supply piping. Should the pump test results fall below the required action range of the code, then additional testing and evaluations will be performed to determine whether the unsatisfactory test results are due to degraded pump performance or through-wall leakage.

For the ESW System, the integrity of the buried piping will be verified during the combination of the quarterly pump testing and the verification that adequate flow is supplied to cooling loads as provided by installed instrumentation. Using the upstream instruments, flow rate for the ESW pump is set at a fixed reference value in accordance with ASME OM Code 2001 Edition through 2003 Addenda and documented in the test record. Downstream critical heat loads supplied by ESW such as the Emergency Diesel Generator Coolers, Control Building Chillers, and the Residual Heat Removal and Core Spray Room Coolers, have installed flow instrumentation. The installed instrumentation represents approximately 90% of the total critical load flow required to be supplied by ESW. Significant through-wall leakage would be evident by a marked decrease in supplied flow to the 90% instrumented downstream loads. Annual trending of the ESW instrumented critical load flow rates compared to the upstream set reference value flow verifies the integrity of the buried piping.

In addition, DAEC proposes to perform visual examination of the ground surface area immediately above each buried section of ESW on a refuel cycle basis in lieu of performing the test required by IWA-5244(b)(l). The visual examinations will be performed only after the subject piping has been in operation at nominal operating conditions for a minimum of 24-hours. The ASME Section XI code only requires a pressure test once each period (Every 3 to 4 years).

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety

Duration of Proposed Alternative

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for DAEC.

Precedents

None