Entergy

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June 21, 2006

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

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Response to NRC Request for Additional Information Regarding Pilgrim Relief Request, PRR-10, Risk-Informed ISI for Class 1, B-F and B-J Welds (TAC NO. MC8293)

REFERENCE:

1. NRC Letter, Request for Additional Information Regarding Pilgrim Relief Request, PRR-10, Fourth 10-Year Inservice Inspection Program Plan, dated May 11, 2006

2. NRC Letter, Individual Plant Examination (IPE) Submittal-Internal Events (Generic Letter 88-20, TAC NO. M74451), dated October 30, 1996

LETTER NUMBER: 2.06.055

Dear Sir or Madam:

The attachment to this letter provides Entergy's response to the NRC Request for Additional Information (Reference 1) in support of PRR-10, regarding Risk-Informed ISI for Class 1, B-F and B-J Welds. These responses are derived from the NRC safety Evaluation Report related to the Pilgrim Individual Plant Examination (Generic Letter 88-20) dated October 30, 1996, (Reference 2).

There are no commitments contained in this letter.

If you have any questions or require additional information, please contact Mr. Bryan Ford, Licensing Manager, at (508) 830-8403.

Sincerely,

Stephen J. Bethay

Stephen J. Bethay

WGL/dm

Attachment: Entergy Response to NRC Request for Additional Information (16 pages)

A047

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Entergy Response to NRC Request for Additional Information (RAI)

NRC RAI 1:

Your original RI-ISI program submittal (letter, dated December 27, 2000, J.F. Alexander (Director, Nuclear Assessment) to U.S. NRC) states on page 3 of 30, "The Pilgrim PSA is presently being refined in order to address the findings of the NRC SER, the BWROG Peer Certification, and to bring the model to the state of the art. These improvements have been considered and will not have an unconservative impact on the conclusions from the RI-ISI evaluation of Class 1 piping."

This implies that changes have been made to the Pilgrim probabilistic safety analysis (PSA) model subsequent to the development of the original RI-ISI program, some of which may have been significant in scope.

a. Please summarize the enhancements and refinements from the NRC SER and from the Boiling Water Reactor Owner's Group (BWROG) Peer Certification (in response to Level A and B Facts and Observations) that have been implemented.

b. Also, please indicate if the Pilgrim PSA has received any subsequent peer reviews, and what, if any, significant issues (e.g. - Level A and B Facts and Observations) emerged as a result of these reviews. If there remains any significant issues with the Pilgrim PSA model that was used to support PRR-10, please summarize your assessment of the impact of resolving these issues on the RI-ISI application for the 4th ISI interval.

Entergy Response:

Improvements made to the Pilgrim PSA since the original RI-ISI program submittal did not result in any changes to the RI-ISI application for the 4th interval.

The NRC Safety Evaluation Report applicable for Pilgrim response to Generic letter (GL) 88-20, dated October 30, 1996 (TAC NO. M74451) determined that the requirements of GL 88-20 were satisfied but found weaknesses in the human reliability analysis area. More recently, the BWROG Peer Review of March 2000 identified several Facts and Observations, including some in the human reliability area. As a result the PNPS PSA model was completely revised. Facts and Observation sheets documented the certification team's insights and potential level of significance. As part of this update, all major issues and observations from the BWROG Peer Review (i.e., Level A, B and C observations) and weaknesses in the human reliability analysis area identified in the NRC SER have been addressed and incorporated into the current PSA Model Update. This PSA update yields a measurably lower core damage frequency (CDF) than the original and the 1995 update. The improved results are due to improved plant performance, replacement of the switchyard breakers, more realistic success criteria based on MAAP runs, and more sophisticated data handling. The major changes in the current PNPS PSA model and plant changes are summarized as follows:

A. Initiating Event

The initiating event frequencies have been updated and included with the current plant data. For example, the loss of offsite power (LOOP) frequency significantly decreased from the original Individual Plant Examination (IPE) frequency of 0.475/yr to the current value of 0.067/yr, which reflects the decreased occurrence of LOOP events since 1990 and replacement of switchyard breakers. Fault tree models were developed to calculate the various support systems initiating event frequencies. In addition, documentation was developed to include a more detailed discussion of initiating event grouping and any

initiating event subsuming.

B. Accident Sequence Evaluation

All the event trees from the original IPE were completely revised for all the analyzed initiating events. All the BWROG Peer Certification Findings and Observations were incorporated into the revised event trees. Major Facts and Observations include the following:

i) Loss-of-Offsite Power (LOOP) Event Tree

- The LOOP event was completely revised to account for the failure modes of high pressure core injection / reactor core isolation cooling (HPCI/RCIC) beyond 8 hours of operation; RPV depressurization on eat capacity temperature limit (HCTL); transfer to the station blackout event tree to address such items as AC recovery at 30 minutes and beyond and premature battery depletion.

ii) Station Blackout (SBO) Event Tree

- Current update reflects the GE load shed calculations and use of the plant SBO procedures for DC load shedding.

iii) Inadvertent Stuck Open Relief Valve (IORV) Event Tree

- The IORV has been modified to include RPV depressurization with 2 safety relief valves (SRVs) given high-pressure injection failure.

iv) LOCA Event Trees

- The PSA update considers both HPCI and RCIC for small break LOCAs.
- Large and Medium LOCAs and subsequent ATWS are modeled as core damage end states in the updated PSA model. Small break LOCAs and ATWS are treated similar to transient induced ATWS.
- Vapor Suppression System LOCAs events are considered.

v) Anticipated Transients without Scram (ATWS) Event Tree

- The revised ATWS event tree reflects the potential MSIVs closure on low RPV level.
- A failure to initiate standby liquid control (SLC) or the failure to perform adequate boron mixing is explicitly modeled.
- The revised Pilgrim ATWS model takes into consideration 'inhibit ADS' and MSIVs bypass issues. In addition, various HRA values take into considerations ATWS accident progressions for RPV and containment conditions as predicted by modular accident analysis program (MAAP).

vi) Loss-of-Containment Heat Removal Sequences

- The revised event trees model the potential impact from containment venting on the low-pressure systems operation. For example, no credit is given for core spray and LPCI, if containment venting is required. In addition, other containment related

phenomena such as high torus temperatures (HPCI operation) and high containment pressures (operations of RCIC and SRVs) are also reflected in the updated event trees models.

- The modeling for loss of containment heat removal was revised to include the various containment related phenomena such as torus temperature impact on HPCI, high drywell pressure impact on RCIC and SRVs, etc.

- The PSA update only considers the direct torus vent path for containment venting.

vii) Interface System Loss-of-Coolant Accident (ISLOCA) Event Tree

- NSAC 154 and NUREG/CR-5124 were used to reassess the ISLOCA analysis.

- Success criteria for low-pressure injection during an ISLOCA are consistent with that used for small LOCAs.

- The revised ISLOCA event tree credits the use of condensate or fire water for large ISLOCA events provided that previous LPCI or core spray operation had occurred to provide initial RPV reflood.

viii) Other Changes

- The revised event trees credit use of feedwater when deemed appropriate.

- CRD system flow into the RPV is credited for sequences that involve containment venting.

- Consistent success criteria were used for RPV depressurization for transients, medium LOCAs, and small LOCAs (i.e. 2 of 4 SRVs are required for LPCI and core spray operation, and 2 of 4 SRVs are required for fire water).

- The revised Pilgrim PSA model is based on Revision 4 of the BWROG emergency plant guidelines (EPGs).

- The core damage definition has been revised to be consistent with the EPRI guidelines on this issue. That is, core damage occurs when peak clad temperatures exceed 2200°F.

- HPCI and RCIC are used based on a 24-hour mission time.

- This version of the Pilgrim PSA update includes an event tree methodology document that explains event tree development.

- The Pilgrim PSA update model reflects the current plant design and description.

- The core damage frequency (CDF) definition is consistent with the PSA Application Guide in order to provide a basis consistent with NRC expectations, and to provide consistency among the BWR community.

- Success criteria for depressurization and MAAP and GOTHIC thermal-hydraulic analyses are properly documented in Appendixes I and L.

C. Thermal - Hydraulic Analysis

The thermal-hydraulic analysis has been completely revised and improved to better support the success criteria. The MAAP4 computer code was used to update and address the many issues raised by the BWROG Peer Certification Team. Specifically, the following:

- A basis was provided for the timing and discharge pressure (flow) adequacy when using the fire water system for successful mitigation during transients and small LOCAs
- Current success criteria for stuck open SRV (SORV) are same as for non-SORV cases. Namely, two SRVs are required for successful RPV depressurization. This is considered for SORV cases.
- Consistent success criteria for RPV depressurization for transients, medium LOCAs and small LOCAs (2 of 4 SRVs for LPCI and core spray operation, and 2 of 4 SRVs for fire water are required).
- Plant-specific calculations were performed to identify the plant response for single or double recirculation pump trip (RPT) failures.
- Verified the appropriateness of this core damage definition.

  In addition to the MAAP4 code, the GOTHIC code was used to predict various room heatup rates for the Reactor Building, Turbine Building, Switchgear room and Battery room.

D. System Analysis

System fault tree models from the original IPE were completely revised to reflect the as-built plant configurations. MAAP analyses were clearly identified in order to support the success criteria of these Level 1 models. More detailed modeling for the logic interlock was included in the system models. A detailed fault tree for the Reactor Protection System was developed based on NUREG/CR-5500, which decreased the failure-to-scram probability from 3.0E-5 to 5.8E-6.

E. Data Analysis

Component failure data, both generic and plant-specific, were reviewed and updated with more recent experience. Plant-specific data were adjusted for industry experience using Bayesian updates. Maintenance unavailabilities were updated to include the maintenance rule records from the system engineers. More recent common cause failure (CCF) data and approach (NUREG/CR-5497) were factored into this update. In addition, the performance of risk significant systems HPCI and RCIC has greatly improved since the original IPE.

F. Human Reliability Analysis

A complete revision of the human reliability analysis (HRA) was performed to identify, quantify and document the pre-initiator and post-initiator human errors (including recoveries). The updated HRA was performed using NUREG/CR-1278 and screening values were only used for low-significance human errors. In addition, a detailed analysis was performed to treat dependencies between post-initiator errors and post-initiator errors.

G. Dependency Analysis

- A complete revision of the internal flooding analysis was developed to systematically address the spatial dependencies.
- Dependency between pre-initiator human errors (such as mis-calibration of instruments) was modeled. In addition, dependencies between multiple post-accident operator actions appearing in the same accident sequence were evaluated.
- Detailed component dependency tables were developed to address all the support systems associated with the modeled systems and components.

H. Structural Response

- The ISLOCA frequency was revised.
- RPV overpressure and Reactor Building capability were discussed and included in the level 2 assessment.

I. Quantification

- The truncation value was lowered to 1.0E-11.
- The HEP dependencies and recovery actions in the cutsets were evaluated.
- The ATWS contribution has decreased due to lower probability of failure to scram based on NUREG/CR-5500.
- The HRA was completely revised to address comments from the BWROG Peer Review that many of the HEPs were not realistic using the previous methodology. In many cases (e.g., failure to perform direct torus vent), the previous HEPs were judged to be overly conservative.

J. Internal Flooding Analysis

The internal flooding analysis from the original IPE was completely revised to include the detailed process, systematical examination of the flood source and evaluation for all the analyzed flooding scenarios.

K. Containment Performance Analysis (Level 2)

Containment performance analysis models and documentation were completely revised from the original IPE. Propagation of Level 1 cutsets to the Level 2 containment event tree was developed. A detailed large early release fraction (LERF) model was developed to ensure that LERF calculations are consistent with the PSA Applications Guide and NRC requirements for Regulatory Guide 1.174. Other salient items incorporated are the following:

- Ensured that mitigating system availability and success accounts for the degraded plant conditions during the core melt progression.
- The containment event tree fault tree models allow credit for AC power recovery post core damage. This ensures that the models do not allow SBO core damage sequences to benefit from AC supported equipment in the Level 2 without explicit consideration of AC power recovery.
- The drywell/torus shell melt-through phenomena were considered as applicable.

- Operator response to key actions incorporating the probability for success given the containment conditions and EOP directions were reassessed.
- Direct torus venting was considered post-core damage.
- A Pilgrim-specific primary containment structural evaluation was included in the Pilgrim containment event tree. This included an assessment in the primary containment structural evaluation of torus failure due to dynamic loading during ATWS scenarios, torus break below the water line, and bellows seal capability.
- A Reactor Building bypass fault tree model was developed to assess impact on LERF assessment.

L. Uncertainty Analysis

An uncertainty analysis was performed for this update.

During this update, individual work packages (event tree, fault tree, human reliability analysis, data, etc.) and internal flooding analysis were provided to each PSA member for independent peer review. The accident sequence packages, system work packages, human reliability analysis and internal flooding analysis were also assigned to appropriate plant personnel for review. For example, event trees, system analyses and fault tree models were forwarded to the applicable plant systems engineers, and the human reliability analysis was assigned to individuals from the plant Operations Training department for review. Similarly, the accident sequence packages, system work packages and containment performance analysis underwent an outside consultant peer review, including review of the fault tree and event tree models, HRA and Level 2 models.

This updated model also went through peer review by the Entergy South PSA group and the review did not identify any significant issues that altered the results.

NRC RAI 2

Page B-55 of the Pilgrim Fourth 10-Year Interval Program Plan (the first page of Relief Request PRR-10) states "the Pilgrim-specific RI-ISI program is summarized in Table 1." However, Table 1 is only partially provided (page B-57). Also, a Table 2 is provided, but the first page of that table is apparently missing. And the second page of Table 2 has the same title as Table 1's title. Please provide a complete set of Tables 1 and 2, and/or otherwise clarify what you had intended to supply as part of PRR-10.

Entergy Response:

Table 1 was mislabeled in the original submittal and there is no Table 2. A complete Table 1 (4 pages) that provides inspection location selection by risk category is provided as an attachment to this letter

NRC RAI 3

Based on the composite of the partial tables provided in PRR-10, it appears that the fourth ISI interval and third ISI interval RI-ISI programs are identical. However, there are two aspects of the fourth interval program that the staff is unable to discern from what was provided:

a) Have there been any changes to the 71 locations selected for non destructive examination (NDE) for the fourth interval relative to the 71 locations selected during the

original RI-ISI program evaluation (within a given system and risk category)? If so, please identify them and give the rationale for those changes.

b) Page B-55 of the Pilgrim Fourth 10-Year Interval Program Plan states "The RI-ISI program has been updated consistent with the intent of NEI-04-05 (Reference 3) and continues to meet EPRI [Electric Power Research Institute] TR-112657 and Regulatory Guide 1.174 risk acceptance criteria."

Please provide a summary of the reviews and other processes that you performed which led to your conclusion that, aside from any location changes (if any) that you describe in your response to RAI 3(a) above, no changes to your RI-ISI program are needed for the fourth 10-year interval, despite changes made to the plant and to your procedures since the original RI-ISI program was implemented.

(Please note that the staff has not reviewed, let alone endorsed, NEI-04-05, as a methodology for performing an evaluation of an RI-ISI program for update.)

Entergy Response:

(a) There has been no change to the 71 locations selected for NDE for the 4th interval relative to the 71 locations selected during the original RI-ISI program evaluation (within a given system and risk category).

(b) The RI-ISI program is a living program and its implementation requires feedback of new relevant information to ensure the appropriate identification of safety-significant piping locations. In July - August 2004, a review of various inputs required to develop the RI-ISI program was conducted using guidance contained in NEI-04-05 to meet Pilgrim commitment to maintain a living program. The efforts consisted of reviewing plant design change documents, procedures, NDE inspection results, plant specific operating history, and industry operating events using the INPO EPIX database, degradation mechanism assessment and PRA changes. No changes were identified that impact the RI-ISI program.

NRC RAI 4

Page 2-37 of the licensee's ISI inspection program plan submitted by letter dated June 29, 2005, states, "RI-ISI socket welds shall receive both a VT-2 and PT examination, at a minimum, each refueling outage." What is meant by PT (pressure test or penetrant test)?

Entergy Response:

"PT" as used here means Penetrant Test.

NRC RAI 5

Of the items selected for RI-ISI in Table 1(2) how many are socket welds? Provide a breakdown of the examination technique to be performed on the items selected for examination, include the frequency of examination. (UT-Appendix VIII, UT-FAC, VT-2/pressure test, etc.)

Entergy Response:

Three (3) of the seventy-one (71) items selected for RI-ISI are socket welds. See Table 2 (4 pages) of this Attachment for examination technique and frequency for the items selected. One hundred percent of the items selected for RI-ISI will be examined in the 4th interval to meet

the required percentage of examinations in accordance with ASME Section XI 1998 Edition/2000 Addenda Table IWB-2412-1, Inspection Program B.

NRC RAI 6

Page 2-38 of the licensee's inservice inspection program plan states the following:

> "Extent" includes 100% of the examination location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the risk-informed evaluation. Areas with acceptable limited examinations, and their bases, shall be documented.

It is not clear what is meant by "acceptable limited examinations," in addition, limited examinations will still need to have a relief request submitted and approved. The paragraph needs to be modified to include, the need to request relief for limited examinations.

Entergy Response:

Acceptable limited examinations are those examinations where less than 10% of the examination volume or required area cannot be examined due to interference by another component or part geometry.

A relief request will be generated for any RI-ISI piping element selection for which greater than 90% examination coverage is not achieved. Consistent with the requirements of code case N-460, an examination will be considered limited if less than or equal to 90% is obtained.

NRC RAI 7

Page B-55, "Code Requirement," states that "...100% of Category B-F welds and 25% of B-J welds for the ASME Class 1 piping greater than 4" NPS be selected for volumetric and/or surface examination." However code changes have changed the requirement to ASME Class 1 piping greater than 1" NPS be selected for volumetric and/or surface examination. The licensee needs to evaluate the effects of the change in scope on its RI-ISI program. Describe how Class 1 piping less than 4" NPS and greater than 1" NPS will be treated.

Entergy Response:

The Pilgrim RI-ISI program for Class 1 piping was developed with the exemptions allowed by IWB-1220(a), (b), (c), and (d). As allowed by IWB-1220(a), a calculation was performed to justify the exclusion of water and steam lines from ISI Class 1 surface and volumetric examination based on normal system makeup. As documented in the calculation, Class 1 piping in water systems with an inside diameter of 1.10" or less, and piping in steam systems with an inside diameter of 2.20" or less qualify for the make-up capacity exemption of IWB-1220(a). Class 1 piping in water systems greater than 1.10" inside diameter and steam systems greater than 2.20" inside diameter are part of the evaluation and analysis process and have been selected for examination when identified through the RI-ISI process.

Table 1
Inspection Location Selection for PNPS RI-ISI Program by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1st Approved RI-ISI Interval		New RI-ISI Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RPV	2 (1)	High (High)	High	TASCS, TT, CC, (FAC)	Medium (High)	B-F	4	4		4	
RPV	2 (2)	High (High)	High	CC, (IGSCC)	Medium (Medium)	B-F	2	1[3]		1[3]	
RPV	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-F	4	1		1	
RPV	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	14	2[4]		2[4]	
RPV	4	Medium	High	None	Low	B-F	5	1		1	
RPV	6	Low	Medium	None	Low	B-F	4	0		0	
MS	2	High	High	TT	Medium	B-J	3	1		1	
MS	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	78	8[5]		8[5]	
MS	4	Medium	High	None	Low	B-J	4	1		1	
MS	5	Medium	Medium	TT	Medium	B-J	3	1		1	
MS	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	4	0		0	
RECIRC	4	Medium	High	None	Low	B-J	66	7		7	
RECIRC	7	Low	Low	None	Low	B-J	4	0		0	
FW	2 (1)	High (High)	High	TASCS, TT, (FAC)	Medium (High)	B-J	19	4		4	
FW	2 (1)	High (High)	High	TT, (FAC)	Medium (High)	B-J	24	4[6]		4[6]	
FW	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	33	4[7]		4[7]	

Table 1 (cont'd)
Inspection Location Selection for PNPS RI-ISI Program by Risk Category

System	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1st Approved RI-ISI Interval		New RI-ISI Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other(2)	RI-ISI	Other(2)
RHR	2	High	High	TASCS	Medium	B-J	8	3		3	
RHR	2	High	High	TT	Medium	B-J	7	1		1	
RHR	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-J	5	0		0	
RHR	4	Medium	High	None	Low	B-J	24	3		3	
RHR	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	2	0		0	
RHR	6	Low	Medium	None	Low	B-J	8	0		0	
RHR	7	Low	Low	None	Low	B-F	1	0		0	
						B-J	3	0		0	
SBLC	4	Medium	High	None	Low	B-J	30	3		3	
SBLC	6	Low	Medium	None	Low	B-J	39	0		0	
RWCU	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	14	2(8)		2(8)	
RWCU	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-J	3	1(9)		1(9)	
RWCU	4	Medium	High	None	Low	B-F	1	0		0	
						B-J	57	6		6	
RWCU	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	1	0		0	
RWCU	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	5	0		0	
RWCU	6	Low	Medium	None	Low	B-F	1	0		0	
						B-J	32	0		0	
RWCU	7	Low	Low	None	Low	B-J	2	0		0	

Table 1 (cont'd) Inspection Location Selection for PNPS RI-ISI Program by Risk Category											
System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1st Approved RI-ISI Interval		New RI-ISI Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RCIC	4	Medium	High	None	Low	B-J	24	3		3	
RCIC	6	Low	Medium	None	Low	B-J	12	0		0	
CS	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	4	2[10]		2[10]	
						B-J	11	0		0	
CS	4	Medium	High	None	Low	B-J	23	3		3	
CS	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	2	0		0	
CS	6	Low	Medium	None	Low	B-J	4	0		0	
HPCI	2	High	High	TASCS	Medium	B-J	2	1		1	
HPCI	4	Medium	High	None	Low	B-J	32	4		4	
HPCI	6	Low	Medium	None	Low	B-J	7	0		0	

Notes to Table 1:

1. Systems are described in the Pilgrim FSAR.

2. The column labeled "Other" is generally 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. PNPS achieved an 11% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.

Notes to Table 1 (cont'd):

3. This weld was selected for examination by both the Generic Letter 88-01 IGSCC Program and the RI-ISI Program. Since crevice corrosion was identified along with IGSCC as a damage mechanism for this weld, the IGSCC examination will include the examination requirements identified in EPRI TR-112657 for crevice corrosion in order to be credited toward both the IGSCC and RI-ISI Programs.

4. These two welds were selected for examination by both the Generic Letter 88-01 IGSCC Program and the RI-ISI Program. Since IGSCC was the only damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

5. Two of these eight welds were selected for examination by both the FAC and RI-ISI Programs. Since FAC is the only damage mechanism identified for these welds, the FAC examinations will be credited towards both programs.

6. Three of the four welds were selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, these welds will be subject to both FAC and RI-ISI examinations.

7. These four welds were selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for these welds, the FAC examinations will be credited toward both programs.

8. These two welds were selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for these welds, the FAC examinations will be credited toward both programs.

9. This weld was selected for examination by both the Generic Letter 88-01 IGSCC Program and the RI-ISI Program. Since IGSCC was the only damage mechanism identified for this weld, the IGSCC examination will be credited toward both programs.

10. These two welds were selected for examination by both the Generic Letter 88-01 IGSCC Program and the RI-ISI Program. Since IGSCC was the only damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

Table 2 Examination Techniques					
Component ID	Component Description	Component Diameter	System ID	ASME Item No	Exam Technique
14R-A-11	PIPE TO VALVE	10 IN.	CS	B9.11	UT-Appendix VIII
14-A-10	PIPE TO ELBOW	10 IN.	CS	B9.11	UT-Appendix VIII
14-A-10A	VALVE TO PIPE	10 IN.	CS	B5.130	UT-Appendix VIII
14-A-3	PIPE TO REDUCER	10 IN.	CS	B5.130	UT-Appendix VIII
14-A-9A	ELBOW TO PIPE	10 IN.	CS	B9.11	UT-Appendix VIII
6-A-12	PIPE TO PIPE	18 IN.	FW	B9.11	UT-Appendix VIII
6-N4D-2	PIPE TO PIPE	12 IN.	FW	B9.11	UT-Appendix VIII
6-A-10	PIPE TO FLUED HEAD	18 IN.	FW	B9.11	UT-Appendix VIII
6-N4D-16	TEE TO PIPE	18 IN.	FW	B9.11	UT - FAC
6-N4A-12	REDUCER TO PIPE	12 IN.	FW	B9.11	UT-Appendix VIII/UT - FAC
6-N4A-13	TEE TO REDUCER	18 IN.	FW	B9.11	UT-Appendix VIII/UT - FAC
6-N4B-11	TEE TO ELBOW	12 IN.	FW	B9.11	UT-Appendix VIII/UT - FAC
6-N4C-11	ELBOW TO PIPE	12 IN.	FW	B9.11	UT - FAC
6-N4C-2	PIPE TO PIPE	12 IN.	FW	B9.11	UT-Appendix VIII
6-N4D-14	PIPE TO REDUCER	18 IN.	FW	B9.11	UT - FAC
6-N4B-8	ELBOW TO PIPE	12 IN.	FW	B9.11	UT-Appendix VIII
6-N4D-15	PIPE TO PIPE	18 IN.	FW	B9.11	UT - FAC
23-O-17	PENETRATION TO PIPE	10 IN.	HPCI	B9.11	UT-Appendix VIII
23-O-18	PIPE TO ELBOW	10 IN.	HPCI	B9.11	UT-Appendix VIII
23-I-17	VALVE TO PIPE	14 IN.	HPCI	B9.11	UT-Appendix VIII
23-I-3	ELBOW TO PIPE	14 IN.	HPCI	B9.11	UT-Appendix VIII

Table 2 Examination Techniques					
Component ID	Component Description	Component Diameter	System ID	ASME Item No	Exam Technique
23-I-16	PIPE TO ELBOW	14 IN.	HPCI	B9.11	UT-Appendix VIII
1-D-15	FLUED HEAD TO PIPE	20-IN.	MS	B9.11	UT-Appendix VIII
1-A-11	PIPE TO ELBOW	20-IN.	MS	B9.11	UT - FAC
1-A-15	FLUED HEAD TO PIPE	20-IN.	MS	B9.11	UT-Appendix VIII
1-B-15	FLUED HEAD TO PIPE	20-IN.	MS	B9.11	UT-Appendix VIII
1-SD-10R	PIPE TO VALVE	3 IN.	MS	B9.21	UT-Appendix VIII
1-C-15	FLUED HEAD TO PIPE	20-IN.	MS	B9.11	UT-Appendix VIII
1-A-16T	VALVE TO PIPE	2 IN.	MS	B9.32	UT-Appendix VIII
1-A-8	ELBOW TO PIPE	20-IN.	MS	B9.11	UT-Appendix VIII
1-A-7	ELBOW TO ELBOW	20-IN.	MS	B9.11	UT-Appendix VIII
1-SD-8R	VALVE TO PIPE	3 IN.	MS	B9.21	UT-Appendix VIII
1-A-12	ELBOW TO PIPE	20-IN.	MS	B9.11	UT - FAC
13-O-19	PIPE TO VALVE	3 IN.	RCIC	B9.21	UT-Appendix VIII
13-O-18	PIPE TO PIPE	3 IN.	RCIC	B9.21	UT-Appendix VIII
13-I-16	VALVE TO PIPE	4 IN.	RCIC	B9.11	UT-Appendix VIII
2R-HB-1	HEADER TO BEND	12 IN.	RECIRC	B9.11	UT-Appendix VIII
2R-HB-4	HEADER TO BEND	12 IN.	RECIRC	B9.11	UT-Appendix VIII
2R-N2K-2	PIPE TO SAFE END	12 IN.	RECIRC	B9.11	UT-Appendix VIII
2R-HA-1	HEADER TO BEND	12 IN.	RECIRC	B9.11	UT-Appendix VIII
2R-HA-4	HEADER TO BEND	12 IN.	RECIRC	B9.11	UT-Appendix VIII

Table 2 Examination Techniques					
Component ID	Component Description	Component Diameter	System ID	ASME Item No	Exam Technique
2R-N2A-2	PIPE TO SAFE END	12 IN.	RECIRC	B9.11	UT-Appendix VIII
2R-N2B-2	PIPE TO SAFE END	12 IN.	RECIRC	B9.11	UT-Appendix VIII
10-O-25	PIPE TO PIPE	20 IN.	RHR	B9.11	UT-Appendix VIII
10R-IA-3	PIPE TO ELBOW	18 IN.	RHR	B9.11	UT-Appendix VIII
10R-IA-4	ELBOW TO PIPE	18 IN.	RHR	B9.11	UT-Appendix VIII
10-O-24	PIPE TO PIPE	20 IN.	RHR	B9.11	UT-Appendix VIII
10R-IA-6	PIPE TO VALVE	18 IN.	RHR	B9.11	UT-Appendix VIII
10R-IA-7	VALVE TO PIPE	18 IN.	RHR	B9.11	UT-Appendix VIII
10R-IB-14	PIPE TO FLUED HEAD	18 IN.	RHR	B9.11	UT-Appendix VIII
2R-N2B-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
6-N4A-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
6-N4B-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
6-N4C-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
14-B-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
2R-N2A-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
6-N4D-1	SAFE END TO NOZZLE	12 IN.	RPV	B5.10	UT-Appendix VIII
RPV-N16B-R-2	SAFE END TO REDUCER	2'	RPV	B5.140	UT-Appendix VIII
1-D-1	NOZZLE TO SAFE END	20-IN.	RPV	B5.10	UT-Appendix VIII
12-O-24	PENETRATION TO PIPE	6 IN.	RWCU	B9.11	UT-Appendix VIII
12-I-5	PIPE TO TEE	6 IN.	RWCU	B9.11	UT - FAC
12-O-28R	PIPE TO ELBOW	6 IN.	RWCU	B9.11	UT-Appendix VIII

Table 2 Examination Techniques					
Component ID	Component Description	Component Diameter	System ID	ASME Item No	Exam Technique
12-O-29R	ELBOW TO PIPE	6 IN.	RWCU	B9.11	UT-Appendix VIII
12-O-30R	PIPE TO ELBOW	6 IN.	RWCU	B9.11	UT-Appendix VIII
12-O-31A	PIPE TO PIPE		RWCU	B9.11	UT-Appendix VIII
12-O-31R	ELBOW TO PIPE	6 IN.	RWCU	B9.11	UT-Appendix VIII
12R-O-7	ELBOW TO PIPE	6 IN.	RWCU	B9.11	UT-Appendix VIII
12-I-4	TEE TO PIPE	6 IN.	RWCU	B9.11	UT - FAC
B-11-75	PIPE TO PIPE	1.5 IN.	SBLC	B9.40	VT-2/Pressure Test & Penetrant Test
B-11-78	PIPE TO ELBOW	1.5 IN.	SBLC	B9.40	VT-2/Pressure Test & Penetrant Test
B-11-79R	ELBOW TO PIPE	1.5 IN.	SBLC	B9.40	VT-2/Pressure Test & Penetrant Test