APPENDIX B

AUGMENTED INSPECTION PROGRAMS

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APPENDIX B

AUGMENTED INSPECTION PROGRAMS

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AUGMENTED INSPECTION PROGRAMS

Introduction

For the purposes of this LGS Unit 1 ISI Program, any examination other than one required by the 1989 Edition of ASME Boiler and Pressure Vessel Code, Section XI, is considered to be augmented. These augmented examinations may be due to additional requirements from NRC Regulatory Guides, Generic Letters, Bulletins, or accelerated implementation of Code rules or regulations. Other reasons for augmented examinations are vendor recommendations, industry committee efforts, or self-imposed inspections.

BWRVIP

The BWR Vessel and Internals Project (BWRVIP) is an association of utilities focused exclusively on BWR vessel and internals issues. PECO Energy is an actively participating member. Through the BWRVIP efforts, a series of inspection and flaw evaluation (I&E) guidelines for safety-related internals has been developed. The members of the BWRVIP have committed to the implementation of the I&E guidelines. As a part of this commitment, it was agreed that once the NRC issues a safety evaluation report (SER) on an I&E guideline, as submitted by the BWRVIP, the licensee or the BWRVIP must inform the NRC of any decision made to not fully implement the guideline, as submitted and approved, within forty-five (45) days of the report approval. Otherwise, the I&E guideline shall be implemented in the fashion noted in the SER. If the NRC staff conditionally approves a BWRVIP document, i.e., issues an SER that provides for material changes to the submitted document, resolution of comments may be required, including potential re-submittal of the BWRVIP document. It is the intention of the BWRVIP that the BWRVIP will inform the NRC staff within forty-five (45) days of SER issuance, if such a situation exists.

In accordance with the BWRVIP commitment, LGS will fully implement BWRVIP I&E guidelines once approved or endorsed by the NRC, if the NRC approves the I&E as submitted. Alternatively, for guidelines that have SER's issued that provide for material changes to the as-submitted document, it is LGS' intent to follow the guidelines, as submitted, until a final agreement is reached that reconciles the NRC/BWRVIP differences. Once endorsed, the guidelines will be instituted within the time frame of the LGS Outage Management process. That is, the guidelines will be incorporated into the next refueling outage for which the scope has not been frozen. It is expected that plant personnel will be aware of pending NRC endorsements, and that upon NRC approval, scope identification would be completed with little delay. BWRVIP documents that have a potential safety impact will be reviewed and may be implemented in an outage for which the scope has already been frozen.
LGS may implement BWRVIP guidelines once approved by the BWRVIP Executive Committee. The guidelines should be implemented in the next outage for which the scope has not been frozen. Deviations from the guidelines may be implemented with appropriate justification and will be identified in the reporting of outage inspection/repair activities. Results of implementation of the BWRVIP I&E guidelines will be provided to NRC with the submittal of ISI data and will be provided to the EPRI BWRVIP Project Manager for entry into an industry database.

Note: LGS endorses and will implement the BWRVIP position that examinations performed prior to the issuance or formal implementation of an I&E guideline can be considered a baseline examination, provided it meets the appropriate BWRVIP baseline criteria.

ANII

The ANII shall be involved with inspections of components within the scope of ASME Section XI. Inspection of components not in the scope of ASME Section XI, upon Owner's request, may be included within the scope of ANII involvement.

Examination Methods

AUGMENTED INSPECTION PROGRAMS

For the purposes of the examinations conducted as part of the augmented program, the following definitions apply.

UT — The use of ultrasonic techniques to perform volumetric examinations. The technique will meet the qualifications of ASME, BWRVIP, or others identified by the cognizant Level III, as appropriate.

PT — The use of liquid penetrant to perform surface examinations. Unless otherwise specified, PT examinations will be conducted using methods employed for Section XI examinations.

VT-3 — This is a visual examination conducted to assess the overall condition of a component as defined in Section XI.

VT-1 — This is a visual examination capable of resolving a 1/32" line as defined in Section XI. It is used to look for evidence of cracking.

MVT-1 — This is a more sensitive visual examination than VT-1. This VT is conducted in such a manner that a 1 mil wire can be resolved. This technique may require cleaning of the surface to be inspected. This is the same technique as the CS VT-1 of BWRVIP-18.

EVT-1 — This is an even more sensitive version of VT-1. For this method, it must be possible to resolve a ½ mil wire. Cleaning of the surface may be required.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-01
USNRC Generic Letter 88-01
Intergranular Stress Corrosion Cracking

I. SCOPE

This augmented inspection program (AUG-01) defines the activities conducted during inservice inspection at LGS Unit 1, pursuant to the examination requirements of USNRC Generic Letter 88-01. This program addresses only those requirements dealing with periodic examination. Specific PECO Energy commitments concerning all of the aspects of USNRC Generic Letter 88-01 are contained in the PECO Energy responses to the letter, Reference II.B.

USNRC Generic Letter 88-01 requires that an augmented inspection program be developed and implemented for certain austenitic stainless steel piping welds and reactor vessel attachments. The technical bases for the NRC staff positions, put forth in the Generic Letter, are detailed in Reference II.D. The applicable requirements of the Generic Letter are summarized below.

A. DESCRIPTION OF USNRC GENERIC LETTER 88-01 CRITERIA USED TO DETERMINE THE EXTENT OF LIMERICK COMPONENTS WITHIN AUG-01 PROGRAM SCOPE

NRC Generic Letter 88-01 applies to piping systems or portions of systems and reactor vessel attachments and appurtenances that meet the following criteria:

1. austenitic stainless steel,
2. four inches or larger in nominal diameter, and
3. contain reactor coolant at a temperature above 200° F, during power operation.

The following LGS Unit 1 systems or portions of systems and components meet the GL 88-01 criteria:

1. Reactor Recirculation System
2. Residual Heat Removal System
3. Core Spray System
4. Reactor Water Clean-up System
5. Reactor Core Isolation Cooling System
6. Jet Pump Instrumentation System
7. Reactor Vessel Stainless Steel Safe Ends > NPS 4

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-01
USNRC Generic Letter 88-01
Intergranular Stress Corrosion Cracking, continued

B. SCOPE OF COMPONENTS WITHIN AUG-01

1. Reactor Recirculation System (Loops A & B)

   Reference: P&ID ISI-M-43, sheet 1

   Scope:

   - NPS 28 Reactor Recirculation Pumps A and B suction piping, from the welds joining the RPV N1 nozzles to safe-ends, through and including the welds to the Recirculation Pump suction nozzles.

   - NPS 28 Reactor Recirculation Pumps A and B discharge piping, from the weld to the Recirculation Pump discharge nozzles, through the NPS 22 headers and including the NPS 12 piping segments, from the headers, to the RPV N2 nozzles to safe-end welds.

   - The weld connecting the NPS 20 RHR piping to the B pump suction, and the welds connecting the NPS 12 RHR piping to the A and B pump discharge.

2. Residual Heat Removal System

   Reference: P&IDs ISI-M-51, sheets 1 and 3

   Scope:

   - NPS 20 RHR supply piping, from the connection at the B loop Reactor Recirculation Pump suction line, up to normally closed inboard containment isolation valve 51-1F077 (reference: line number DCA-105).

   - NPS 12 RHR return piping, from valves HV-51-1F050A and B, to the Reactor Recirculation Pump A and B discharge piping (reference: line number DCA-104).

   - NPS 12 RHR LPCI injection, from valves 51-1F065A, B, C, and D, to the RPV and including the RPV N17nozzles to safe-end welds (reference: line number DCA-318).

3. Core Spray System

   Reference: P&ID ISI-M-52, sheet 1

   Scope:

   - NPS 12 Core Spray Sparger supply piping, from inboard valve 52-1F007A and B, up to and including the NPS 10 RPV N5 nozzles to safe-end welds (reference: line number DCA-319, 320).

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-01
USNRC Generic Letter 88-01
Intergranular Stress Corrosion Cracking, continued

4. Reactor Water Clean-Up System

Reference: P&ID ISI-M-44, sheets 1 and 2

Scope:

- NPS 6 RWCU pump suction piping inside containment, including the connection at the Reactor Recirculation suction line and primary containment penetration X-14 (reference: line number DCA-101).

- NPS 4 RWCU from 6"x4" reducer on pump suction header to "A" RWCU pump inlet (reference: line number DCC-103).

- NPS 6 RWCU pump suction piping outside of containment (reference: line number DCB-102 and DCC-103). See Note 1.

- RWCU 4 discharge piping, from the 4" x 3" reducers in the RWCU "B" and "C" pump discharge header, to the tube side inlet of the Regenerative heat exchanger (reference: line number DCC-101). See Note 1.

- NPS 4 RWCU "A" pump discharge to and including 4x4x4 tee in common RWCU pump discharge header (reference: line number DCC-101) see note 2.

- RWCU piping from the tube side outlet of the Regenerative heat exchanger to the tube side inlets of the Non-Regenerative heat exchangers (reference: line number DCC-102). See Note 1.

- RWCU return piping from the shell side outlet of the Regenerative heat exchanger to RWCU valve HV-44-1F039 (reference: line number DCC-104 and ECC-105). See Note 1.

Note 1: RWCU system piping welds outside primary containment, (reference line Nos. DCC-101,-102,-103,-104 and ECC-105) are classified ASME Class 3 non-Nuclear Safety Related and as such, are not subject to NDE per ASME Section XI. A representative sample of 5% of the total population of these piping welds was examined at each of two (2) refueling outage beginning with 1R04 such that a total of 10% of the population was examined. Subsequent to this PECO Energy has received approval to implement GL 88-01 Supplement 1 per Reference II.H. These welds currently meet USNRC Criteria 1, 2 and 3 as provided in Reference II.C and are following Schedule A, No IGSCC Inspection Required. Refer to the Examination Program, Section IV, for details on the USNRC Criteria and Examination Schedules.

Note 2: Per GL 88-01, these welds currently meet USNRC Criteria 1, 2, and 3 as provided in Reference II.C. No IGSCC inspection required.

5. Reactor Core Isolation Cooling System

AUGMENTED INSPECTION PROGRAMS

Reference: P&ID ISI-M-49 sheet 1

Scope:

- The welds at the pipe connection to the stainless steel flow element FE-49-N016 (reference: line number DBA-107).

6. Jet Pump Instrumentation System

Reference: P&ID ISI-M-42, sheet 1

Scope:

- The circumferential welds greater than 4" NPS between the RPV nozzle N8 safe-end to safe-end extension, and the safe-end extension to the jet pump instrumentation penetration seal.

7. Reactor Vessel Stainless Steel/Inconel Safe Ends

The RPV attachments and appurtenances within the scope of this response are limited to stainless steel/inconel safe-ends > 4" NPS, attached to RPV nozzles.

Reference: P&ID ISI-M-41, sheet 1
P&ID ISI-M-42, sheet 1
P&ID ISI-M-43, sheet 1

Scope:

- The stainless steel/inconel safe-ends attached to RPV nozzles N1, N2, N5, N8, N9, and N17. (Safe-ends for the N5 and N17 nozzles are inconel)

Note: Safe-end attachment to RPV nozzles N1, N2, N5, and N17 have been previously identified within the scope description of the systems associated with these connections.

II. REFERENCES

A. Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, dated January 25, 1988.

B. LGS Unit 1 responses to NRC Generic Letter 88-01, dated August 2, 1988, April 28, 1989, May 30, 1989, September 11, 1989 and June 8, 1990.

C. Generic Letter 88-01, Supplement 1, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping, dated February 4, 1992.

D. NUREG-0313, Revision 2, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, January 1988.

E. USNRC SER for Docket Nos. 50-352 and 50-353, Evaluation of NRC Generic Letter 88-01 Response, Philadelphia Electric Company Limerick Generating Station, Unit s 1 and 2 (TAC Nos. 69143 and 69144), dated March 6,1990.

AUGMENTED INSPECTION PROGRAMS

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F. CM-6 USNRC SER for Docket Nos. 50-352 and 50-353, Evaluation of Philadelphia Electric Company's Submittals Responding to NRC Generic Letter 88-01, Limerick Generating Station, Unit s 1 and 2 (TAC Nos. 69143 and 69144), dated October 22, 1990. (T02670)

G. CM-1 USNRC Letter to PECO Energy, Indication in Reactor Vessel Recirculation Riser Nozzle to Safe End Weld, Limerick Generating Station , Unit 1, (TAC No. M83078), dated April 9 1992. (T01980) also (T03995).

H. USNRC Evaluation for Docket Nos. 50-352 and 50-353, IGSCC Inspection Plan of RWCU Piping Welds Outboard of the Primary Containment Isolation Valves; Limerick Generating Station, Unit s 1 and 2 (TAC Nos. M92754 and M92755), dated February 7,1996.

III. GENERAL

NRC Generic Letter 88-01 requires that each pressure retaining circumferential butt weld, that is within scope , be assigned to a category. The available categories are Category A through Category G. The assignment of these categories is based on the degree to which the weld is susceptible to Intergranular Stress Corrosion Cracking (IGSCC). Category A welds are least susceptible, Category G welds are most susceptible. NRC Generic Letter 88-01 and NUREG-0313 Rev. 2 provide details on the determination of IGSCC category. The examination frequency for each of the scope welds is determined by the IGSCC category that is assigned to the weld . This is explained in more detail under the section of this augmented program document entitled examination schedules.

IV. EXAMINATION PROGRAM

A. Examination Methods and Personnel

PECO Energy is committed to complying with the NRC Staff positions on examination methods and personnel as delineated in NRC Generic Letter 88-01. The examination method to be performed will be the ultrasonic (UT) type volumetric method. For UT examinable ASME Class 1 and 2 welds, the IGSCC examinations will be performed in accordance with the requirements contained in the 1989 Edition of ASME Section XI for the ASME class of the weld. For UT examinable ASME Class 3 and non-class welds, the requirements in Section XI for Class 2 welds will apply. Details of the volumetric examination method may be upgraded as practical to ensure that the examinations will be effective. The personnel performing the IGSCC volumetric examinations will be qualified for such examinations by a formal program approved by the NRC.

When complete examination of a selected weld is found to be impractical due to component/plant configuration, another weld will be selected for examination. If alternative welds which fulfill the original selection criteria are not available, the originally selected weld will be examined to the maximum extent possible. Alternate examinations may be performed as applicable. The limited examination described above shall be documented and reported in the ISI Summary Report.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-01
USNRC Generic Letter 88-01
Intergranular Stress Corrosion Cracking, continued

B. Examination Schedules

The examination frequencies in this augmented program conform to the USNRC staff positions provided in Generic Letter 88-01. With exception of the RWCU system outside containment, the frequency of examination depends on the IGSCC Category that the weld is assigned to. The examination frequencies are as follows:

IGSCC Category	Examination Extent and Schedule
A	25% every 10 years (at least 12% in 6 years)
B	50% every 10 years (at least 25% in 6 years)
C	All within next 2 refueling cycles and then all every 10 years (at least 50% in 6 years)
D	All every 2 refueling cycles
E	50% next refueling outage, then all every 2 refueling cycles
F	All every refueling outage
G	All next refueling outage

The examination frequency for the RWCU system, outboard of the second Primary Containment Isolation valve, conforms to the USNRC staff positions provided in Generic Letter 88-01 Supplement 1 and Reference II.H. The frequency of examination depends on the extent of compliance with Criteria 1, 2 and 3 as follows:

RWCU Piping Outboard of Second Primary Containment Isolation Valves[1]	
IGSCC Category	Examination Extent and Schedule
Meets Criteria 1, 2 and 3 or piping made of resistant material and meets Criteria 1	No IGSCC inspection required; Schedule A
Meets Criteria 1, but does not meet Criteria 2 or 3	At least 2% of the welds or 2 welds every refueling outage whichever sample is larger; Schedule B
Does not meet Criterion 1	At least 10% of the welds every refueling outage; Schedule C

USNRC Criteria for RWCU Piping:

1. Satisfactory completion of all Generic letter 89-I0 required actions.
2. No IGSCC detected in RWCU piping welds inboard of the second isolation valves (on-going GL 88-0I inspection).
3. No IGSCC detected in RWCU piping welds outboard of the second isolation valves after inspecting a minimum of 10% of the susceptible piping welds.

NOTE:
1. If IGSCC was found during inspection the sample expansion and methods of mitigation including replacement should be discussed with the staff.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-01
USNRC Generic Letter 88-01
Intergranular Stress Corrosion Cracking, continued

The IGSCC category assigned depends on such factors as:

- Whether stress improvement is performed.
- Whether cracks are known to exist in the weld.
- Whether the weld is reinforced by overlay.
- If corrosion resistant cladding has been approved.
- What the base and weld materials are.
- Whether the weld has been UT examined utilizing methods and personnel as stipulated in the Generic Letter.

Since some of these factors can change, the IGSCC Category assigned to a particular weld is also subject to change.

A UT indication in Reactor Recirculation nozzle to safe end weld VRR-1RD-1A N2H has resulted in this weld being classified as IGSCC Category F. Approval to use the Mechanical Stress Improvement Process (MSIP) for IGSCC mitigation was obtained and MSIP was applied to the weld in 1992, during 1R04. A post MSIP UT was performed on the weld prior to Unit 1 startup from 1R04 as required by GL 88-01. The weld shall be UT examined for each of the next four (4) refueling outages starting with 1R05 and ending with 1R08. After the 1R08 UT, the weld may be upgraded to IGSCC Category E and follow the examination schedule for Category E provided that no adverse change in the UT indication is found.
CM-1

C. Weld Selection

Where the augmented inspection program required examination of a sample of applicable welds, the size and content of the sample was determined from the total population of circumferential welds subject to the program requirements.

The selection of welds for examination under this augmented inspection program has been coordinated with the selection of welds for examination under the ISI Program, (i.e. if a weld requiring augmented examination is selected for Inservice Examination, it was also selected for augmented examination). The examination requirements of both the ISI Program and the Augmented Program are therefore satisfied simultaneously, to the extent that those requirements overlap (i.e., a single volumetric examination performed to satisfy all augmented requirements, and the volumetric examination requirements of the ISI Program). This selection philosophy has been reviewed and deemed to yield a representative sample of the welds requiring the augmented examinations.

The total population of welds subject to examination under this augmented inspection program are identified in the ISI Program Tables which are included in the tables section of this document.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-01
USNRC Generic Letter 88-01
Intergranular Stress Corrosion Cracking, continued

V. EXAMINATION RESULTS

A. Sample Expansion

If one or more Category A, B, C, or D welds are found to be cracked, or if additional cracks or significant crack growth is discovered in a Category E weld during the interval, a sample expansion plan will be invoked. The sample expansion plan utilized will be as put forth in the Staff Position on Sample Expansion of NRC Generic Letter 88-01, including Supplement 1.

If one or more of the RWCU welds inboard of the primary containment isolation valves have confirmed IGSCC indications, then an additional sample of RWCU welds outboard of the primary containment isolation valves shall be selected and examined based on the USNRC Criteria provided in Section IV.B for these welds.

B. Flaw Evaluation

Flaws exceeding the acceptance criteria of IWB-3500 of ASME Section XI will be evaluated, then either repaired, replaced, or deemed acceptable for continued operation. Repairs or replacements will be documented in the Owners Report for Repairs and Replacements as required by ASME Section XI. Evaluations of flaws for continued operation will be performed in accordance with the criteria in IWB-3600 of ASME Section XI. For aspects of flaw evaluation which are not contained in IWB-3600, the requirements in NUREG-0313 Revision 2 will be used in conjunction with IWB-3600.

The USNRC Staff Position on flaw evaluations requires the above referenced criteria for acceptance and evaluation is in accordance with the 1986 Edition of ASME Section XI. Later Editions of ASME Section XI up to and including the 1995 Edition with 1996 Addenda may be used subject to engineering evaluation that an acceptable level of margin against failure of low toughness materials such as fluxed welds (SAW, SMAW) is provided and secondary stresses are considered.

C. Repairs/Mitigation

PECO Energy intends to follow the USNRC staff positions on repair and mitigation techniques, as may be applicable to the scope of this program. In addition to the use of the guidance provided in the staff positions and the referenced ASME Code, PECO Energy will assess the effect such repair or mitigation techniques may have on the overall system (e.g. shrinkage, stiffness, increased dead weight, etc.), as defined in Supplement 1 to Generic Letter 88-01.

VI. REPORTS/RECORDS

A. NRC Notification

If any flaws are identified which do not meet the acceptance criteria for continued operation (referenced above under flaw evaluation), the NRC will be duly notified of the disposition of the affected flaws. NRC approval of the disposition for each flaw exceeding the criteria will be obtained before operation is resumed.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-02
NUREG-0619
BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking

I. SCOPE

This augmented inspection program (AUG-02) defines the examination requirements of USNRC NUREG-0619 applicable to the LGS Unit 1 Feedwater nozzles/spargers as modified by the USNRC approved alternative prepared by the BWR Owners Group, Topical Report GENE-523-A71-0594-A, Revision 1, Alternate BWR Feedwater Nozzle Inspection Requirements, dated August 1999.

Augmented examinations associated with the Control Rod Drive Return Line (CRDRL) nozzle (N9)/piping system are not required at LGS Unit 1. The CRDRL nozzle (one nozzle) has been cut and capped and the CRDRL eliminated. Augmented examinations per NUREG-0619 are not applicable.

II. REFERENCES

A. NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (November 1980) with Generic Letter 81-11 (February 20, 1981).

B. PECO Energy Letter of September 2, 1982, J. S. Kemper to Darrell G. Eisenhut (USNRC).

C. GE-NE-A00-0542-3, Evaluation of GE's BWR Feedwater Nozzle UT Technique, dated July 16, 1998.

D. SIR-00-035, Fracture Mechanics Evaluation for the Feedwater Nozzles at Limerick Generating Station, dated March 2000.

E. GE-NE-523-A71-0594-A, Alternate BWR Feedwater Nozzle Inspection Requirements, dated May 2000.

III. GENERAL

In order to facilitate early detection of the initiation of feedwater nozzle thermal fatigue cracking and thereby limit crack growth within the bounds of approved repair methods NUREG-0619 requires the implementation of a plant specific inspection program in accordance with Section 4.3 of the NUREG. As an alternative to the inspection program recommended in NUREG-0619, the Reference E topical report provides the following:

A. Accept the ultrasonic testing (UT) as the basis to eliminate supplemental liquid penetrant testing of the feedwater nozzle inside radius.

B. Lengthen the time interval between routine UT of the feedwater nozzle inside radius.

C. Reduce the inspection area of the feedwater nozzle inside radius.

For LGS Unit 1, the alternative implements existing ASME Section XI Code requirements.

AUGMENTED INSPECTION PROGRAMS

IV. FEEDWATER NOZZLE AUGMENTED INSPECTION PROGRAM

The LGS Unit 1 examination program requirements shall be based on Reference E, Table 6-1, and the following table parameters:

A. Triple-sleeve spargers with two piston-ring seals, unclad nozzles configuration.

B. Growth to allowable flaw depth is 34.6 years as determined in Reference D.

C. UT Method 3, Automated, full RF Recording (no threshold) as determined in Reference C.

The inspection interval for the UT of nozzle zones 1 and 2 shall be once every 10 years. The inspection interval for the UT of nozzle zone 3 may be extended to once every 20 years. No inspections are required for nozzle zone 4. The inspection interval for the UT of nozzle to safe end weld zone 5 shall be in accordance with ASME Section XI Table IWB-2500-1 for Examination Category B-F.

Visual examination of the feedwater spargers shall be performed once every four (4) refueling cycles in accordance withNUREG-0619 requirements. This examination shall include the entire sparger with special attention given to the junction point of the sparger arms and the flow nozzles.

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated in the same manner as Code-required examinations.

VI. REPORTS/RECORDS

Examination records and reports shall be prepared, maintained and submitted to the USNRC (as required) in the same manner as Code-required examinations.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-03
BWRVIP-18
BWR Core Spray Internals

I. SCOPE

This augmented program (AUG-03) specifies the inspections to be conducted on core spray piping and spargers inside the reactor vessel at LGS Unit 1 to meet the requirements of BWRVIP-18.

This augmented inspection program incorporates from its previous version the examination requirements of USNRC IE Bulletin 80-13 applicable to the core spray sparger and core spray supply header piping.

II. REFERENCES

A. CM-5 IE Bulletin No. 80-13, Cracking in Core Spray Spargers, dated April 4, 1980 transmitted to PECO Energy from Boyce H. Grier 4/4/80 for information. No written response required. (T02668) also (T03843)

B. GE SIL No. 289, Core Spray Sparger Visual Inspection Revision 1, dated May 2 1980.

C. GE SIL No. 289, Supplement 1, Core Spray Piping Visual Inspection Revision 1, dated March 15, 1989.

D. BWRVIP-18, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines, EPRI Report TR-106740, dated July 1996.

E. Letter from Carl Terry (BWRVIP Chairman) to C. E. Carpenter (USNRC), BWRVIP Response to NRC Safety Evaluation of BWRVIP-18, dated January 11, 1999.

III. GENERAL

BWRVIP-18 specifies inspection of core spray internals including piping, spargers, nozzles and brackets.

In developing BWRVIP-18, the BWRVIP considered existing SIL's, RICSIL's, NRC, and other BWRVIP documents. All current inspection recommendations associated with safety function of the core spray internals are contained in BWRVIP-18. Other recommendations are considered superseded.

IV. EXAMINATION PROGRAM

There are no Section XI requirements for the Core Spray Internals. This examination program is designed to comply with the provisions of BWRVIP-18 and any related plant-specific evaluations. Welds that have been solution annealed are exempt from inspection. Detailed inspection requirements for each location, and the total population of items subject to examination by this augmented inspection program, are located in ISI Program Tables for the RPV which is included in this specification.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-03
BWRVIP-18
BWR Core Spray Internals, continued

A. Baseline Inspections

Baseline inspections are the first inspections that satisfy the BWRVIP-18 guidelines, even if they were performed prior to issuance of the guidelines. The piping baseline is to be performed on all circumferential piping welds. Visual or UT methods are acceptable. If the inspection method is visual, then EVT-1 visual examination s required to be used. Supplemental UT may be needed if flaws are detected. The inspection method employed affects the reinspection requirements.

The sparger baseline is location-dependent. LGS Unit 1 is a geometry-tolerant plant, as defined in BWRVIP-18. For this class of plant, the sparger baseline involves an MVT-1 visual examination (MVT-1 is the same as CSVT-1 visual examination in BWRVIP-18) of critical locations and a VT-3 visual examination of the less critical nozzle welds. Per Reference E, the BWRVIP has agreed for simplicity and uniformity to make the sparger inspection requirements the same for Geomety-critical and Geometry-tolerant plants. Except for baselines already completed using MVT-1(like LGS Unit 1) the inspection method for sparger welds S1, S2 and S4 shall be EVT-1 and shall be VT-1 for the S3 and SB welds.

The baseline examination for the piping and sparger brackets (see Reference E) is MVT-1 visual examination without cleaning, although the need for cleaning to assure a good inspection should be evaluated. If indications are noted, then EVT-1 visual examination and cleaning are to be used to evaluate the indication.

Welds with limited accessibility, i.e., hidden welds, are to have the accessible portions inspected to the fullest extent. Hidden welds should be evaluated using the guidance in BWRVIP-18.

Repair baseline inspections should confirm the function of the repair. The repaired weld need only be inspected if it is depended on to provide integrity to the repair. Additional guidance is contained in BWRVIP-18.

B. Reinspection

The reinspection strategy involves the use of "target welds" for piping and the spargers. Reinspection frequency is dependent on the baseline inspection method. The method used for reinspection can be different from the baseline, provided that baseline requirements for the method chosen are satisfied.

Piping reinspection is to occur every other refueling outage, if the baseline were accomplished using UT, or every refueling cycle, if the baseline were performed visually. The reinspection sample includes all creviced and tee box-to-pipe welds, all welds with existing flaws, and a rotating sample of 25% of the piping butt welds (P4a-P4d, in BWRVIP-18) such that four reinspections would cover all welds. The reinspection for the sparger in a geometry-tolerant plant is performed using visual techniques used for the baseline. The scope includes any previously cracked locations and a rotating 25% sample of the sparger welds.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-03
BWRVIP-18
BWR Core Spray Internals, continued

Reinspection of piping and sparger brackets should use the VT-1 examination method. If no cracking is detected, then reinspection every two cycles (50% rotating sample) is sufficient. If cracking is detected, then reinspection frequency of the flawed location and other locations should be based on the flaw evaluation.

Hidden welds should be inspected to the extent practical at a frequency as if they are piping welds. Repairs should be examined as specified in BWRVIP-18.

C. Scope Expansion

If one or more flaws are found during the baseline, scope expansion is not needed, since the baseline is a 100% inspection scope. If flaws are detected during the reinspection, all remaining similar locations are to be examined during that outage.

D. Flaw Evaluation

BWRVIP-18 contains loading information in Section 4, and flaw evaluation methodology in Section 5 and Appendix A. When flaws are detected, the flaw evaluation must consider the impact of the flawed location on another location. Realistic yet conservative assumptions about the condition of uninspected regions should be made. Adjustment to reinspection frequencies should be considered, based on the flaw.

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations shall be performed as described in paragraph IV.D above.

VI. REPORTS/RECORDS

There are no Code-required inspection or reporting requirements for the core spray internals. However, LGS and BWRVIP have committed to supply inspection, evaluation, and repair results to NRC and to EPRI. Therefore all reports and records shall be prepared and maintained in accordance with the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These results will be forwarded to NRC with the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

Upon completion of visual examinations, as required by IE Bulletin 80-13, a detailed report of the results of the examinations and corrective actions taken (if any) shall be submitted to the Nuclear Regulatory Commission in accordance with 10CFR50.4[1].

In the event of identified cracks, an evaluation report shall be submitted to USNRC Nuclear Reactor Regulation for review and approval prior to return to operation[2].

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-03
BWRVIP-18
BWR Core Spray Internals, continued

NOTE:

1. Examinations, as required by this augmented inspection program, are typically performed during a normal refueling outage in which scheduled ASME Section XI examinations are also performed. ASME Section XI requires that a Summary Report be filed, within 90 days of completion of the inservice examination, with jurisdictional enforcement and regulatory authorities. Due to the similar nature of these activities, reports as required by this augmented inspection program may be submitted in conjunction with the ISI Summary Report.

2. Twenty four (24) Hour Prompt Notification shall be made to the Nuclear Regulatory Commission, Director-Region I, upon completion of an evaluation that has determined that a relevant flaw indication is a crack in accordance with 10CFR50.4 If no flaw indications are detected then the use of the ISI Summary Report shall be sufficient to report the examination results.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-04
BWRVIP-41
BWR Jet Pump Assembly

I. SCOPE

This augmented program (AUG-04) specifies the inspections to be conducted on Jet Pump Assemblies at LGS Unit 1 to meet the requirements of BWRVIP-41.

This augmented inspection program incorporates from its previous version the examination requirements of USNRC NUREG/CR-3052 applicable to the jet pump hold-down beam assemblies and Augmented Inspection Program No. AUG-12 for the recommended examinations of the jet pump sensing lines.

II. REFERENCES

A. NUREG/CR-3052, Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure, dated November, 1984.

B. CM-4 IE Bulletin No. 80-07, BWR Jet Pump Assembly Failure, dated April 4, 1980, and including Supplement No. 1 dated May 13, 1980. (T02668)

C. GE SIL No. 330, Jet Pump Beam Cracks, dated June 9, 1980.

D. GE SIL No. 330 Supplement 2, Jet Pump Beam Cracks, dated October 1993.

E. GE RICSIL No. 065, Jet Pump Cracks, dated December 1993.

F. GE SIL No. 551, Jet Pump Riser Brace Cracking, dated February 1993.

G. BWRVIP-41, BWR Jet Pump Assembly Inspection and Evaluation Guidelines, EPRI Report TR-108728, dated October 1997.

III. GENERAL

BWRVIP-41 specifies inspection of selected jet pump assembly components. The inspection method, extent and frequency is a function of the relative safety significance of a given location. Each location was ranked as high (H), medium (M), or low (L). Additionally, the IGSCC susceptibility of the material at a given location was used as a factor.

In developing BWRVIP-41, the BWRVIP considered existing SIL's, RICSIL's, and NRC and other BWRVIP documents. All current inspection recommendations associated with safety function of the jet pump assembly are contained in BWRVIP-41. Other recommendations are considered superseded. Additionally, because BWRVIP-41 supersedes SIL No. 420, the requirements of AUG-12 have been incorporated into AUG-04. However, BWRVIP-41 concludes that the items that were examined in accordance with the recommendations of SIL No. 420, the jet pump sensing lines, do not have any adverse safety consequences associated with their failure, and recommends that no examination of these items be performed. Thus, the examinations that were moved to AUG-04 from AUG-12 have been deleted.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-04
BWRVIP-41
BWR Jet Pump Assembly, continued

The total population of items subject to examination by this augmented inspection program are located in ISI Program Tables. The location identification from BWRVIP-41 is used as a generic component identification number. The specific component is identified by adding the jet pump number to the generic identification number; items associated with the riser are identified by adding both jet pump numbers to the generic identification number.

IV. EXAMINATION PROGRAM

There are no Section XI requirements for the Jet Pump Assembly. This examination program is designed to comply with the provisions of BWRVIP-41 and plant-specific evaluations that are beyond the scope of BWRVIP-41.

BWRVIP-41 uses terminology not routinely used in inspection programs. Locations were assigned a priority of high, medium, or low. Inspection recommendations are based on this priority and refer to a time period called an "inspection cycle". For the purposes of BWRVIP-41, an inspection cycle is equal to six (6) years.

A. Baseline Inspections

Baseline inspections are the first inspections that satisfy the BWRVIP-41 guidelines, even if they were performed prior to issuance of the guidelines. The baseline requirements are dependent on the priority classification. This does not apply to beam assemblies, because they have unique criteria.

High Priority - The baseline for these locations is to be completed within the first inspection cycle (6 years) from the time the inspections begin. At least 50% of these locations are to be inspected during the first outage of implementing BWRVIP-41.

Medium and Low Priority - The baseline for these locations is to be completed within two (2) inspection cycles (12 years). At least 50% of the baseline is to be inspected during the first inspection cycle.

B. Reinspection

Reinspections (except beam assemblies) are required for all locations that received a baseline, but at a less frequent interval.

High Priority - The reinspection is to be completed within two (2) inspection cycles, with 50% of the locations being inspected during each inspection cycle.

Medium and Low Priority - The reinspection is to be performed at a rate of 25% of the population each future inspection cycle.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-04
BWRVIP-41
BWR Jet Pump Assembly, continued

C. Scope Expansion

If one or more flaws, e.g., cracks, wear, bolt loosening, etc., are found during the baseline or the reinspection of a specific location, all of the remaining locations of the same type, i.e., all locations with the same number/ID, on all other jet pump assemblies should be inspected during the same refueling outage unless the flaw can be correlated to a specific event. Additionally, the effect that degradation of one location has on others should be considered when determining if scope expansion is warranted.

D. Flaw Evaluation

BWRVIP-41 contains loading information in Section 4 and flaw evaluation methodology in Section 5. When flaws are detected at a certain location, the evaluation must consider the impact of one location on another. Realistic yet conservative assumptions about the condition of uninspected regions should be made. Adjustment to reinspection frequencies should be considered based on the flaw.

E. Repaired Locations

BWRVIP-41 does not have specific recommendations regarding repairs and their inspections. LGS will develop repairs in accordance with plant procedures. Inspection and reinspection criteria will be determined as part of the repair process.

The ISI Program Table for the RPV identifies the inspection criteria for each location. The detailed bases for both the need for inspection and the frequency is in BWRVIP-41.

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations shall be performed as described in paragraph IV.D, above.

VI. RECORDS/REPORTS

There are no Code-required inspection or reporting requirements for the jet pump assembly. However, LGS and BWRVIP have committed to supply inspection, evaluation, and repair results to NRC and to EPRI. Therefore, all reports and records shall be prepared and maintained in accordance with the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These results will be forwarded to NRC, with the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-05
USNRC Mechanical Engineering Branch (MEB) Technical Position MEB 3-1
NUREG 0800 No Break Boundaries

I. SCOPE

This augmented inspection program (AUG-05) defines the mandatory examination requirements of USNRC Mechanical Engineering Branch Technical Position MEB 3-1, identified in NUREG 0800, applicable to LGS Unit 1 high energy piping between containment isolation valves and the first outboard restraint for which no breaks are postulated. Referred to as "no break" boundaries in this program.

Examination requirements, as detailed in this document, are exclusive of those ASME Section XI inservice inspection requirements for the identified portions of systems listed in Table AUG-05-1. However, where possible, individual examinations performed may be used to satisfy both requirements. See ISI Program Tables for common components.

II. REFERENCES

A. NUREG 0800, Standard Review Plan, USNRC Mechanical Engineering Branch Technical Position MEB 3-1.

B. Limerick Generating Station UFSAR, Section 3.6, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping.

III. GENERAL

The above reference prescribes that cracks or breaks need not be postulated for containment isolation piping provided that certain stress criteria are met and all pipe welds are volumetrically examined during each inservice inspection interval.

IV. SCOPE OF EXAMINATIONS

100% of all circumferential and longitudinal welds within the boundaries described in Table AUG-05-1 shall be volumetrically examined. The specific welds within these boundaries are listed in the ISI Program Tables with appropriate reference to this Augmented Inspection Program (AUG-05).

V. FREQUENCY OF EXAMINATIONS

Examinations shall be performed once each inspection interval - Interval Distribution (ID).

VI. DISCUSSION

Examinations of Class 1 welds shall be in accordance with Subsection IWB of ASME Section XI.

Examinations of Class 2 and 3 welds, and non-classed welds shall be in accordance with Subsection IWC of ASME Section XI.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-05
USNRC Mechanical Engineering Branch (MEB) Technical Position MEB 3-1
NUREG 0800 No Break Boundaries

TABLE AUG-05-1
NO BREAK BOUNDARIES

System	Description	ISI P&ID
Main Steam	From the upstream pipe-to-elbow weld on the first elbow upstream of the inboard isolation valves HV-41-1F022A, B, C, D through penetrations X-7A, B, C, D and outboard isolation valves HV-41-1F028A, B, C, D to the downstream pipe-to-elbow weld on the first downstream elbow.	ISI-M-41
Feedwater	From the downstream pipe-to-valve weld on inboard check valves 41-1F010A, B through ISI-M-55 penetration X-9A, B and the first two outboard check valves HV-41-1F074A, B and HV-41-1F032A, B upstream to and including the first weld outside the reactor building. This includes NPS 16 branch lines DBB-103 and DBB-104 downstream to valves HV-41-109A, B including the upstream pipe-to-valve weld and the 8" DBB-103 branch line up to and including the upstream pipe-to-valve weld on valve 55-1058.	ISI-M-41
HPCI	From the upstream pipe-to-tee weld, upstream of inboard isolation valve HV-55-1F002 through penetration X-11 and outboard isolation valve HV-55-1F003, downstream to and including the upstream pipe-to-elbow weld on the third outboard elbow.	ISI-M-55
RWCU	From and including the pipe-to-elbow upstream weld on the second upstream elbow through inboard isolation valve HV-44-1F001, penetration X-14, outboard isolation valve HV-44-1F004 to and including the downstream elbow-to-pipe weld on the first elbow downstream of valve HV-44-1F040.	ISI-M-44
RCIC	From and including the downstream elbow-to-pipe weld on the second inboard elbow through inboard isolation valve HV-49-1F007, downstream through penetration X-10 and outboard isolation valve HV-49-1F008, to and including the pipe to pipe weld downstream of the third elbow.	ISI-M-49

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-06
Outboard Feedwater Check Valves HV-41-1F074A and B

I. SCOPE

This augmented inspection program (AUG-06) defines the examination requirements to assure that the materials of fabrication of the 24" Feedwater check valves HV-41-1F074A and B provide an acceptable margin of safety, relative to fracture toughness of the reactor containment pressure boundary applicable to LGS Unit 1.

Examination requirements, as detailed in this document, are exclusive of those ASME Section XI inservice inspection requirements for the identified components within the scope of this document.

II. REFERENCE

10CFR50, Appendix A, General Design Criteria (GDC) 51.

III. GENERAL

General Design Criteria (GDC) 51, Fracture Prevention of Containment Pressure Boundary, requires that under operating, maintenance, testing, and postulated accident conditions:

A. The ferritic materials of the containment pressure boundary behave in a non-brittle manner and

B. The probability of rapidly propagating fracture is minimized.

IV. EXAMINATION REQUIREMENTS

100% surface examination of both the internal and external valve body surfaces on valves HV-41-1F074A and B.

V. EXAMINATION FREQUENCY

These augmented examinations are only required to be performed once. They have been completed during the first refueling outage. Therefore, this program has been satisfied and is considered closed.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-07
SIL No. 455
Recommendation for Additional ISI of Alloy 182 Nozzle Weldments

I. SCOPE

This augmented inspection program (AUG-07) defines the specific examination requirements of General Electric Company (GE) Nuclear Services Information Letter (SIL) No. 455, as applicable to LGS Unit 1. This SIL addresses the occurrence of and recommended actions for detection of intergranular stress corrosion cracking (IGSCC) in alloy 182 RPV nozzle to safe end welds.

Examination requirements, as detailed in this document, are exclusive of those ASME Section XI Inservice Inspection requirements for the subject welds; however, where possible, individual examinations performed may be used to satisfy both requirements. See ISI Program Tables for common components.

II. REFERENCES

A. GE SIL No. 455, ISI of Additional Alloy 182 Welds, Revision 2, dated January 29, 2001.

B. NUREG-0313, Revision 2, Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.

C. Augmented Inspection Program No. AUG-01: NRC Generic Letter 88-01, Intergranular Stress Corrosion Cracking.

III. GENERAL

Recent ultrasonic examinations (UT) of Alloy 182 RPV nozzle to safe end welds (i.e. welds designs which incorporate alloy 182 welds and/or weld butters) at several BWR facilities have resulted in the detection of cracking which appears to have initiated as IGSCC in the alloy 182 weld butter, and has in many cases, propagated into the low alloy steel of the RPV nozzle.

LGS Unit 1 has several RPV nozzle to safe end welds which incorporate alloy 182 as either weld material, weld buttering or both. These welds are detailed in Table AUG-07-1.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-07
SIL No. 455
Recommendation for Additional SI of Alloy 182 Nozzle Weldments, continued

IV. EXAMINATION PROGRAM

The minimum examination requirements for the subject welds are in accordance with USNRC NUREG-0313, AUG-01 (References B and C), and ASME Section XI. Briefly, this means that the subject welds fall within the scope of USNRC NUREG-0313 and the AUG-01 program and as such are subject to the personnel, procedure, and examination frequency, etc. requirements of these mandatory documents. Also, the subject welds are within the scope of ASME Section XI and are subject to the requirements of Examination Category B-F.

As a result of the occurrence of through wall leakage from previously identified non-relevant indications, GE has developed enhanced automated UT techniques for Alloy 82 and Alloy 182 welds and has issued SIL No. 455 Revision 2 which supersedes and voids SIL Nos. 455 Revision 1 and 455 Revision 1 Supplement 1. This SIL also closes RICSIL No. 082.

In addition to the above, GE SIL No. 455 recommends that the following actions be considered in preparing for future examinations. These are summarized below:

A. Design Evaluation

1. Review the as-built designs of nozzle-to-safe-end welds containing Alloy 182 or a combination of Alloy 182 and Alloy 82 for the following configurations:

   - Recirculation inlet and outlet nozzles
   - Core spray nozzles
   - Low pressure coolant injection (LPCI) nozzles
   - Jet pump instrumentation nozzles
   - Control rod drive hydraulic return nozzles
   - Feedwater nozzles
   - Other small bore nozzles

2. Include the review of weld fabrication records, fabrication techniques, repair history, materials, welding processes, and radiographic records. Identify all significant UT indications from prior inspections that are not classified. This data will assist in accurate assessments of future UT data. In-process welding repairs should receive special attention as excessive amounts of repair work could lead to initiation of flaws.

B. UT Technique Selection and Qualification

1. Perform automated UT scanning and recording on Alloy 182 and Alloy 82 welds whenever practical. The equally overlapping scan lines of automated UT provide redundant coverage of the outside scanning surface, and the data is recorded and retained automatically. Due to the complex geometry associated with these configurations and the number of different angles required to interrogate the material volume, it is strongly recommended that automated data collection systems be used to allow the data to be analyzed offline. Also, digitized data can only be acquired with automated methods. The main benefits of digitized data are that: (1) the data has

improved accuracy; (2) the data can be archived; and (3) multiple evaluations can be performed by different personnel.

2. The NRC has stated that the use of automated scanning is highly desirable. GE does not recommend manual techniques; however, if manual techniques are used, special care should be exercised to assure that coupling and coverage are adequate. Slow, methodical scanning rates and continual observation of the UT oscilloscope are essential to maximize the crack detection probability for this type of indication.

3. Where possible, the techniques intended for use, whether automated or manual UT, should be calibrated and qualified using a realistic mockup of the weld configuration that contains suitable reflectors in the areas of concern.

C. UT Examination Plan

1. When preparing for upcoming examinations, data collected previously using automated equipment should be reanalyzed with current techniques, where possible. Due to improvements in analysis equipment over the past several years, the results of reanalysis may help in preparing for future inspections and developing associated contingencies.

2. Given recent experiences of larger-than-expected flaws, plants should consider performing fracture mechanics analyses to establish allowable flaw sizes prior to scheduled examinations. Contingency activities for weld overlay repair should be considered as well.

D. UT Performance and Data Evaluation

1. For those designs in which the Alloy 182 weld material ties into the nozzle bore cladding, UT examinations of the Alloy 182 material and the affected LAS nozzle should be performed in this extended area. These examinations can be performed during scheduled ISI of the Alloy 182 nozzle butters.

2. Use 45-degree and 60-degree refracted longitudinal waves for crack detection and sizing in the Alloy 182 material and the low alloy material. This should be performed at a gain level such that the small signals received from the inside surface, which is sometimes referred to as an ID roll, are at approximately 10% of full screen height. Scanning should be performed with the sound beams directed both axially and circumferentially. A 45-degree shear wave should also be used to examine the low alloy material if suspect indications are detected with the refracted longitudinal wave search units.

3. Examination of the upper regions (toward the OD) of the weld that are outside the ASME Code examination volume (inner 1/3 of piping thickness) should be considered, especially in welds where previous indications have been dispositioned as non-relevant or due to geometry. This extended examination region, essentially the full weld volume, is intended to identify indications from mid-wall and deeper flaws where reduced detectability of the base signals may exist.

4. Identify significant UT indications that are not classified as known geometric or metallurgical conditions. These indications should receive further analysis.

5. Identify changes in indication characteristics from prior examinations. Changes that cannot be explained should receive additional evaluation.

AUGMENTED INSPECTION PROGRAMS

6. Do not apply UT techniques used for Intergranular Stress Corrosion Cracking (IGSCC) directly to Dissimilar Metal Welds such as Alloy 182 and Alloy 82 welds and weld butters. Techniques used for Dissimilar Metal welds should take into consideration the multiple weld interfaces, unusual weld geometries typically present, and the tendency for SCC to occur within the weld and weld butter.

7. Consider the possibility of mode conversion or reflection in the evaluation of multiple indications occurring in the same general location. A single flaw may cause UT indications that appear to be separated by a significant distance in the axial direction. An incorrect characterization can be made if mode conversion is not considered. Mode conversion causes the mode, direction and velocity of the sound beam to change as the incident sound beam strikes an interface at certain angles. If not accounted for, the indication could be positioned in an incorrect location.

8. Make a direct comparison of digitized data from UT indications with digitized data taken on previous inspections, where available.

The PECO Energy UT procedures meet the additional requirements of GE SIL No. 455, Rev. 2.

V. EXAMINATION RESULTS

All examination results shall be evaluated in accordance with NUREG-0313, AUG-01 (References B and C) and ASME Section XI.

VI. REPORTS/RECORDS

Reporting and record-keeping requirements shall be in accordance with NUREG-0313, AUG-01, and ASME Section XI.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-07
SIL No. 455
Recommendation for Additional ISI of Alloy 182 Nozzle Weldments, continued

TABLE AUG-07-1
RPV Nozzle to Safe End Alloy 182 Weldments

Component ID No.	Description	Type
VRR-1RS-1A N1A	Recirculation Outlet	Alloy 182 weld / weld butter
VRR-1RS-1B N1B	Recirculation Outlet	Alloy 182 weld / weld butter
VRR-1RD-1B N2A	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1B N2B	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1B N2C	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1B N2D	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1B N2E	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1A N2F	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1A N2G	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1A N2H	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1A N2J	Recirculation Inlet	Alloy 182 weld / weld butter
VRR-1RD-1A N2K	Recirculation Inlet	Alloy 182 weld / weld butter
DCA-319-1 N5A	Core Spray	Alloy 182 weld butter
DCA-320-1 N5B	Core Spray	Alloy 182 weld butter
RPV-1IN N8A	Jet Pump Instrumentation	Alloy 182 weld/weld butter
RPV-1IN N8B	Jet Pump Instrumentation	Alloy 182 weld/weld butter
RPV-1IN N9	CRD Return nozzle to cap	Alloy 182 weld/weld butter
DCA-318-2 N17A	LPCI	Alloy 182 weld butter
DCA-318-1 N17B	LPCI	Alloy 182 weld butter
DCA-318-3 N17C	LPCI	Alloy 182 weld butter
DCA-318-4 N17D	LPCI	Alloy 182 weld butter

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-08
Extended Examination Volume for Code Category B-D

I. SCOPE

This augmented inspection program (AUG-08) is no longer required . Refer to Program No. AUG-02 for scope of examinations.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-09
Examination of the RPV Closure Head Lifting Lugs

I. SCOPE

This augmented inspection program (AUG-09) defines the specific examination requirements for the RPV closure head lifting lugs, as applicable to LGS Unit 1.

II. REFERENCES

None

III. GENERAL

ASME Section XI Code Examination Category B-H, Integral Attachments for Vessels, requires a surface examination of LGS Unit 1 RPV integral attachments. Code Table IWB-2500-1, Note (1) limits integral attachment examinations to those attachments that meet the following conditions:

A. The attachment is on the outside surface of the pressure retaining component;

B. The attachment provides component support as defined in NF-1110;

C. The attachment base material design thickness is 5/8 in. or greater; and

D. The attachment weld joins the attachment either directly to the surface of the vessel or to an integrally cast or forged attachment to the vessel.

Per the above criteria, the RPV closure head lifting lugs are excluded from the examination requirements of Code Category B-H.

IV. EXAMINATION PROGRAM

There are four (4) closure head lifting lugs on the LGS Unit 1 RPV. Due to the importance of the closure head lifting lugs to refueling operations, and the relative magnitude of weight they are required to carry, PECO Energy has determined that routine nondestructive examination of the lug attachment welds is in order.

As such, the closure head lifting lugs shall be optionally examined in accordance with requirements.

V. EXAMINATION RESULTS

All examination results shall be evaluated in accordance with ASME Section XI.

VI. REPORTS/RECORDS

Reporting and record-keeping requirements shall be in accordance with ASME Section XI.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-10
UFSAR Table 3.2-1
Non-Q Reactor Pressure Vessel Internal Components

I. SCOPE

This augmented inspection program (AUG-10) defines the examination requirements applicable to certain LGS Unit 1 reactor pressure vessel internal components, as committed to in the LGS UFSAR Table 3.2-1.

Examination requirements, as detailed in this document, are exclusive of those ASME Section XI inservice inspection requirements for the identified components within the scope of this document.

II. REFERENCE

Limerick Generating Station Updated Final Safety Analysis Report (UFSAR).

III. GENERAL

ASME Section XI, 1989 Edition, Examination Category B-N-1, requires a visual examination, VT-3, of areas/spaces above and below the reactor core that are made accessible for examination by removal of components during normal refueling outages. This requirement, as previously interpreted by PECO Energy, included not only the accessible areas/spaces in the reactor pressure vessel itself, but also those nuclear safety related (Q-listed) reactor pressure vessel internal components which occupy that space. Subsequent to this the BWR Vessel and Internals Project (BWRVIP) has performed a safety assessment and developed a set of inspection and evaluation guidelines for these Q-listed non-Code components. The reactor internals, whose safety function requires conformance to 10CFR50, Appendix B, quality standards are summarized in UFSAR Table 3.2-1 and had been previously included as Examination Category B-N-1 components in the LGS Unit 1 ISI Program. Based on the BWRVIP safety assessment, the inspection requirements for these Q-listed non-Code components are addressed as Augmented ISI Programs in NE-042.

Per UFSAR Table 3.2-1, Note (62), certain reactor internal components are non-nuclear safety related (non-Q-listed) nor are they subject to 10CFR50, Appendix B. However, PECO Energy, recognizing the importance of these components, has committed to including the subject components for examination in the LGS Unit 1 ISI Program.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-10
UFSAR Table 3.2-1
Non-Q Reactor Pressure Vessel Internal Components, continued

IV. EXAMINATION PROGRAM

Per UFSAR Table 3.2-1, Note (62), the following components are not safety related, not Q-listed and not under 10CFR50, Appendix B;

A. steam dryer,
B. shroud head and steam separator assembly,
C. in-core guide tubes/guide tube stabilizers,
D. differential pressure and liquid control lines inside the RPV (excluding those portions that are part of the reactor coolant pressure boundary and are Q-listed),
E. fuel orifices,
F. feedwater spargers,
G. jet pump instrument lines, and
H. surveillance specimen holders.

These components, per this augmented inspection program, are included for examination in the LGS Unit 1 ISI Program. Where an Augmented ISI Program for these non-nuclear safety related components has been established to address requirements from USNRC Regulatory Guides, Generic Letters, Bulletins, or accelerated implementation of Code rules and regulations or has been established to address vendor recommendations, industry committee efforts, or self-imposed inspections requirements then the inspection requirements of AUG-10 (this augmented inspection program) shall be limited to those requirements of the specific Augmented ISI Program for the component. Otherwise, the above components shall be subject to inspection requirements modeled on ASME Section XI, 1989 Edition, Table IWB-2500-1, Examination Category B-N-1.

NOTE: The integral attachment welds to the above components shall be subject to inspection requirements of ASME Section XI, 1989 Edition, Table IWB-2500-1, Examination Category B-N-2.

V. EXAMINATION RESULTS

All examination results shall be evaluated/dispositioned in accordance with the rules of ASME Section XI or the specific Augmented ISI Program.

VI. REPORTS/RECORDS

All records/reports shall be prepared/maintained in accordance with the rules of ASME Section XI and plant procedures or the specific Augmented ISI Program.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-11
SIL No. 409
Incore Dry Tube Cracks

I. SCOPE

This augmented inspection program (AUG-11) defines the specific examination requirements of General Electric Company (GE) Nuclear Services Information Letter (SIL) No. 409, as applicable to LGS Unit 1. This SIL provides information/ recommendations relative to cracks found in BWR Intermediate Range Monitor (IRM) and Source Range Monitor (SRM) instrumentation dry tubes.

Examination requirements, as detailed in this document, are exclusive of any ASME Section XI inservice inspection requirements for the identified components within the scope of this document.

II. REFERENCE

GE SIL No. 409, Incore Dry Tube Cracks, Revision 1, dated July 31, 1986.

III. GENERAL

Examinations of IRM/SRM dry tubes at several BWRs have resulted in cracking and/or crack indications observed in a number of IRM/SRM instrumentation dry tubes. All of the observed cracks are within the top two (2) feet of the dry tube assembly, primarily in the perforated tube adjacent to either the weld between the tube and the guide plug or the weld between the tube and the primary pressure boundary. (See SI drawing XI-BN-5, Page 1)

The cracking is considered to be caused by a combination of crevice corrosion cracking and irradiation assisted stress corrosion cracking (IASCC), while crack initiation time is strongly dependent on BWR water chemistry (i e. water conductivity).

The LGS Unit 1 IRM/SRM instrumentation dry tubes are the original BWR/2-6 design and as such are susceptible to the cracking described.

Crack initiation time and growth rate for the LGS Unit 1 configuration are dependent on time in use, water quality, and loading variations (e.g., flow induced vibration, bumping during fuel movements). Recommended visual examinations shall be in accordance with Section IV.

IV. EXAMINATION PROGRAM

There are four (4) SRM and eight (8) IRM dry tube assemblies in the LGS Unit 1 reactor pressure vessel. Visual examination (VT-1) of the top two (2) feet of these dry tubes is recommended in accordance with Table AUG-11-1.

V. EXAMINATION RESULTS

Visual examination results shall be documented/dispositioned in accordance with ASME Section XI.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-11
SIL No. 409
Incore Dry Tube Cracks, continued

VI. REPORTS/RECORDS

All reports/records associated with the examinations of this augmented program shall be prepared/maintained in accordance with ASME Section XI and plant procedures.

TABLE AUG-11-1 INCORE DRY TUBE
RECOMMENDED INSPECTION PROGRAM

	Water conductivity	
	Meets EPRI Guidelines[1]	Does Not Meet EPRI Guidelines[2]
SRM/IRM Dry Tubes	4/2[2,3,4]	2/1[2,3,5]

NOTES:

1. EPRI water conductivity guidelines appear in EPRI NP 3589 SR LD for the cumulative service of dry tubes.

2. X/Y - Visual examination should be performed during the "Xth" refueling outage after dry tube installation. Subsequent visual examinations should be performed every "Yth" refueling outage.

3. The SRM/IRM dry tubes are located between the Top Guide and Core Plate and are not accessible during a normal refueling outage. Removal of an adjacent fuel cell is required to provide access for remote visual examination.

4. The SRM/IRM dry tubes are ASME Section XI B-N-1 components and are required to be examined when accessible. Access permitting, the Code frequency of examination meets or exceeds the requirements of this augmented program. However, this program requires a VT-1 examination be performed in lieu of a VT-3.

5. The SRM/IRM dry tubes are ASME Section XI B-N-1 components and are required to be examined when accessible. The requirements for subsequent examinations, given water quality, are more restrictive than that required by the Code. Consideration should be given to flow induced vibrations, bumping during fuel movements and time since last visual examination in scheduling these components for examination.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-12
SIL No. 420
Inspection of Jet Pump Sensing Lines

This augmented program (AUG-12) was originally required by the requirements of General Electric Nuclear Services Information Letter (SIL) No. 420, Inspection of Jet Pump Sensing Lines, dated March 28, 1985. The requirements of SIL No. 420 were superseded by BWRVIP-41, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines, dated October 1997. The recommendations of BWRVIP-41 are covered in AUG-04, Jet Pump Assembly. The requirements of AUG-12 have been superseded by and are incorporated into AUG-04. This augmented examination program, AUG-12, is NO LONGER REQUIRED.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program

I. SCOPE

This augmented inspection program (AUG-13) defines the mandatory examination and testing requirements for snubbers. This program has been prepared to satisfy the Surveillance Requirements of LGS Unit 1 Technical Specification 3/4.7.4.

All snubbers installed on the reactor coolant system and all other nuclear safety related systems are subject to the requirements of this augmented inspection program. Snubbers installed on non-nuclear safety related systems are also within the scope of this program and may be excluded only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

Examination and testing requirements of this augmented inspection program apply to the snubber assembly which includes the snubber body and attachments out to and including the load pins and their retainers per Figure AUG-13-1 (See Attachment 4). Snubber support components beyond this defined space are outside of the scope of this augmented inspection program.

A complete listing of all snubbers within the scope of this augmented inspection program is provided in the AUG-13 Augmented Inspection Program Tables.

II. REFERENCES

A. ASME/ANSI Operations and Maintenance Standard OM-1987 with OMc-1990 Addenda, Part 4 (including additional industry/committee studies).

B. LGS Unit 1 Technical Specification 3/4.7.4, Snubbers

III. DEFINITIONS

A. Activation - the parameter that verifies restraining action.

B. Application Induced Failure - failures resulting from environmental conditions or application of the snubber for which it has not been designed or qualified.

C. Breakaway Force - the minimum applied force required to initiate extension or retraction of the snubber.

D. Defined Test Plan Group - a population of snubbers having similar design or application characteristics selected for testing in accordance with the 13.3 percent or 37 testing sample plan.

E. Design or Manufacturing Failure - failures resulting from a potential defect in manufacturing or design that give cause to suspect other similar snubbers. This includes failures of any snubber(s) that fails to withstand the environment or application for which it was designed.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

F. Drag Force - the force required to maintain the snubber movement at a constant velocity prior to activation.

G. Equipment Dynamic Restraint (Snubber) - a device which provides restraint to a component or system during a sudden application of forces but allows essentially free motion during thermal movement.

H. Examination Group - a composition of snubbers which have been selected to be examined.

I. Examination - the performance of visual observations for impaired functional ability due to physical damage, leakage, corrosion or degradation from environmental or operating conditions.

J. Failure Mode Group - a composition of snubbers whose failure and potential for the same failure is similar.

K. Inaccessible Snubbers - those snubbers that are in a high radiation area or other conditions that would render it impractical for the snubbers to be examined under normal plant operating conditions without exposing plant personnel to undue hazards.

L. Isolated Failure - the nature of the failure does not lend other snubbers to be suspect. For example, failures resulting from damage during installation or shutdown (i.e., dropping equipment or tools on the snubber, missing pins, etc.).

M. Maintenance - replacement of parts, adjustments, and similar actions which do not change the design of the snubber, taken to prevent deficiencies in the function of a snubber.

N. Maintenance, Repair, Installation Induced Failures - failures which result from damage during maintenance, repair, or installation activities, the nature of which lends other snubbers to be suspect.

O. Mechanical Snubbers - devices in which load is transmitted entirely through mechanical components.

P. Modification - alteration in the design of a snubber to improve its suitability for a given environment or application

Q. Normal Operating Conditions - operating conditions during reactor startup, operating at power, hot standby, reactor cooldown to cold shutdown.

R. Operability Testing - measurement of parameters that verify snubber operability.

S. Operating Temperature - the temperature of the environment surrounding a snubber at its installed plant location during the phase of plant operation for which the snubber is required.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

T. Qualitative Testing - that testing performed to establish the functioning of a parameter without determining the specific measure of the parameter, similar to go/no-go gauging.

U. Quantitative Testing - that testing performed to establish the specific measure or the limit of the functioning of a parameter, such as that required to establish that a parameter is functioning within a specified range.

V. Release Rate - the rate of the axial snubber movement under a specified load after activation of the snubber took place.

W. Repair - replacement of parts and similar actions which do not change the design of the snubber, taken to correct deficiencies in the function of a snubber.

X. Replacement Snubber - any snubber other than the snubber immediately previously installed at the location.

Y. Swing Clearance - the movement envelope within which the snubber must operate without restriction, from the cold installed position to the hot operating position.

Z. Test Temperature - the temperature of the environment surrounding the snubber at the time of the test.

AA. Unacceptable Snubbers - those snubbers which do not meet examination or testing requirements.

BB. Unexplained Failure - failures that cannot be categorized as design or manufacturing, maintenance, repair, installation, application induced, or isolated. This includes all failures for which the cause of the failure cannot be determined.

IV. GENERAL SNUBBER EXAMINATION AND TESTING REQUIREMENTS

Snubbers are installed on nuclear safety and non-nuclear safety related systems at LGS Unit 1 to ensure the continued structural integrity of the reactor coolant system and other nuclear safety related systems following a seismic or other event initiating dynamic loads. As such, assurance of the ability of these snubbers to perform as designed through examination and testing is imperative.

Requirements for examination and testing of snubbers are addressed by regulatory and industry groups in plant Technical Specifications, ASME Section XI, ASME / ANSI, OM-1987 with OMc-1990 Addenda and INPO good practices. This augmented inspection program, prepared by PECO Energy, is intended to provide a comprehensive program which demonstrates the operability of applicable LGS Unit 1 snubbers and effectively addresses both regulatory and industry concerns regarding snubber examination and testing.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

This augmented inspection program constitutes the "Surveillance Requirements" section of plant Technical Specification 3/4.7.4., Requirements of Technical Specification 3/4.7.4, other than surveillance requirements, still apply.

A. Responsibility

PECO Energy, as Owner of LGS Unit 1, is responsible for the preparation and implementation of this program including:

1. Implementation of the requirements of this program in accordance with site administrative procedures and the Quality Assurance Program.

2. Qualification of personnel performing the examinations and tests.

3. Preparation of all necessary written procedures for complying with the requirements of this program.

4. Collection and retention of all design and operating information necessary for the performance of the examination and testing program. This information shall be available for use during implementation of the program.

B. Procedures

Examinations, tests, and maintenance or repair activities shall be performed in accordance with written procedures.

C. Examination and Test Results

The results of all examination and testing shall be documented and shall include as a minimum:

1. Manufacturer's model number, serial number, type, unique location identification and/or PECO Energy identification of the snubber, as applicable.

2. Pertinent examination and test data.

3. Identification and disposition of nonconformances.

4. Information to identify the test/examination performed, procedure used, and date.

5. Test equipment used.

6. Acceptability of test/examination results.

7. Identification of examination and test personnel.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

D. Personnel Qualifications

Test Personnel who are required to witness, perform, and/or evaluate the snubber testing shall be qualified in accordance with site administrative procedures. Inspection personnel performing and evaluating visual examinations shall be qualified for VT- 3 visual examination in accordance with ASME Section XI 1989 Edition or PECO Energy approved equivalent.

E. Instrumentation and Test Equipment

Instrumentation and test equipment used to verify snubber performance shall have the range and accuracy necessary to demonstrate conformance to specific examination or test requirements.

All instruments and test equipment used in performing the examination and testing program shall be calibrated and controlled in accordance with site administrative procedures.

F. Snubber Maintenance or Repair

Snubbers within the scope of this program shall not be subjected to maintenance or repair prior to examination and/or testing specifically for the purpose of meeting the examination and/or testing requirements. The preventative or corrective actions required by the LGS Quality Assurance Program shall supersede this requirement.

G. Post Maintenance Examination and Testing

Maintenance activities which can alter the snubber's intended function shall be evaluated by considering the effects of the maintenance on the snubber's ability to meet the examination and testing acceptance criteria. Snubbers which undergo maintenance activities which could alter the snubber's ability to perform its intended function shall be examined and tested in accordance with the applicable requirements of Section V of this appendix. The requirements selected shall ensure that the function(s) which may have been affected are verified by the examination or tests to be acceptable.

The site administrative procedures governing maintenance activities shall address these requirements.

H. Snubber Repair, Replacement, or Modification

All snubbers within the scope of this program shall be repaired, replaced or modified in accordance with ASME Section XI and site administrative procedures.

Repair activities which can alter the snubber's intended function shall be evaluated by considering the effects of the repair on the snubber's ability to meet the examination and testing acceptance criteria. Snubbers which undergo repair activities which could alter the snubber's ability to perform its intended function shall be examined and tested in accordance with the applicable requirements of Section V of this appendix.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

Replacement or modified snubbers shall be visually examined in accordance with the requirements of Section V of this appendix.

Visual examinations or operability testing, as may be required above, shall be addressed in site administrative procedures governing ASME Section XI repair/replacement activities.

I. Deletion of Unacceptable Snubbers

When unacceptable snubbers are deleted (based on analysis of the affected piping system), the deleted snubber(s) shall, nevertheless, be considered in its respective failure mode group; and the effect of the corrective action taken, for the balance of the failure mode group, shall apply. For example, for the purposes of the applicable corrective action, the deletion of the snubber may be considered the same as replacement with a snubber qualified for the application.

J. Transient Dynamic Event

If a transient dynamic event occurs which may affect operability, the affected system(s) and associated snubbers shall be reviewed and any appropriate corrective action taken. Any corrective actions taken are independent of the examination and testing requirements of this program.

K. Supported Component(s)/System Evaluation

An engineering evaluation shall be performed of component(s) and/or system(s) on which an unacceptable snubber is installed for possible damage to the supported system and/or component.

V. <u>SNUBBER EXAMINATION AND TESTING PROGRAM</u>

Each snubber within the scope of this augmented inspection program shall be demonstrated operable by performance of the program requirements as detailed in this Section.

Certain snubbers may be waived in part or totally from the requirements of this program (on a case-by-case basis), provided technical justification for the deviation be filed with regulatory authorities prior to implementation of the deviation.

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

A. Visual Examination

Visual examination for operational readiness is required for snubbers with the number of snubbers and the frequency of reexamination being determined by the number of unacceptable snubbers within a group and the corrective action taken.

Visual examination shall be performed to identify physical damage, leakage, corrosion, or degradation from environmental exposure or operating conditions. External features which may indicate operability of the snubber shall be examined. An examination checklist shall be prepared for this purpose.

The initial visual examinations performed in accordance with this augmented program shall be implemented during the first refueling outage following regulatory acceptance of this program.

1. Examination Documentation

   The following documentation is necessary to support implementation and verification of the visual examinations:

   a. Examination procedures and checklists verifying examination and as-found conditions.

   b. Examination records.

   c. Thermal movement inspection records.

   d. Records of nonconformance and corrective actions that are required.

2. Snubber Categorization

   Snubbers may be categorized and grouped as accessible and inaccessible; these groups may be considered separately for the purpose of visual examination. Determination of accessible/inaccessible snubber groups and plans for separate or joint application of program requirements by group shall be made and documented prior to initiating examinations for a given examination interval. Once determined, groups shall be used throughout the examination interval and shall not be changed.

3. Examination Sample Size

   The initial and all subsequent visual examinations shall include all (100%) of the snubbers of all groups as may have been established in 2) above.

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

4. Examination Frequency

   The initial inservice examination of all snubbers shall be started not less than two months after attaining 5% reactor power operation and shall be completed within 12 calendar months after attaining 5% reactor power operation. Subsequent examination intervals shall be as follows:

   a. The second inservice examination shall be conducted at the first refueling outage. No schedule change in accordance with Table AUG-13-1 (See Attachment 4) is required.

   b. The third inservice examination shall be conducted at the second refueling outage.

   c. Subsequent examination intervals shall be in accordance with Table AUG-13-1 (See Attachment 4).

5. Outage-Based Visual Examinations

   Table AUG-13-1 (See Attachment 4), Refueling Outage-Based Visual Examination Table, provides the permissible number of unacceptable snubbers allowed, for various snubber populations or groups, to continue with the normal examination frequency schedule. In addition, Table AUG-13-1 details all corrective actions to be taken and provides examination frequency adjustments to be made, based on the number of unacceptable snubbers found during the visual examination.

6. Visual Examination Acceptance Criteria

   Visual examinations shall verify conformance of the snubber installation to the following requirements:

   a. Must Restrain Movement.

      Snubbers shall be installed such that when activated, piping/component movement is restrained. Visual observation of loose fasteners, corroded or deformed members, or detection of disconnected components or other conditions that might interfere with the proper restraint of movement requires evaluation. Snubbers which are determined to be incapable of restraining movement shall be considered unacceptable.

   b. Must Permit Thermal Movement.

      Snubbers shall be installed in such condition that thermal movement of the piping/component is not restricted to the extent that unacceptable over stressing could develop. Observed binding, misalignment, and/or deformation may be indicative of such a situation, and shall be evaluated. Snubber installations determined to excessively restrict piping/component thermal movement shall be considered unacceptable.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

c. Design-Specific Observations.

Snubbers shall be free of defects that may be generic to particular designs, as may be detected by visual examination. Visual examination anomalies which indicate potential impaired operability of the snubber(s) may be resolved by operability testing in accordance with the Section V, Paragraph A.7 of this appendix.

7. Visual Inspection Results Evaluation

Discrepant conditions identified during visual examinations shall be evaluated and dispositioned in accordance with Table AUG-13-2 (See Attachment 4). Snubbers which have been evaluated as being incapable of performing their function, or having conditions which if left uncorrected, could affect the capability of the snubber performing its function will be counted as visual failures for the purpose of calculating the next inspection interval.

8. Operability Test Evaluation

Any snubber(s) found to be unacceptable as a result of visual examination may be operability tested in accordance with the requirements of Section V, Paragraph B of this appendix. Results may be used to evaluate the snubber as acceptable, provided that testing can show the unacceptable condition did not affect operability.

B. Operability Testing

Operability testing for operational readiness is required to be performed on representative samples of snubbers, based on the sampling plans provided herein. The number of snubbers to be tested is determined by the sampling plan chosen and the corrective actions prescribed by that sampling plan. Additional samples taken, based on the number of unacceptable snubbers found, is also determined by the specific sampling plan chosen.

Testing, as required by this augmented program, shall be implemented during the first refueling outage following regulatory acceptance of this program.

1. Testing Documentation

The following documentation is necessary to support implementation and verification of operability testing:

a. Operability testing procedures.

b. Previous test records.

c. Nonconformance results, evaluations, and corrective actions.

d. Defined test plan grouping.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

2. Operability Testing Requirements

The following general requirements apply to all operability testing performed:

a. Operability testing loads

Snubbers shall be tested at a load sufficient to verify the operating parameters specified in Section V, Paragraph B.4, of this appendix. Testing at less than rated load must be correlated to operability parameters at rated load.

b. Test correction factors

Differences may exist between the installed operating conditions and the conditions under which a snubber is tested. In such cases, correction factors shall be established and test results shall be correlated to operating conditions as appropriate.

c. Test-as-found.

Operability testing should be performed on snubbers in their "as-found" condition, to the fullest extent practical, for all snubber parameters to be tested.

d. Test restrictions.

Testing methods utilized shall not alter the condition of the snubber such that the test results no longer represent snubber parameters prior to testing.

e. In situ testing.

Where desirable, in situ operability testing (i.e., testing with the snubber installed in its permanent location) may be utilized provided test methods and equipment have been approved by PECO Energy.

f. Bench testing.

Operability testing may be performed by removal of the snubber and bench testing, provided test methods and equipment have been approved by PECO Energy. Following reinstallation of the snubber, a visual examination in accordance with the applicable requirements of Section V, Paragraph A of this appendix, shall be performed.

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

g. Subcomponent testing.

Where snubber physical size, test equipment limitations, or snubber inaccessibility prevent the use of either in situ testing or bench testing, the snubber subcomponents shall be tested and reassembled in accordance with PECO Energy approved procedures.

h. Correlation of indirect measurements.

Testing methods may be used which measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters, through established methods.

i. Parallel and multiple installations.

The snubbers of parallel and/or multiple installations shall be identified and counted individually.

j. Fractional sample sizes

All fractional sample sizes shall be rounded up to the next integer.

3. Qualitative Testing.

Qualitative testing may be used in lieu of quantitative measurements in meeting the operability test acceptance criteria of this document, following review and approval of this method, by regulatory authorities. Sufficient data, based upon service history or life cycle testing, shall be obtained to demonstrate the ability of the parameter in question to be within specification over the life of the snubber (e.g. demonstration that activation takes place without measurement of the activation level). A test report shall be prepared for each snubber exempted from quantitative operability testing requirements. The test report shall verify the parameter was within specifications to allow exemption of the snubber from quantitative testing of the parameter.

4. Operability Testing Acceptance Criteria

Operability testing shall verify conformance to the following requirements:

a. The force that will initiate motion (breakaway force), the force that will maintain low velocity (drag force), or both, as required by the test procedure, are within specified limits, both in tension and in compression.

b. Activation is within the specified range of time, velocity, or acceleration in both tension and compression.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

c. Release rate, where applicable, is within the specified range in tension and compression. For units specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be demonstrated.

5. Operability Testing Failure Evaluation

Snubbers that do not meet the operability testing acceptance criteria for quantitative testing or qualitative testing shall be evaluated to determine the cause of failure, using test failure mode groups.

a. Test failure mode groups

Unacceptable snubber(s) shall be categorized into test failure mode group(s). Test failure mode group(s) shall include all unacceptable snubbers that have a given failure mode, and all other snubbers subject to the same failure mode. The following failure modes shall be used:

- Design, manufacturing
- Application induced.
- Maintenance, repair, installation.
- Isolated.
- Unexplained.

b. Test failure mode group boundaries

Once a test failure mode group has been established, any snubber(s) in that test failure mode group will not be part of the defined test plan groups from which the snubber(s) originated except as noted in (c) below. The new test failure mode group will remain as defined until corrective action has been completed.

Note that for the 37 testing sample plan, established failure mode groups once separated from the defined test plan group(s), are referred to as "independent" test failure mode groups.

c. Snubbers in more than one test failure mode group

In the event that a snubber(s) becomes included in more than one test failure mode group, it shall be counted in each failure mode group in which it is unacceptable and shall be subject to the corrective action of each test failure mode group.

d. Additional failure mode group review

Once the operability test requirements are satisfied for a given defined test plan group, then any additional failure mode group review or testing shall not require any subsequent testing on the defined test plan group.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

6. Defined Test Plan Groups.

   Defined test plan groups shall be determined prior to initiating testing. These groups shall encompass all snubbers and shall be based on similarities of design or application. That is, snubbers may be grouped by size, type, design, application, or other means as determined by engineering evaluation.

7. Operability Testing Interval

   Testing in accordance with the selected sampling plan shall be performed each refueling outage for each defined test plan group.

8. Operability Testing Sampling Plan Selection.

   Testing shall be conducted for each defined test plan group using one of the following sampling plans:

   a. 13.3% testing sample plan
   b. 37 testing sample plan

   The plan to be used for each defined testing plan group of snubbers shall be selected before testing begins for the test interval. Once selected, the plan shall be used throughout the test interval for that defined test plan group and any failure mode group that is determined from the original defined test plan group.

9. Operability Testing Corrective Action and Continued Testing

   Snubbers that do not meet the operability testing acceptance criteria for quantitative testing or qualitative testing shall be subject to corrective action(s), with its indicated impact on continued testing. Selection of the corrective action shall be governed by the sampling plan which is used. Any maintenance, repairs, replacements or modifications shall meet the requirements of this program.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

10. The 13.3% Testing Sample Plan

When the 13.3% testing sample plan is chosen for a defined test group, the following criteria shall apply:

a. Initial test sample lot size and composition for a test interval.

For the first sample lot tested, a representative/random sample of 13.3% of the snubbers in the defined test plan group shall be selected. As far as practical, the sample selected shall include the various designs, configurations, operating environments, range of sizes, capacity of snubbers, etc. The first sample lots tested shall be a composite based on the ratio of each particular category to the total number of snubbers in the defined test plan group. Sample lot selection from the representative categories of snubbers shall be random.

b. Additional test(s) lot size in the same test interval.

For any snubber(s) determined to be unacceptable as a result of testing, an additional sample of at least 1/2 the size of the initial sample lot shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where C is the total number of snubbers found to be unacceptable; or all snubbers in the failure mode group have been tested. (The testing of additional samples by this criteria is also required for snubbers determined to be unacceptable in any additional test lot.)

c. Additional test lots composition in the same test interval.

As far as is practical, the additional samples shall include:

- Snubbers of the same manufacturer's design.
- Snubbers immediately adjacent to those found unacceptable.
- Snubbers from the same piping system.
- Snubbers from other piping systems that have similar operating conditions such as temperature, humidity, vibration, and radiation.
- Snubbers which are previously untested.

d. Subsequent test interval population selection.

For subsequent refueling outages, each representative sample shall be selected in accordance with a), b), and c) above, from the total population of the defined test plan group.

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

e. Sample plan corrective action.

The 13.3% sample plan corrective actions are dependent upon the assigned failure mode group as follows:

Design, manufacturing, maintenance, repair, installation and application induced test failure modes

- All snubbers in a test failure mode group shall be replaced or modified in accordance with Section IV, Paragraph H of this document, and categorized as acceptable.

  OR

- The unacceptable snubbers in the test failure mode group shall be replaced, or repaired to the original qualified condition. The number of unacceptable snubbers shall determine the additional test lots of Section V, Paragraph B.10.b.

  OR

- The unacceptable snubbers in an application induced test failure mode group shall be replaced or repaired to an acceptable condition. All snubbers in this group shall be categorized as acceptable provided the environment or applications are compatible with the design parameters.

Isolated test failure mode

The unacceptable snubbers in this test failure mode group shall be replaced or repaired in accordance with Section IV, Paragraph H of this appendix and categorized as acceptable.

Unexplained test failure mode

The unacceptable snubbers in this test failure mode group shall be replaced or repaired in accordance with Section IV, Paragraph H of this appendix. The number of unacceptable snubber(s) shall determine the additional testing lots in accordance with Section V, Paragraph B.10.b.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

11. The 37 Testing Sample Plan

When the 37 testing sample plan is chosen for a defined test group, the following criteria shall apply:

a. Initial sample size and composition.

The initial sample shall consist of 37 snubbers selected randomly for each defined test group which utilizes the 37 testing sample plan.

b. Additional defined test plan group testing

For any snubber(s) determined to be unacceptable as a result of testing, additional samples shall be selected such that the following test plan equation is satisfied (See Attachment 4, Figure AUG-13-2):

$\mathbf{N \geq 36.49 + 18.18\ C}$ where

N = Total number of snubbers tested which were selected from the defined test plan group

and

C = Total number of unacceptable snubbers found in the defined test plan group (excluding those in independent test failure mode groups) plus one for each independent test failure mode group.

Additional samples shall be selected in a random manner from the remaining population of the defined test plan group. Snubbers in test failure mode groups shall be separated and should not be included in the additional sample(s).

c. Independent failure mode group testing

Once a test failure mode group has been established, it shall be separated for continued testing apart from the defined test plan group. It is then identified as an independent test failure mode group.

For an independent test failure mode group, the number of unacceptable snubbers which define the test failure mode group shall determine the additional testing in the test failure mode group in accordance with the following equation (See Attachment 4 Figure AUG-13-2):

$\mathbf{N \geq 36.49 + 18.18\ C}$ where

AUGMENTED INSPECTION PROGRAMS

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Snubber Examination and Test Program, continued

N = Initial defined test plan lot of 37 tested plus all those selected and tested from the independent test failure mode group.

and

C = Total number of unacceptable snubbers in the independent test failure mode group.

In addition, the following criteria shall apply to additional testing in an independent test failure mode group:

- Snubbers are selected in a random manner from the independent test failure mode group.

- Any additional unacceptable snubbers found in the independent test failure mode group shall be counted for continued testing only for that independent test failure mode group.

- Testing completion is in accordance with the equation in c) above.

d. The 37 testing sample plan corrective action

The following corrective action shall apply:

- All unacceptable snubbers in the defined test plan group shall be replaced or repaired in accordance with Section IV, Paragraph H of this appendix to the original qualified condition. These unacceptable snubbers shall remain categorized as unacceptable for the purpose of additional testing per the 37 testing sample plan, Section V.B.13.b of this appendix.

- The unacceptable snubber(s) in a test failure mode group shall be replaced or repaired in accordance with Section IV, Paragraph H of this appendix to the original qualified condition. These unacceptable snubbers shall be used in determining the requirements for additional testing per the 37 testing sample plan, Section V.B.11.c of this appendix.

VI. SERVICE LIFE MONITORING

The service life of all snubbers shall be monitored to ensure that the service life is not exceeded between augmented examination/testing intervals. The maximum expected service life for various seals, springs, and other critical parts shall be extended of shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be operable. Replacements shall meet the requirements of Section IV, Paragraph H of this appendix.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

VII. REPORTS/RECORDS

All reports/records associated with the examinations/testing of this augmented program shall be prepared/maintained in accordance with ASME Section XI and site administrative procedures.

Records of service life monitoring shall be maintained in accordance with LGS Unit 1 Technical Specification 6.10.3.

Details of examinations and tests conducted under this program need not be included in the ASME Section XI Summary Report and Form NIS-1. An abstract of examinations completed may be provided in the Summary Report. Form NIS-2 shall be included in the Summary Report for any Section XI repairs or replacements performed on the snubbers within the scope of this program.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-14
Balance of Plant
Snubbers Examination Program

I. SCOPE

This augmented inspection program (AUG-14) defines the PECO Energy examination requirements for snubbers installed on non-nuclear safety related systems which are not subject to the examination and testing requirements of Technical Specification 3/4.7.4.

The examination requirements of this augmented inspection program apply to the snubber assembly which includes the snubber body and attachments out to and including the load pins and their retainers (See Attachment 4 Figure AUG-13-1). Like the Tech Spec Snubber Program support components beyond this defined space are outside of the scope of this augmented inspection program.

A complete listing of all snubbers within the scope of this augmented inspection program is provided in the Augmented Inspection Program No. AUG-14 Tables.

II. REFERENCES

None

III. GENERAL

Snubbers installed on non-nuclear safety related systems whose failure or failure of the system on which they are installed would have no adverse effect on any safety related system are excluded from the examination and testing requirements of Technical Specification 3/4.7.4. However, at LGS Unit 1, all snubber assemblies (both nuclear safety related and non-nuclear safety related) were procured and installed to the quality standards of nuclear safety related snubbers. As such, it is PECO Energy's plan to continue to maintain all snubber assemblies to quality standards. Therefore, the purpose of this augmented inspection program is to provide examination requirements necessary for continued assurance of the quality of the subject non-Technical Specification snubbers.

IV. EXAMINATION REQUIREMENTS

All (100%) of the snubbers within the scope of this augmented inspection program shall be visually examined.

The visual examination, performed to assess the general mechanical and structural condition of The snubber, shall identify any physical damage, leakage, corrosion, or degradation from environmental exposure or operating conditions. In addition, the snubber shall be visually examined for operability. Snubber physical (attributes) features which indicate functional adequacy or could affect snubber operability shall also be visually examined.

Examinations shall be performed every ten years for those snubbers not examined during refueling outages.

Examinations shall be performed utilizing procedures and personnel qualified in accordance with the ISI Program (see Section 3.3.1).

PROGRAM No. AUG-14
Balance of Plant
Snubbers Examination Program, continued

V. EXAMINATION RESULTS

Visual examination results shall be documented and evaluated in accordance with ISI Program requirements. In addition, the snubber installation shall be evaluated for conformance to the following requirements:

A. Must Restrain Movement.

Snubbers shall be installed such that when activated, piping/component movement is restrained. Visual observation of loose fasteners, corroded or deformed members, or detection of disconnected components or other conditions that might interfere with the proper restraint of movement requires evaluation. Snubbers which are determined to be incapable of restraining movement shall be considered unacceptable.

B. Must Permit Thermal Movement.

Snubbers shall be installed in such condition that thermal movement of the piping/component is not restricted to the extent that unacceptable over stressing could develop. Observed binding, misalignment, and/or deformation may be indicative of such a situation, and shall be evaluated. Snubber installations determined to excessively restrict piping/component thermal movement shall be considered unacceptable.

C. Design-Specific Observations.

Snubbers shall be free of defects that may be generic to particular designs, as may be detected by visual examination.

Visual examination anomalies which indicate potential impaired operability of the snubber(s) may be resolved by operability testing.

Snubbers evaluated as unacceptable shall be unacceptable for service until such time as the snubber is deemed acceptable via repair, replacement, engineering evaluation or operability test. Note that the discovery of an unacceptable snubber per this augmented inspection program does not initiate additional examinations, nor does the total number of unacceptable snubbers impact examinations and testing of Augmented Inspection Program No. AUG-13, Technical Specification 3/4.7.4, Snubber Examination and Test Program.,

VI. REPORTS/RECORDS

All reports/records associated with the examinations of this augmented program shall be prepared/maintained in accordance with ASME Section XI and site administrative procedures.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-15
SIL No. 433
Shroud Head Bolt Cracks

I. SCOPE

This augmented inspection program (AUG-15) defines the specific examination recommendations of General Electric Company (GE) Nuclear Services Information Letter (SIL) No. 433, as applicable to LGS Unit 1. This SIL addresses the occurrence of intergranular stress corrosion cracking (IGSCC) of RPV shroud head bolting and the GE recommended actions in light of this problem.

Any shroud head bolts which have been replaced with the new design (BWR/6) bolt are outside of the scope of this augmented inspection program.

II. REFERENCES

A. GE SIL No. 433, Shroud Head Bolt Cracks, dated February 7, 1986.

B. GE SIL No. 433, Supplement 1, Shroud Head Bolt Failures, dated September 15, 1993.

III. GENERAL

Shroud head bolt cracking has been observed at several BWR facilities. This cracking, which occurs in the NiCrFe Alloy 600 shaft in a creviced region, has been confirmed to be crevice accelerated IGSCC. Crack initiation and growth rate are dependent on time in use, loading, and particularly, on water quality.

GE SIL No. 433, Supplement 1, identified a new crack location in the pre-BWR/6 shroud head bolts different from the creviced collar area. The new crack location is approximately 68 inches above the bottom of the bolt at the weld connection between the lower portion of the Inconel 600 rod and the stainless steel type 304 stud portions of the bolt.

Regardless of the crack location, failed bolts provide the same response: i.e. they will fail to unlatch/latch the tee section or hold the 50 ft-lb preload.

The shroud head bolts are non-nuclear safety related. There is no safety concern associated with a failure of these bolts. There is not a loose parts concern during power operations because a failed bolt will be captured in-place. The failure of 1 or 2 bolts is not sufficient to allow differential pressure to lift the shroud head during plant operation. Plant specific analysis may be performed to justify plant operation with more than 2 failed bolts. While failure of the shroud head bolt(s) is not a safety concern, loss of bolt integrity may result in an unnecessary challenge to plant equipment; therefore, augmented examination of the LGS Unit 1 shroud head bolting is recommended.

Note that the GE design "replacement" shroud head bolt incorporates modifications to eliminate the collar crevice and utilizes a more IGCSS resistant material; augmented examination of any replacement shroud head bolts is not required.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-15
SIL No. 433
Shroud Head Bolt Cracks, continued

IV. EXAMINATION REQUIREMENTS

A visual inspection (non-ASME Code), to verify that the bolt tee section is unlatched from the shroud lugs before removing the shroud head, should be performed if the pre-BWR/6 shroud head bolts have not been UT'd in the past 3 years. This visual verification of shroud head bolt unlatching is performed each refuel outage per the reactor pressure vessel disassembly procedure. This visual verification is a positive form of functional testing; performed at a higher frequency then the UT of the pre-BWR/6 bolts.

Straight beam ultrasonic examination of all LGS Unit 1 original design (pre-BWR/6) shroud head bolts should be performed once each inspection interval (i.e. once in every 10 years). The UT procedure utilized shall be capable of detecting IGSCC in the given bolt configuration. Ultrasonic examinations are conducted with the subject bolts in place on the shroud head/separator, following removal of the assembly and storage in the equipment storage pool.

V. EXAMINATION RESULTS

All failed bolts, and bolting evaluated as cracked, shall be considered for replacement with the new design replacement bolts.

If cracked bolts cannot be replaced or if bolt status is indeterminate, an engineering evaluation shall be performed to assess safety concerns and potential risk of damage to other plant equipment.

VI. REPORTS/RECORDS

All reports/records associated with the examinations of this augmented program shall be prepared/maintained in accordance with ASME Section XI and plant procedures.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-16
SIL No. 462
Shroud Support Access Hole Cover Cracks

I. SCOPE

This augmented inspection program (AUG-16) defines the specific examination recommendations of General Electric Company (GE) Nuclear Services Information Letter (SIL) No. 462, as applicable to LGS Unit 1. This SIL addresses the intergranular stress corrosion cracking (IGSCC) of the shroud support access hole covers and the GE recommended actions regarding susceptibility/routine examination.

II. REFERENCES

GE SIL No. 462, Revision 1, Access Hole Cover Cracking, March 22, 2001.

III. GENERAL

This Revision 1 to SIL No. 462 supersedes and voids SIL No. 462, SIL No. 462 Supplement 1, SIL No. 462 Supplement 2 Revision 1 and SIL No. 462 Supplement 3.

AHCs are present in GE BWR/3-/6 plants. Current AHC designs can be grouped into four categories. These categories of AHC design are listed, generally, from the most to the least susceptible configuration for stress corrosion cracking (SCC).

1. Thin Alloy 600 AHC plates connected to the SSP with a creviced Alloy 82/182 weld.
2. Full thickness Alloy 600 AHC plates connected to the SSP with a creviced Alloy 82/182 weld.
3. AHC arrangements where one Alloy 600 AHC plate is seal welded (no crevice) top and bottom with Alloy 182 or 82 weld metal and the other AHC is a "top-hat" configuration designed to eliminate the nickel alloy crevices.
4. AHC repair assemblies that replace the welded AHC with a multi-bolted plate.

GE Nuclear Energy recommends that owners of GE BWR/3-/6 plants consider the following:

A. Inspection Methods

Use inspection methods that are capable of detecting both circumferential and radial crack indications in the AHC plate/assembly, connecting weld and adjacent Shroud Support Plate (SSP).

B. Inspection Schedule

The inspection schedule is a function of the AHC design, plant water chemistry, inspection experience and the previous inspection method(s). The recommendations in this SIL are applicable to BWRs that maintain their water chemistry in a manner consistent with BWR industry guidelines.

1. The inspection schedule for a plant with an installed AHC bolted repair should be in accordance with the AHC repair hardware inspection schedule that was specified at the time of the repair.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-16
SIL No. 462
Shroud Support Access Hole Cover Cracks, continued

2. The inspection schedule for a plant with AHC designs as described in categories 1, 2, and 3 under Discussion should be as follows:
   - For a NWC plant where the previous inspection was top surface VT -1 only and no crack indications were found, subsequent inspections, either top surface VT-1 or UT, should be performed during a refueling outage within 4 years of the previous inspection.
   - For a NWC plant where the previous inspection was UT and no crack indications were found, subsequent inspections, either top surface VT -1 or UT, should be performed during a refueling outage within 6 years of the previous inspection.
   - For a plant with an effective program of HWC or NobleChem©/HWC, a baseline UT inspection should be conducted according to the recommendation for subsequent inspections as noted above (dependent on the previous inspection method). Once the baseline has been established and no crack indications are found, future top surface VT-1 inspections should be conducted every 8 years and future UT inspections should be conducted every 12 years. Note that an effective HWC program is considered one where hydrogen injection availability is 80% or better at levels resulting in ECP<–230 mv-SHE on the upper and lower surfaces of the AHC. In those cases where a HWC plant does not strictly meet these criteria, an appropriate inspection schedule can be established by performing a plant specific susceptibility analysis.

Note that the inspection frequencies above are for the case where no crack indications are found. If indications are found, the inspection frequency will depend upon the structural analysis results if the AHC is left as-is. Recommended action 1 would apply to a subsequently repaired AHC.

C. Inspection Analysis

Bounding analyses should be performed before the inspection to establish allowable flaw sizes to facilitate rapid disposition of identified indications during the outage. If circumferential or radial indications are found that exceed the assumptions used in the bounding analyses an additional analysis should be performed to verify structural margin. Depending on the extent of indications found, particularly in the radial direction, repair contingencies should be considered.

The LGS Unit 1 design includes two (2) access holes in the shroud support plate (group 3, above). These were utilized for access to the lower plenum during construction and were subsequently closed by welded access hole covers. As reported in SIL No. 462, cracking in the access hole cover plate attachment weld has been detected in a BWR/4. The cracking occurred in the heat affected zone of the creviced Alloy 600 access hole cover plate and is attributed to crevice accelerated IGSCC.

The LGS Unit 1 design has eliminated the creviced Alloy 600 plate configuration on both access hole covers. However, unlike Unit 2 the Unit 1 design has not eliminated the crevice in the 180 high hat AHC at the between the Alloy 600 AHC and the 316L adapter ring.

AUGMENTED INSPECTION PROGRAMS

SIL No. 462
Shroud Support Access Hole Cover Cracks, continued

Although 316L in itself is considered an IGSCC resistant material, the creviced weld configuration adversely affects this design's overall susceptibility to IGSCC. As such, augmented examination is recommended.

IV. EXAMINATION REQUIREMENTS

The probability of IGSCC occurring in the LGS Unit 1 configuration is considered low and is not expected to occur early in plant operation. The AHC design is not conducive to performing UT due to the lack of centering lugs for the UT delivery system. Accordingly, LGS will continue to perform visual, VT-1, inspections from the top surface.

While GE recommends for BWR's with group 3 AHC's and NobleChem/HWC, augmented visual (VT-1) examinations every eight (8) years following a baseline UT, the LGS Unit 1 design permits a ten (10) year reinspection frequency without baseline UT. As an alternate LGS may perform a visual VT-1 from the top surface every six (6) years in conjunction with the visual EVT-1 of the shroud support welds H8 and H9 per Augmented ISI Program No. AUG-25.

Review of plant water chemistry and/or the incidence of IGSCC in other RPV internal components may necessitate a revised frequency of this examination.

This is an Owner's Augmented Inspection Program. The recommended inspections shall be performed at PECO Energy's discretion as access permits.

V. EXAMINATION RESULTS

Examination results generated from this augmented inspection program shall be recorded and evaluated in accordance with applicable plant procedures.

VI. REPORTS/RECORDS

All reports/records associated with the examinations of this augmented program shall be prepared/maintained in accordance with ASME Section XI and plant procedures.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-17
SIL No. 474
Steam Dryer Drain Channel Cracking

I. SCOPE

This augmented inspection program (AUG-17) defines the specific examination recommendations of General Electric Company (GE) Nuclear Services Information Letter (SIL) No. 474, as applicable to LGS Unit 1. This SIL reports the occurrence of cracking in the drain channel to steam dryer skirt attachment welds and the related GE examination recommendations.

II. REFERENCES

A. GE SIL No. 474, Steam Dryer Drain Channel Cracking, October 26, 1988.

B. PECO Energy memorandum dated 3/7/90, T. A. Shea to distribution, RE: Clarification of the ISI Coordination Group's Position on RPV Inservice Inspection Issues.

III. GENERAL

The LGS Unit 1 steam dryer is not a nuclear safety related component; its function is to improve the quality of the steam before it leaves the reactor vessel. The steam dryer drain channels channel water runoff from the dryer back into the reactor pressure vessel. Cracking has been discovered at several BWR/4, 5 and 6 plants in the drain channel to dryer skirt attachment welds, both the horizontal and vertical welds. GE analysis indicates that crack initiation was due to high cycle fatigue.

The subject cracking is not a safety concern; however, if extreme cracking would occur, steam quality could become severely degraded and could potentially damage balance of plant components. Failed drain channels could result in loose parts and potentially damage RPV internal components. As such, augmented examination is recommended to ensure steam dryer reliability.

IV. EXAMINATION REQUIREMENTS

Visual (VT-1) examination of the LGS Unit 1 steam dryer drain channel attachment welds is recommended every refueling outage.

Note that the steam dryer has been included for routine visual (VT-3) examination in accordance with Augmented Inspection Program No. AUG-10. Examinations, as required by this Augmented Inspection Program No. AUG-17, represent an increase in both examination sensitivity and frequency for select areas of the steam dryer.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-17
SIL No. 474
Steam Dryer Drain Channel Cracking, continued

V. EXAMINATION RESULTS

Examination results generated from this augmented inspection program shall be recorded and evaluated in accordance with applicable plant procedures.

Any cracking detected shall be evaluated for repair to preclude any further crack growth.

VI. REPORTS/RECORDS

All reports/records associated with the examinations of this augmented program shall be prepared/maintained in accordance with ASME Section XI and plant procedures.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-18
Revision 1
RHR Heat Exchanger Pressure Retaining Bolting

The requirements of this Augmented Inspection Program have been completed.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-19
Weld Centerline Markings

I. SCOPE

This augmented inspection program (AUG-19) defines the actions committed to by PECO Energy in response to an NRC open item regarding LGS Unit 1 weld centerline marking.

Augmented Inspection Program No. AUG-19 applies to full penetration butt welds within the scope of this ISI Program of which ultrasonic examination (UT) is the specified method of examination. That is, this program applies to Class 1 and 2 welds selected for ISI UT examination during the inspection interval. In addition, any full penetration butt weld, regardless of Class, requiring ultrasonic examination by any augmented inspection program (e.g. AUG-01), or welds requiring ultrasonic examination as a result of additional samples taken following unacceptable ISI examination, shall be subject to this AUG-19 program.

II. REFERENCES

CM-2 PECO Energy letter J. S. Kemper to R. W. Starostecki (USNRC) dated August 30, 1984, Open Items Report for PECO Energy Limerick Generating Station, Unit 1. (T02666)

III GENERAL

During the preservice inspection (PSI) activities on LGS Unit 1, USNRC Region I identified an open item regarding the lack of weld centerline marking on some welds subject to PSI ultrasonic examination (UT). In response to this item, PECO Energy committed to implement a program during the first inservice inspection interval to remedy this situation. Augmented Inspection Program No. AUG-19 is established to provide adequate weld centerline marking of welds previously not marked during PSI. Note that many welds have already been marked. Therefore, the purpose of this program is to verify whether adequate markings exist and to provide weld centerline marking in those cases where markings are lacking.

IV. AUG-19 PROGRAM

Prior to performing ultrasonic examination of welds within the scope of this program, the weld centerline marking shall be checked and reworked, as needed, in accordance with the following requirements:

The weld crown shall be measured and the dimensions recorded. Reference marks shall be permanently stamped or vibro-etched on each side of the weld in the base metal of the component outside the examination area. The marks shall be placed in adjacent pairs, nominally three inches from the weld centerline in four locations around the circumference, 90 degrees apart, and shall be completed before weld preparation. Where three inches of access is not available on the base metal because of geometric limitations, the location shall be at the nearest practical distance provided that the marks are placed at equal distances from the weld centerline.

Weld marking shall be performed in accordance with Specification P-305, General Welding Requirements for Limerick Generating Station Units 1 and 2, Appendix L.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-19
Weld Centerline Markings

V. EXAMINATION RESULTS

Not applicable.

VI. REPORTS/RECORDS

Not applicable.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-20
BWRVIP-76
BWR CORE SHROUD

I. SCOPE

This augmented inspection program (AUG-20) describes the inspection program for the core shroud welds in LGS Unit 1. This program is based on the guidance developed by the BWRVIP in response to shroud cracking in multiple plants. Plant-specific evaluations provide additional bases for the inspection program.

II. REFERENCES

A. GE SIL No. 572, Core Shroud Cracks, Revision 1, dated October 4, 1993.

B. USNRC Information Notice 93-79, Core Shroud Cracking at Beltline Region Welds in Boiling Water Reactors.

C. GE Nuclear Energy Report No. GE-NE-523-148-1193, BWR Core Shroud Evaluation, dated April 1994.

D. USNRC Generic Letter 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors, dated July 25, 1994.

E. CM-8 USNRC SER for Docket No. 50-352, Response to Generic Letter 94-03 Limerick Generating Station, Unit 1 (TAC No. M90099), dated March 7, 1995. (T03848)

F. BWRVIP-01, BWR Core Shroud Inspection and Flaw Evaluation Guideline, Revision 2 , EPRI Report TR-107079, dated October 1996.

G. BWRVIP-07, Guidelines for Reinspection of BWR Core Shrouds, EPRI Report TR-105747, dated February 1996.

H. BWRVIP-63, Shroud Vertical Weld Inspection and Evaluation Guidelines, EPRI Report TR-113170, dated June 1999.

I. BWRVIP-76, BWR Core Shroud Inspection and Flaw Evaluation Guideline, EPRI Report TR-114232, dated November 1999.

III. GENERAL

Shroud cracking detected during 1992 and 1993 in various BWR's led to industry and USNRC action. The BWR industry formed the BWR Vessel and Internals Project (BWRVIP) to focus industry resources. A series of assessments and evaluations was performed by the industry and evaluated by the USNRC. Finally, BWRVIP-01 was issued, which was then approved by the USNRC. It addressed baseline inspections of shrouds based on a variety of factors, including age, water chemistry, and materials of construction. During this same time, NRC issued GL 94-03, requiring BWR licensees to provide inspection information and assurance that shroud integrity was maintained.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-20
BWRVIP-76
BWR CORE SHROUD, continued

The LGS Unit 1 Core Shroud assembly was fabricated by Sun Ship Company. Figure AUG-20-1 illustrates the general configuration of the shroud, as well as weld locations within the shroud assembly.

PECO Energy has reviewed the materials, fabrication and operational histories of the LGS Unit 1 core shroud and has submitted this information to the USNRC in our response to GL 94-03. The LGS Unit 1 plant-specific susceptibility factors are summarized below:

A. The dryer/separator support, top guide support, core support plate rings, are fabricated from welded 304L stainless steel, forged ring segments, with carbon contents of ≤ 0.026%. The upper, middle and lower shroud shells are fabricated from welded, rolled 304L stainless steel plates, with carbon contents ≤ 0.024%. The shroud support cylinder is fabricated from Alloy 600 (Inconel 600).

B. Welding of the shroud plates and rings for circumferential welds H1 – H6 was accomplished by submerged arc welding using ER308L filler metal. Welding of the bi-metallic weld, H7, was accomplished by gas metal arc welding using filler metal 82. Weld residual stress levels resulting from these fabrication processes are high.

C. LGS Unit 1 operated with low reactor coolant ionic content levels during the initial years of operation. The initial five year average coolant conductivity for LGS Unit 1 was 0.150 μS/cm, which is considerably lower than the average for other U.S. BWRs (where the conductivities range from ~0.123 μS/cm to 0.717 μS/cm, and average ~0.340 μS/cm).

D. As of August 25, 1994, LGS Unit 1 has operated for 6.4 cumulative years at full power, which is below the median for U.S. BWRs (the range is 3.7 years - 17.8 years, with a median of 10.8 years).

A safety assessment of the LGS Unit 1 core shroud has been performed in accordance with Reference C. The LGS Unit 1 core shroud materials, fabrication and operational factors place it in susceptibility Category "B" per BWRVIP-01, Reference F.

The baseline augmented examinations of the LGS Unit 1 core shroud were performed during the first refueling outage after achieving 8 EFPY of operations (1R06). The scope of the baseline examinations was in accordance with BWRVIP-01, Reference F, for susceptibility Category B plants except that the L_{min} examination method was not used for this baseline. The baseline examination attempted to examine 100% of the accessible length of Category B required welds. $L_{min(s)}$ was calculated prior to the examination as an aid in planning the examination and for the purpose of developing flaw acceptance criteria.

The BWRVIP issued BWRVIP-07, which specified reinspection frequencies for shrouds based on the amount of cracking detected during the baseline inspection. This was submitted for NRC review and has been the source of much debate between industry and NRC. NRC, in a letter issued April 27, 1998, suggested that BWRVIP add additional conservatism to its reinspection frequencies contained in BWRVIP-07.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-20
BWRVIP-76
BWR CORE SHROUD, continued

As a result of NRC comments, BWRVIP issued BWRVIP-76, which incorporates and supersedes BWRVIP-01, BWRVIP-07 and BWRVIP-63. BWRVIP-76 defines categories of core shrouds and identifies inspections, inspection intervals, generic acceptance standards, and evaluation procedures for horizontal and vertical welds of repaired and non-repaired core shrouds of all categories.

BWRVIP-76 makes several changes to earlier BWRVIP documents. These changes incorporate generic approaches and provide a unified and NRC-accepted approach for ensuring the integrity of core shrouds. These changes include the following.

A. Increasing the inspection sample to 100% of accessible regions.
B. Eliminating the distinction between baseline and subsequent inspection requirements.
C. Providing procedures to determine inspection intervals.
D. Condensing and simplifying the inspection strategies and evaluation procedures for horizontal and vertical welds in repaired and non-repaired core shrouds.

Reexamination scope and frequency for the LGS Unit 1 core shroud shall be in accordance with BWRVIP-76.

IV. EXAMINATION PROGRAM

LGS Unit 1 shroud welds shall be inspected consistent with BWRVIP-76. A baseline inspection of the LGS Unit 1 Core Shroud was performed during 1R06 in accordance with BWRVIP-07 requirements for a Category B Plant. Volumetric examinations of weld H3, H4, H5 and H7 were performed on 100% of the accessible length. Repairs were not required and the shroud remains classified as Category B. BWRVIP-76 does not require the inspection of vertical core shroud welds of Category B shrouds.

In performing the inspection of the horizontal core shroud welds, the preferred inspection method is UT volumetric inspection or a two-sided surface inspection as identified and approved in BWRVIP-03, or both. BWRVIP-76 contains generic acceptance standards in Figure 2-2. In the event that these acceptance standards cannot be met, a weld- or plant-specific evaluation procedure may be used. The bases for the generic acceptance standards and inspection intervals are presented in BWRVIP-76, Appendix C, and the plant-specific evaluation procedure in presented in Appendix D.

100% of the accessible regions of welds H3, H4, H5, and H7 are to be inspected. If more than 50% of the length of a weld is inspected, and the observed cracking is less than 10% of the inspected length, then the weld is acceptable for continued service. BWRVIP-76 defines the inspection interval as a function of the amount of cracking observed and the stress values for core shroud faulted loading conditions. If less than 50% of the length of a weld is inspected, and the observed cracking is less than 10% of the inspected length, then a plant-specific evaluation must be performed per Appendix D. If the observed cracking of any weld is greater than 10%, then the core shroud is re-classified as Category C, and the inspection requirements of that category, as identified in BWRVIP -76, apply.

BWRVIP-76, Table 2-1, determines examination frequency, whether six or ten years, or as determined by plant-specific analysis, as a function of the percentage of observed cracking and of the stress value of the faulted loading condition.

PROGRAM No. AUG-20
BWRVIP-76
BWR CORE SHROUD, continued

V. EXAMINATION RESULTS

The core shroud is a core support component. Therefore, it is to be inspected and evaluated in accordance with Section XI. The BWRVIP program has replaced Code-required VT-3 visual examination with more stringent examination requirements that meet and exceed Code requirements. All shroud inspection results shall be recorded and evaluated as are Code-required examinations. Additional evaluation criteria developed by the BWRVIP using Code margins is documented in BWRVIP-01 and shall be considered in flaw evaluation.

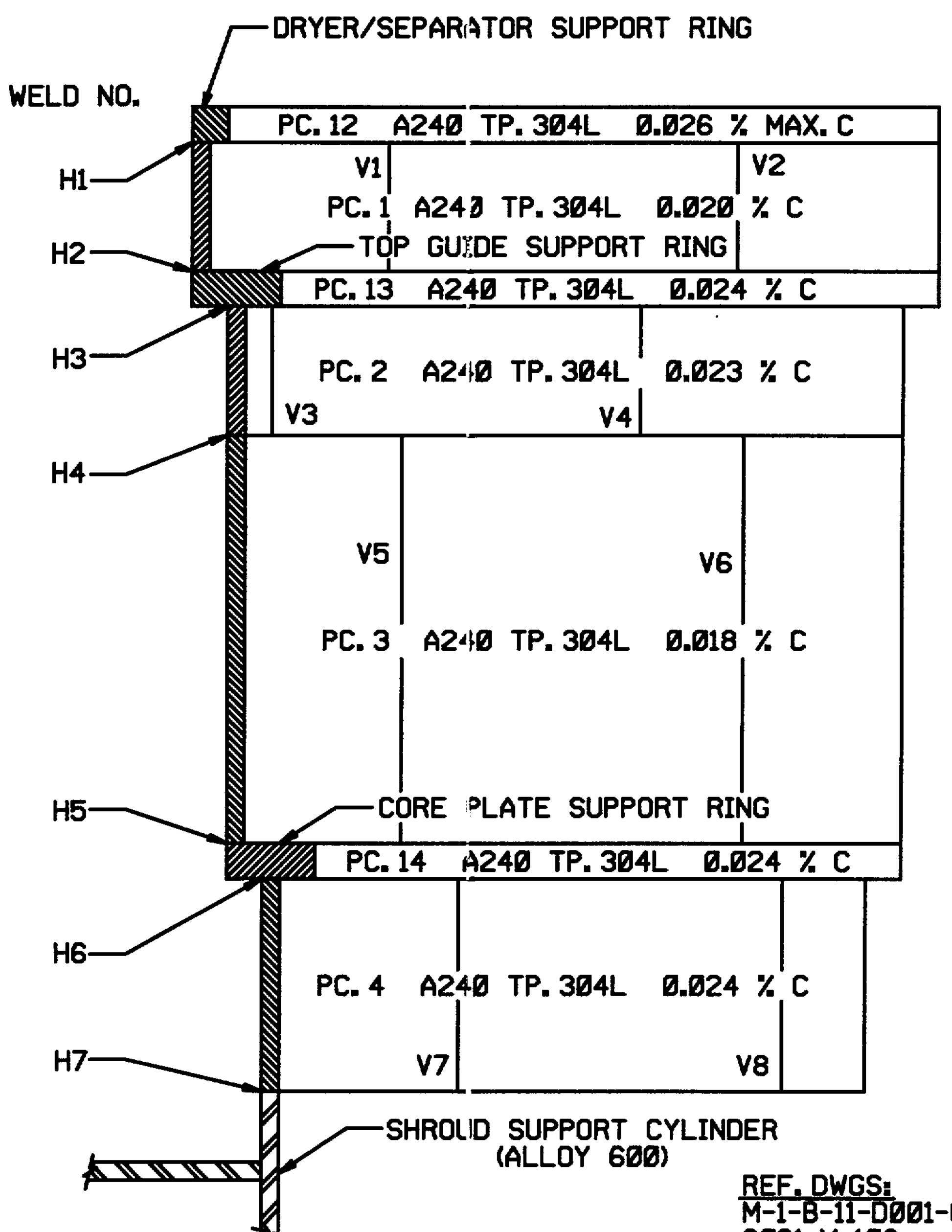
VI. REPORTS/RECORDS

All reports and records shall be prepared and maintained in accordance with the 1989 Edition of Section XI, Specification NE-042, and plant procedures. LGS and BWRVIP have committed to supply inspection, evaluation and repair results to NRC and to EPRI. These results will be forwarded to NRC as part the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-20
BWRVIP-76
BWR CORE SHROUD, continued
FIGURE AUG-20-1

REACTOR PRESSURE VESSEL - SHROUD
LIMERICK GENERATING STATION
UNIT 1



REF. DWGS:
M-1-B-11-D001-C-24-2
8031-M-108

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-21
BWRVIP-26
BWR Top Guide

I. SCOPE

This augmented inspection program (AUG-21) applies to the top guide in LGS Unit 1. The top guide is designed to provide lateral support for the fuel assemblies; thus, it is part of the core support structure. It is designated as a safety-related component and classified in accordance with ASME Section XI. This program is based on BWRVIP-26 recommendations for the BWR/4,5 with Aligner Pin Assemblies plus Wedges configuration and provides appropriate inspection requirements to assure safety function integrity of the top guide. Thus it may be used to satisfy ASME Section XI requirements.

II. REFERENCES

A. GE SIL No. 554, Top Guide Cracking, April 6, 1993.

B. GE RICSIL No. 059, Top Guide Crack Indications, May 31, 1991.

C GE SIL No. 588, Revision 1, Top Guide and Core Plate Cracking, May 18, 1995.

D. BWRVIP-06, Safety Assessment of BWR Reactor Internals, October 1995.

E. BWRVIP-26, BWR Top Guide Inspection and Flaw Evaluation Guidelines, EPRI Report TR-107285, dated December 1996.

III. GENERAL

As reported in RICSIL No. 059, cracking was identified in the top guide of a GE BWR/2 reactor. The cracking was found at the bottom of an non-notched area of the 304 SS top guide egg crate structure. Because of its proximity to fuel assemblies, the lower portion of the top guide beams is exposed to significant amounts of irradiation. This increases susceptibility to irradiation assisted stress corrosion cracking (IASCC).

Top guide cracking in the LGS Unit 1 configuration, should it occur, is not expected to occur early in plant operation. In accordance with the recommendations of SIL-554, visual examination (VT-1) was performed starting with the plant's eighth year of operation or LGS Unit 1's Fifth Refueling Outage (1R05). VT-1 visual examination was performed on a sample of grid locations where fuel and blade guides were removed. No cracking was detected on the top guide. In accordance with the recommendations of the SIL, had cracking been detected, then additional ultrasonic examination would be performed of top guide beam intersections in locations of the highest fluence.

Subsequent to the publication of SIL-554, however, BWRVIP-06 made the determination that no safety consequences would result from the failure of the top guide beams. Even multiple failures would not prevent control rod insertion. BWRVIP-06 did not recommend examination of the top guide.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-21
BWRVIP-26
BWR Top Guide, continued

The BWRVIP-26 guideline recommends inspection of three locations applicable to LGS Unit 1. Two locations are the aligner pins and sockets (locations 2 and 3 in BWRVIP-26) in the top guide and shroud. However, these examinations are not necessary for plants that have wedges installed that can carry the lateral load in the event the aligner pins fail. LGS has such wedges. Thus, there are no required inspections for the LGS Unit 1 aligner pins and sockets.

The third location is the hold-down assemblies, i.e., the C-clamps (location 9). Examination of the C-clamps is required only for those plants whose vertical loads exceed the top guide weight. This calculation has not been performed at LGS. Therefore, as a conservative measure, LGS will perform the visual examinations recommended by BWRVIP-26.

IV. EXAMINATION PROGRAM

BWRVIP-26, for locations 2 and 3, requires no examination of the aligner assemblies, as the aligner hardware is redundant to the wedges between the top guide and the shroud.

BWRVIP-26, for location 9, requires a VT-3 visual examination of each C-clamp assembly. There is no concern for failure of the C-clamps. However, the welds that attach the clamps to the top guide are creviced, and could crack, possibly resulting in loss of C-clamp function. There could be safety consequences for plants where the top guide could lift under accident conditions.

V. EXAMINATION RESULTS

Examinations results shall be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations shall be performed using IWB-3000 of the 1989 Edition of Section XI and the guidance set forth in Section 4 of BWRVIP-26.

VI. REPORTS/RECORDS

LGS and BWRVIP have committed to supply inspection, evaluation and repair results to the USNRC and to EPRI. Therefore, if examinations are performed on the top guide, all reports and records shall be prepared and maintained per the 1989 Edition of ASME Section XI, Specification NE-042, and plant procedures. These results will be forwarded to NRC as an attachment to the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-22
USNRC IE Bulletin Nos. 95-02 and 96-03
RHR and CS Suction Strainers

I. SCOPE

This augmented inspection program (AUG-22) defines the recommended inspection requirements for compliance with the commitments made in response to USNRC IE Bulletin 95-02 and IE Bulletin 96-03, applicable to LGS Unit 1 Residual Heat Removal and Core Spray System suction strainers.

II. REFERENCES

A. CM-9 USNRC IE Bulletin No. 95-02, Unexpected Clogging of a Residual Heat Removal (RHR) Pump Suction Strainer While Operating in Suppression Pool Cooling Mode. (T03645)

B. USNRC IE Bulletin No. 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors.

C. PECO ENERGY letter of Nov. 1, 1996, G. A. Hunger to U.S. Nuclear Regulatory Commission.

D. PECO ENERGY letter of Apr. 3, 1997, G. A. Hunger to U.S. Nuclear Regulatory Commission.

E. PECO ENERGY letter of Oct. 6, 1997, G. A. Hunger to U.S. Nuclear Regulatory Commission.

III. GENERAL

PECO ENERGY committed to include the Residual Heat Removal and Core Spray system suction strainers in the ISI program in response to IE Bulletin 96-03. The concern of IE Bulletin 96-03 and PECO Energy's response was to ensure that the Residual Heat Removal and Core Spray systems remain operable and ready for use in case of a large break DBA. The suction strainers on these systems are located in the suppression pool and are normally inaccessible. They were replaced with strainers sized to accept the maximum credible amount of fibrous insulation which could be damaged and blown/washed into the suppression pool without compromising the NPSHA of the system pumps. Periodic inspection of the strainers will ensure that the strainers are neither physically degraded nor partially clogged with debris from normal plant operation.

In addition, in response to I.E. Bulletin 95-02, PECO Energy committed to a program for monitoring sludge accumulation on the suppression pool floor. Examinations of the suppression pool floor will be conducted during the two refueling cycles associated with LGS replacement strainer installations, after which time the sludge accumulation rate will be determined. If the amount of sludge material discovered during these inspections does not exceed the design basis assumptions for the strainers (i.e. 100 lbs/year), the inspection interval will be increased to every other refueling outage. Any changes in the inspection frequency will be appropriately evaluated to ensure there is no adverse impact on plant operations.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-22
USNRC IE Bulletin Nos. 95-02 and 96-03
RHR and CS Suction Strainers, continued

IV. EXAMINATION PROGRAM

The specific details of the inspection program to be implemented are not provided in the letter to the USNRC. However, the intent of the commitment was to provide assurance that the strainers are structurally sound, and not partially clogged or physically degrading. Visual inspections are adequate to provide this assurance. The condition of each individual strainer is representative of the others, since they all experience the same demineralized water environment and are made of the same corrosion resistant material. Indications of degradation or clogging shall be evaluated by Engineering. Evidence of clogging would be coverage of more than 1% of the strainer open area by anything more than a light coating of loose material. All (100%) of the strainer assemblies shall be inspected during each interval for general structural condition, and one strainer module (screen) in each system inspected for debris during every other refueling outage.

V. EXAMINATION RESULTS

Examination results generated from this augmented inspection program shall be recorded and evaluated in accordance with applicable plant procedures and the 1980W81 Section XI requirements

VI. REPORTS/RECORDS

All reports/records associated with the examinations of this augmented program shall be prepared and maintained in accordance with ASME Section XI and plant procedures.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-23
BWRVIP-25
BWR Core Plate

I. SCOPE

This augmented inspection program (AUG-23) applies to the core plate in LGS Unit 1. The core plate is designed to provide lateral support for the fuel assemblies, control rod guide tubes, and in-core instrumentation. It also provides vertical support for the peripheral fuel assemblies; thus, it is part of the core support structure. It is designated as a safety-related component and classified in accordance with ASME Section XI. This program is based on BWRVIP-25 recommendations for the BWR/4 Type I without wedges configuration and provides appropriate inspection requirements to assure safety function integrity of the core plate. Thus it may be used to satisfy ASME Section XI requirements.

II. REFERENCES

A. GE SIL 558, Rev. 1, Top Guide and Core Plate Cracking, dated May 18, 1995.

B. BWRVIP-25, BWR Core Plate Inspection and Flaw Evaluation Guidelines, EPRI Report TR- dated December 1996.

III. GENERAL

The BWRVIP-25 guideline recommends examination of one location applicable to LGS Unit 1. This is of the core plate rim hold-down bolts. There are 34 rim hold-down bolts in each unit at LGS.

Examination of the rim hold-down bolts is not necessary for plants with wedges that can carry the lateral load in the event the bolts fail. LGS does not have such wedges. However, should wedges be installed, either alone or as part of a shroud repair, if ever implemented, this bolt examination could be eliminated.

IV. EXAMINATION PROGRAM

There are two examination options in BWRVIP-25. The first option is a UT volumetric examination of the rim hold-down bolts performed from the top of the bolts. The second option is to perform an EVT-1 visual examination, as defined in BWRVIP-25, from below the core plate. The visual examination would involve dismantling RPV internals in some manner to gain access. For both techniques, the initial sample size is 50% (17) rim hold-down bolts. Should cracking be detected, the sample size should be increased to 100%.

The BWRVIP-25 document does not have specific reinspection criteria nor a requirement to examine the second 50%, except when cracking is detected in the initial baseline. Rather, it states: "..a reinspection schedule should be developed, based on plant-specific analyses that consider plant geometry, number of bolts, loading conditions and inspection experience. Note that good inspection results combined with good operating experience of BWR bolts and the degree of redundancy of the hold-down bolts may justify elimination of any reinspection."

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-23
BWRVIP-25
BWR Core Plate, continued

It should be noted that the preferred inspection method at LGS is the UT. However, the UT technique has not yet been developed and qualified by the BWRVIP. Further, as shown at ISI Figure BN-7-3, a keeper is installed over each nut that precludes access for a UT examination. Therefore, UT examination of the hold-down bolts is not a viable option at LGS.

The alternative examination method is visual examination. However, the risks associated with dismantling RPV components to perform such a visual examination of the bolts from below the core plate, as recommended by BWRVIP-25, are not warranted. This is especially true since there is no evidence to date of a problem within the BWR industry. Therefore, visual examination of the bolting from below the core plate is not planned.

If examinations are conducted, then they will be performed using the EVT-1 examination method on the accessible portion of the top of the bolts. These examinations are at LGS' discretion.

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations should be performed using the guidance set forth in Section 4 of BWRVIP-25, and IWB-3000 of the 1989 Edition of Section XI.

VI. REPORTS/RECORDS

LGS and BWRVIP have committed to supply inspection, evaluation, and repair results to the USNRC and to EPRI. Therefore, all reports and records shall be prepared and maintained per the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These results will be forwarded to NRC as an attachment to the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-24
BWRVIP-27
BWR Standby Liquid Control System

I. SCOPE

This augmented inspection program (AUG-24) applies to the N10 penetration in the LGS Unit 1 reactor vessel bottom head.

II. REFERENCES

A. BWRVIP-27, BWR Standby Liquid Control System Core Plate ΔP Inspection and Flaw Evaluation Guidelines, EPRI Report TR-107286, dated April 1997.

B. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, no Addenda.

III. GENERAL

The successful function of the standby liquid control (SLC) system is important to achieve reactor shutdown. The SLC system would function adequately when initiated, so long as boron is injected into the RPV. The focus of BWRVIP-27 is on the region where the ΔP/SLC nozzle penetrates the bottom head. However, at LGS, boron is not injected through the ΔP/SLC nozzle, N10, but through the two core spray nozzles, N5A and N5B. Thus, the recommendations of BWRVIP-27 are not applicable.

IV. EXAMINATION PROGRAM

The guidelines for core spray internal piping and spargers, Augmented Inspection Program No. AUG-03 (BWRVIP-18) shall be followed.

V. EXAMINATION RESULTS

The guidelines for core spray internal piping and spargers, Augmented Inspection Program No. AUG-03 (BWRVIP-18) shall be followed.

VI. REPORTS/RECORDS

The guidelines for core spray internal piping and spargers, Augmented Inspection Program No. AUG-03 (BWRVIP-18) shall be followed.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-25
BWRVIP-38
Shroud Support Structure

I. SCOPE

This augmented inspection program (AUG-25) applies to the shroud support structure in LGS Unit 1. The shroud support structure consists of the shroud support plate, shroud support cylinder and 14 support legs. The support plate and legs are also integrally welded to the RPV. The shroud support structure is designed to provide lateral support for the core. It also provides vertical support for the peripheral fuel assemblies; thus, it is part of the core support structure. It is designated as a safety-related component and classified in accordance with ASME Section XI. This program is based on BWRVIP-38. It provides appropriate inspection requirements to assure safety function integrity of the shroud support structure. It may be used to satisfy ASME Section XI requirements.

II. REFERENCES

A. BWRVIP-38, BWR Shroud Support Inspection and Flaw Evaluation Guideline, EPRI Report TR-108823, dated September 1997.

III. GENERAL

The shroud support plate is inspected per ASME Section XI, Examination Category B-N-2, Item No. B13.40. ASME Section XI requires a VT-3 visual examination of accessible surfaces once every 10 years. The requirements of BWRVIP-38 also specify an inspection of the shroud support plate to shroud support cylinder weld, H8, and the shroud support plate to reactor pressure vessel pad weld, H9, using either UT volumetric examination from either the RPV outside surface or the annulus, EVT-1 visual examination from the annulus, or ET eddy current examination from the annulus. Operating loads and flaw tolerances, as described in Section 5 of BWRVIP-38 dictate the amount of inspection. Examinations conducted prior to the issuance of the BWRVIP guideline can be credited toward the baseline, if they meet the criteria set forth in BWRVIP-38.

IV. EXAMINATION PROGRAM

Using Table 5-1 of BWRVIP-38, the load multiplier for LGS Unit 1 is 1.40. Using this value, and Figure 5-1 for weld H8 and Figure 5-2 for weld H9, the amount of inspection required for each Unit 1 weld is 10% of the circumferential length. The inspection method to be employed is EVT-1. The reinspection frequency is 6 years. While these examinations are intended to be in addition to the VT-3 of 100% of the accessible portions of the shroud support plate over the second ten-year interval a review of the accessibility of the shroud support structure shows that the EVT-1 examinations of welds H8 and H9 in the area between the access holes and the jet pumps at 0° and 180° RPV azimuth constitutes essentially 100 of the accessible surfaces. Accordingly, the BWRVIP-38 examinations, which use a more sensitive NDE method and are at an higher inspection frequency, may be used to satisfy ASME Section XI requirements.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-25
BWRVIP-38
Shroud Support Structure, continued

The BWRVIP-38 inspections are focused on the top portion of the shroud support structure. This is acceptable based on the conservative assumptions regarding cracking in the shroud support structure including the support legs in the lower plenum. Although the shroud support structure exhibits a very large flaw tolerance, EVT-1 will not be able to detect cracking on the far side of welds H8 and H9. Assuming a bounding crack growth rate of $5x10^{-5}$ inches/hour the reinspection frequency has been set to a maximum of six years for EVT-1. In the event that flaws are detected the sample expansion criteria of BWRVIP-38 shall be used. The extent to which inspections are to be performed in the lower plenum for the far side of welds H8, H9 and the support legs depends on the extent of flaws detected in the near side of welds H8 and H9 and any plant specific analysis required to justify continued plant operation.

Flaw evaluations of H8 and H9 should be performed using guidance from Section 5 and Appendix A of BWRVIP-38. Additionally, H9 flaw evaluations must satisfy IWB-3000 of the 1989 Edition of Section XI for at least the portion of the weld that is part of the Class 1 boundary.

BWRVIP-38 specifies that, if flaws are detected, an effective flaw length is to be determined and compared to the flaw tolerance value (i.e. 1%) for the weld in question, using Section 5. If flaw tolerances are satisfied, then the inspection is complete. If flaw tolerance is not satisfied, then the examination scope expansion should continue until the flaw tolerance criteria are satisfied. If flaw tolerance criteria are not satisfied, then a plant-specific evaluation must be performed. Section XI, IWB-3000, must be used in the flaw evaluation until NRC has approved use of BWRVIP-38 for use in lieu of the Code.

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated as Code-required examinations. H9 is a weld to the Class 1 pressure boundary and H8 is an integral weld of the welded core support structure.

VI. REPORTS/RECORDS

All reports and records shall be prepared and maintained per the 1989 Edition of Section XI, Specification NE-042, and plant procedures. LGS and the BWRVIP have committed to supply inspection, evaluation, and repair results to the USNRC and to EPRI. These results will be forwarded to the USNRC as part the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-26
BWRVIP-42
Low Pressure Coolant Injection Coupling

I. SCOPE

This augmented inspection program (AUG-26) applies to the low pressure coolant injection (LPCI) couplings in LGS Unit 1.

II. REFERENCES

A. BWRVIP-42, LPCI Coupling Inspection and Flaw Evaluation Guidelines, EPRI Report TR-108726, dated December 1997.

B. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, no Addenda.

III. GENERAL

BWRVIP-42 recommends that a visual examination of critical locations of each LPCI coupling be performed to address the possibility of cracking over the long term. These locations are classified as high priority, low priority or no inspection locations.

IV. EXAMINATION PROGRAM

BWRVIP-42, Table 3-1, identifies the welds and items to be examined, the safety/inspection priority and the examination frequencies. The examination method is determined based on the safety/inspection priority. At least one third of all high priority locations shall be examined within four (4) years (2R) of the baseline inspection with 100% of the high priority locations inspected within 12 years (6R). 100% of the low priority locations shall be inspected within 12 years (6R) of the baseline inspection

If no flaw indications are detected during the re-inspection then the re-inspection is complete. If flaws are detected during the re-inspection then scope expansion and/or flaw evaluations shall be performed in accordance with BWRVIP-42 requirements.

V. EXAMINATION RESULTS

Examination results generated from this augmented inspection program shall be documented and evaluated in the same manner as Code-required examinations. Chapter 5 of BWRVIP-42 may be considered for use in evaluations to determine structural and pressure boundary integrity of observed flaws.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-26
BWRVIP-42
Low Pressure Coolant Injection Coupling, continued

VI. REPORTS/RECORDS

There are no Code-required reporting requirements for the visual examinations. However, LGS and the BWRVIP have committed to supply inspection, evaluation, and repair results to the USNRC and to EPRI. Therefore all reports and records shall be prepared and maintained per the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These results will be forwarded to the USNRC as an attachment to the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-27
BWRVIP-47
Lower Plenum

I. SCOPE

This augmented inspection program (AUG-27) applies to the lower plenum region in LGS Unit 1.

II. REFERENCES

A. BWRVIP-47, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines, EPRI Report TR-108727, dated December 1997.

III. GENERAL

The BWRVIP-47 guideline documents the evaluation of all the lower plenum safety-related components. The evaluation includes the control rod drive (CRD) housings and stub tubes, control rod guide tubes, orificed fuel supports, in-core housing, guide tube, and dry tubes assemblies. BWRVIP-47 recommends a sample inspection of the three welded locations on the control rod guide tubes and guide tube/fuel support alignment pin-to-core plate weld, and the pin, itself. All other locations are either adequately addressed with Code pressure tests or do not warrant inspection.

IV. EXAMINATION PROGRAM

A 10% sample of the CRD guide tube sleeve-to-alignment lug welds (CRGT-1) population is to be examined within 12 years (6R), with 5% (1/2 of the sample) to be examined within 6 years (3R). The examination technique is VT-3 visual examination. Alternatively, if the activity of reinstalling/realigning the orificed fuel support verifies the pin and lug integrity, then the inspection is not required. This may be the preferred option, since the examination of the other two (2) CRD guide tube locations will require removal and reinstallation of the fuel support and involves the same sample size.

The CRD guide tube body-to-sleeve weld (CRGT-2) and CRD guide tube base-to body weld (CRGT-3) also require a 10% sample of the guide tube population, with 5% (1/2 of the sample) to be examined within six years. The examination technique is EVT-1 visual examination.

The guide tube and fuel support alignment pin-to-core plate weld and the pin, itself (FS/GT-ARPIN-1), are to be examined using the same sample criteria. A 10% sample of the population is to be examined within 12 years (6R), with 5% (1/2 of the sample) to be examined within 6 years (3R). The examination technique is VT-3 visual examination. As is the case with CRGT-1, verification of the pin and lug integrity as part of the reinstallation of the orificed fuel support is an acceptable alternative to the VT-3 visual examination.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-27
BWRVIP-47
Lower Plenum, continued

The scope expansion criteria are the same for each inspection location. They are:

A. If one or more flaws are found, similar locations in an additional 5% of the total population of guide tubes or pins must be examined. These additional inspection locations must be from the immediately surrounding area of the flawed component and must be previously unexamined.

B. If flaws are found during the additional examinations defined in A above, then the expansion criteria given in A is repeated until no new flaws are found.

There are no reinspection criteria at this time. BWRVIP-47 states that "Baseline inspection results will be reviewed by the BWRVIP and, if deemed necessary, reinspection recommendations will be developed later."

V. EXAMINATION RESULTS

Examination results should be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations should be performed using acceptable methodology. The guidance set forth in IWB-3000 of the 1989 Edition of Section XI is acceptable.

VI. REPORTS/RECORDS

There are no Code reporting requirements. However, LGS and BWRVIP have committed to supply inspection, evaluation, and repair results to NRC and to EPRI. Therefore all reports and records shall be prepared and maintained per the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These results will be forwarded to NRC with the Code-required submittal of ISI data. The data will also be provided to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-28
BWRVIP-48
Vessel ID Attachment Weld

I. SCOPE

This augmented inspection program (AUG-28) applies to the reactor pressure vessel (RPV) internal (ID) attachment welds in LGS Unit 1.

II. REFERENCES

A. BWRVIP-48: BWR Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, EPRI Report TR-108724, dated February 1998.

III. GENERAL

The BWRVIP-48 guideline documents the evaluation of all the reactor vessel ID attachments. The result of the evaluation is that most attachments require no additional inspections beyond those required by Section XI. BWRVIP-48 does recommend more sensitive visual examinations of four (4) sets of attachments. These are the attachments for the jet pump riser braces, the core spray piping brackets, the feedwater brackets, and the steam dryer support brackets. The feedwater brackets and the steam dryer support brackets are subject to examination only if they are attached with a furnace-sensitized stainless steel, which is susceptible to IGSCC.

The more sensitive visual examination is a modified VT-1 (MVT-1). Because it is a more sensitive examination, performance of MVT-1 meets the requirements of Section XI and will be so applied at LGS, i.e., performance of the single MVT-1 visual examination will be applied to both Section XI and BWRVIP examination requirements. Ultrasonic inspections are recommended for all indications of suspected flaws to determine if the flaw has propagated into the RPV base metal.

IV. EXAMINATION PROGRAM

A. Jet Pump Riser Brace Brackets

BWRVIP-48 recommends that jet pump riser brace pad welds and heat-affected zones (HAZ) be examined within 12 years (6R), with 50% of the welds to be examined within the next 6 years (3R). This coincides with the inspection frequency of the jet pump riser brace weld to weld pad described in BWRVIP-41 and AUG-04. The examination method to be used is MVT-1 visual examination.

Performing the examinations over 12 years, instead of the Code ten-year interval, would require relief request approval by the NRC. Therefore, these examinations at LGS will be planned and scheduled in accordance with BWRVIP recommendations during the second ten-year interval in accordance with Section XI. If the NRC approves BWRVIP-48 and allows the 12 year interval as an alternative to Section XI, then this augmented inspection program may be revised at that time.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-28
BWRVIP-48
Vessel ID Attachment Weld, continued

Reinspection guidance for the jet pump riser braces weld pads is beyond the scope of this augmented inspection program. The Code presently requires reinspection each ten-year interval. BWRVIP recommends a 25% sample during each subsequent 6 year (3R) cycle following the initial 12 year term. Either way, the reinspection will not occur during the second ten-year interval, so will be addressed in the next update of the ISI program, i.e., for the third inspection interval.

B. Core Spray Piping Brackets

BWRVIP-48 requires a baseline of the primary and supplemental bracket weld pads and weld HAZ during the next outage, which is the first outage of implementing these criteria. The examination method specified is MVT-1 visual examination. As discussed above, this satisfies Section XI, regarding examination method.

The BWRVIP-48 reinspection frequency is 100% every four refueling cycles (4R). These reinspections may be scheduled with core spray examinations of AUG-03. Section XI requires reinspection once each ten-year interval and allows the examinations to be spread over time or to be deferred to the end of the interval. Therefore, meeting the BWRVIP-48 criteria will exceed the reinspection requirements of Section XI.

C. Feedwater Brackets

BWRVIP-48 specifies that the feedwater bracket weld pads and weld HAZ be examined at the frequency required by Section XI, i.e., once every 10 years, (5R). However, BWRVIP-48 requires that the examination be performed using the more sensitive MVT-1 visual examination in lieu of the VT-1 visual examination of Section XI. This is due to the potential for IGSCC.

Reinspection frequency is that of Section XI, i.e., once per 10 years.

D. Steam Dryer Support Brackets

BWRVIP-48 specifies that the steam dryer support bracket welds and HAZ be examined at the frequency required by Section XI, i.e., once per 10 years. However, BWRVIP-48 requires that the examination be performed using the more sensitive MVT-1 visual examination in lieu of the VT-1 visual examination of Section XI. This is due to the potential for IGSCC.

Reinspection frequency is that of Section XI, i.e., once per 10 years.

E. Scope Expansion Criteria

The scope expansion criteria of BWRVIP-48 requires the examination of all remaining locations of the same type, e.g., core spray bracket attachment welds and associated HAZ, during the same outage, unless the flaw can be correlated to a specific event that would not affect other locations. Use of this criteria will be evaluated on a case-by-case basis and, if employed, will require relief request approval by NRC. Otherwise, the standard scope expansion criteria of Section XI, IWB-2430, will be used.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-28
BWRVIP-48
Vessel ID Attachment Weld, continued

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations shall be performed using IWB-3000 of the 1989 Edition of Section XI. Alternative methodology may be used with NRC approval.

VI. REPORTS/RECORDS

Each of the locations described in this augmented inspection program is required to be examined in accordance with Section XI. The examination results will be recorded and maintained according to the rules of 1989 Edition of Section XI, Specification NE-042, and plant procedures. The results will be supplied to NRC as part of the Code-required ISI data submittal. Additionally, LGS and BWRVIP have committed to supply inspection, evaluation, and repair results to EPRI. Therefore, these results will be forwarded to the EPRI Project Manager for BWRVIP activities.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-29
BWRVIP-49
Instrument Penetrations

I. SCOPE

This augmented inspection program (AUG-29) applies to reactor pressure vessel (RPV) instrument penetrations in LGS Unit 1.

II. REFERENCES

A. BWRVIP-49: Instrument Penetrations Inspection and Flaw Evaluation Guidelines, EPRI Report TR-108695, dated March 1998.

B. ASME Boiler and Pressure Vesse Code, Section XI, 1989 Edition, no Addenda.

III. GENERAL

The BWRVIP-49 demonstrated that there are no safety concerns associated with degradation of RPV instrument penetrations. The BWRVIP determined that these penetrations, specifically, the HAZ associated with each penetration was susceptible to stress corrosion cracking (SSC). However, in evaluating the consequences of this potential cracking at an instrument penetration, the BWRVIP concluded that the failure of a single penetration would not affect the ability of the plant to achieve safe shutdown.

LGS is already required by Section XI to perform an examination of the RPV instrument penetrations of concern by means of a VT-2 visual examination. A primary pressure boundary leak test is performed during each refueling outage. Thus, the instrument penetrations are effectively inspected for leakage after each refueling cycle. Cracking of any instrument penetration is not detected until there is a leak that is need of correction. Section XI provides the means to detect, evaluate, and correct leaks. Therefore, the BWRVIP has determined that additional inspection of these instrument penetrations is not required.

IV. EXAMINATION PROGRAM

Visual examination of RPV instrument penetrations shall be performed in accordance with the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These examinations meet or exceed the requirements of BWRVIP-49. There are no additional examinations required by BWRVIP-49.

V. EXAMINATION RESULTS

Examination results shall be documented and evaluated in the same manner as Code-required examinations. Flaw evaluations shall be performed in accordance with IWB-3000 of the 1989 Edition of Section XI. These penetrations are within the Class 1 boundary.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-29
BWRVIP-49
Instrument Penetrations, continued

VI. REPORTS/RECORDS

There are no examinations that are required by the BWRVIP. However, there are Code-required reporting requirements that are associated with the Code-required VT-2 visual examination. Therefore, all reports and records shall be prepared and maintained per the 1989 Edition of Section XI, Specification NE-042, and plant procedures. These results will be forwarded to NRC as part of the Code-required submittal of ISI data.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-30
BWRVIP-05
Reactor Pressure Vessel Shell Weld

I. SCOPE

This augmented inspection program (AUG-30) addresses the specific steps taken by LGS Unit 1 to satisfy the NRC augmented examination requirements mandated by the Code of Federal Regulations, Title 10, Part 50, paragraph 50.55a(g)(6)(ii)(A), including alternatives agreed to by the USNRC and PECO Energy.

II. REFERENCES

A. BWRVIP-05: BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations, dated March 1998.

B. ASME Boiler and Pressure Vessel Code, Section XI, 1986 Edition, no Addenda.

C. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, no Addenda.

D. NRC Information Notice 97-63, Status of NRC Staff's Review of BWRVIP-05, Revision 0, dated August 7, 1997, and Revision 1, dated May 17, 1998.

E. Relief Request No. RR-01, Rev. 2, Table IWB-2500-1, Pressure-Retaining Welds in Reactor Vessel.

III. GENERAL

The Section XI ISI Program requirements for LGS Unit 1 for the first inspection interval were those of the 1986 Edition. The Code required examination of all circumferential and all longitudinal reactor pressure vessel (RPV) shell welds during the first inspection interval. A rule change to 10CFR50.55a regulation, in 1992, required examination of all RPV shell welds by the end of the inspection interval in effect at the time the regulation was promulgated. This rule became effective during the first ten-year ISI interval for LGS Unit 1. Accordingly, LGS Unit 1 was required to satisfy this rule by the end of the first inspection interval or to propose an alternative examination program for NRC approval. The rule required examination of "essentially 100%" of all vessel shell welds by the end of the first inspection interval.

Subsequent to the issuance of the regulation, technical bases provided by the BWRVIP, and NRC reviews of those bases, led the NRC to allow deferral of the circumferential weld inspections required by 10CFR50.55a(g)(6)(ii)(A). In September 1995, the BWRVIP submitted a set of recommendations for RPV shell weld examinations to the NRC. These recommendations, contained in report BWRVIP-05, eliminated the inspection of reactor vessel circumferential welds.

AUGMENTED INSPECTION PROGRAMS

PROGRAM No. AUG-30
BWRVIP-05
Reactor Pressure Vessel Shell Weld, continued

The NRC and the BWRVIP are currently engaged in the process of revising the regulatory requirement in consideration of the BWRVIP recommendations. As part of this process, the NRC performed an independent assessment of the technical bases provided by the BWRVIP. The initial conclusions of this assessment led the NRC to issue Information Notice (IN) 97-63. The information contained in IN97-63 and supplemental information provided by the NRC to the BWRVIP in an August 8, 1997, meeting gave the criteria necessary to justify a deferral of the RPV circumferential weld examinations for 40 months or 2 fuel cycles, whichever is greater. Based on this information and confirmation of its applicability to LGS Unit 1, PECO Energy submitted a request for technical alternative pursuant to the provisions of 10CFR50.55a(a)(3)(i) as documented in Relief Request Number RR-01. Therefore, the requirement to examine essentially 100%, i.e., at least 90%, of each RPV shell weld, Examination Category B-A, Item Number B1.10, at LGS Unit 1, before the end of the current inspection interval, is modified. The requirement for Unit 1 became to examine welds to the maximum extent practical, as allowed in RR-01.

The examinations of these welds for LGS Unit 1 will be conducted in accordance with this technical alternative approved by the NRC.

IV. EXAMINATION PROGRAM

The examinations RPV shell welds, Examination Category B-A, Item Number B1.10, at LGS Unit 1, will be conducted in accordance with RR-01, the technical alternative approved by the NRC. The planned alternative program to the 90% coverage of each weld uses the recommendations of BWRVIP-05 as a basis for doing no additional examinations beyond the described "best effort" approach.

V. EXAMINATION RESULTS

Examination results generated from this augmented inspection program shall be recorded and evaluated in accordance with the 1989 Edition of the Code and applicable plant procedures.

VI. REPORTS/RECORDS

All reports and records associated with the examinations of this augmented program shall be prepared and maintained in accordance with the 1989 Edition of Section XI, Specification NE-042, and plant procedures. They will be submitted to the NRC as part of the Code-required ISI submittal. The results will also be provided to the EPRI project manager for BWRVIP activities.

APPENDIX C

REFERENCE DRAWINGS

TABLE OF CONTENTS

REFERENCE DRAWINGS

ASME SECTION XI BOUNDARY P&ID's

P&ID NUMBER	SYSTEM NAME
ISI-M-01, Sh. 1	Main Steam
ISI-M-08, Sh. 1	Condensate & Refueling Water Storage
ISI-M-11, Sh. 1,1A,2,3,4	Emergency Service Water
ISI-M-12	RHR Service Water
ISI-M-13, Sh. 1	Reactor Enclosure Cooling Water
ISI-M-15, Sh. 1,15,16	Compressed Air (Service Air)
ISI-M-20, Sh. 3,4,4A,5,6,7,8	Fuel & Diesel Oil Storage and Transfer
ISI-M-26, Sh. 1,2,4	Plant Process Radiation Monitoring
ISI-M-40, Sh. 1	MSIV Leakage Control System - Abandoned in Place per MOD P00017-1
ISI-M-41, Sh. 1,2	Nuclear Boiler
ISI-M-42, Sh. 1,2,5	Nuclear Boiler Vessel Instrumentation
ISI-M-43, Sh. 1,2	Reactor Recirculation Pump
ISI-M-44, Sh. 1,2	Reactor Water Clean-up
ISI-M-45, Sh. 1	Clean Up Filter Demineralizer
ISI-M-46, Sh. 1	Control Rod Drive Hydraulic - Part A
ISI-M-47, Sh. 1	Control Rod Drive Hydraulic - Part B
ISI-M-48, Sh. 1	Standby Liquid Control
ISI-M-49, Sh. 1	Reactor Core Isolation Cooling
ISI-M-50, Sh. 1	RCIC Pump Turbine
ISI-M-51, Sh. 1,2,3,4	Residual Heat Removal
ISI-M-52, Sh. 1,2,2A	Core Spray
ISI-M-53, Sh. 1,2	Fuel Pool Cooling and Clean-up
ISI-M-55, Sh. 1	High Pressure Coolant Injection
ISI-M-56, Sh. 1	HPCI Pump Turbine
ISI-M-57, Sh. 1,2,3	Containment Atmospheric Control
ISI-M-58, Sh. 1,2	Hydrogen Recombiner
ISI-M-59, Sh. 1,2	Primary Containment Instrument Gas
ISI-M-60, Sh. 1	Primary Containment Leak Testing
ISI-M-61, Sh. 1	Liquid Radwaste Collection
ISI-M-87, Sh. 4,5	Drywell Chilled Water
ISI-M-90, Sh. 1,2	Control Structure Chilled Water

REFERENCE DRAWINGS

ASME SECTION XI CLASS 1 AND 2 ISI ISOMETRICS

SYSTEM	CLASS 1	CLASS 2
Residual Heat Removal	01-01	01-02
	01-101	01-102
	01-04	01-03
	01-104	01-103
	01-07A	01-03A
	01-107A	01-103A
	01-09A	01-05
	01-109A	01-105
	01-11	01-06
	01-111	01-106
		01-06A
		01-106A
		01-07
		01-107
		01-08
		01-108
		01-09
		01-109
		01-10
		01-110
		01-12
		01-112
		01-13
		01-113
		01-14
		01-114
		01-16
		01-116
		01-17
		01-117
		01-22
		01-122
		01-23
		01-123
		01-26
		01-126
		01-127
		01-128
		01-129
		01-130
High Pressure Coolant Injection	02-01	02-101A

REFERENCE DRAWINGS

ASME SECTION XI CLASS 1 AND 2 ISI ISOMETRICS

SYSTEM	CLASS 1	CLASS 2
	02-101	02-02
		02-102
		02-03
		02-103
		02-04
		02-104
		02-05
		02-105
		02-06
		02-106
		02-07
		02-107
		02-08
		02-108
		02-09
		02-109
Main Steam	03-01	03-03
	03-101	03-103
	03-02	03-05
	03-04	03-105
	03-104	03-06
		03-106
Core Spray	04-01	04-02
	04-101	04-102
	04-04	04-03
	04-104	04-103
		04-05
		04-105
		04-06
		04-106
		04-07
		04-107
		04-08
		04-108
		04-109
		04-110
		04-111
Feedwater	05-01	05-02
	05-101	05-102

REFERENCE DRAWINGS

ASME SECTION XI CLASS 1 AND 2 ISI ISOMETRICS

SYSTEM	CLASS 1	CLASS 2
	05-03 05-103	05-04 05-104 05-05(Class 4 Augmented only) 05-06(Class 4 Augmented only)
Reactor Core Isolation Cooling	06-01 06-101	06-02 06-102 06-03 06-103 06-04
Reactor Recirculation	07-01 07-101 07-02 07-102 07-03 07-103 07-04 07-104	
Reactor Water Clean Up	08-01 08-101 08-02(includes Non-Classed piping) 08-102 08-03 08-103 08-04 08-104 08-05 (Augmented only) 08-06 (Augmented only) 08-07 (Augmented only) 08-08 (Augmented only)	
Control Rod Drive		09-01 09-101 09-02 09-102
Standby Liquid Control	11-01 11-101 11-02	

REFERENCE DRAWINGS

ASME SECTION XI CLASS 1 AND 2 ISI ISOMETRICS

SYSTEM	CLASS 1	CLASS 2
	11-102	
	11-03	
	11-103	
RPV Vent	12-01	

REFERENCE DRAWINGS

COMPONENTS, REACTOR PRESSURE VESSEL AND RPV APPURTENANCES

COMPONENT	CLASS 1	CLASS 2
Recirculation Pumps	XI-1P-201	
RHR Pumps		XI-1P-202
RHR Heat Exchangers		XI-1E-205 Page 1
RCIC Pump		XI-10P-203
HPCI Pumps		XI-10P-204
Core Spray Pumps		XI-1P-206
Reactor Pressure Vessel	XI-BA-1 - Page 1 XI-BA-2 - Page 1 XI-BA-3 - Page 1 XI-BA-4 - Page 1 XI-BA-5 - Page 1 XI-BA-6 - Page 1 XI-BA-7 - Page 1 XI-BD-1 - Page 1 XI-BD-2 - Page 1 XI-BD-3 - Page 1 XI-BE-1 - Page 1 XI-BE-2 - Page 1 XI-BE-3 - Page 1 XI-BE-4 - Page 1 XI-BE-5 - Page 1 XI-BF - Page 1 XI-BF-1 - Page 1 XI-BF-2 - Page 1 XI-BF-3 - Page 1 XI-BF-4 - Page 1 XI-BF-5 - Page 1 XI-BF-6 - Page 1 XI-BF-7 - Page 1 XI-BF-8 - Page 1	
Reactor Pressure Vessel, continued	XI-BF-9 - Page 1	

REFERENCE DRAWINGS

COMPONENTS, REACTOR PRESSURE VESSEL AND RPV APPURTENANCES

COMPONENT	CLASS 1	CLASS 2
	XI-BF-17- Page 1	
	XI-BG - Page 1	
	XI-BH-1 - Page 1	
	XI-BH-1 - Page 2	
	XI-BH-2 - Page 1	
	XI-BH-3 - Page 1	
	XI-BH-4 - Page 1	
	XI-BN - Page 1	
	XI-BN-1 - Page 1	
	XI-BN-1 - Page 2	
	XI-BN-1 - Page 3	
	XI-BN-2 - Page 1	
	XI-BN-2 - Page 2	
	XI-BN-2 - Page 3	
	XI-BN-3 - Page 1	
	XI-BN-3 - Page 2	
	XI-BN-4 - Page 1	
	XI-BN-4 - Page 2	
	XI-BN-4 - Page 3	
	XI-BN-4 - Page 4	
	XI-BN-4 - Page 5	
	XI-BN-4 - Page 6	
	XI-BN-4 - Page 7	
	XI-BN-5 - Page 1	
	XI-BN-5 - Page 2	
	XI-BN-6 - Page 1	
	XI-BN-6 - Page 2	
	XI-BN-6 - Page 3	
	XI-BN-7 - Page 1	
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	XI-BN-7 - Page 3	
	XI-BN-7 - Page 4	
	XI-BN-8 - Page 1	
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	XI-BN-8 - Page 3	
	XI-BN-8 - Page 4	
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	XI-BN-8 - Page 6	
	XI-BN-8 - Page 7	
	XI-BN-9 - Page 1	
	XI-BN-9 - Page 2	
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	XI-BN-11 - Page 1	
	XI-BN-11 - Page 2	

REFERENCE DRAWINGS

COMPONENTS, REACTOR PRESSURE VESSEL AND RPV APPURTENANCES

COMPONENT	CLASS 1	CLASS 2
	XI-BN-12 - Page 1	
	XI-BN-12 - Page 2	
	XI-BN-13 - Page 1	
	XI-BN-14 - Page 1	
	XI-BNN - Page 1	
	XI-FA-1 - Page 1	
	XI-FA-1 - Page 3	
	XI-FA-2 - Page 1	
	XI-FA-2 - Page 2	
	XI-FA-2 - Page 3	
	XI-RPV-1 - Page 1	
	XI-RPV-1 - Page 1	
	XI-RPV-1 - Page 2	
	XI-RPV-1 - Page 3	
	XI-RPV-1 - Page 4	
RPV App	XI-RPV-1N	

REFERENCE DRAWINGS

ASME CLASS 3 FABRICATION ISOMETRIC DRAWINGS
FOR COMPONENT SUPPORTS AND INTEGRAL ATTCAHMENTS

SYSTEM	DRAWING NUMBER
Emergency Service Water	HBC-081-01
	HBC-081-02
	HBC-082-01
	HBC-082-02
	HBC-082-03
	HBC-083-01
	HBC-083-02
	HBC-084-01
	HBC-084-02
	HBC-138-01
	HBC-138-02
	HBC-138-03
	HBC-143-01
	HBC-143-02
	HBC-143-03
	HBC-147-01
	HBC-147-02
	HBC-147-03
	HBC-152-01
	HBC-152-02
	HBC-152-03
	HBC-158-01
	HBC-159-01
	HBC-166-01
	HBC-192-01
	HBC-192-02
	HBC-192-03
	HBC-192-04
	HBC-193-01
	HBC-193-02
	HBC-193-03
	HBC-193-04
	HBC-194-01
	HBC-194-02
	HBC-194-03
	HBC-194-04
	HBC-195-01
	HBC-195-02
	HBC-195-03
	HBC-195-04
	HBC-238-01 (ends at Unit 2 Boundary)
	HBC-243-01 (except from FW51 to FW3)
Emergency Service Water, continued	HBC-247-02

REFERENCE DRAWINGS

ASME CLASS 3 FABRICATION ISOMETRIC DRAWINGS
FOR COMPONENT SUPPORTS AND INTEGRAL ATTCAHMENTS

SYSTEM	DRAWING NUMBER
	HBC-252-01 (except from FW50 to Unit 1/Unit 2 tie-in)
	HBC-266-01
	HBC-270-01
	HBC-292-01 (ends at Unit 2 N-5 Boundary) (2 Shts)
	HBC-292-02 (ends at Unit 2 N-5 Boundary) (2 Shts)
	HBC-292-03 (ends at Unit 2 N-5 Boundary) (2 Shts)
	HBC-292-04 (ends at Unit 2 N-5 Boundary) (2 Shts)
	HBC-293-01 (ends at Unit 2 N-5 Boundary)
	HBC-293-02 (ends at Unit 2 N-5 Boundary)
	HBC-293-03 (ends at Unit 2 N-5 Boundary)
	HBC-293-04 (ends at Unit 2 N-5 Boundary)
	HRC-001-01
	HRC-002-02
	C-1078
Main Steam	GBC-101-01
	GBC-101-02
	GBC-101-03
	GBC-101-04
	GBC-101-05
	GBC-101-06
	GBC-101-07
	GBC-101-08
	GBC-101-09
	GBC-101-10
	GBC-101-11
	GBC-101-12
	GBC-101-13
	GBC-101-14
	GBC-116-01
	GBC-116-02
	GBC-116-03
	GBC-116-04
	GBC-116-05
	GBC-116-06
	GBC-116-07
	GBC-116-08
Main Steam, continued	GBC-116-09
	GBC-116-10

REFERENCE DRAWINGS

ASME CLASS 3 FABRICATION ISOMETRIC DRAWINGS
FOR COMPONENT SUPPORTS AND INTEGRAL ATTCAHMENTS

SYSTEM	DRAWING NUMBER
	GBC-116-11
	GBC-116-12
	GBC-116-13
	GBC-116-14
RHR Service Water	GBC-102-01
	GBC-103-01
	GBC-104-01
	GBC-106-01
	GBC-110-01
	HBC-091-01
	HBC-091-02
	HBC-091-03
	HBC-091-04
	HBC-091-05
	HBC-091-06
	HBC-091-07
	HBC-091-08
	HBC-091-09
	HBC-091-10
	HBC-091-11
	HBC-091-12
	HBC-091-13
	HBC-091-14
	HBC-091-15
	HBC-091-16
	HBC-091-17
	HBC-091-18
	HBC-091-19
	HBC-180-01
	HBC-181-01
	HBC-182-01
	HBC-183-01
	HBC-280-01 except from FW1 to Unit 1/Unit 2 tie-in
	HBC-282-01 except form Unit1/Unit2 tie-in to HBC-282-2
	HBC-507-01
	HBC-507-02
	HBC-507-03
RHR Service Water, continued	HBC-507-04
	HBC-507-05
	HBC-507-06

REFERENCE DRAWINGS

ASME CLASS 3 FABRICATION ISOMETRIC DRAWINGS
FOR COMPONENT SUPPORTS AND INTEGRAL ATTCAHMENTS

SYSTEM	DRAWING NUMBER
	HBC-507-07
	HBC-507-08
	HBC-507-09
	HBC-507-10
	HBC-507-11
	HBC-507-12
	HBC-507-13
	HBC-507-14
	HBC-507-15
	HBC-507-16
	HBC-507-17
	HBC-507-18
	HBC-508-01
	HBC-508-02
	HBC-509-01
	HBC-509-02
	HBC-537-01
	HBC-563-01

REFERENCE DRAWINGS

ASME SECTION XI UT CALIBRATION BLOCKS

CAL. BLOCK No	MATERIAL	DRAWING No. (8031-)
LIM-30-.625-CS	SA333Gr6	M-246A-258
LIM-30-.375-CS	SA155GrKC70	M-246A-61
LIM-28-SS-O'LAY	SA312Gr304	M-246A-275
LIM-22-SS-O'LAY	SA312Gr304	M-246A-275
LIM-20-SS-O'LAY	SA312Gr304	M-246A-275
LIM-28-2.72-SS316	SA358T316L	M-246A-256
LIM-28-1.285-SS316	SA403WP316	M-246A-256
LIM-28-1.076-SS316	SA358T316	M-246A-256
LIM-26-1.013-CS	SA106GrC	M-246A-256
LIM-26-.928-CS	SA106GrC	M-246A-60
LIM-24-1.812-CS-F	SA350LF2	M-246A-256
LIM-24-1.812-CS	SA106GrB	M-246A-60
LIM-24-.375CS	SA106Grb	M-246A-61
LIM-22-1.22-CS	SA234WPB	M-246A-256
LIM-22-1.009-SS316	SA358T316	M-246A-207
LIM-20-1.031-CS	SA333Gr6	M-246A-60
LIM-20-1.031-SS	SA403WP304-W	M-246A-256
LIM-20-.903-SS316	SA358T316L	M-246A-258
LIM-20-.733-SS	SA376T304	M-246A-260
LIM-20-.375-CS	SA106GrB	M-246A-61
LIM-20-.500-CS	SA106GrB	M-246A-260
LIM-18-.500-CS	SA106GrB	M-246A-258
LIM-18-.938-CS	SA106GrB	M-246A-260
LIM-18-.375CS	SA106GrB	M-246A-61
LIM-16-1.219-CS	SA106GrC	M-246A-60
LIM-16-.500-CS	SA333Gr6	M-246A-258
LIM-16-.375-CS	SA106GrB	M-246A-61
LIM-14-1.093-CS	SA106GrB	M-246A-256
LIM-14-.937-CS	SA106GrB	M-246A-258
LIM-14-.750-CS	SA106GrB	M-246A-260
LIM-14-.375 CS	SA106GrB	M-246A-61
LIM-12-.843-CS	SA106GrB	M-246A-258
LIM-12-.843-SS316	SA358T316L	M-246A-258
LIM-12-.688-CS	SA333Gr6	M-246A-258
LIM-12-.688-SS	SA312T304	M-246A-260
LIM-12-.688-SS316	SA358T316L	M-246A-258

REFERENCE DRAWINGS

ASME SECTION XI UT CALIBRATION BLOCKS

CAL. BLOCK No	MATERIAL	DRAWING No. (8031-)
LIM-12-.586SS	SA358Tp304	M-246A-260
LIM-12-.375-CS	SA106GrB	M-246A-61
LIM-10-.843-CS	SA106GrB	M-246A-258
LIM-10-.718-CS	SA106GrB	M-246A-258
LIM-10-.593-CS	SA106GrB	M-246A-258
LIM-10-.593-SS316	SA358T316L	M-246A-258
LIM-10-.365-CS	SA106GrB	M-246A-61
LIM-8-.906-CS	SA106GrB	M-246A-258
LIM-8-.719-CS	SA106GrB	M-246A-246
LIM-8-.594-CS	SA106GrB	M-246A-246
LIM-8-.500-CS	SA106Grb	M-246A-258
LIM-6-.562-CS	SA106GrB	M-246A-258
LIM-6-.432-CS	SA106GrB	M-246A-61
LIM-6-.432-SS	SA376T304	M-246A-61
LIM-6-.280-CS	SA106GrB	M-246A-61
LIM-6-.432-SS316	SA312T316L	M-246A-276
LIM-4-.437-CS	SA106GrB	M-246A-276
LIM-4-.337-CS	SA106GrB	M-246A-61
LIM-4-.337-SS304	SA312T304	M-246A-208
LIM-3-.438-CS	SA106GrB	M-246A-230
LIM-SP-515-CS	SA515Gr70	M-246A-194
LIM-SP-350-CS	SA350LF2	M-246A-194
LIM-SP-333-CS	SA333Gr6	M-246A-194
LIM-SP-182-SS316	SA182F316	M-246A-232
LIM-SP-181-CS	SA181Gr2	M-246A-194
LIM-SP-105-CS	SA105Gr2	M-246A-194
LIM-3-STUD-CS	SA540 B-24CL2	M-246A-218
LIM-3-STUD-II-CS	SA540 B-24CL2	M-246A-278
LIM-1.00-P	SA516-Gr70	M-246A-277
LIM-F-1.18-CS	SA516-Gr70	M-246A-211
LIM-F-.812-CS	SA516-Gr70	M-246A-277
LIM-RHR-HT-EX-IR	SA105-CL2	XI-1E-205
RPV STD. No. 1A	SA533	M-246B-184
RPV STD. No. 2	SA533	M-246B-39
RPV STD. No. 3	SA533	M-246B-127
RPV STD. No. 4	SA533	M-246B-128
RPV FLANGE SEAL	A36	M-246B-254
MAIN STEAM ZONE 3	A285	M-246B-188

REFERENCE DRAWINGS

ASME SECTION XI UT CALIBRATION BLOCKS

CAL. BLOCK No	MATERIAL	DRAWING No. (8031-)
RECIRC OUT ZONE 3	A285	M-246B-188
FEEDWATER ZONE 3	A285	M-246B-188
PART NO. 1 THRU 21	SA508	M-246B-2-(1)-8
S/S NO. 4	SA508	M-246B-157
RPV CLOSURE STUD	SA540	M-246B-115
NUT NO. B61	SA540	M-246B-37
LIM-2-.218SS	SA312TP316L	M-246A-276
LIM-1.5-.200-SS	SA312TP316L	M-246A-276
BF-9-CB-1	SA508	M-246B-163
LIM-2		M-246B-188

REFERENCE DRAWINGS

CONTAINMENT INSERVICE INSPECTION BOUNDARY DRAWINGS

DRAWING NUMBER	TITLE
C-002-00096	Exterior Bulkhead Assembly 2'-6" x 6'-0" Personnel Lock
C-002-00139	12'-0" I.D. Equipment Door for Personnel Lock
C-002-00243	Top Head, Flange, Bolting Rings and Seal Plate Assembly
C-002-00247	Top Head Dollar Plates and Access Manhole Shop Assemblies
C-0247	Reactor Building Units 1 and 2, Primary Containment General Arrangement
C-0262	Reactor Building Primary Containment Downcomer Arrangement
C-0270	Reactor Building Units 1 & 2 Primary Containment MSRV Discharge Pipe and Downcomer Bracing
C-0276	Liner Plate Requirements General Outline
C-0279	Liner Plate Requirements Penetration Schedule
C-0283	Reactor Building Units 1 & 2, Liner Plate Requirements, Suppression Pool Details
C-0287	Reactor Building Units 1 & 2, Liner Plate Requirements, Drywell Penetration Sections and Details
C-0288	Reactor Building Units 1 & 2, Liner Plate Requirements Drywell Penetration Details
C-0293	Reactor Building, Primary Containment Downcomer Details
C-0294	Diaphragm Floor Plan and Details
C-0776	Liner Plate Attachment Acceptance Criteria
C-0867	Drywell Wall Sections – Details Units 1 and 2
C-0936	Seismic Stabilizer Plan and Details
C-0943	Drywell Interior Structural Steel Beam Connection Details
E-0370	Schematic Diagram Primary Containment Vacuum Relief Valve Assembly Test 1 & 2 Units
M-0057	P&ID Containment Atmospheric Control
M-0060	P&ID Primary Containment Leak Testing
M-0061	P&ID Liquid Radwaste Collection
M-0391	Flued Head Details

SELECTION CRITERIA

APPENDIX D

SELECTION CRITERIA

TABLE OF CONTENTS

SELECTION CRITERIA

1.0 GENERAL

The Code provides mandatory rules and requirements for the inservice inspection of Class 1, 2, and 3 components and their supports. Included in these Code requirements are rules for the selection of specific components for examination (i.e. not all components subject to Code requirements require examination during the inspection interval.) This document addresses these Code rules, as they apply to LGS Unit 1, and provides the specific selection basis utilized for all LGS Unit 1 nonexempt components and their supports.

The selection bases documented in Attachments B, C, and D represents the initial component selections for the inservice inspection interval. Appendix D provides the information necessary to support the initial ISI component sample selections.

These selection bases represent the minimum number of component selections necessary to satisfy the Code requirements. When performing calculations to determine examination sample size, fractional numbers shall be rounded up or down to the nearest whole number as appropriate. In the event that proration calculations indicate essentially zero selections, a minimum of one (1) selection shall be made. When, due to good engineering judgement, selections made are in excess of Code requirements, it is so noted in the selection basis.

These selection bases are not required to be revised as a result of changes in plant configuration from repairs, replacements, or modifications throughout the inspection interval. Repaired, replaced or modified components shall have a Section XI baseline inspection performed as part of the ASME Repair and Replacement Program. The baseline inspections are sufficient to satisfy the ISI requirements for those initially selected components that have been repaired, replaced or modified but had not yet been examined during the inspection interval. Further, it is not the intent to revise these selection bases when an initially selected component, with its ISI requirements completed for the inspection interval, is subsequently deleted during the inspection interval due to plant repairs, replacements or modifications. The Section XI baseline inspections of the new component performed as part of the ASME Repair and Replacement Program are sufficient to ensure continued conformance to Code requirements. Due to the requirement to perform baseline inspections on repaired, replaced or modified components, a reconciliation of these selection bases with changes in plant configuration is only required as part of the ISI Program update at the beginning of each inspection interval.

1.1 Multiple Component Concept

The multiple component concept is used frequently in the selection of components in a variety of Code Examination Categories. Basically, for a group of like components (i.e., multiple components), typically equipment, of similar design and performing a similar function; the Code requires examination of at least one component from the group of these multiple components. Where the multiple component concept is applied for selections it is so noted under the appropriate Code Examination Category. Each group of components established is assigned a group number for purposes of identification of the group. A complete listing of all multiple component group numbers may be found in Attachment A of this Appendix.

1.2 Selection Optimization

Sometimes, especially in the cases of Class 1 and 2 welds, augmented inspection programs require examinations which duplicate the Code examination methods required for that same component. Where possible, selections shall be made to optimize examinations performed, yet still meet the selection requirements of the Code.

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1.3 Definitions

1.3.1 Random selections – Random selections pertain to those selections made purely at random and not based on any Code required parameter(s), such as component size, configuration, stress etc.

1.3.2 Structural Discontinuity (SD) – Structural discontinuities generally include pipe to fitting/valve weld joints such as elbows, tees, reducers, flanges, etc., and pipe branch fitting connections to the main piping run.

1.3.3 Terminal End (TE) - Terminal ends are the extremities of piping runs. Generally these connect the piping to structures and components such as in-line anchors, flued heads at penetrations and nozzles in vessels and pumps, each of which acts as a rigid restraint or provides at least two (2) degrees of restraint to piping thermal expansion.

Also for branch piping, the connection to the run piping branch fitting or tee may be considered a TE if the ratio of the run piping OD to the branch piping OD is ≥ 3 to 1.

2.0 LGS Unit 1 ISI COMPONENT SELECTION BASIS

Information presented in Appendix D is organized by component Class and Code Examination Category (as found in Table IWX-2500-1 of the Code); Class 1, 2, and 3 component supports are discussed separately. The specific selection criteria applies to all components within the Code Examination Category unless otherwise stated. All categories are discussed as follows:

2.1 Class 1 Code Examination Categories Involving Sample Selection
2.2 Class 2 Code Examination Categories Involving Sample Selection
2.3 Class 3 Code Examination Categories Involving Sample Selection
2.4 Class 1, 2, and 3 Component Supports

2.1 Class 1 Code Examination Categories Involving Sample Selection

2.1.1 Examination Category B-E, Pressure Retaining Partial Penetration Welds in Vessels

In accordance with Table IWB-2500-1, twenty five percent (25%) of all partial penetration welds in the reactor pressure vessel nozzles, control rod drive nozzles, and vessel instrumentation nozzles shall be selected for examination. Selections shall be evenly distributed among the different types of partial penetration welds, by selecting twenty five percent (25%) by Code Item Number. The following represents the initial selection by Code Item Number.

Item Number	Population	Number Selected
B4.11	1	1
B4.12	185	46
B4.13	65	16
Total	251	63

2.1.2 Examination Category B-G-1, Pressure Retaining Bolting, Greater Than 2" in Diameter

2.1.2.1 Pumps, Item Numbers B6.180 through B6.200 inclusive

Selection of the Reactor Recirculation pump bolting shall be in accordance with Table IWB-2500-1, Note 3, "For ... pumps, ... examinations are limited to components selected for examination under Examination Categories... B-L-2. Referring to Examination Category B-L-2,

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Pump Casings, the multiple component concept applies and examinations are limited to one (1) of two (2) Reactor Recirculation pumps. The Reactor Recirculation pumps are multiple component group number 20.

2.1.3 Examination Category B-G-2, Pressure Retaining Bolting, 2 in. and Less in Diameter

2.1.3.1 Piping, CRD Housings Item Numbers B7.50 and B7.80, respectively

In accordance with Table IWB-2500-1, all Class 1 bolting in piping in Code Examination Category B-J shall be selected for examination.

Since CRD housing bolting is only required to be examined when a housing is disassembled, selection of individual housings for examination is not practical. All CRD housings are considered selected; however, examinations are only required in the event the housings are disassembled.

2.1.3.2 Valves, Item Number B7.70

Selection of Class 1 valve bolting (2 in. and less in diameter) shall be in accordance with Table IWB 2500-1, Note 2, "For ... valves, examinations are limited to components selected for examination under Examination Categories ... B-M-2." Referring to Examination Category B-M-2, Valve Bodies, the multiple component concept applies and examinations are limited to one (1) valve in a multiple component group. See Examination Category B-M-2 (Section 2.1.8) and Attachment A of this Appendix for further discussion of valve groupings.

2.1.4 Examination Category B-H, Integral Attachments for Vessels

Selections shall be made in accordance with Section 9.2 and Table 9.2-1 of this ISI Program. Per footnote 4 of the Table, a minimum of 10% of the reactor vessel integrally welded attachments shall be selected for examination.

See Attachment D for the specific weld totals.

2.1.5 Examination Category B-J, Pressure Retaining Welds in Piping

The extent (percentage) of Class 1, Category B-J welds selected for examination shall be in accordance with ASME Section XI, 1989 Edition. Dissimilar metal welds on portions of piping not including nozzle to safe end welds, which are assigned to Code Category B-F for piping shall also be included in the Category B-J weld population.

The welds selected for examination shall consist of a 25% representative sample of each system and shall include circumferential, branch connection, and socket welds not exempted by IWB-1220. The examination sample shall include all terminal ends connected to vessels and other terminal ends and weld joints connected to other components where stress levels exceed either of the following limits:

1. Primary plus secondary stress intensity range of 2.4 Sm
2. Cumulative usage factor (u) of 0.4

The sample shall include additional piping welds so that the total population of B-J welds selected for examination equals 25% of the B-J welds.

A portion of the longitudinal welds intersecting any of the selected circumferential welds shall also be examined in accordance with ASME Code Case N-524.

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See Attachment B for specific weld totals.

2.1.6 Examination Category B-K-1, Integral Attachments for Piping, Pumps, and Valves

Selection of Class 1 integrally welded attachments shall be in accordance with Section 9.2 and Table 9.2-1 of this ISI Program which is based on ASME Code Case N-509.

See Attachment D for the specific weld totals.

2.1.7 Examination Category B-L-2, Pump Casings

Selection of Class 1 Reactor Recirculation pump casings for examination shall be in accordance with Table IWB-2500-1, Note 1, "Examinations are limited to at least one pump in each group of pumps performing similar functions in the system. The multiple component concept applies and examinations are limited to one (1) of two (2) Reactor Recirculation pumps. The Reactor Recirculation pumps are multiple component group number 20.

2.1.8 Examination Category B-M-2, Valve Bodies

Selection of Class 1 valve bodies subject to the requirements of Examination Category B-M-2 shall be in accordance with Table IWB-2500-1, Note 3, "Examinations are limited to at least one valve within each group of valves that are of the same size, constructional design (such as globe, gate, or check valves) and manufacturing method, and that perform similar functions in the system (such as containment isolation and system over pressure protection)." The multiple component concept applies and twenty one (21) valve groupings have been established (See Attachment A for details of valve groupings). Examinations are limited to one (1) valve per group.

2.2 Class 2 Code Examination Categories Involving Sample Selection

2.2.1 Examination Category C-A, Pressure Retaining Welds in Pressure Vessels

Selection of Examination Category C-A welds in the Residual Heat Removal (RHR) heat exchangers shall be in accordance with IWC-2500-1, Note 3, "In the case of multiple vessels of similar design, size, and service (such as ... heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels." Therefore, the multiple component concept applies and examinations shall be performed on one vessel or distributed among the two (2) RHR heat exchangers. The RHR heat exchangers are multiple component group number 23.

2.2.2 Examination Category C-B, Pressure Retaining Nozzle Welds in Vessels

Selection of Examination Category C-B welds in the RHR heat exchangers shall be in accordance with IWC-2500-1, Note 4, "In the case of multiple vessels of similar design, size, and service (such as heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels." Therefore the multiple component concept applies and examinations shall be performed on one vessel or distributed among the two (2) RHR heat exchangers. The RHR heat exchangers are multiple component group number 23.

SELECTION CRITERIA

2.2.3 Examination Category C-C, Integral Attachments for Vessels, Piping, Pumps and Valves

Selection of Examination Category C-C integrally welded attachments shall be in accordance with Section 9.2 and Table 9.2-2 of this ISI Program which is based on ASME Code Case N-509. A minimum of 10% of the integral attachments of each Item Number shall be examined.

See Attachment D for the specific weld totals.

2.2.4 Examination Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping

Selection of Examination Category C-F-1 pressure retaining welds shall be in accordance with IWC-2500-1, Note 2:

"The welds selected for examination shall include 7.5%, but not less than 28 welds, of all austenitic stainless steel of high alloy welds not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Category C-F-1. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

A. the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt austenitic stainless steel or high alloy welds in each system (i.e., if a system contains 30% of the nonexempt welds, then 30% of the nondestructive examinations required by Examination Category C-F-1 shall be performed on that system);

B. within a system, the examinations shall be distributed among terminal ends and structural discontinuities... prorated, to the degree practicable, on the number of nonexempt terminal ends and structural discontinuities in that system; and

C. within each system, examinations shall be distributed between line sizes prorated to the degree practicable."

As used herein, the term "to the degree practicable" shall mean within a count of one (1), however the total Code Category examination sample size shall not be less than 7.5%, or 28 welds, whichever is greater. If the total population subject to examination is less than or equal to 28 welds, then the total population shall be selected for examination. In systems which include terminal ends, a minimum of one (1) terminal end shall be selected, even in the event that proration indicates less than one (1) selection.

Currently there are less than 28 LGS Unit 1 welds in Examination Category C-F-1, therefore, all welds shall be selected for examination. Appendix D, Attachment C contains a tabular listing of Examination Category C-F-1 weld totals.

2.2.5 Examination Category C-F-2, Pressure Retaining Welds in Carbon or Low Alloy Steel Piping

Selection of Examination Category C-F-2 pressure retaining welds shall be in accordance with IWC-2500-1, Note 2:

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"The welds selected for examination shall include 7.5%, but not less than 28 welds, of all carbon and low alloy steel welds not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Category C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

A. the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt carbon and low alloy steel welds in each system (i.e., if a system contains 30% of the nonexempt welds, then 30% of the nondestructive examinations required by Examination Category C-F-2 shall be performed on that system);

B. within a system, the examinations shall be distributed among terminal ends and structural discontinuities... prorated, to the degree practicable, on the number of nonexempt terminal ends and structural discontinuities in that system; and

C. within each system, examinations shall be distributed between line sizes prorated to the degree practicable."

As used herein, the term "to the degree practicable" shall mean within a count of one (1), however the total Code Category examination sample size shall not be less than 7.5%, or 28 welds, whichever is greater. If the total population subject to examination is less than or equal to 28 welds, then the total population shall be selected for examination. In systems which include terminal ends, a minimum of one (1) terminal end shall be selected, even in the event that proration indicates less than one (1) selection.

Appendix D, Attachment C contains tables which illustrate all prorations as required by (A), (B), and (C) above for all Examination Category C-F-2 welds. These prorations indicate the total number, type, and nominal pipe size of welds needed to be selected within each system.

2.3 Class 3 Code Examination Categories Involving Sample Selection

2.3.1 Examination Categories D-A, D-B and D-C

Integral attachment welds for Class 3 vessels, piping, pumps and valves shall be inaccordance with Section 9.1 and Table 9.2-3 of this ISI Program which is based on ASME Code Case N-509

2.4 Class 1, 2, and 3 Component Supports

2.4.1 Non-piping Component Supports

Selection of Class 1, 2 and 3 component supports shall be in accordance with Section 9.1 and Table 9.1-1of this ISI Program which is based on ASME Code Case N-491-1.

Non-piping component supports shall be selected for examination in accordance with IWF-2510:

A. Component supports selected for examination shall be the supports of those components that are required to be examined under IWB, IWC,IWD.

B. For multiple components within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined.

Basically, for those non-piping components which are not a part of a multiple component group (e.g., RPV, RCIC pump, HPCI pumps), all supports are selected for examination. Where the

SELECTION CRITERIA

multiple component concept applies, (e.g., Reactor Recirculation pumps (group 20), Core Spray pumps (group 21), RHR pumps (group 22), RHR heat exchangers (group 23)), the supports of only one of the components in the group is selected for examination.

2.4.2 Piping Component Supports

A. The total number of supports required to be selected for examination is determined by the ASME Section XI classification of the support, (i.e., Class 1, 2, or 3).

Class 1 25% of the nonexempt population
Class 2 15% of the nonexempt population
Class 3 10% of the nonexempt population

B. All supports are assigned to one of the following types - Anchor, Snubber, Rigid, and Variable. See paragraph 9.1.5.2 for "type" definitions.

C. Selections shall be distributed among the required number of supports determined in (a) above by system and type, prorated by the number of supports of each type within each system.

Attachment D illustrates the specific LGS Unit 1 support populations and sampling plan prorations resulting in the total number and types of supports selected in each system.

SELECTION CRITERIA
Attachment A

MULTIPLE COMPONENT GROUPS
CLASS 1 VALVES

VALVE GROUP	VALVE NUMBER	ISI FIGURE #	VENDOR	VENDOR PRINT #	VALVE SIZE & TYPE	MATERIAL	BOLTING	
							BONNET	HINGE
1	HV-41-1F022A,B,C,D HV-41-1F028A,B,C,D	03-01, 04 03-01, 04	Atwood & Morrill Atwood & Morrill	M1-B21-F022-C9.2 M1-B21-F028-C7.2	26" Globe 26" Globe	SA-216WCB (Cast) SA-216WCB (Cast)	Yes Yes	N/A N/A
2	PSV-41-1F013A,B, C,D,E,F,G,H,J,K,L,M, N,S	03-01, 04	Target Rock	M1-B21-F013-B2.1	6"x10" Relief	SA-105 (Forged)	Yes (2 sets) Yes (inlet flng.)	N/A N/A
3	HV-41-1F011A,B	05-01, 03	Anchor Darling	P-105A-1	24" Gate	SA-352LCB (Cast)	No (press. seal)	N/A
4	41-1F010A,B HV-41-1F074A,B	05-01, 03 05-01, 03	Atwood & Morrill Atwood & Morrill	P-105B-1 P-116-1	24" Check 24" Check	SA-325LCB (Cast) SA-352LCB (Cast)	No (press.seal) No (press. sela)	Yes Yes
5	HV-43-1F023A,B HV-43-1F031A,B	07-01, 02 07-01, 02	Lunkenheimer Lunkenheimer	M1-B32-F023-B-1.4A M1-B32-F031-B-1.4	28" Gate 28" Gate	SA-351CF8M (Cast) SA-351CF8M (Cast)	Yes Yes	N/A N/A
6	HV-44-1F001 HV-44-1F004	08-02 08-02	Anchor Darling Anchor Darling	P-107A-69 P-107A-69	6" Globe 6" Globe	SA-351CF8M (Cast) SA-351CF8M (Cast)	No (press. seal) No (press. seal)	N/A N/A
7a	44-1F027	08-02	Anchor Darling	P-107A-87	6" Gate	SA-351CF8M (Cast)	No (press. seal)	N/A

SELECTION CRITERIA
Attachment A

MULTIPLE COMPONENT GROUPS CLASS 1 VALVES								
VALVE GROUP	VALVE NUMBER	ISI FIGURE #	VENDOR	VENDOR PRINT #	VALVE SIZE & TYPE	MATERIAL	BOLTING BONNET	 HINGE
7b	HV-44-1F100	08-04	Velan	P-107B-6	6" Gate	SA-182-F316 (Forged)	No (press. seal)	N/A
7c	HV-C-44-1F105	08-02	Anchor Darling	P-107A-69	6" Globe	SA-351CF8M (Cast)	No (press. seal)	N/A
8a	51-1F060A,B	01-01, 04	Anchor Darling	P-107A-86	12" Gate	SA-351CF8M (Cast)	No (press. seal)	N/A
8b	51-1F077	01-11	Anchor Darling	P-107A-1	20" Gate	SA-351CF8M (Cast)	No (press. seal)	N/A
8c	HV-51-1F017A,B,C,D	01-02, 05, 07, 09	Anchor Darling	M1-E11-F017-C-2.6BC	12" Gate	SA-352LCB (Cast)	Yes	N/A
9	HV-51-1F015A,B	01-03, 05	Anchor Darling	M1-E11-F015-C-2.2	12"Globe	SA-351CF8M (Cast)	Yes	N/A
10	HV-51-1F041A,B,C,D HV-51-1F050A,B	01-01, 04, 07, 09 01-01, 04	Atwood & Morrill Atwood & Morrill	M1-E11-F041-B-3.3 M1-E11-F050-B-1.2	12" Check 12" Check	SA-352LCB (Cast) SA-352CF8M (Cast)	Yes Yes	Yes Yes
11a	51-1F065A,B,C,D	01-01, 04, 07, 09	Velan	P-107B-1	12" Gate	SA-182F316 (Forged)	No (press. seal)	N/A
11b	HV-51-1F008 HV-51-1F009	01-11 01-11	Velan Velan	P-107B-57 P-107B-57	20" Gate 20" Gate	SA-182F316 (Forged) SA-182F316 (Forged)	No (press. seal) No (press. seal)	N/A N/A

SELECTION CRITERIA
Attachment A

MULTIPLE COMPONENT GROUPS
CLASS 1 VALVES

VALVE GROUP	VALVE NUMBER	ISI FIGURE #	VENDOR	VENDOR PRINT #	VALVE SIZE & TYPE	MATERIAL	BOLTING	
							BONNET	HINGE
12	HV-52-108	04-04	Anchor	P-107A-19	12" Check	SA-351CF8M (Cast)	No (press. seal)	Yes
13	HV-52-1F005	04-01	Anchor Darling	M1-E21-F005-B-2.5	12" Gate	SA-352LCB (Cast)		N/A
14	HV-52-1F006A,B	04-01, 04	Atwood & Morrill	M1-E21-F006-B-2.3	12" Check	SA-352LCB (Cast)		Yes
15	52-1F007A,B	04-01, 04	Velan	P-107B-1	12" Gate	SA-182F316 (Forged)	No (press. seal)	N/A
16	HV-55-1F002 HV-55-1F003	02-01 02-01	Anchoring Darling Anchoring Darling	P-104C-34 P-104C-34	10" Globe 10" Globe	SA-216WCB (Cast) SA-216WCB (Cast)	No (press. seal) No (press. seal)	N/A N/A
17	Reserved for future use							
18	Reserved for future use							
19	Reserved for future use							

SELECTION CRITERIA
Attachment A

MULTIPLE COMPONENT GROUPS
EQUIPMENT

Equipment Group	Component ID	Description
20	1AP-201 1BP-201	Reactor Recirculation Pumps
21	1AP-206 1BP-206 1CP-206 1DP-206	Core Spray Pumps
22	1AP-202 1BP-202 1CP-202 1DP-202	RHR Pumps
23	1AE-205 1BE-205	RHR Heat Exchangers

EXAMINATION CATEGORY: B-J
WELD SELECTION TABLES

GRAND TOTAL NUMBER OF WELDS: 1069
PERCENT TO BE EXAMINED: X 25 %
OVERALL SAMPLE SIZE: 267

SYSTEM	TOTAL WELDS (B-F AND B-J CATS.)	TOTAL B-F CATEGORY WELDS IN THE B-J POPULATION	TOTAL B9.11	TOTAL B9.12	TOTAL B9.21	TOTAL B9.31	TOTAL B9.32	TOTAL B9.40	TOTAL TE-VESSEL WELDS	TOTAL OTHER TE WELDS	OTHER WELDS EXCEEDING STRESS OR USAGE (SFL WELDS)	TOTAL TE-VESSEL + TE WELDS + SFL WELDS
CS	97	14	31	52	0	0	0	0	2	2	10	14
FW	96	12	83	0	0	1	0	0	6	2	25	33
HPCI	23	0	23	0	0	0	0	0	0	2	0	2
MS	111	8	84	0	0	18	1	0	4	32	8	44
RCIC	30	4	16	0	10	0	0	0	0	2	0	2
RHR	231	32	102	96	0	1	0	0	4	11	28	43
RPV-APP	28	1	11	0	5	0	0	11	13	0	0	13
RR	259	0	83	148	12	8	2	6	12	0	0	12
RWCU	130	3	78	0	38	0	5	6	1	3	22	26
SLC	64	0	0	0	2	0	0	62	0	2	8	10
	1069	74	511	296	67	28	8	85	42	56	101	199

EXAMINATION CATEGORY: C-F-1
WELD SELECTION TABLES PART 1

GRAND TOTAL NUMBER OF WELDS:	16	
PERCENT TO BE EXAMINED:	X 7.5 %	
OVERALL SAMPLE SIZE:	16	(NOT LESS THAN 28 OR DO 100%)

SYSTEM	TOTAL WELDS	PERCENT OF GRAND TOTAL	SYSTEM SAMPLE SIZE	SYSTEM TOTAL TE'S	SYSTEM TOTAL SD'S	TE % OF TOTAL TE + SD	SD % OF TOTAL TE + SD	TE REQ'D SAMPLE	SD REQ'D SAMPLE	OTHER REQ'D SAMPLE	TE'S SELECTED	SD'S SELECTED	OTHER SELECTED
CS	1	6.25%	1.00	0	0	0.00%	0.00%	0	0	1	0	0	1
RHR	13	81.25%	13.00	0	8	0.00%	100.00%	0	8	0	0	8	0
RWCU	2	12.50%	2.00	0	2	0.00%	100.00%	0	2	0	0	2	0
	16	100.00%	16.00										

NOTES:

1. Since the total number of welds requiring examination is less than or equal to 28, then a minimum of 28 non -exempt welds (or the total population) are subject to the rules of Code Examination Category C-F-1, and are selected.

2. Code Examination Category C-F-1 includes all non-exempt Class 2 austenitic stainless steel, high alloy, dissimilar metal welds.

3. The C-F-1 weld counts for total population, overall and system sample size include welds not required to be examined per C-F-1 rules. (i.e. piping welds < 3/8" nominal wall.) all other weld counts include only those welds subject to examination.

EXAMINATION CATEGORY: C-F-1
WELD SELECTION TABLES PART 2

SYSTEM	LINESIZE................	6"	8"	10"	12"	14"	16"	18"	20"	22"	24"	26"	30"	SYSTEM TOTALS	TOTAL LINESIZES
CS	TOTAL WELDS..........	0	0	0	1	0	0	0	0	0	0	0	0	1	1
	TOTAL SELECTED...	0	0	0	1	0	0	0	0	0	0	0	0	1	
RHR	TOTAL WELDS..........	6	0	0	2	0	0	0	1	4	0	0	0	13	4
	TOTAL SELECTED...	6	0	0	2	0	0	0	1	4	0	0	0	13	
RWCU	TOTAL WELDS..........	2	0	0	0	0	0	0	0	0	0	0	0	2	1
	TOTAL SELECTED...	2	0	0	0	0	0	0	0	0	0	0	0	2	
													GRAND TOTAL WELDS........	16	
													GRAND TOTAL SELECTED..	16	

NOTES:
1. The C-F-1 weld counts contain only those non-exempt Class 2 austenitic stainless steel, high alloy or dissimilar metal welds subject to examination per C-F-1 rules, (i.e. does not include piping welds< 3/8" nominal wall.)

EXAMINATION CATEGORY: C-F-2
WELD SECLECTION TABLES PART 1

GRAND TOTAL NUMBER OF WELDS: 1580
PERCENT TO BE EXAMINED: X 7.5%
OVERALL SAMPLE SIZE: 118.5 (NOT LESS THAN 28 OR DO 100%)

SYSTEM	TOTAL WELDS	PERCENT OF GRAND TOTAL	SYSTEM SAMPLE SIZE	SYSTEM TOTAL TE'S	SYSTEM TOTAL SD'S	TE % OF TOTAL TE + SD	SD % OF TOTAL TE + SD	TE REQ'D SAMPLE	SD REQ'D SAMPLE	TE'S SELECTED	SD'S SELECTED
CRD	53	3.35%	3.98	0	43	0.00%	100.00%	0	4	0	4
CS	243	15.38%	18.23	9	173	4.95%	95.05%	1	17	1	17
FW	28	1.77%	2.10	2	26	7.14%	92.86%	0	2	1	2
HPCI	203	12.85%	15.23	6	179	3.24%	96.76%	0	15	1	15
MS	230	14.56%	17.25	4	104	3.70%	100.00%	1	17	1	17
RCIC	129	8.16%	9.68	4	82	4.65%	95.35%	0	9	1	9
RHR	692	43.80%	51.90	16	494	3.14%	96.86%	2	50	2	50
RWCU	2	0.13%	0.15	4	0	100.00%	0.00%	0	0	1	0
	1580	100.00%	118.50								

NOTES:

1. Class 2 dissimilar metal welds are included in Code Examination Category C-F-1, and are not included in C-F-2 weld count.

2. The C-F-2 weld counts include all non-exempt Class 2 carbon steel/low alloy welds including those welds not required to be examined per C-F-2 rules. (i.e. piping welds < 3/8" nominal wall).

3. In system where terminal ends are available for selection , at least one (1) terminal end shall be selected, even in the proration indicates essentially zero selections.

EXAMINATION CATEGORY: C-F-2
WELD SECLECTION TABLES PART 2

SYSTEM	LINESIZE...............	6"	8"	10"	12"	14"	16"	18"	20"	22"	24"	26"	30"	SYSTEM TOTALS	TOTAL LINESIZES
CRD	TOTAL WELDS..........	0	45	8	0	0	0	0	0	0	0	0	0	53	2
	TOTAL SELECTED...	0	3	1	0	0	0	0	0	0	0	0	0	4	
CS	TOTAL WELDS..........	9	0	30	61	56	87	0	0	0	0	0	0	243	5
	TOTAL SELECTED...	1	0	2	4	4	7	0	0	0	0	0	0	18	
FW	TOTAL WELDS..........	1	13	0	0	0	8	0	0	0	6	0	0	28	4
	TOTAL SELECTED...	0	1	0	0	0	1	0	0	0	0	0	0	2	
HPCI	TOTAL WELDS..........	0	33	10	57	51	26	[illegible]	19	0	0	0	0	203	7
	TOTAL SELECTED...	0	3	1	4	4	2	0	1	0	0	0	0	15	
MS	TOTAL WELDS..........	24	3	0	0	20	0	10	0	0	0	173	0	230	5
	TOTAL SELECTED...	2	0	0	0	1	0	1	0	0	0	13	0	17	
RCIC	TOTAL WELDS..........	101	10	18	0	0	0	0	0	0	0	0	0	129	3
	TOTAL SELECTED...	8	1	1	0	0	0	0	0	0	0	0	0	10	
RHR	TOTAL WELDS..........	49	32	0	15	4	57	346	62	1	76	0	50	692	10
	TOTAL SELECTED...	4	2	0	1	0	4	26	5	0	6	0	4	52	
RWCU	TOTAL WELDS..........	2	0	0	0	0	0	0	0	0	0	0	0	2	10
	TOTAL SELECTED...	1	0	0	0	0	0	0	0	0	0	0	0	1	

GRAND TOTAL WELDS........ 1580

GRAND TOTAL SELECTED.. 119

NOTES:
1. The C-F-2 weld counts contain only those non-exempt Class 2 carbon steel/low alloy welds subject to examination per C-F-2 rules, (i.e. does not include piping welds < 3/8" nominal wall).

			ANCHORS	SNUBBERS	RIGID RESTRAINTS	VARIABLE SUPPORTS	CONSTANT SUPPORTS
CLASS 1	CS	SYSTEM TOTALS..........	2	8	0	4	0
CLASS 1	FW	SYSTEM TOTALS..........	2	16	0	18	0
CLASS 1	HPCI	SYSTEM TOTALS..........	1	6	0	3	0
CLASS 1	MS	SYSTEM TOTALS..........	4	13	4	10	0
CLASS 1	RCIC	SYSTEM TOTALS..........	1	7	3	4	0
CLASS 1	RHR	SYSTEM TOTALS..........	7	50	3	24	0
CLASS 1	RR	SYSTEM TOTALS..........	0	30	6	8	6
CLASS 1	RWCU	SYSTEM TOTALS..........	2	42	21	11	0
CLASS 1	SLC	SYSTEM TOTALS..........	2	0	37	1	0
		TOTAL FOR ALL CLASS 1 SYSTEMS...					

NOTE:
ASME Code requires a minimum of 25% of Item F1.10, supports, be inspected each interval.

			ANCHORS	SNUBBERS	RIGID RESTRAINTS	VARIABLE SUPPORTS	CONSTANT SUPPORTS
CLASS 2	CRD	SYSTEM TOTALS..........	0	0	35	0	0
CLASS 2	CS	SYSTEM TOTALS..........	2	28	53	25	0
CLASS 2	FW	SYSTEM TOTALS..........	0	0	3	0	0
CLASS 2	HPCI	SYSTEM TOTALS..........	6	30	45	14	0
CLASS 2	MS	SYSTEM TOTALS..........	0	6	67	5	0
CLASS 2	RCIC	SYSTEM TOTALS..........	2	16	44	7	0
CLASS 2	RHR	SYSTEM TOTALS..........	12	158	151	72	0
		TOTAL FOR ALL CLASS 2 SYSTEMS...					

NOTE:
ASME Code requires a minimum of 15% of Item F1.20, supports, be inspected each interval.

			ANCHORS	SNUBBERS	RIGID RESTRAINTS	VARIABLE SUPPORTS	CONSTANT SUPPORTS
CLASS 3	ESW	SYSTEM TOTALS..........	47	7	315	8	0
CLASS 3	MS	SYSTEM TOTALS..........	39	50	47	49	0
CLASS 3	RHR SW	SYSTEM TOTALS..........	10	12	401	13	0
		TOTAL FOR ALL CLASS 3 SYSTEMS...					

NOTE:
ASME Code requires a minimum of 10% of Item F1.30, supports, be inspected each interval.

EXAMINATION CATEGORY F-A ITEM NO. F1.40 SELECTION TABLE

		ANCHORS	SNUBBERS	RESTRAINTS	SUPPORTS	SUPPORTS
CS	SYSTEM TOTALS..........	0	0	4	0	0
HPCI	SYSTEM TOTALS..........	0	0	1	0	0
RCIC	SYSTEM TOTALS..........	1	0	0	0	0
RHR	SYSTEM TOTALS..........	3	0	16	1	0
RPV	SYSTEM TOTALS..........	0	8	1	0	0
RR	SYSTEM TOTALS..........	2	12	0	6	0
TOTAL FOR ALL CLASS SYSTEMS...						

NOTE:
ASME Code requires a minimum of 100% of Item F1.40, supports other than piping supports, be inspected each interval .

EXAMINATION CATAGORIES
B-K, C-C, AND D-A
CODE CASE N-509 INTEGRALLY
WELDED ATTACHMENTS

	SYSTEM	CAT. B-K TOTAL WELDS	CAT. C-C TOTAL WELDS	CAT. D-A TOTAL WELDS
VESSELS	RPV	10	0	0
	RHR	0	16	0
PIPING	CRD	0	6	0
	CS	0	5	0
	ESW	0	0	75
	HPCI	0	14	0
	MS	4	1	47
	RCIC	0	4	0
	RR	4	0	0
	RHR	1	49	0
	RHR SW	0	0	35
PUMPS	RCIC	0	4	0
	RR	8	0	0
VALVES	NONE	0	0	0

NOTE:
1. Only systems having a component in the above population categories are listed.
2. Category D-A count does not include MS per N-509, Table 2500-1, Category D-A, Note 3, not subject to corrosion.
3. Category B-K selection for RPV shall include 10% of the length of welds FR and CG, examined fron RPV OD, and one stabilizer bracket attachment weld.
4. Total weld count shown for RHR vessels and RR pumps. Multiple Component Selection criteria applies. See Attachment A for component groups.

ATTACHMENT 1

This attachment previously contained a report titled:

Limerick Generating Station - Unit 1

Inservice Inspection Program
First Ten Year Inspection Interval
Code Edition Upgrade
1980/W81 to 1986

EXAMINATION RECONCILIATION REPORT

Prepared By:
Gilbert / Commonwealth, Inc.

In order to have a common Code Effective date with the LGS Unit 2 ISI Program, PECO Energy elected to perform a midterm program upgrade of the LGS Unit 1 ISI Program from ASME Section XI 1980 Edition with Addenda through Winter 1981 to the 1986 Edition.

The upgrade was implemented following the third refueling outage, 1R03, which was the first outage of the second period of the first ten year interval.

The Reconciliation Report was prepared to establish the extent of completion credit to be applied from the 1980/w81 Program to the 1986.

Since LGS Unit 1 has completed its first ten year inspection interval, the Reconciliation Report is no longer required.

ATTACHMENT 2

ASME SECTION XI
ISI PROGRAM POSITION PAPERS

TABLE OF CONTENTS

POSITION PAPER No. PSL-92-001

Revision 1

Subject: This ISI program position has been developed to provide guidance for the implementation of specific pressure testing requirements of the ASME Section XI Code.

Reference: This position is intended to be used in conjunction with the 1986 Edition of ASME Section XI.

Applicability: This position applies only to systems and components classified as "Class 1" for the purposes of applying the Section XI Code. See Position Paper PSL-92-002 for discussion of Class 2 and Class 3 systems.

Discussion: Deleted from Revision 3 of the Program. This position paper is NO LONGER REQUIRED. Position was incorporated into ASME Code Case N-566. See Appendix A, Relief Request Table RR-12-5 for details.

Position Paper No. PSL-92-002
Revision 1

Subject: This ISI program position has been developed to provide guidance for the implementation of specific pressure testing requirements of the ASME Section XI Code.

Reference: This position is intended to be used in conjunction with the 1986 Edition of ASME Section XI.

Applicability: This position applies only to systems and components which are classified as "Class 2 or 3" for the purposes of applying the Section XI Code. This position is not applicable to Class 1 components, except as discussed in PSL-92-001.

Discussion: Deleted from Revision 3 of the Program. This position paper is NO LONGER REQUIRED. Position was incorporated into ASME Code Case N-566. See Appendix A, Relief Request Table RR-12-5 for details.

Position Paper No. PSC-92-003
Revision 0
Supersedes Position Statement IPS # 87-002

Subject: This ISI program position provides guidance for the implementation of specific component support examination requirements of the ASME Section XI Code.

Reference: This position is for use in conjunction with the 1980 Edition of ASME Section XI, through the 1989 Edition.

Applicability: This position applies to all component supports which are within the jurisdiction of the Section XI Code. It pertains to nonintegrally attached supports which have bolted or other mechanical connections buried beneath the component insulation.

Discussion: ASME Section XI (subparagraph IWF-1300(e)) allows the component support visual examination boundary to extend from the surface of the component insulation, provided the nonintegrally attached support either carries the weight of the component or serves as a structural restraint in compression. This rule assumes that loss of integrity of the bolting or other mechanical connection buried beneath the insulation, and not accessible for visual examination, will become obvious on the external surface of the insulation. Therefore, the subject component support must be carrying the weight of the component at all times.

Position: If a component support required to be examined, is a nonintegrally attached support, and contains a mechanical connection which is buried beneath the component insulation, the insulation need not be removed provided the support carries the weight of the component (either in tension or compression), or acts as a restraint in compression during normal plant operations (per design calculations). Snubbers do not qualify for this position, since they do not carry the weight of the component.

If the bolted or other mechanical connection of a nonintegrally attached support is able to be examined without removing the insulation, then this position does not apply, since the insulation does not need to be removed.

Examples:

1. Component supports which qualify for use of this position paper.
   - spring hangers
   - rod hangers
   - spring supports
   - vertically oriented struts (above or below component)
   - frame supports or restraints (containing bolted or other mechanical connections) which bear the weight of the component
   - restraints acting in compression during normal operations

2. Component supports which do not qualify for use of this position paper.
   - snubbers
   - horizontal struts (not acting in compression)

Summary:

1. Component supports must carry the weight of the component (in tension or compression) or act as a restraint in compression during normal operations to qualify for this position paper.

2. Snubbers do not qualify for this position paper.

3. If the mechanical connection is not buried within the component insulation, then the insulation need not be removed.

Reference:

1. ASME Section XI Interpretation No. IN92-010, dated March 10, 1992.
2. ASME Section XI Interpretation No. IN92-006B, dated March 10, 1992.
3. ASME Section XI Position Statement IPS# 87-002, dated November 28, 1988.

Position Paper No. PSC-92-004
Revision 0
Supersedes Position Statement IPS # 91-001

Subject: This ISI program position provides guidance for the implementation of specific component support examination requirements of the ASME Section XI Code.

Reference: This position may be used in conjunction with the 1980 Edition of ASME Section XI, through the 1989 Edition. It may also be used in conjunction with the Limerick (LGS) 1 & 2 component supports examination program conducted in accordance with Section 9.1 of this ISI Program.

Applicability: This position may be applied to all component supports which are within the jurisdiction of the station ISI or alternate ISI Program (LGS) for component supports examination. It shall not be applicable to the examination and testing programs for snubbers, which are conducted in accordance with the plant Technical Specifications.

Discussion: ASME Section XI, subparagraph IWF-2430(a) requires additional examinations of component supports, if the results of regularly scheduled ISI examinations of component supports require corrective measures in accordance with the provisions of IWF-3000. Paragraph IWF-3122 provides four methods for acceptance of the results of examinations: IWF-3122.1 Acceptance by Examination; IWF-3122.2 Acceptance by Repair; IWF-3122.3 Acceptance by Replacement; and IWF-3122.4 Acceptance by Evaluation or Test.

Component support examination results which do not satisfy first-line screening or acceptance criteria, do not automatically require repair or replacement as described in subparagraphs IWF-3122.2 and IWF-3122.3 respectively. Generally, such unacceptable component supports may be found acceptable for continued/intended service via evaluation or test, as described in subparagraph IWF-3122.4. This acceptance by evaluation or test is not considered a corrective measure.

Occasionally, component support examination results found acceptable by evaluation or test may still require some minor rework of the support, often to avoid making unnecessary changes to the applicable design documents. Such rework, after the support has been determined to be acceptable for continued/intended service, is also not considered a corrective measure.

Position: Component supports found acceptable for continued/intended service[1] by evaluation or test (even though examination results deviate from screening/acceptance criteria), are not considered supports requiring corrective measures as referenced in IWF-2430a. Accordingly, additional examinations need not be performed. Additionally, minor rework of a component support after it's acceptance for continued service by evaluation or test, is also not considered a corrective measure, and does not require additional examinations to be performed.

Component supports found unacceptable for continued/intended service after evaluation or test do require additional examinations to be performed. Likewise, component supports requiring automatic rework, repair, or replacement as a result of examination results, will also require additional examinations to be performed.

1. For the purpose of this position paper, "continued/intended service" is defined as a condition within the design basis of the component, and the condition will not compromise the long term use of the component.

Examples:

1. A variable spring hanger has an as-found spring load setting which deviates from the design drawing specified setting by more than the tolerance allowed in the examination procedure. After evaluation, the as-found load setting is determined to be acceptable for intended and continued service, however the design drawing setting would then have to be revised. In lieu of revising the design drawings, the variable spring hanger is field adjusted (reworked) to the design load.

2. Examination of a variable spring hanger reveals that the spring can load scale (provided for measuring load setting) is missing. Subsequent evaluation of linear measurements recorded, indicate that the as-found spring position represents an acceptable load setting, and the support is acceptable for its intended service. Minor rework is conducted however, to attach a new load scale to the spring can. Additional examinations of IWF-2430a are not required in this situation.

3. Examination of a rigid sway strut reveals a completely cracked weld between the strut body and the extension tube. Evaluation (if conducted) reveal that the support would not fulfill its intended function in a tension mode. Repair of the support is required. Additional examinations are required in this situation.

Summary:

1. Component supports with unacceptable examination results which are found to be acceptable for continued/intended service by evaluation or test, do not require the examination of additional supports.

2. Minor rework performed on component supports found acceptable for continued/intended service need not be considered corrective measures. Additional examinations do not need to be performed.

Reference:

1. ASME Section XI Interpretation No. XI-1-86-30 dated April 30, 1986.
2. ASME Section XI Position Statement IPS# 91-001, dated December 30, 1991.

Position Paper No. PSL-95-001
Revision 0

Subject: This ISI Program Position provides guidance for the implementation of examination requirements of the ASME Section XI Code for components added to, or deleted from the facility during the inspection interval.

Reference: This position may be used in conjunction with the 1980 Edition of ASME Section XI, through the 1995 Edition.

Applicability: This Position Paper applies to Limerick Generating Station, and is applicable to the examination requirements imposed by ASME Section XI through the ISI Program. It is not applicable to the examination requirements imposed by other source documents through the Augmented Inspection Programs.

This position supercedes position PSC-95-001, Revision 1 in its entirety.

Discussion: ASME Section XI, sub-subparagraphs IWB, IWC, IWD, IWE-2412, and subparagraph IWF-2410 discuss implementation of the required examinations during the course of the ten year inspection interval, using Inspection Program B. The referenced Code Editions do not contain discussions on how to deal with components which may be added to, or removed from the facility during the course of the inspection interval. Since components are frequently added to, or deleted from the total population of items subject to ISI examination as a result of plant modifications, a standard approach for dealing with these components is needed.

Position: Added Components:

When components (e.g. items or welds) are added to the facility (and the ISI Program) during the course of the ten year inspection interval (i.e. the added components increase the total population of items subject to ISI examination), the following guidance will apply for determining the selection of components for ISI examination during the remainder of the inspection interval:

The selection bases are not required to be revised as a result of changes in plant configuration from repairs, replacements, or modifications throughout the inspection interval. Repaired, replaced or modified components shall have a Section XI baseline inspection performed as part of the ASME Repair and Replacement Program. The baseline inspections are sufficient to satisfy the ISI requirements for those initially selected components that have been repaired, replaced or modified but had not yet been examined during the inspection interval. Due to the requirement to perform baseline inspections on repaired, replaced or modified components, a reconciliation of the selection bases with changes in plant configuration is only required as part of the ISI Program update at the beginning of each inspection interval.

Deleted Components:

When components (e.g. items or welds) are removed from the facility (and deleted from the ISI Program) during the course of the ten year inspection interval (i.e. the deleted components decrease the total population of items subject to ISI examination), the following guidance will apply for determining the selection of components for ISI examination during the remainder of the inspection interval:

- If the items or welds have been examined as part of the ISI Program prior to their removal from the plant, credit toward satisfaction of the ten year inspection interval requirement shall be retained.

- If the items or welds have not been selected and scheduled for examination during the inspection interval, and their deletion from the program reduces the total number of examinations required to be performed during the interval, then components may be deselected. Scheduled examinations for these deselected components may be cancelled.

- If the items or welds have not yet been examined, but were selected and scheduled to be examined at a later date, reselections may be required. The need for reselection would be based on the remaining population of components within the specific Examination Category and Item Number. If the total examination requirement for the remaining population of components does not reduce, then alternate components must be selected and scheduled for examination in lieu of those which have been deleted.

Summary:

1. Component (items or welds) added to the ISI Program during the inspection interval, need not be scheduled for ISI examination during the remainder of the inspection interval since the added components will receive a baseline inspection as part of the ASME Section XI Repair and Replacement Program.

2. Components which are repaired during the course of an inspection interval (i.e. no impact on total population of components subject to examination), are not applicable to this position.

3. Completion credit shall be retained for components which have been examined, to satisfy ISI

Program requirements, prior to removal from the facility.

4. Components which are deleted from the facility during the course of the inspection interval (i.e. reduce the total population of components subject to ISI examination) can be directly reconciled with the total inspection interval examination requirements.

   NOTE - Periodic system pressure testing is not affected by the addition/deletion of components. Previously completed periodic system pressure tests need not be reperformed following additions/deletions of components, provided breeches of the pressure retaining boundary are pressure tested as part of the modification causing these breeches. The original schedule, for subsequent periodic pressure tests of the affected components, may be maintained.

Reference:

1. ASME Section XI Code, 1992 Edition, through the 1994 Addenda.
2. ASME Code Foreword, 1992 Edition, including the 1992 Addendum.

Position Paper No. PSC-98-001
Revision 0

Subject: This ISI Program position provides guidance for the performance of ASME Section XI pressure testing of the Reactor Coolant Pressure Boundary.

Reference: This position is for use in conjunction with the 1980 Edition of ASME Section XI, through the 1995 Edition.

Applicability: This position applies to the:

1. ASME Section XI, Table IWB-2500-1, Periodic pressure tests of systems comprising the RCPB following each refueling outage.
2. ASME Section XI, IWA-5214/Code Case N-416-1, Repair and Replacement pressure tests of affected RCPB components.
3. Procedure A-C-26, non-ASME Code, Post Maintenance Tests (PMT), of the RCPB systems or components.

Discussion: This position was originally documented in a white paper prepared for use during 1R07 and 2M23 (both in 1998) at Limerick Generating Station. The paper, which has been presented to and accepted by the LG Leadership Team with copies submitted to the LG USNRC Resident Inspector and Region I, is reprinted here in its original format.

Position:

PECO ENERGY
LIMERICK GENERATING STATION
PEACH BOTTOM ATOMIC POWER STATION

ASME SECTION XI PRESSURE TESTING AND CORE CRITICALITY

EXECUTIVE SUMMARY

A 1996 revision to 10CFR50 Appendix G, "Fracture Toughness Requirements" for the reactor coolant pressure boundary (RCPB), included an explicit prohibition against completing ASME Section XI pressure tests after initiating core criticality (i.e. nuclear heat may not be used to achieve pressurization). This prohibition was the result of issues involving pressure testing at Plant Hatch, and related backfit appeals by NUBARG in the late 1980's and early 1990's. This issue resurfaced in 1997 when Quad Cities Station performed an ASME Section XI leakage test of the "RPV" after startup from a refueling outage (i.e. the OPS HYDRO was performed using nuclear heat). The NRC issued Information Notice 98-13 to remind licensees of the prohibition on core criticality prior to the completion of Section XI pressure tests.

Although 10CFR50 Appendix G applies to the entire RCPB (i.e. all ASME Class 1 systems), it is important to note that the specific prohibition applies only to ASME Section XI pressure testing of the "Reactor Vessel". It is also noted that during the 1996 Rule Making process the ACRS and NEI both took specific exception to the Staff's prohibition.

At issue are the three (3) types of pressure tests that include the RPV in the boundary of the test:

1. ASME Section XI, Table IWB-2500-1, Periodic pressure tests of <u>systems</u> comprising the RCPB following each refueling outage.
   - The ASME Code clearly requires that these pressure tests be performed "prior to plant startup" from each refueling outage.
   - Core criticality (i.e. nuclear heat) <u>may not</u> be used to achieve pressurization.

2. ASME Section XI, IWA-5214/Code Case N-416-1, Repair and Replacement pressure tests of affected RCPB <u>components</u>.
   - The ASME Code allows the component(s) to be tested "<u>prior to or immediately upon return to service</u>".
   - Per the ASME Code, either flood-up and pressurization from external sources (non-nuclear heat) or core criticality (nuclear heat) <u>may</u> be used for pressurization.

3. Procedure A-C-26, non-ASME Code, Post Maintenance Tests (PMT), of the RCPB systems or components.
   - Pressure tests, performed solely to meet A-C-26 PMT requirements, are not subject to the prohibitions of 10CFR50 Appendix G.
   - By procedure, either flood-up and pressurization from external sources (non-nuclear heat) or core criticality (nuclear heat) may be used for pressurization.

In all applicable documentation, the USNRC Staff has identified this as a compliance issue. However, the Staff's basis for their position that ASME Section XI pressure testing provides assurance that GDC-14 requirements to design, fabricate, erect and test the RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture have been met, cannot withstand technical challenge. Extensive research by the ASME has shown that Section XI pressure testing of the RCPB is adequate for determining leak tight integrity but is of little to no value for determining structural integrity. RCPB structural integrity is best measured by the periodic performance ASME Section XI surface and volumetric nondestructive examinations and best protected by adherence to T.S. operating limits.

The Staff's risk determination also cannot withstand technical challenge. A review of the risks involved in performing ASME Section XI pressure tests by flood-up and pressurization from external sources (non-nuclear heat) has concluded that this type of testing represents the most significant challenge to plant safety. Although there is no USQ involved with the conduct of the pressure test; the margin to brittle fracture is reduced; the highest operating stresses are imposed; an additional thermal cycle is imposed; fatigue usage is increased; infrequently used/abnormal system alignments are required and; the potential for violations of Plant Technical Specifications is increased.

Based on our analysis we concur with the USNRC Staff that the issue of ASME Section XI pressure testing and core criticality is one of compliance and not an issue involving nuclear safety provided we remain in compliance with the applicable Codes, Standards and Regulations. Although permitted by the ASME Code, to ensure compliance with 10CFR50 Appendix G, for ASME Section XI Repair and Replacement pressure tests (Type 2 tests, above) to the RPV shell (pressure retaining membrane excluding bolting), out to and including the safe ends, and interior attachment welds to the RPV shell (pressure retaining membrane only, internal components excluded) shall not use core criticality (nuclear heat) to achieve pressurization.

A review of previous and current pressure testing activities at both PB and LG has determined that PECO Energy, as summarized above, has complied and shall remain in compliance with all requirements related to the issue ASME Section XI pressure testing and core criticality, including the 1996 revision to 10CFR50 Appendix G.

Prepared by:

R. E. Ciemiewicz

R. J. McCall

R. E. Simpson

D. L. Schmidt

BACKGROUND

Effective January 18, 1996, the NRC issued a final rule (Ref. 2) which revised Appendix G of 10 CFR Part 50. This revision included a clarification to explicitly prohibit the initiation of core criticality prior to completing ASME Section XI required pressure tests and leak tests. Of the reactor vessel Specifically, Paragraph IV.2.d (of Appendix G) states: "Pressure tests and leak tests of the **"Reactor Vessel"** that are required by Section XI of the ASME Code must be completed before the core is critical" [emphasis added]. This explicit prohibition was apparently deemed necessary as a result of issues which arose on the topic in the late 1980's and early 1990's. The issues involved ASME Section XI pressure testing at Plant Hatch, and related backfit appeals by NUBARG. Recently, this topic has resurfaced via the issuance of USNRC Information Notice (IN) 98-13 (Ref. 4). This IN described an event which occurred at the Quad Cities Station in 1997. The event involved performance of the ASME Section XI required post refueling outage pressure test after initiation of core criticality. This pressure test is required by the Code in order to complete Examination Category B-P visual examination requirements. Core criticality prior to completion of this pressure test was clearly prohibited by 10CFR50 Appendix G at the time of the Quad Cities event The IN was intended to remind licensees of the prohibition on core criticality prior to the completion of Section XI pressure tests.

PRESSURE TESTING REQUIREMENTS

ASME Section XI (Editions applicable to PECON, Ref. 1) requires the following pressure testing of the reactor coolant pressure boundary (RCPB):

a) A system leakage test following each refueling outage, prior to plant start-up (Examination Category B-P). A system hydrostatic test must be substituted for this system leakage test once every ten years, at or near the end of the interval

The ASME Code clearly requires these tests to be performed "prior" to plant startup.

Flood-up and pressurization from external sources (non-nuclear heat) is required.

Core criticality (i.e. nuclear heat) may not be used to achieve pressurization.

Once this test is performed, the Code requirement is satisfied for the entire fuel cycle.

b) A pressure test (hydrostatic, leakage, functional, or inservice) following certain Section XI repairs and replacements (IWA-5214 & Code Case N-416-1).

Only required when a non exempt (from pressure testing) ASME Section XI repair or replacement has been performed.

Test boundary may be limited to the specific component repaired or replaced.

Code allows the component(s) to be tested "prior to or immediately upon return to service".

Code does not specify the method for achieving pressurization. Therefore, flood-up or core criticality may be used for pressurization.

10CFR50, Appendix G applies to the pressure boundary of the RCPB. Therefore, limitations on the Code pressure tests shall apply only to Code repairs and replacements of the RPV shell (out to and including the nozzle safe ends) and interior attachments to the shell. Either flood-up or core criticality may be used for pressurization to conduct testing of Code repairs and replacements of RPV internal components.

Repair or replacement of bolting of the RCPB is under the jurisdiction of the Code rules, however bolting is exempt from the Code pressure testing requirements. Therefore, flood-up or core criticality may be used for pressurization to conduct testing of RCPB mechanical connections associated with bolting repair or replacement.

Additionally, Procedure A-C-26, non-Code, Post Maintenance Tests (PMT) of the RCPB may be needed.

Opening / closing of the RCPB at a mechanical connection, without repairing or replacing pressure boundary components, is not a Code repair or replacement. Therefore, Code pressure testing does not apply.

Pressure tests performed solely to meet A-C-26 PMT requirements are not subject to the prohibition of 10CFR50, Appendix G.

Either flood-up or core criticality may be used for pressurization to perform PMT pressure tests.

The removal of a fuel bundle from the RPV during a planned or forced maintenance outage is not a Code repair or replacement, nor is it a refueling outage activity. Therefore Code repair /

replacement and periodic pressure testing is not required.

The Section XI Code rules for pressure testing do not contain direction regarding core criticality. When asked the specific question regarding use of nuclear heat (via the Code Inquiry Process), the Code Committee's official response was that the Code did not address such issues (Ref. 10).

USNRC STAFF BASIS

In the NRC response to the NUBARG appeal (Ref. 5), the NRC denied the appeal and provided further clarity as to their intent regarding the prohibition on core criticality. The basis of the Staff position is as follows:

Defense in Depth: Any "testing" of a barrier used for the prevention of accidental release of fission products is inappropriate using nuclear power (i.e. core critical)

Examinations: The quality of examinations conducted under the conditions of core criticality are potentially questionable, considering the temperature, radiological, and access conditions.

PECON RECOMMENDATIONS

As required by 10 CFR 50, Appendix G, core criticality shall not be initiated before the completion of the following ASME Section XI required pressure tests:

1) Post refueling outage leakage or periodic hydrostatic test (Examination Category B-P),
2) Pressure tests required by ASME Section XI following repairs or replacements associated with the reactor vessel shell (nozzle safe ends inward).

All other pressure tests of the RCPB can be conducted after initiation of core critical.

Examples- Replacement of Main Steam Relief Valves following the completion of the Examination Category B-P testing.

ASME Section XI repair or replacement of mechanical connections associated with lines connecting to the reactor vessel (e.g. head vent line).

Subsequent non-refueling outage breaches of the RCPB, where no ASME Section XI repair or replacement has been performed (e.g. change-out of leaking fuel element).

PECON BASIS

The explicit prohibition of 10CFR50 Appendix G applies to ASME Section XI pressure testing of the reactor vessel. Other ASME Section XI pressure tests required for components outside the reactor vessel (but within the RCPB) are not subject to the prohibition.

Defense in Depth: Following completion of the periodic pressure test required by ASME Section XI, Examination Category B-P or IWA-5214, requiring sub-critical core conditions prior to conducting other pressure tests of the RCPB would result in additional thermal cycling of the reactor vessel. This would represent an unnecessary challenge to the vessel from both a fatigue usage and brittle fracture margin perspective.

Examinations: Examinations of portions of the RCPB are reasonable to perform successfully, even under core critical conditions, since access and ambient temperatures are not significantly different prior to and following criticality. Radiation exposure for the smaller scope of examinations performed at low power levels is not a concern.

Operational: Maintaining applicable Mode conditions (i.e. no core criticality) to conduct all pressure tests of the RCPB can result in unnecessary cycling of the RCPB and unnecessary operation of associated components due to Mode limitations. This can contribute to degradation of the structural components, which is contradictory to the goal of safe operation.

Test Performance: Verification of leak tight integrity may require the benefit of full thermal expansion, which is not achievable with artificial heating. Installation of alternate heating sources, to achieve full thermal expansion, is not warranted.

REGULATORY RISK

The Final Rule's prohibition explicitly applies only to the "reactor vessel", and only during required Section XI pressure testing. The ASME Section XI Code requirements do not explicitly prohibit core criticality. Current Code requirements and Interpretations (Ref. 11 through13) for pressure testing imply core critical conditions may be used (i.e. Inservice Pressure Test, by definition, requires system to be operating). Additionally, the ACRS response to the proposed Final Rule (Ref. 6) included a recommendation to revisit the prohibition, as they felt it could not be justified in terms of risk. Nevertheless, after review of all documents associated with this issue, NRC Staff challenge for use of nuclear heat to conduct any pressure testing of any portion of the RCPB, cannot be ruled out. Undocumented discussions with the Staff and industry experts indicated some disagreement about the appropriate extent of the subject prohibition.

REFERENCES

1. ASME Boiler & Pressure Vessel Code, Section XI, Rules For Inservice Inspection Of Nuclear Power Plant Components;
   1980 Edition w/ Addenda through Winter 1981 (PBAPS),
   1986 Edition (LGS).
   1995 Edition w/ 1996 AddendumCode Edition currently proposed for NRC endorsement per 62FR63892, dated 12/3/97

2. Federal Register, (60FR65456, dated December 19, 1995); Final Rule 10 CFR Part 50, "Fracture Toughness Requirements for Light Water Reactor Pressure Vessels".

3. SECY-95-205, Revisions to Regulatory Requirements for Reactor Pressure Vessel Integrity in 10 CFR Part 50, dated August 4, 1995.

4. NRC Information Notice 98-13: Post-Refueling Outage Reactor Pressure Vessel Leak Testing Before Core Criticality, dated April 20, 1998.

5. NRC letter: James M. Taylor, NRC Executive Director for Operations, to Messrs. Reynolds and Stenger of the Nuclear Utility Backfitting and Reform Group (NUBARG), dated February 2, 1990.

6. Advisory Committee on Reactor Safeguards (ACRS) letter to J. M. Taylor, "Proposed Final Rule and Regulatory Guide for Fracture Toughness Requirements for Light Water Reactor Pressure Vessels", dated June 16, 1995.

7. ASME Section XI Code Case N- 515, Class 1 Mechanical Joint Pressure Tests Section XI, Division 1.

8. ASME Section XI Code Case N-508-1, Rotation of Serviced Snubbers and Pressure Relief Valves for the Purpose of Testing Section XI, Division 1.

9. Draft Regulatory Guide DG-1050 (Revision 12 to Regulatory Guide 1.147), May 1997.

10. ASME Section XI Code Inquiry XI-1-86-53 dated 2/11/97.

11. ASME Section XI Code Case N-416-1, Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2 and 3.

12 ASME Section XI Code Case N-498-1, Alternative Rules for 10-Year System Hydrostatic Testing for Class 1, 2 and 3 Systems.

13. ASME Section XI Code Inquiries:

a. XI-1-83-25 dated 10/27/83, Use of steam for pressure test
b. XI-1-83-37R2 dated 9/19/89, hydro testing repair & replacements
c. XI-1-83-70R dated 1/26/90, leakage test disassembled components
d. XI-1-86-13R dated 6/10/91, pressure tests disassembly / reassembly
e. XI-1-86-34 dated 9/18/86, Examination Category B-P pressurizing medium
f. XI-1-89-08 dated 11/14/88, pressure test for replacement of bolts
g. XI-1-89-15 dated 5/17/89, pressure tests Class 1 components
h. XI-1-89-33 dated 2/7/90, pressure testing after disassembly
i. XI-1-92-23 dated 3/16/92, Examination Category B-P pressurizing medium
j. XI-1-92-30 dated 5/22/92, IWA-5214 pressure tests
k. XI-1-92-65 dated 2/7/94, pressure test, replacement of bolting
l. XI-1-95-50 dated 11/12/96, required system pressure tests

POSITION PAPER No. PSL-00-001

Snubber Operability and LCO Log

The purpose of this engineering evaluation is to document the process that is currently being used to control the Tech Spec 3/4.7.4, Snubber, Allowed Outage Time and to verify that these additional administrative controls meet the requirements of approved Operations and Maintenance procedures.

The structure of snubber related Work Orders as planned in PIMS does not provide for efficient control of the Tech Spec 3/4.7.4, Snubber, Allowed Outage Time. Accordingly the Snubber LCO Log, a LAN based system similar to a Narrative Log, has been developed for this purpose.

From this review it was concluded:

Snubber related Work Order Activities should be statused in PIMS in accordance with the Maintenance/Outage Schedule.

The actual equipment (snubber) status shall be documented in the Snubber LCO Log by entries made by the Work Group.

The LCO/TS Action and ST/RT Logs in the MCR require only a single entry at the beginning of the Snubber Inspection (outage) and may remain open until the end of the Inspection at which time they may be closed.

The start of the Snubber AOT shall be the "Pin Out" time entered in the Snubber LCO Log.

The end of the Snubber AOT shall be the "VT-3 SAT" time entered in the Snubber LCO Log.

The PMT for all snubbers removed for testing or maintenance work is a VT-3 Visual Examination after reinstallation.

An entry in the Snubber LCO Log documenting the satisfactory completion of the PMT is sufficient to return the snubber to operable status.

BASES:

NOM-C-8.4, Unavailable Equipment/Equipment Release

Q or Tech Spec snubber activities do not require a clearance (Section 4.4). Levels of administrative control, comparable to a Clearance, are provided by the Work Order Activity Description, LCO/TS Action Log and Snubber LCO Log.

Unavailability Review (Section 5.3) – Unavailability Reviews have been completed for 100% of the snubber population. Engineering and at least two (2) members of Shift Management performed the review. Unavailability Reviews do not require re-review each outage unless there is a Modification involving the addition/deletion of snubbers or other supports within the Snubber Inspection System (SIDS No.). Details of the review are documented in Work Order Activity Description "Special Clearance Requirements".

Two (2) members of Shift Management, one of whom is the Control Room Supervisor, shall authorize snubber Work Order Activities.

Shift Management authorization of snubber Work Order Activities releases the snubber to the Work Group but shall not be considered as starting the TS 3/4.7.4 Allowed Outage Time (AOT). See NOM-C-11.1, Operability, below.

Shift Management authorization of the first snubber Work Order Activity for the Snubber Inspection shall require an entry to be made in the Regulatory Action (LCO/TS Action) Log. See NOM-L-6.4, Section 12.1, Regulatory Action, below.

NOM-C-11.1, Operability

Affected SSC's are identified by the Snubber Inspection System (SIDS No.) and Snubber LCO Log No.

SSC's required to be operable, remain operable with the snubber removed during the 72 hour Allowed Outage Time (AOT) provided by TS 3/4.7.4. See USNRC position, below.

The snubber AOT shall start with the "Pin Out" time recorded in the Snubber LCO Log and shall end with the "VT-3 SAT" time also entered in the Snubber LCO Log.

USNRC Position on Snubber 72 Hour Action Statement:

If a snubber is removed from its installation for testing, the action requirement for the supported system is not applicable, as long as the 72 hours limit is not exceeded. The snubber could be removed one at a time and replaced, or all snubbers are removed at the same time, and replaced as a group at the same time. The Tech Spec operability requirements do not require consideration of supported system redundancy or impact until the snubbers are out of service in excess of 72 hours. The supported system will have to be declared inoperable if the 72 hours limit is exceeded.

If a snubber is found to be inoperable, the action statement requires that an engineering evaluation of the supported system be performed or that the supported system be declared inoperable immediately.

The engineering evaluation is to determine if the supported system was affected by the inoperable snubber. This does not relate to the capacity of the attached system to withstand a seismic event.

If the results of the evaluation show, prior to the end of the 72 hours, that the supported system was made inoperable by the inoperable snubbers and those snubbers were not restored or replaced, then the supported system Tech Specs should be entered then, and not at the end of the 72 hours.

Note: when a snubber is removed from service for the purpose of testing, an engineering evaluation is not required.

When a supported system LCO is not met due to a support system LCO not being met, only the support system, and not the supported system LCO actions are required to be entered.

NOM-C-8.1, Equipment Status Control and Documentation

The Plant Information Monitoring System (PIMS), Regulatory Action (LCO/TS Action) Log and the Snubber LCO Log shall be used to control and document the status of snubber activities on SSC's required to be operable.

The most timely information is provided by the Snubber LCO Log.

NOM-L-6.4-1, ST/RT Status Log
Requires only one entry per ST per Snubber Inspection (OUTAGE). The entry may remain open for the duration (until the end) of the Inspection at which time they may be closed.
ST-4-103-000-0, Generic Snubber Visual Inspection
ST-4-103-301-1(2), Snubber Functional Test
ST-1-103-300-1(2), 24 Month Snubber Functional Test Program
ST-1-103-990-1(2), Snubber Service Life Monitor

NOM-C-6.2, Narrative Logs / Scope of Entry

The Snubber LCO Log is analogous to Narrative Logs for the CRS, ACRS, the ROs, and the PRO.

The Snubber LCO Log is available to be viewed in the MCR by CRS, ACRS, the ROs, and the PRO on the LAN which is accessible from terminals in the MCR.

The Snubber LCO Log includes the following information for each snubber:
Outage number
Snubber number
Work Order No.
Date and time the snubber was removed for testing or maintenance
Calculates and displays the 72 hr AOT
Date and time snubber was replaced
Status of visual/functional examinations, SAT/UNSAT
The supported system
Other affected systems

The Snubber LCO Log is updated by the Work Group (Station Maintenance). This pdate will always precede updates in PIMS W/O Completion Remarks (CREM)

NOM-L-6.4, Section 12.1, Regulatory Action

Upon authorization of the first TS Work Order for snubber inspections/maintenance, Shift Management shall make an entry in the LCO/TS Action Log.

The entry shall direct the reviewer to the Snubber LCO Log for the status of a specific snubber.

Only one (1) entry in the LCO/TS Action Log is required for all inspections/maintenance to be performed on TS snubbers.

The LCO/TS Action Log entry may be closed when the last TS snubber Work Order is in "ACT COMPLETE" status in PIMS at the end of the Refueling Outage.

AG-CG-26.4, Work Order (W/O) Work Performance

Snubber Work Order Activities may be taken to "SCHED" or "INPROG" status in PIMS any time following authorization by Shift Management. "SCHED" or "INPROG" status in PIMS shall not be considered as starting the TS 3/4.7.4 Allowed Outage Time (AOT).

The entering, by the Maintenance Work Group, of the snubber "PIN OUT" time in the Snubber LCO Log shall constitute:
the start of the 72 Hour Allowed Outage Time
Prior notification to Shift Management of the actual start of work.

See NOM-C-11.1, Operability, above.

AG-CG-26.6, Post Maintenance/Modification Testing (PMT)

Snubber Post Maintenance Testing is included in the PIMS Component "C" Type Activity for the removal/maintenance/reinstallation of a particular snubber.

Completion of the Post-Installation Verification section of procedures M-200-043, M-200-044 or M-200-045, as applicable to the size and make of the specific snubber shall be considered as completion of the PMT for that snubber.

NOM-C-8.5, Equipment Return to Service

A member of Shift Management shall authorize the return of equipment or systems back to an Operable status. after confirmation of the following information as appropriate:

A snubber is considered returned to service and operable status following the satisfactory completion of the PMT based on the following:

Proper housekeeping and equipment condition has been verified by the PMT.

Completion of a Work Order search is not required since the PMT is required to be completed satisfactorily and deferral of the VT-3 Visual examination is not permitted.

The PMT is the final required Surveillance Test for that snubber and its satisfactory completion indicates the snubber is in surveillance.

Required Independent/Double verifications have been completed as specified by individual procedure and NOM-C-9.1 and NOM-C-9.4.

The system/equipment has been walked down as appropriate to verify that it can be safely operated to fulfill its design function.

The snubber is considered a support system and does not require other auxiliary and support systems for operability.

No Operating Procedures, including COLs, are required to return equipment/systems to service.

No compensatory actions other than completing the snubber work within the snubber 72 Hour AOT are taken or needed to be closed out.

Position Paper No. PSL-01-001
Revision 0

Successive Inspections

Subject: This ISI program position provides guidance for determining the practicality of performing component examinations during successive inspection intervals following the sequence of examinations established during the first inspection interval.

Reference: This position may be used in conjunction with the 1980 Edition of ASME Section XI, through the 1995 Edition.

Applicability: This position may be applied to all Class 1, 2, 3 and MC component and their supports subject to the successive inspection requirements of ASME Section XI, IWX-2420.

Discussion: ASME Section XI, IWX-2420 requires that, "to the extent practical ", the sequence of component examinations established during the first inspection interval be repeated during each successive inspection interval.

Position: There are a number of factors that affect the practicality of repeating the sequence of component examinations established during the first inspection interval. The list of factors below is not intended to be all-inclusive rather it serves to illustrate the potential need to modify the sequence of examinations in a manner that optimizes, safety, radiological, scaffolding, insulation removal, or other considerations.

1. Revised ASME Code requirements that may result from the 10-year update of the ISI Program.
2. Revised Regulatory requirements that may result from new or revised generic issues or rule making.
3. Plant repairs, replacements or modifications that may result in the addition or deletion of components or new baseline examinations.
4. An increased length of the typical fuel cycle that results in a decrease in the number of refueling outages in a 10-year inspection interval.
5. The reselection of components to maximize examination coverage or to address safety and radiological issues.

Position Paper No. PSL-01-001
Revision 0, continued

Summary:

Factors exist that add, delete or otherwise change the components requiring examination. These factors affect the inspection strategy established during the first inspection interval and thus the practicality of repeating the sequence of examinations during subsequent inspection intervals. To the extent practical, the sequence of component examinations established during the first inspection interval shall be repeated during each successive inspection interval. However, the practicality of the inspection strategy may make it necessary to modify or otherwise establish a new sequence of examinations in a manner that optimizes, safety, radiological, scaffolding, insulation removal, or other considerations. In all cases, existing, modified or new inspection sequences shall satisfy the interval percentage requirements established by the ASME Code.

Reference:

1. ASME Section XI Code Case No. N-624, Successive Inspections, dated May 7, 1999.

ATTACHMENT 3

ASME SECTION XI
SUBSECTION IWE AND IWL
ACCEPTANCE CRITERIA

Purpose: This attachment describes in detail the acceptance criteria for the inspection of Class MC and Class CC components.

Criteria:

For Class MC components, if flaws are detected, flaw evaluations shall be performed in accordance with the Code, IWE-3000, as detailed by the following standards.

1. Standards for Examination Category E-A, Containment Surfaces

   A. IWE-3510.1 Visual Examinations – General

      This requirement states that," (a) the General Visual Examination shall be performed by, or under the direction of, a Registered Professional Engineer or other individual knowledgeable in the requirements for design, inservice inspection, and testing of Class MC and metallic liners of Class CC components. The examination shall be performed either directly or remotely, by an examiner with visual acuity sufficient to detect evidence of degradation that may affect either the containment structural integrity or leak tightness".

      IWE-3510.1 does not provide any specific requirements other than visual acuity, for the qualification of personnel performing the General Visual Examination. Personnel performing the General Visual Examination should be familiar with or have training on the type of containment being examined. The LGS QA program is followed for the qualification of the inspection personnel performing the General Visual Examination. It is acceptable to use Visual, VT-3 certified personnel for the General Visual Examinations if they meet the requirements of IWE-3510.1, and have had additional instruction on containment inspection as specified by the individual directing the program required under IWE-3510.1.

   B. IWE-3510.2 and IWE-3510.3 Visual Examinations – VT-3

      IWE-3510.2 and IWE-3510.3 address acceptance criteria for Visual, VT-3 examination of coated and non-coated surfaces.

2. Standards for Examination Category E-B, E-C, E-D, and E-F

   IWE-3511, IWE-3512, IWE-3513, and IWE-3514 address acceptance criteria for Categories E-B, E-C, E-D, and E-F.

3. IWE-3515 Standards For Examination Category E-G, Pressure-Retaining Bolting

   Acceptance criteria for visual examination of pressure-retaining bolting are addressed in IWE-3515.1. It states that bolting materials shall be examined in accordance with the material specification for defects that may cause the bolted connection to violate either the leak tight or structural integrity. Material specifications only provide for material properties, e.g., tensile strength, hardness, etc., and fabrication discontinuities. For inservice bolting, acceptance criteria specific to inservice discontinuities, e.g., corrosion, cracking, etc., are most applicable. The utility can use the material specifications or develop an alternative for acceptance criteria. Performing an evaluation to show that the Visual, VT-1 acceptance criteria of Examination Category B-G-1 or B-G-2 is equivalent or better than the material specification may be the best alternative for establishing pressure-retaining bolting acceptance criteria.

4. Reportable Conditions for Coatings on Class MC Components

If evidence of degradation is detected during the general visual examination, perform a detailed visual examination and determine (by a visual comparison to the applicable ASTM standard) if any of the conditions listed below are present. Any one of the conditions listed below shall be identified to Engineering.

A. Blistering GREATER THAN size No. 6 (medium) as specified in ASTM D 714
B. Checking GREATER THAN standard No. 2 as specified in ASTM D 660
C. Cracking GREATER THAN standard No. 6 as specified in ASTM D 661
D. Flaking GREATER THAN standard No. 6 as specified in ASTM D 772
E. Rusting EQUAL TO OR GREATER THAN Grade 7 as specified in ASTM D 610

NOTE: Rust staining, accumulated dirt, or dirt containing iron compounds should not be confused with the actual rust involved. Only rusting of the pressure-retaining material is to be considered.

5. Reportable Conditions for Class MC Components

If evidence of degradation is detected during the general visual examination, PERFORM a detailed visual examination and DETERMINE if any of the conditions listed below are present. Any one of the conditions listed below shall be identified to Engineering.

A. Excessive Corrosion/Pitting (generally represented by dark discoloration (red/brown), spalling from swelling, rust ejection, deep pits, and/or other severe manifestation)
B. Deep Gouges or Dents (excluding fabrication or installation marks)
C. Excessive Wear (generally represented by shiny surfaces, ridges, evidence of motion, or other material wastage)
D. Bulging of the Liner (a separation of the metallic liner from the reinforced concrete structure)
E. Other Damage, Deformation, or Degradation (any condition (e.g., cracks, arc strikes, tears, broken welds, etc.) that is not listed above and may be detrimental to the material condition within the examination boundary).

NOTE: Degradation that is not detrimental to the pressure-retaining boundary, such as general corrosion or light surface pitting is acceptable and need NOT be recorded.

For Class CC - Concrete Containment Components:

1. Subarticle IWL-2510 states that "Concrete surface areas, including coated areas, except those exempted by IWL-1200 (b), shall be VT-3C visual examined for evidence of conditions indicative of damage or degradation, such as defined in ACI 201.1 R-92, in accordance with IWL-2310 (b). Selected areas, such as those that indicate suspect conditions, shall receive a VT-1C examination in accordance with IWL-2310 (a).

   Although this subarticle mandates that examination of concrete surfaces for evidence of conditions indicative of damage or degradation and a detailed examination of suspect areas, no prescriptive criteria is specified for recording conditions that may be indicative of damage or degradation or what constitutes a suspect condition.

   The objective of this part of the ISI Program is to establish a consistent approach in:

   A. recording conditions that should be monitored;

   B. defining what conditions constitute a suspect areas that require a detailed or VT-1C examination, and;

   C. defining conditions that are to be investigated by the IWL Responsible Engineer.

   Since Subarticle IWL-2510 references the use of ACI 201.1R-92, a similar format to that outlined in this document was used. As an additional aide, the corresponding identifier (e.g., A.1, A.2.1) and applicable photograph(s) specified in document were referenced.

2. ***A.1 CRACKS*** - a complete or incomplete separation, of either concrete or masonry, into two or more parts produced by breaking or fracturing. The different types (e.g., pattern, checking, hairline, D-cracking) of cracking are illustrated by photographs in ACI 201.1R-92 (see Figures(A.1.6a-f, A.1.1, A.1.3)

   Cracking of the concrete cover is a common mechanism for any concrete structure. This condition is normally a result of normal expansion and contraction that occurs within the concrete due to variations in temperature and stress.

   Passive cracks observed in the concrete cover are acceptable for continued service and do not warrant a review by the IWL Responsible Engineer. Passive cracks are defined as those having an absence of growth (when compared to the baseline examination results) and absence of other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup).

   Cracking may contribute to the corrosion of the reinforcing steel or evidence of fatigue. To assure these concerns are recorded and monitored, cracking greater than 1 mm (0.04") in maximum width should be recorded during the baseline examination. The threshold of 1 mm in maximum width is consistent with the guidance provided in ACI 349.3R-96.

   Cracking meeting the recordable threshold should be monitored to determine if it is passive or active. Passive cracks are defined as those having an absence of growth (when compared to the baseline examination results) and absence of other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup).

If the crack is determined to be passive, further review by the IWL Responsible Engineer is not warranted.

If for some reason a crack becomes active (e.g., a change in crack width or length when compared to the baseline exam data), this condition should be classified as suspect and a detailed or VT-1C examination should be performed to determine the magnitude and extent. If confirmed by this examination, the IWL Responsible Engineer should investigate this change.

In the unlikely event, if a crack degrades to the point that reinforcement steel is exposed and severe corrosion is present, signs of distress would also be observed during the examination. Typical signs of this type of distress would be corrosion staining emerging from the crack or bulging of the cover caused by corrosion buildup. Since both of these conditions are evidence of potential structural degradation, these conditions should be classified as suspect and a detailed or VT-1C examination should be performed to determine the magnitude and extent. If confirmed by this examination, the IWL Responsible Engineer should also investigate either of these conditions.

Table 10.8-1
Reportable Conditions for Class CC Components
A.1 Cracks

Thresholds		Action To Be Taken
Recordability:	Any crack that visually appears to be greater than 1 mm (0.04") in maximum width.	Record the condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining emerging from the crack, active changes in crack width or length (when compared to the baseline exam), or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup).	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Corrosion staining, active changes in crack width or length, or other degradation mechanisms at the crack is confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

3. **A.2 DETERIORATION**

A. ***A.2.1 Distortion*** - any abnormal deformation of concrete from its original shape. This condition is illustrated by photograph in ACI 201.1R-92 (see Figure A.2.10).

Distortion of the concrete structure would be a result of abnormal loading conditions (e.g., earthquake, water hammer) and the damage would be primary concentrated in the concrete cover. However, internal structural degradation may be possible.

For this reason, any abnormal deformation should be recorded and investigated by the IWL Responsible Engineer.

Table 10.8-2 Reportable Conditions for Class CC Components A.2.1 Distortion		
Thresholds		Action To Be Taken
Recordability:	Any abnormal deformation of concrete from its original shape.	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Any abnormal deformation of concrete from its original shape.	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Any abnormal deformation of concrete from its original shape.	Investigate the condition to determine if further evaluation or repair is warranted.

B. ***A.2.2 Efflorescence (Leaching)*** - a deposit of salts, usually white, formed on a surface, the substance having emerged from below the surface. This condition is illustrated by photograph in ACI 201.1R92 (see Figure A.2.12).

Efflorescence (also referred to as leaching) is caused by exposure of the concrete to flowing or penetrating water that results in the leaching of certain salts, including calcium hydroxide, from the concrete paste. This condition normally occurs at locations of high moisture penetration and flow, such as cracks.

Since leaching normally occurs at cracks, further degradation of the crack may occur and may contribute to the corrosion of the reinforcing steel. As stated earlier, corrosion of the reinforcement steel could potentially affect the structural integrity of the concrete containment. Thus, any leaching should be recorded and monitored. Since corrosion of the reinforcing steel is the main concern, any evidence of corrosion staining emerging from the degraded surface or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup) has been established as the threshold for classifying this condition as a suspect condition. At this point, a detailed or VT-1C examination should be performed. If the detail or VT-1C examination confirms corrosion staining, an investigation by the IWL Responsible Engineer should be performed.

Table 10.8-3
Reportable Conditions for Class CC Components
A.2.2 Efflorescence (Leaching)

Thresholds		Action To Be Taken
Recordability:	Any leaching.	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining emerging from the degraded surface or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup).	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Corrosion staining or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup) are confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

C. ***A.2.3 Popout*** - the breaking away of small portions of a concrete surface due to internal pressure that leaves a shallow, typical conical depression. This condition is illustrated by photographs in ACI 201.1R92 (see Figures A.2.19, A.2.19.1, A.2.19.2, A.2.19.3).

Popouts on the concrete cover would not compromise the structural integrity of the containment. For this reason, they should be acceptable for continued service and a review by the IWL Responsible Engineer would not warranted.

However, the concrete cover at this affected area may continue to degrade and expose the reinforcing steel. For this reason, any popout that visually appears to be greater than 50 mm (2.00") in diameter or equivalent surface area should be recorded and monitored. The threshold of 50 mm (2.00") is consistent with the guidance provided in ACI 349.3R-96.

Since corrosion of the reinforcement steel could potentially affect structural integrity of the concrete containment, any evidence of corrosion staining emerging from the degraded surface and/or exposed reinforcing steel detected during the general or VT-3C examination should be considered suspect and a detailed or VT-1C examination should be performed. If corrosion staining and/or exposed reinforcing steel is confirmed by the detail or VT-1C examination, an investigation by the IWL Responsible Engineer should be performed.

Table 10.8-4
Reportable Conditions for Class CC Components
A.2.3 Popout

Thresholds		Action To Be Taken
Recordability:	Any popout that visually appears to be greater than 50 mm (2.00") in diameter or equivalent surface area.	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining emerging from the popout and/or there is evidence of exposed reinforcing steel.	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Corrosion staining and/or exposed reinforcing steel is confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

D. ***A.2.4 Scaling*** *(including peeling)* – local flaking or peeling away of the near surface portion of concrete or mortar. Scaling may be loss of coarse aggregate particles, as well as mortar. This condition is illustrated by photographs in ACI 201.1R92 (see Figures A.2.20.1a, A.2.20.1b, A.2.20.2a, A.2.20.2b, A.2.20.3a, A.2.20.3b, A.2.20.4a, A.2.20.4b).

Similar to a popout, scaling of the concrete cover will not compromise the structural integrity of the containment. For this reason, scaling would be acceptable for continued service and a review by the IWL Responsible Engineer would not be warranted.

Since the concrete cover could continue to degrade in this affected area and expose the reinforcing steel, scaling that visually appears to be greater than 30 mm (1.125") in depth should will be recorded and monitored. The threshold of 30 mm (1.125") is consistent with the guidance provided in ACI 349.3R-96.

Any evidence of corrosion staining emerging from the degraded surface and/or exposed reinforcing steel that is detected during the general or VT-3C examination should be considered suspect and a detailed or VT-1C examination should be performed. If corrosion staining and/or exposed reinforcing steel is confirmed by the detail or VT-1C examination, a review by the IWL Responsible Engineer should be performed.

Table 10.8-5
Reportable Conditions for Class CC Components
A.2.4 Scaling (including peeling)

Thresholds		Action To Be Taken
Recordability:	Any scaling that visually appears to be greater than 30 mm (1.125") in depth.	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining is emerging from the scaling and/or there is evidence of exposed reinforcing steel.	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Corrosion staining and/or exposed reinforcing steel is confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

E. ***A.2.5 Spall*** - a fragment, usually in the shape of a flake, detached from a larger mass by a blow, by the action of weather, by pressure, or by expansion within the large mass. A spall is normally a circular or oval depression or in some cases elongated depression over a reinforcing bar. This condition is illustrated by photographs in ACI 201.1R-92 (see Figures A.2.21.1, A.2.21.2).

A spall on the concrete cover would not compromise the structural integrity of the concrete containment. Thus, spalls would be acceptable for continued service and a review by the IWL Responsible Engineer would not be warranted.

Since the concrete cover could continue to degrade in this affected area and expose the reinforcing steel, any spall that visually appear to be 20 mm (0.750") or more in depth and 200 mm (8.00") or greater in any dimension should be recorded and monitored. The threshold of 20 mm (0.750") or more in depth and 200 mm (8.00") or greater in any dimension is consistent with the guidance provided in ACI 349.3R-96.

Since corrosion of the reinforcement steel could potentially affect structural integrity, any evidence of corrosion staining emerging from the degraded surface and/or exposed reinforcing steel that is detected during the general or VT-3C examination should be considered suspect and a detailed or VT-1C examination should be performed. If corrosion staining and/or exposed reinforcing steel is confirmed by the detail or VT-1C examination, a review by the IWL Responsible Engineer should be performed.

Table 10.8-6
Reportable Conditions for Class CC Components
A.2.5 Spall

	Thresholds	Action To Be Taken
Recordability:	Any spall that visually appears to be 20 mm (0.750") or more in depth and 200 mm (8.00") or greater in any dimension.	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining emerging from the spall and/or there is evidence of exposed reinforcing steel.	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Corrosion staining and/or exposed reinforcing steel is confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

F. ***A.2.6 Stalactite*** - a downward pointing formation, hanging from the surface of concrete, shaped like an icicle.

G. ***A.2.7 Stalagmite*** - stalagmite are similar to stalactite with the exception they are an upward formation.

Stalactite and/or stalagmite are caused by exposure of the concrete to flowing or penetrating water that results in the leaching of certain salts, including calcium hydroxide, from the concrete paste. This condition normally occurs at locations of high moisture penetration and flow, such as cracks.

Similar to leaching, this mechanism normally occurs at cracks. Since degradation of the crack may occur and contribute to the corrosion of the reinforcing steel, either of these conditions should be recorded and monitored.

Since corrosion of the reinforcing steel is the main concern, any evidence of corrosion staining emerging from the degraded surface or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup) has been established as the threshold for classifying this condition as a suspect condition. At this point, a detailed or VT-1C examination should be performed. If the detailed examination or VT-1C examination confirms corrosion staining, an investigation by the IWL Responsible Engineer should be performed.

Table 10.8-7
Reportable Conditions for Class CC Components
A.2.6 Stalactite / A.2.7 Stalagmite

Thresholds		Action To Be Taken
Recordability:	Any stalactite or stalagmite.	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining emerging from the degraded surface or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup).	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Corrosion staining or other degradation mechanisms at the crack (e.g., bulging caused by corrosion buildup) are confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

H. ***A.2.8 Corrosion*** – Disintegration or deterioration of concrete or reinforcement by electrolysis or by chemical attack. This condition is illustrated by photograph in ACI 201.1R-92 (see Figure A.2.5).

Since conditions associated with concrete disintegration and/or deterioration has already been discussed, no further discussion of this condition will be provided.

Under most conditions, concrete provides adequate protection of embedded materials against corrosion. The protective value of the concrete is attributable to its high alkalinity. The degree to which concrete will provide this protection is a function of the quality of the concrete, the depth of the concrete cover, and good construction practices.

Corrosion of the reinforcing steel is not limited to cracked concrete surfaces. Corrosion can occur in uncracked surfaces when the concrete cover over the steel is insufficient.

In both cases, rust stains can be observed in the pores of the concrete and in small cracks at the surface during the early stages of corrosion. As the corrosion advances, prominent cracking of the concrete in a direction parallel to the reinforcement and delamination of the concrete will occur. In severe cases, spalling down to the level of the reinforcement will occur.

Since excessive corrosion of the reinforcing steel may be evidence of potential structural degradation, any corrosion staining emerging from the concrete and/or other evidence of corrosion (e.g., bulging caused by corrosion buildup) should be recorded and established as the suspect condition threshold. If the detail or VT-1C examination confirms either condition, an investigation by the IWL Responsible Engineer should be performed.

Table 10.8-8
Reportable Conditions for Class CC Components
A.2.8 Corrosion

Thresholds		Action To Be Taken
Recordability:	Evidence of corrosion staining emerging from the concrete surface or other evidence of corrosion (e.g., bulging caused by corrosion buildup).	Record condition and any supplemental information necessary to identify the location of the area for future monitoring.
Suspect Condition:	Evidence of corrosion staining emerging from the concrete surface or other evidence of corrosion (e.g., bulging caused by corrosion buildup).	Perform a detailed or VT-1C examination to determine the magnitude and extent of the suspect condition and record results.
IWL Responsible Engineer Review:	Evidence of corrosion staining emerging from the concrete surface or other evidence of corrosion is confirmed.	Investigate the condition to determine if further evaluation or repair is warranted.

ATTACHMENT 4

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program

TABLE AUG-13-1
Refueling Outage-Based Visual Examination Table

Population or Group Note 1	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A for Extended Interval Notes 2 and 3	Column B Maximum for Same as Previous Interval Notes 2 and 4	Column C for Interval Reduction by 1/3rd Notes 2,5,6
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or Greater	29	56	109

NOTES:

1. Interpolation between population or group sizes and the number of unacceptable snubbers is permissible. Use the next lower integer found for the permissible number of unacceptable snubbers.

2. The basic interval shall be the normal fuel cycle up to 24 months. The examination interval may be as great as twice the fuel cycle (Note 3) or as small as 1/3rd of the fuel cycle (Notes 5b and 6). The maximum (previous interval) value used to determine the next examination interval shall be one normal fuel cycle. The examination intervals may vary by ± 25% to coincide with the actual outage.

3. If the number of unacceptable snubbers is equal to or less than the value in Column A, then the next examination interval may be increased to twice the past examination interval, i.e., the next exam according to the former interval may be skipped. When the former interval is the refueling cycle, the snubbers may be examined only every other refueling cycle interval so long as the results of the visual examination meet the requirements of Column A. The snubbers that are installed at locations where the snubbers were unacceptable at the previous examination shall be examined during the skipped refueling outage.

4. If the number of unacceptable snubbers exceeds the value in Column A, but is equal to or less than the number in Column B, then the next visual examination shall be conducted at the same interval as the immediately preceding interval. When the former interval is the refueling cycle the next interval is the current refueling cycle.

5. If the number of unacceptable snubbers exceeds the number in Column B, but is equal to or less than the value in Column C, then one of the following shall apply:

   a. A review and evaluation to justify continued use of the snubbers shall be performed. The previous examination interval may then be used. When the former interval is the refueling cycle the next interval is the current refueling cycle,

   OR

   b. The next examination interval shall be decreased by 1/3rd of the previous examination interval or in accordance with the interpolation between Columns B and C, in proportion to the exact number of unacceptable snubbers.

6. If the number of unacceptable snubbers exceeds the value in Column C, then the corrective actions and justifications of Note 5a shall be performed and the examination interval shall be decreased to 1/3rd of the previous interval.

PROGRAM No. AUG-13
Technical Specification 3/4.7.4
Snubber Examination and Test Program

TABLE AUG-13-2
Snubber Visual Examination Discrepancy Report Disposition

Condition[9]	Potential Affect on Operability	Action Required to Determine Visual Discrepancy Report Action / Disposition[8]				Snubber Repair or Replacement
		Stroke Test[11]	Functional Test[11]	Engineering Review		
				Evaluation	Sign-off on D. R.	
Cold Set / Hot Set Out of Tolerance Not Bottomed Out	No	No	No	7	7	1, 6
Cold Set / Hot Set Out of Tolerance Bottomed Out	Yes	Yes	3	Yes	Yes	1, 2, 5, 6
Body or Transition Tube Dent > 1/8”	Yes	Yes	3	2	3	2, 5
Body or Transition Tube Arc Strike	Yes	Yes	3	2	2	2, 5, 6 Blend and Paint
Telescoping Cylinder Heavy Rust on OD	Yes	Yes	3	3	3	2, 5, 6 Clean
Internal Corrosion	Yes	Yes	3	3	3	2, 5
Internal Torque Broken	No	No	No	No	No	5
PSA-35s and PSA-100s Center Torque Broken	No	No	No	No	No	4
End Cap / Transition Tube Broken Torque	No	No	No	No	No	4
Spherical Bearing Displacement > 1/16”	No	No	No	No	No	4

TABLE AUG-13-2
Snubber Visual Examination Discrepancy Report Disposition

Condition[9]	Potential Affect on Operability	Action Required to Determine Visual Discrepancy Report Action / Disposition[8]		Engineering Review		Snubber Repair or Replacement
		Stroke Test[11]	Functional Test[11]	Evaluation	Sign-off on D. R.	
Spherical Bearing Frozen will not Rotate[10]	Yes	Yes	3	Yes	Yes	2, 5, 6 Clean, Free or Replace Bearings
Spherical Bearing Frozen will Rotate[10]	No	No	No	No	No	No
Clamp or End Bracket Wear Marks Caused by Binding	Yes	Yes	3	Yes	Yes	1, 2, 5, 6
Mechanical Interference or Misalignment Indication of Binding and/or Bending	Yes	No	Yes	Yes	Yes	1, 2, 5, 6
Mechanical Interference or Misalignment No Indication of Binding	No	No	No	Yes	Yes	6 Unless Accept-As-Is
Clamp Nut Loose or Inadequate Thread Engagement	No	No	No	No	No	6
Clamp Loose and Affecting Snubber Movement	Yes	Yes	3	2	3	1, 2, 5, 6 Tighten Clamp
PSA-1/4, 1/2, 1, 3, 10, Lisega 3018 Used as Step	Yes	Yes	3	2	3	2, 5
PSA-35, 100, PSB Used as Step	No	No	No	No	Yes	No
Load Pin Bent, Worn or Missing	Yes	Yes	3	Yes	Yes	1, 4, 6

TABLE AUG-13-2
Snubber Visual Examination Discrepancy Report Disposition

Condition[9]	Potential Affect on Operability	Action Required to Determine Visual Discrepancy Report Action / Disposition[8]				Snubber Repair or Replacement
				Engineering Review		
		Stroke Test[11]	Functional Test[11]	Evaluation	Sign-off on D. R.	
Washers Missing or Incorrect	No	No	No	No	No	4
Load Pin Retainers or Cotter Pins Missing or Improperly Installed	No	No	No	No	No	6
Safety Wire Incorrect	No	No	No	No	No	4
Dust Cover Screws Loose or Missing	No	No	No	No	No	6
Weld Indications	Yes	No	Yes	Yes	Yes	1, 2, 5, 6
Serial or ID Number Incorrect / Missing	No	No	No	No	No	No
ST Data Sheet Typo / Error	No	No	No	No	No	No
Hydraulic Snubber Evidence of Fluid Leakage from Reservoir Level Still Satisfactory	No	Yes	3	2	2	5
Hydraulic Snubber Evidence of Fluid Leakage from Reservoir Level Unsatisfactory	Yes	Yes	Yes	Yes	Yes	5
Condition Re-verified as Not Discrepant	No	No	No	No	No	No

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Snubber Examination and Test Program

TABLE AUG-13-2
Snubber Visual Examination Discrepancy Report Disposition, continued

NOTES:

1. Potential Structural Rework depending on Engineering evaluation.
2. Required only if unacceptable Functional Test Results obtained.
3. Required only if Stroke Test fails.
4. Correct condition per applicable maintenance procedures.
5. Rebuild or Replace per Note 4.
6. Correct condition per Work Order instructions.
7. Not required if acceptable per Design Drawing or within range of thermal movement previously determined as acceptable. Reference applicable documentation on Discrepancy Report.
8. Special cases involving a combination of conditions or based on the judgement of Engineer may result in additional evaluation or testing than noted above.
9. Conditions not addressed will be reviewed on a case-by-case basis and will require sign-off by Engineering.
10. Rotation shall be defined as follows:
    a. For PSA-1/4, ½, Lisega 3018 - hand force only, no mechanical assistance.
    b. PSA-1, 3, 10 - may be rotated with assistance of standard screw driver or equivalent (maximum overall length 13").
    c. PSA-35, 100 - may be rotated with assistance of heavy duty screw driver or equivalent (maximum overall length 29"). Care shall be taken not to damage snubber or affect torqued connections.
11. Testing as listed may be waived with concurrence from Engineering.

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Technical Specification 3/4.7.4
Snubber Examination and Test Program, continued

Figure AUG-13-1
Snubber Scope of Examination/Testing

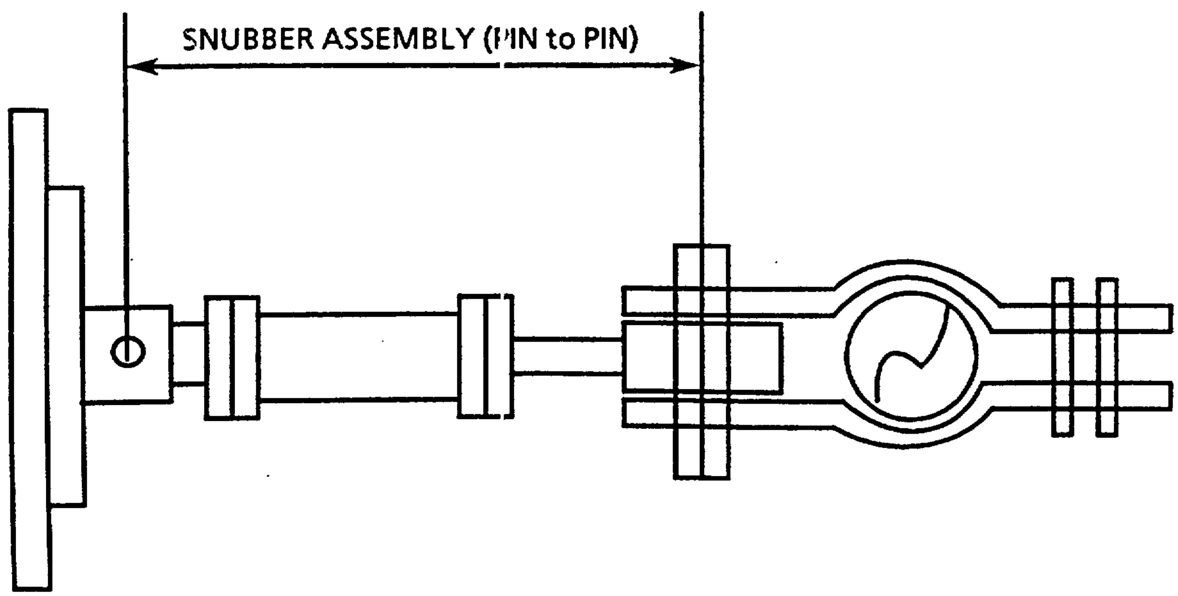

FIGURE AUG-13-2
THE 37 TESTING SAMPLE PLAN