March 11, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer and Senior Vice President
Exelon Generating Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 – EVALUATION OF RELIEF REQUESTS I3R-02, I3R-05, I3R-06, I3R-07, I3R-08, I3R-09, I3R-10, I3R-11, I3R-12, ASSOCIATED WITH THE THIRD INSERVICE INSPECTION INTERVAL (TAC NOS. MD5200 AND MD5201)

Dear Mr. Pardee:

By letter dated March 6, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070660108), as supplemented by letter dated November 8, 2007, (ADAMS Accession No. ML073170370), Exelon Generating Company, LLC submitted various proposed requests for relief from, and alternatives to, the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a, concerning the Third 10-year inservice inspection (ISI) programs at Limerick Generating Station, Units 1 and 2 (LGS).

Based on the information provided, the Nuclear Regulatory Commission (NRC) staff concludes that compliance with the specified American Society of Mechanical Engineers (ASME) Boiler *and Pressure Vessel Code* (Code), Section XI, requirements relating to relief requests I3R-06, I3R-07, and I3R-08 is impractical. Therefore, the Commission grants relief pursuant to 10 CFR 50.55a(g)(6)(i) for the Third 10-year ISI interval at each unit. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Regarding relief requests I3R-02, I3R-05, I3R-09, I3R-10, I3R-11, and I3R-12, the NRC staff concludes that the proposed alternatives will provide an acceptable level of quality and safety. Therefore, the NRC staff authorizes the use of the proposed alternatives pursuant to 10 CFR 50.55a(a)(3)(i) for the Third 10-year ISI interval at each unit.

The Third 10-year ISI interval began on February 1, 2007, and is scheduled to conclude on January 31, 2017, for LGS, Units 1 and 2.

Documentation of the NRC staff review and evaluation is contained in the enclosed safety evaluation.

If you have any questions, please contact the Limerick Project Manager, Mr. Peter J. Bamford, at 301-415-2833.

Sincerely,

/ra/

Harold K. Chernoff, Chief
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosure: As stated

cc w/encl: See next page

If you have any questions, please contact the Limerick Project Manager, Mr. Peter J. Bamford, at 301-415-2833.

Sincerely,

/ra/

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INSERVICE INTERVAL RELIEF REQUESTS

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated March 6, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070660108), as supplemented by letter dated November 8, 2007, (ADAMS Accession No. ML073170370), Exelon Generating Company, LLC (the licensee) submitted various proposed requests for relief from, and alternatives to, the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a, concerning the Third 10-year Inservice Inspection (ISI) programs at Limerick Generating Station (LGS or Limerick), Units 1 and 2. The Third 10-year ISI interval began on February 1, 2007, and is scheduled to conclude on January 31, 2017, for LGS, Units 1 and 2.

2.0 REGULATORY EVALUATION

Inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3, components is to be performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (Code) and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). As stated in 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety; or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3, components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI (ASME Code), "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b), 12-months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ASME Code of record for LGS, Units 1 and 2, Third 10-year ISI interval is the 2001 edition through 2003 addenda of the ASME Code, Section XI.

Enclosure

The LGS, Units 1 and 2, Risk Informed (RI) ISI program was developed in accordance with the methodology contained in the Electric Power Research Institute's (EPRI's) topical report EPRI TR-112657, Rev. B-A, which was reviewed and approved by the NRC staff. This RI-ISI program was originally submitted to the NRC by letter dated March 15, 2002 (ADAMS Accession No. ML020860479), and approved for use in the Second 10-year ISI interval by letter dated March 3, 2003 (ADAMS Accession No. ML030620491) as an alternative pursuant to 10 CFR 50.55a(a)(3)(i).

3.0 TECHNICAL EVALUATION

3.1 Relief Request I3R-02, revision 0

By letter dated March 6, 2007, the licensee requests NRC authorization to continue the implementation of an RI-ISI piping program for the Third 10-year ISI interval at LGS, Units 1 and 2. The scope of the RI-ISI program is limited to the inspection of ASME Code Class 1 and 2 piping (Categories B-F, B-J, C-F-1, and C-F-2 welds). The licensee considered relevant information since the development of the original program and reviewed and updated the RI-ISI program.

The licensee is requesting relief to use the proposed RI-ISI program plan in the Third 10-year ISI interval instead of the ASME Code, Section XI program for piping. An acceptable RI-ISI program plan is expected to meet the five key principles discussed in NRC Regulatory Guide (RG) 1.178, dated September 2003; NRC NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, chapter 3.9.8; and EPRI TR-112657, as stated below.

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes result in an increase in Core Damage Frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (51 FR 30028).
5. The impact of the proposed change should be monitored by using performance measurement strategies.

The first principle is met in this relief request because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(a)(3)(i) and therefore, an exemption request is not required. The second and third principles require assurance that the alternative program is consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained, respectively. Assurance that the second and third principles are met is based on the application of the approved methodology and not on the particular inspection locations selected. The methodology used to develop the RI-ISI program for the Third 10-year inspection interval is unchanged from the methodology approved for use in the Second 10-year inspection interval, and therefore, the second and third principles are met.

The RI-ISI program is a living program that requires periodic updating and that, as a minimum, risk ranking of piping segments will be reviewed on an ASME period basis. In the submittal dated March 6, 2007, and as supplemented by letter dated November 8, 2007, the licensee provided a summary of the changes that have occurred after the original implementation of the RI-ISI program. These include:

- Transition from the 1989 edition to the 2001 edition through the 2003 addenda of ASME Code, Section XI;
- Limited examination coverage which resulted in changes to the location of several examinations to increase code examination coverage;
- Original risk impact assessment update;
- Plant modifications resulting in an increase in the number of welds and other modifications affecting plant risk; and
- RI-ISI category reclassifications due to an updated Probablistic Risk Assessment model that affected the number and location of the required examinations

As described in Section 3.2.1 of EPRI TR-112657, the RI-ISI program scope is determined by the ASME inspection program scope. As a result of the above changes, the number of high-risk category weld examinations increased from 41 to 62 for LGS Unit 1 and from 46 to 63 for LGS Unit 2. The number of medium risk category weld examinations increased from 55 to 79 for LGS Unit 1 and from 51 to 82 for LGS Unit 2.

As was done in the original implementation of the RI-ISI program, the provisions listed in Table 1, "Examination Category R-A, Risk-Informed Piping Examinations," contained in ASME Code Case N-578-1 will be used to supplement the requirements of EPRI TR-112657, Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods." The licensee clarified how Code Case N-578-1 will be used by letter dated November 8, 2007, stating that EPRI TR-112657 does not identify the examination volumes for components without a degradation mechanism and it does not specify examination volumes and methods for socket welds. LGS has requested to use the examination methods from Code Case N-578-1 instead of the methods from EPRI TR-112657, except that the volumetric method will be used to examine intergranular stress corrosion cracking (IGSCC), and the VT-2 (visual) examination method will be used to examine socket welds in accordance with the provisions of Code Case N-578-1, Table 1. The examination figures specified in Section 4 of EPRI TR-112657 will be used to determine the examination volume based on the degradation mechanism and component configuration.

A comparison of the requirements of EPRI TR-112657 to those of Code Case N-578-1 indicates that the only differences are the inspection volumes, the methods specified for welds with no degradation method and for socket welds. The NRC staff finds the use of the inspection volumes as specified in EPRI TR-112657 to be acceptable as this is consistent with the approved methodology of EPRI TR-112657. EPRI TR-112657 does not identify the examination volumes or method for components without a degradation mechanism and it does not specify examination volumes or methods for socket welds. The NRC staff finds the use of volumetric methods to examine components without a degradation mechanism, as specified in Code Case N-578-1, a conservative approach and therefore acceptable. The NRC staff finds that performance of a VT-2 examination of socket welds each refueling outage, as required by Code Case N-578-1, will ensure an acceptable level of quality and safety as compared to a surface examination conducted once per ISI interval, as required by the ASME Code.

The licensee reported in letter dated November 8, 2007, that the original risk impact assessment was updated to confirm the change in risk was maintained within the acceptance guidelines. The original methodology of the calculation was not changed, and the change in risk was simply re-assessed using the initial 1989 Section XI program prior to RI-ISI and the new element selection for the Third 10-year interval RI-ISI program. This same process has been maintained in each revision to the Limerick RI-ISI report that has been performed to date.

Using this process, the change in risk for Unit 1 was 2.21×10^{-08} for delta-core damage frequency (delta-CDF) and -1.25×10^{-09} for delta-large early release frequency (delta-LERF). For Unit 2, the values were 2.68 x 10^{-08} for delta-CDF and 1.02 x 10^{-09} for delta-LERF. These values are all within the 1.00 x 10^{-06} and 1.00 x 10^{-07} acceptance criteria referenced in RG 1.178 for delta-CDF and delta-LERF, respectively.

Development of an acceptable RI-ISI program is primarily achieved through the risk-ranking and the inspection location selection processes. Estimates of the change in CDF and LERF is a final phase intended to provide additional assurance that aggregate changes in risk will be acceptable. Although the ASME inspection program may change slightly when developed from the new code of record, the accuracy of the change in risk calculations do not warrant developing a new ASME program for the new Code of Record simply to be used as a new baseline and then discarded. Therefore, the NRC staff finds the comparison of the risk estimate between the RI-ISI program proposed in the submittal and the ASME program based on the Code of Record from which relief was granted in NRC letter dated March 3, 2003, is appropriate and acceptable. No deviation from the risk acceptance criteria were identified and NRC staff finds that the process provides assurance that the fourth key principle is met. Section 3.6.6.1 of EPRI TR-112657 states, in part, that the service history, susceptibility review and ongoing industry events reviews assure that the industry trends are being monitored to assure that if an unexpected or new mechanism is identified, or a new component is identified as susceptible to an existing degradation mechanism, the RI-ISI program will be updated to reflect that change. The program update will incorporate any additional inspections mandated by the NRC, as well as those inspections deemed appropriate by the industry groups addressing the specific issues.

As stated in letter dated November 8, 2007, all dissimilar metals (DM) welds, as characterized in ASME Code, Section XI IWA-9000, have been evaluated for failure potential and consequence of failure along with the other non-exempt piping. The piping segments containing the DM welds were classified into the appropriate RI-ISI categories, and appropriate elements were selected per the category requirements for examination during the third inspection interval.

DM welds that are susceptible to IGSCC (i.e., IGSCC Categories B through G, as applicable) and not subject to other degradation mechanism(s) are removed from the RI-ISI program population. They are contained in the Limerick ISI Augmented Program 01, "USNRC Generic Letter 88-01, Intergranular Stress Corrosion Cracking," and are subject to the inspection requirements of BWRVIP-75-A "BWR [Boiling-Water Reactor] Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules." Furthermore, all DM welds classified as Category A (resistant material) per BWRVIP-75-A are included in the RI-ISI program. Removal of a well defined population of welds whose sole degradation mechanism is targeted by an approved augmented program was accepted by the NRC staff by letter dated March 3, 2003, and is therefore acceptable.

As stated by the licensee, the Third interval RI-ISI program will be a continuation of the current application and will continue to be a living program. The monitoring program developed for the second interval RI-ISI program was found to be acceptable by the NRC staff as documented in letter dated March 3, 2003. The program and requirements are still applicable to the Third interval RI-ISI program. Therefore, the NRC staff concludes that the RI-ISI program continues to be a living program and that the fifth key principle is met.

Based on the above discussion, the NRC staff finds that the five key principles of risk-informed decision making are ensured by the licensee's proposed Third 10-year RI-ISI interval program plan and therefore, the proposed program for the Third 10-year ISI inspection interval is acceptable.

Based on the information provided in the licensee's submittals, the NRC staff has determined that the proposed alternative provides an acceptable level of quality and safety, and, therefore, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the Third 10-year ISI inspection interval at LGS Units 1 and 2.

3.2 <u>Relief Request I3R-05, revision 0</u>

3.2.1 Component for Which Relief is Requested

All LGS, Units 1 and 2, safety-related ASME Code Class 1, 2 and 3 snubbers.

3.2.2 Code Requirements

The ASME Code, Section XI, article IWF-5000, provides inservice inspection requirements for snubbers.

Paragraphs IWF-5200(a) and IWF-5300(a) require that snubber preservice and inservice examinations be performed in accordance with OM-4, using the VT-3 visual examination method described in IWA-2213.

Paragraphs IWF-5200(b) and IWF-5300(b) require that snubber preservice and inservice tests be performed in accordance with OM-4.

Paragraphs IWF-5200(c) and IWF-5300(c) require that integral and nonintegral attachments for snubbers, including lugs, bolting, pins, and clamps, be examined in accordance with the requirements of Subsection IWF.

3.2.3 Licensee's Proposed Alternative

The licensee proposes to use LGS, Units 1 and 2, Technical Specification (TS) 3/4.7.4, "Snubbers," to perform visual examinations and functional testing of ASME Code Class 1, 2 and 3 snubbers in lieu of meeting ASME Code, Section XI requirements.

3.2.4 Licensee's Basis for Requesting Relief

The licensee states that the required TS program for the ISI classified snubbers overlaps with the ASME Code, Section XI requirements and thus the ASME Code requirements present an unnecessary redundancy without a compensating increase in the level of quality and safety.

The purpose of the Augmented Inspection Program described in the LGS, Units 1 and 2, TS 3/4.7.4 is to assure and demonstrate operational readiness and structural integrity of snubbers through testing and examination. The TS snubber visual examination program, which includes all safety related snubbers, incorporates the alternate snubber visual examination requirements delineated in USNRC Generic Letter (GL) 90-09, "Alternate Requirements for Snubber Visual Inspection Intervals and Corrective Actions". The examinations are performed by qualified personnel and meet the intent of the inspections and tests of ASME Code, Section XI. The TS functional testing program is based on the ASME/ANSI OMc-1990 Addenda to the ASME/ANSI OM-1987 Edition, Part 4, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)".

LGS, Units 1 and 2, has procedures in place to implement the program as described in the TS 3/4.7.4. These procedures include Corporate procedures ERAA-330-004, "Visual Examination of

Snubbers," ER-AA-330-010, "Snubber Functional Testing," and ER-AA-330-011, "Snubber Service Life Monitoring." Station surveillance test procedures are used to implement the visual examination, functional testing, and service life monitoring requirements for snubbers. The general requirements of the ASME Code, Section XI, subsection IWA, such as examination methods, personnel qualifications, etc. still apply.

As required by the ASME Code, Section XI, subsections IWF-5200(c) and IWF-5300(c), the examination of snubber integral and nonintegral attachments, including bolting and load pins, will be performed in accordance with subsection IWF. The examination of snubber welded attachments will be performed in accordance with the ASME Code, Section XI, subsections IWB, IWC, and IWD welded attachment examination requirements (e.g., Examination Categories B-K, C-C, and D-A).

3.2.5 NRC Staff Evaluation of Relief Request I3R-05

The licensee requested relief from the requirements of the ASME Code, Section XI, paragraphs IWF-5200(a) and (b), and IWF-5300(a) and (b). The licensee proposed that the inservice visual examinations and functional testing of ASME Code Class 1, 2 and 3 snubbers be performed in accordance with the requirements of the Limerick TS 3/4.7.4 in lieu of meeting the requirements in the ASME Code, Section XI, paragraphs IWF-5200(a) and (b), and IWF-5300(a) and (b).

The applicable edition of Section XI of the ASME Code for the Limerick Units 1 and 2 Third 10-year ISI interval is the 2001 edition through 2003 addenda. The ASME Code, Section XI, paragraphs IWF-5200(a) and (b), and IWF-5300(a) and (b), reference OM-4, 1987 edition with OMa-1988 addenda.

ASME Code, Section XI, paragraphs IWF-5200(a) and IWF-5300(a) require that snubber preservice and inservice examinations be performed in accordance with OM-4, using the VT-3 visual examination method described in IWA-2213. Paragraphs IWF-5200(b) and IWF-5300(b) require that snubber preservice and inservice tests be performed in accordance with OM-4.

Paragraphs IWF-5200(c) and IWF-5300(c) require that integral and non-integral attachments for snubbers, including lugs, bolting, pins, and clamps, be examined in accordance with subsection IWF. The licensee states that the examination of snubber welded attachments will be performed in accordance with the ASME Code, Section XI, subsections IWB, IWC, and IWD welded attachment examination requirements (e.g., Examination Categories B-K, C-C, and D-A). Visual examiners, who are qualified to the applicable rules of ASME Code, Section XI, article IWA-2000, "Examination and Inspection," will perform the examinations of the safety related snubbers.

ASME Code, Section XI, Table IWA-1600-1, states that OM-4 shall be the 1987 edition with OMa-1988 addenda. OM-4 specifies the requirements for visual examination (paragraph 2.3) and functional testing (paragraph 3.2). The licensee proposes to use TS 3/4.7.4 for inservice visual examination and functional testing of all safety-related snubbers (pin-to-pin). A visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. To verify that a snubber can operate within specific performance limits, the licensee performs functional testing that typically involves removing the snubber and testing it on a specially designed stand or bench. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber operability.

Limerick TS 3/4-7.4 incorporates GL 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." GL 90-09 acknowledges that the visual inspection schedule (as contained in OM-4) is excessively restrictive and that licensees with large snubber populations have spent a significant amount of resources and have subjected plant personnel to unnecessary radiological exposure to comply with the visual examination requirements. GL 90-09 states that its alternative schedule for visual inspection provides the same confidence level as that provided by OM-4.

As mentioned above, the applicable OM-4 shall be the 1987 edition with OMa-1988 addenda. The licensee states that Limerick TS 3/4.7.4 incorporated the use of OM-4 with the OMc-1990 addenda on May 11, 1992, through Amendment No. 54 to the Limerick Unit 1 Facility Operating License and Amendment No. 19 to the Limerick Unit 2 Facility Operating License, to update the snubber examination and testing requirements. The NRC staff finds that changes made under Amendment No. 54 for Limerick Unit 1 and Amendment No. 19 for Limerick Unit 2, by using OMc-1990 addenda, are still applicable and are equivalent to OM-4 with OMa-1988 addenda requirements. Therefore, all the references and comparisons made in this safety evaluation are based on OM-4 1987 edition with OMa-1988 addenda.

TS 3/4.7.4 defines inservice examination requirements: (1) visual examination; (2) visual examination interval frequency; (3) method of visual examination; (4) subsequent examination intervals; and (5) inservice examination failure evaluation. Inservice operability testing requirements are also defined: (1) inservice operability or functional test; (2) initial snubber sample size; (3) additional sampling; (4) failure evaluation; (5) test failure mode groups; and (6) corrective actions for the 10 percent sample and 37 sample plans that are similar to those provided by OM-4. OM-4 requirements and TS 3/4.7.4 criteria are compared and summarized as follows:

Inservice Examination Requirements

(1) Visual Examination

TS 3/4.7.4, SR 4.7.4.c, requires that visual inspections shall verify that: (1) the snubber has no visible indications of damage or impaired operability; (2) attachments to the foundation or supporting structure are secure; and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are secure. The visual examination per SR 4.7.4.c verifies visible indication of damage or impaired operability of snubbers as well as its attachments and anchorages. OM-4, paragraph 2.3.1.1, requires snubber visual examinations to identify impaired functional ability due to physical damage, leakage, corrosion, or degradation. The snubber visual examination requirements in TS 3/4.7.4 are considered to be equivalent to snubber visual examination requirements of OM-4 paragraphs 2.3.1.1. Therefore, this alternative provides an acceptable level of quality and safety.

(2) Visual Examination Interval Frequency

TS Table 4.7.4-1 provides snubber visual inspection interval frequency requirements which are different than the OM-4 visual inspection interval requirements. Table 4.7.4-1 incorporates the visual inspection interval frequency as specified in GL 90-09. GL 90-09 acknowledges that the visual inspection interval frequency (as contained in OM-4) is excessively restrictive and that licensees with large snubber populations have spent a significant amount of resources and have subjected plant personnel to unnecessary radiological exposure to comply with the visual examination requirements. GL 90-09 states that its alternative schedule (interval frequency) for visual inspection provides the same confidence level as that provided by OM-4. Therefore, this alternative provides an acceptable level of quality and safety.

(3) Method of Visual Examination

IWF-5300(a) requires that inservice examination be performed in accordance with OM-4, using the VT-3 visual examination method described in IWA-2213. IWA-2213 states that VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearance, settings, and physical displacements, and to detect discontinuities and imperfections, such as loss of integrity at bolts and welded connections, loose or missing parts, debris, corrosion, wear, or erosion. VT-3 includes examinations for conditions that could affect operability or functional adequacy of snubbers and constant load and spring type supports.

TS SR 4.7.4.c, requires that visual inspections shall verify that: (1) the snubber has no visible indications of damage or impaired operability; (2) attachments to the foundation or supporting structure are secure; and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are secure.

The intent and scope of IWA-2213 and TS 3/4.7.4 are essentially equal, although the IWA-2213 wording is more detailed than the TS in listing specific items to be included. However, these items are intuitive to meeting the TS requirements and are more specifically addressed in the implementing procedure, which closely parallels the IWA-2213 list. Also the TS makes no distinction between integral and non-integral attachments. All are included in the examination to verify overall structural integrity. Further, as specified in letter dated November 8, 2007, visual examinations are performed using qualified personnel who are specifically qualified to a limited VT-3 certification (IWA-2350), and this limited certification meets the requirements of an ASME Code, Section XI, VT-3 certification (IWA-2213) for snubber examination. Exelon Procedure TQ-AA-122, “Qualification and Certification of Nondestructive (NDE) Personnel,” controls this certification.

Therefore, the intent and scope of Limerick TS visual inspection requirements are equivalent to the OM-4, VT-3 examination requirements and the NRC staff finds the licensee’s method of snubber visual inspection provides an acceptable level of quality and safety, and is acceptable.

(4) Subsequent Examination Intervals

TS Table 4.7.4-1 establishes subsequent snubber visual inspection intervals based on the number of unacceptable snubbers discovered, in lieu of OM-4 paragraph 2.3.2 requirements. These requirements are equivalent to the guidance provided in GL 90-09, which has been approved for use by the NRC. Therefore, the NRC staff finds that the subsequent examination intervals contained in TS Table 4.7.4-1 provide an acceptable level of quality and safety and is acceptable.

(5) Inservice Examination Failure Evaluation

OM-4, paragraph 2.3.4.1 requires that snubbers not meeting examination criteria be evaluated to determine the cause of unacceptability. Paragraph 2.3.4.2 states that snubbers found unacceptable may be tested in accordance with the requirements of paragraph 3.2.
TS SR 4.7.4.c, states that snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and/or (2) the affected snubber is functionally tested

in the as-found condition and determined operable per acceptance criteria of the SR 4.7.4.f. The licensee program is considered to be equivalent to the requirements of OM-4. Therefore, the NRC staff finds that the TS's inservice examination failure evaluation requirements provide an acceptable level of quality and safety.

<u>Inservice Operability Testing Requirements</u>

(1) Inservice Operability Test

TS SR 4.7.4.f states that the snubber functional test is to verify: (1) activation is achieved within specified range in both tension and compression; (2) bleed rate, or release rate where required, is present in both tension and compression, within the specified range (hydraulic snubbers); (3) the force required to initiate or maintain motion is within the specified range in both direction of travel (mechanical snubbers); and (4) the ability to withstand load without displacement. The licensee states that, generally, snubbers shall be functionally tested in a bench test. OM-4, paragraph 3.2.1.1, Operability Test, states that snubber operational readiness tests verify activation, release rate, and breakaway force or drag force by either an in-place or bench test. The NRC staff finds that the TS requirements are considered to be equivalent to the snubber operability test requirements of OM-4 paragraph 3.2.1. Therefore, the TS functional test requirements provide an acceptable level of quality and safety.

(2) Snubber Sample Size

TS SR 4.7.4.e, Functional Tests, states that snubbers shall be functionally tested using the following sample plans: (1) at least 13.3 percent of the total population of a snubber type; or (2) a representative sample of 37 snubbers of a snubber type. The sample plan(s) shall be selected for each type prior to the test period and cannot be changed during the test period. OM-4, Section 3.2.3, requires either a 10 percent testing sampling plan, a "37 testing sample plan," or a "55 testing sample plan." The licensee's 13.3 percent testing sample is more conservative than the 10 percent testing sample plan as specified in OM-4, whereas the licensee's 37 testing sample plan is similar to the 37 sample plan as specified in OM-4. As a result, the number of snubbers tested during outages are considered to be equivalent to the OM-4 requirements. Therefore, the TS requirements for snubber sample size provide an acceptable level of quality and safety.

(3) Additional Sampling

(a) For 10 percent testing sample plan

TS SR 4.7.4.e.1 requires that for each snubber of that type that does not meet the functional test acceptance criteria of SR 4.7.4.f, an additional sample of at least one-half the size of the initial sample shall be tested. OM-4, paragraph 3.2.3.1(b), requires that an additional sample size must be at least one-half the size of the initial sample size of the "defined test plan group" of snubbers. That is, for a 13.3 percent sample program, an additional 6.65 percent of the same type of snubber in the overall population would need to be tested. Therefore, TS 3/4.7.4 requirements for additional sampling when using the 13.3 percent testing sample plan for LGS are considered acceptable.

(b) For 37 testing sample plan

OM-4, paragraph 3.2.3.2(b), states that for any snubber(s) determined to be unacceptable as a result of testing, an additional random sample of at least one-half the size of initial sample lot shall be tested until the total number tested (N) is equal to the initial sample size multiplied by the factor 1+ C/2, where C is total number of snubbers found to be unacceptable. For a 37 sample plan, this is represented as an equation, N = 37(1 + C/2), in Appendix C of the OM-4 Code. The TS Figure 4.7.4-1, "Sample Plan 2 for Snubber Functional Test" requirement is the same, as it requires a representative random sample of each test group to satisfy the equation C = 0.055N - 2.007, where N = the number tested, and C = the number of unacceptable snubbers. For the initial sample (C=0), this equation gives N = 36.5 snubbers, rounding up to 37. The licensee states that if one snubber fails to meet the acceptance criteria, then an additional random sample of 19 snubbers of the failed type will be tested, which is equivalent to one-half of the initial sample. Therefore, TS SR requirements for additional sampling when using the 37 testing sample plan are considered acceptable.

(4) Inservice Operability Failure Evaluation

OM-4, paragraph 3.2.4.1, requires that snubbers not meeting the operability testing acceptance criteria in paragraph 3.2.1 shall be evaluated to determine the cause of the failure. The cause of failure evaluation requires a review of information related to other unacceptable snubbers and a determination of whether other snubbers of similar design would require further examination. SR 4.7.4.g, "Functional Test Failure Analysis," states that an engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers, irrespective of type, which may be subject to the same failure mode. Therefore, the NRC staff finds that the TS SR requirements related to inservice operability failure evaluation are considered to be equivalent to the OM-4 requirements.

(5) Test Failure Mode Groups

OM-4, paragraph 3.2.4.2, requires that unacceptable snubber(s) be categorized into failure mode group(s). A test failure mode group shall include all unacceptable snubbers that have a given failure mode, and all other snubbers subject to the same failure mode. SR 4.7.4.e states that if during the functional testing, additional testing is required due to failure of snubbers, the unacceptable snubbers may be categorized into failure mode group(s). A failure mode group shall include all unacceptable snubbers that have a given failure mode and all other snubbers subject to the same failure mode. Once a failure mode group has been established, it can be separated for continued testing apart from the general population of snubbers. However, all unacceptable snubbers in the failure mode group shall be counted as one unacceptable snubber for additional testing in the general population. Therefore, the TS SR requirements are considered to be equivalent to the OM-4 requirements, and are acceptable.

(6) Inservice Operability Testing Corrective Actions for 10 percent sample or 37 sample plan

OM-4, paragraphs 3.2.5.1 and 3.2.5.2, requires that unacceptable snubbers be adjusted, repaired, modified, or replaced. SR 4.7.4.h states that snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers which have repairs which might affect functional test results shall be tested to meet the functional test criteria before installation. Therefore, the NRC staff finds that the TS SR corrective actions associated with unacceptable snubbers at LGS are considered to be equivalent to the OM-4 requirements.

Based on the above discussions, the NRC staff finds that snubber inservice visual examinations and functional testings, conducted in accordance with TS 3/4.7.4, provide reasonable assurance of snubber operability and provide a level of quality and safety equivalent to that of the ASME Code, Section XI, Subarticles IWF-5200(a) and (b), and IWF-5300(a) and (b). Therefore, the NRC staff finds the licensee's proposed alternative provides an acceptable level of quality and safety with respect to snubber inservice visual inspection and functional testing.

3.3 Relief Request I3R-06, revision 0

3.3.1 Component for Which Relief is Requested

Residual Heat Removal (RHR) Heat Exchanger Pressure-Retaining Shell Circumferential Welds (Shell-to-Flange Welds, two heat exchangers per unit)

3.3.2 Code Requirements

ASME Code, Section XI, Table IWC-2500-1 states that the heat exchanger shell welds require a volumetric examination in accordance with the examination requirements illustrated in ASME Code, Section XI, Figure IWC-2500-1.

ASME Code, Section XI, Table IWC-2500-1 examinations are limited to 100 percent of the pressure-retaining shell circumferential welds at gross structural discontinuities of one (1) heat exchanger (or the equivalent of one heat exchanger) during the inservice inspection interval.

Code Case N-498-4, "Alternative Requirements for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems, Section XI, Division 1," proposes as an alternative to the ASME Code requirements.

ASME Code Case N-460, "Alternative Examination Coverage for Class 1 and Class 2 Welds", as an alternative approved for use by the NRC in RG 1.147, Revision 14, "Inservice Inspection Code Case Acceptability, Section XI, Division 1," states that a reduction in examination coverage due to part geometry or interference for any ASME Code Class 1 or 2 weld is acceptable provided that the reduction is less than 10 percent, i.e., greater than 90 percent examination coverage is obtained.

3.3.3 Licensee's Proposed Alternative Examination

The ASME Code required volumetric examination for the subject welds in both LGS, Units 1 and 2 RHR heat exchangers will be performed to the maximum extent possible based on the obstructions and geometric constraints. Additionally, a VT-2 visual examination during system pressure testing per ASME Code, Section XI, Examination Category C-H, will be performed on the heat exchangers (once during each period) to verify leak tight integrity of these welds.

3.3.4 Licensee's Basis for Relief Request

The licensee requests relief pursuant to 10 CFR 50.55a(g)(5)(iii), on the basis that conformance with these code requirements is impractical as conformance would require extensive structural modifications to the RHR heat exchangers.

ASME Code, Examination Category C-A, Item Number C1.10, requires the volumetric examination of the equivalent of the welds of one heat exchanger per Unit. Due to physical limitations of the heat exchanger flange, a complete examination of the welds is limited to approximately 87.5 percent coverage of the ASME Code-required volume via transverse scans from the shell side of the weld.

The physical limitations impacting volumetric examinations also impact the use of surface examination techniques. Performance of a surface exam would involve significant radiation exposure and man hours spent on disassembly/re-assembly. This evolution would also create the potential for a leakage path of reactor coolant.

Thus, compliance with the applicable ASME Code requirements can only be accomplished by redesigning and re-fabricating the subject heat exchangers. The licensee proposes that the ASME Code requirements are impractical in accordance with 10 CFR 50.55a(g)(5)(iii).

3.3.5 NRC Staff Evaluation of Relief Request I3R-06

The ASME Code requires volumetric examination of essentially 100 percent of the weld length of the RHR pressure-retaining shell circumferential welds. The licensee is unable to obtain the ASME Code examination requirements on the basis that conformance to the ASME Code is impractical as it would require extensive modifications to the LGS, Units 1 and 2 RHR heat exchangers.

For the RHR heat exchanger shell-to-flange weld, the volumetric examination is limited due to access restrictions from the flange bolting protruding through the vessel flange. The bolting configuration prohibits the licensee from performing the ASME Code-required volumetric examinations. Using transverse scans from the shell side of the subject weld, the licensee was able to examine 87.5 percent of the volume of each RHR heat exchanger weld. The licensee noted that the limitations are also applicable to surface examination techniques. The licensee also considered disassembling of the flange mechanical connection, in order to complete the examination. However, this would result in significant radiation exposure to the licensee's personnel. Also, disassembly and reassembly of the RHR components could damage the components and possibly cause a leakage path for reactor coolant. Based on drawings provided by the licensee and description of the licensee's basis for relief, the NRC staff determined that in order for the licensee to perform the ASME Code-required volumetric and surface examinations, the subject RHR heat exchangers would have to be redesigned. The redesign of the subject RHR heat exchangers to allow a 100 percent inspection would be impractical.

As an alternative, the licensee proposed to perform the ASME Code examinations to the maximum extent possible based on the obstructions and geometric constraints described in the request for relief and illustrated in the drawings provided by the licensee. As noted above, the licensee was able to examine approximately 87.5 percent of the volume of each RHR heat exchanger weld. If significant service-induced degradation were occurring, there is reasonable assurance that evidence of it would be detected by these volumetric examinations. The licensee also proposed to perform VT-2 visual examinations during system pressure testing per ASME Code, Section XI, Examination Category C-H, on the subject heat exchangers once during each ISI period to verify the absence of leakage from the subject welds. The NRC staff determined that this combination of proposed alternatives provide reasonable assurance of the integrity of the RHR heat exchangers and components.

3.4 Relief Request I3R-07, Revision 0

3.4.1 Component for Which Relief is Requested

Pressure-Retaining Casing Welds in Residual Heal Removal and Core Spray Pumps Embeded in Concrete.

The following is a listing of the LGS pumps affected by this relief request:

1(2)A-P202 are the Unit 1 and Unit 2 "A" Residual Heat Removal Pumps
1(2)B-P202 are the Unit 1 and Unit 2 "B" Residual Heat Removal Pumps
1(2)C-P202 are the Unit 1 and Unit 2 "C" Residual Heat Removal Pumps
1(2)D-P202 are the Unit 1 and Unit 2 "D" Residual Heat Removal Pumps
1(2)A-P206 are the Unit 1 and Unit 2 "A" Core Spray Pumps
1(2)B-P206 are the Unit 1 and Unit 2 "B" Core Spray Pumps
1(2)C-P206 are the Unit 1 and Unit 2 "C" Core Spray Pumps
1(2)D-P206 are the Unit 1 and Unit 2 "D" Core Spray Pumps

3.4.2 Code Requirements

ASME Code, Section XI, Table IWC-2500-1, states that the pump casing welds require a surface examination in accordance with the examination requirements illustrated in Figure IWC-2500-8. The multiple-component concept from ASME Code, Section XI, Table IWC-2500-1, applies, and examinations are limited to either 100 percent of the welds of one of four RHR pumps and one of four Core Spray pumps (per unit), or distributed among any of the pumps of that same group (per unit) with similar design, size, function, and service in the system. The examination may be performed from either the inside or outside surface of the component.

ASME Code Case N-460, as an alternative approved for use by the NRC RG 1.147, Revision 14, states that a reduction in examination coverage due to part geometry or interference for any ASME Class 1 or 2 weld is acceptable provided that the reduction is less than 10 percent, i.e., greater than 90 percent examination coverage is obtained.

3.4.3 Licensee's Proposed Alternative Examination

In the event the subject welds become accessible upon disassembly of any one of the pumps, the welds will be surface examined from the inside surface or a VT-1 visual examination will be performed for that particular pump group to the maximum extent practicable based on the obstructions and geometric constraints. The examination method will be determined by LGS based on radiation environment data at the time access is enabled. Additionally, a VT-2 visual examination during system pressure testing per ASME Code, Section XI, Examination Category C-H, will be performed once each period by examining the surrounding area (exposed areas around the pumps where the pump casing join/merge with the concrete) for evidence of leakage in accordance with ASME Code, Section XI, IWA-5241(b).

3.4.4 Licensee's Basis for Relief Request

The licensee requests relief pursuant to 10 CFR 50.55a(g)(5)(iii) on the basis that conformance with these ASME Code requirements (ASME Code Examination Category C-G, surface examination on the equivalent of one RHR pump and one Core Spray pump per Unit) is impractical as conformance would require extensive structural modifications to these pumps. This is because the welds on each of the four RHR and four Core Spray pumps per Unit are encased in concrete and are inaccessible for surface examination.

Destruction of the concrete resulting in unnecessary cost and radiation exposure to perform the surface weld examinations is impractical without a compensating increase in safety. Access to the affected welds can only be achieved through disassembly of the pump and removal of the pump internals in order to perform the required surface examinations from the inside surface of the welds. This effort, in the absence of any other necessary pump maintenance, represents a significant expenditure of man-hours and radiation exposure to plant personnel, without a compensating increase in plant safety.

Therefore, compliance with the applicable ASME Code requirements can only be accomplished by redesigning and re-fabricating the subject pumps. This effort would be impractical in accordance with 10 CFR 50.55a(g)(5)(iii).

3.4.5 NRC Staff Evaluation of Relief Request I3R-07

The ASME Code, Section XI, requires that pump casing welds receive a surface examination in accordance with the examination requirements illustrated in ASME Code, Section XI, Figure IWC-2500-8. The multiple component concept applies to pumps, and examinations are limited to either 100 percent of the welds of one pump per unit, or distributed among any of the pumps of that same group, per unit, with similar design, size, function, and service in the system. The subject examination may be performed from either the inside or outside surface of the component. A surface examination in accordance with the examination requirements illustrated in ASME Code, Section XI, Figure IWC-2500-8. Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief on the basis that conformance with the ASME Code requirements are impractical.

Based on drawings provided by the licensee in its letter dated November 7, 2007, the NRC staff determined that it is impractical for the licensee to perform the ASME Code-required examination because the welds on each of the subject pumps are encased in concrete and are inaccessible for surface examination. Furthermore, to access the affected welds, the licensee would have to disassemble the subject pumps, remove the pump internals, and would have to perform the ASME Code-required surface examinations from the inside surface of the subject pumps. This would expose the licensee’s personnel to excessive radiation exposure just to disassemble the subject pumps to perform the examination. In order for the licensee to perform ASME Code-required examinations, the subject pumps would be required to redesigned, which would cause a burden on the licensee. Therefore, the NRC staff has determined that the ASME Code requirements are impractical to perform.

As an alternative, the licensee has proposed to examine the subject pump welds when the pumps are disassembled. The subject welds will then be surface examined from the inside surface or a VT-1 visual examination will be performed for that particular pump group to the maximum extent practicable based on the obstructions and geometric constraints. The licensee will also perform a VT-2 visual examination during system pressure testing per ASME Code once each period by examining the surrounding exposed areas around the pumps where the pump casing join/merge with the concrete for evidence of leakage in accordance with the ASME Code. The NRC staff has determined that the licensee’s proposed alternative provides reasonable assurance of the integrity of the subject pumps.

3.5 Relief Request I3R-08, Revision 0

3.5.1 Component for Which Relief is Requested

Reactor Pressure Vessel (RPV) Head Flange Seal Leak Detection System (pressure testing)

3.5.2 Code Requirements

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item Number B15.10 requires all pressure-retaining components be subject to a system leakage test and a VT-2 visual examination in accordance with ASME Code, Section XI, IWB-5220. This pressure test is to be conducted prior to plant startup following each reactor refueling outage. The pressure-retaining boundary for the test conducted at or near the end of each inspection interval shall be extended to all Class 1 pressure-retaining components per ASME Code, Section XI, IWB-5222(b).

ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item 15.30 requires a system leakage test each refueling outage to be conducted prior to plant start up following a reactor refueling outage.

Code Case N-498-4 proposes, as an alternative to the ASME Code, requirements for system leakage testing.

3.5.3 Licensee's Proposed Alternative Examination

A VT-2 visual examination on the ASME Code, Class 1 portion of the RPV Flange Leak Detection Line will be performed during each refueling outage when the RPV head is off and the head cavity is flooded above the vessel flange. The static head developed with the leak detection line filled with water will allow for the detection of any gross indications in the line. This examination will be performed each refueling outage as per the frequency specified by ASME Code, Section XI, Table IWB-2500-1.

3.5.4 Licensee's Basis for Relief Request

The licensee requests relief pursuant to 10 CFR 50.55a(g)(5)(iii), on the basis that pressure testing the RPV Flange Leak Detection Line is impractical.

The RPV Head Flange Leak Detection Line is separated from the reactor pressure boundary by one passive membrane, a silver-plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange. The purpose of this line is to indicate failure of the inner flange seal O-ring via a High Pressure Alarm in the Main Control Room.

The configuration of this system precludes manual testing while the vessel head is removed. Plugging the tap to allow system pressurization is prohibited by the tap's configuration, small size and the high test pressure requirement (approximately 1045 pounds per square inch gauge (psig). When the vessel head is installed, an adequate pressure test cannot be performed due to the fact that the inner O-ring is designed to withstand pressure in one direction only and pressurization would likely damage the O-ring.

Pressure testing of this line during the ASME Code, Class 1, System Leakage Test, is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. Purposely failing the inner O-ring to perform the ASME Code-required test would require purchasing a new set of O-rings, additional time and radiation exposure to de-tension the reactor vessel head, install the new O-rings, and then resetting and re-tensioning the reactor vessel head. The licensee considers this to impose an undue hardship and burden.

3.5.5 NRC Staff Evaluation of Relief Request I3R-08

ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item 15.30 requires a system leakage test each refueling outage to be conducted prior to plant start up following a reactor refueling outage. The licensee is unable to perform the ASME Code required pressure testing because of the design of the RPV flange leak detection line. Therefore, the licensee requested relief pursuant to 10 CFR 50.55a(g)(5)(iii).

The RPV head flange leak detection line is separated from the reactor pressure boundary by one passive membrane, a silver-plated O-ring located on the vessel flange. In addition, a second O-ring is located on the opposite side of the tap in the vessel flange. The subject line is required during plant operation and indicates the failure of the inner flange seal O-ring. Failure of the inner O-ring would result in a high pressure alarm in the plant's main control room. The licensee considered performing a manual test with the RPV head removed; however, the configuration of this system precludes manual testing while the vessel head is removed.

The configuration of the vessel tap, combined with the small size of the tap and the high test pressure requirement of approximately 1045 psig, prevents the licensee from temporarily plugging the tap. In addition, with the vessel head installed, the inner O-ring can only withstand pressure in one direction. In addition, due to the groove that the O-ring sits in and the pin/wire clip assembly as noted the licensee's Figure I3R-08.1 contained in the March 6, 2007 submittal, pressurization in the opposite direction into the recessed cavity and retainer clips would more than likely damage the O-rings. Therefore, an adequate pressure test cannot be performed.

The licensee noted that pressure testing of this line during the ASME Code, Section, XI, Class 1 system leakage test is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. If these tests were performed, they would require purposefully failing the inner O-ring and would require replacing the O-rings exposing the licensee's personnel to unnecessary radiation exposure. The NRC staff has determined that based on the drawings and the licensee's basis, the ASME Code requirements are impractical. In order for the licensee to perform ASME Code-required examinations the RPV head would be required to redesigned, which would cause a burden on the licensee. Therefore, the ASME Code requirements are impractical to perform.

As an alternative, the licensee proposed to perform a VT-2 visual examination on the ASME Code, Class 1 portion of the RPV flange leak detection line during each refueling outage when the RPV head is removed and the head cavity is flooded above the vessel flange. The static head developed with the leak detection line filled with water will allow for the detection of any gross indications in the line. The VT-2 examination will be performed each refueling outage as specified by ASME Code, Section XI, Table IWB-2500-1. The NRC staff has determined that the licensee's proposed alternative provides reasonable assurance of the integrity of the RPV flange leak detection line.

3.6 Relief Request I3R-09, revision 0

3.6.1 Component for Which Relief is Requested

Drywell Pressure Instrumentation from Penetrations X-22, X-30B, X-40E, and X-50A

3.6.2 Code Requirements

The 2001 edition through 2003 addenda of the ASME Code, Section XI, paragraph IWD-2500, states that components shall be examined and pressure tested as specified in Table IWD-2500-1. Table IWD-2500-1, Examination Category D-B, Item Number D2.10 requires performance of VT-2 visual examination during system leakage tests. The licensee requested relief from performance of system leakage tests and VT-2 visual examination requirements specified in IWD-2500-1

3.6.3 Licensee's Proposed Alternative

LGS TSs require channel checks every 12 hours to verify drywell pressure instrumentation operability. This is performed by verifying proper pressure readings. A significant tubing leak will cause an improper reading, and will be corrected and retested. The tubing and components are also included in the Integrated Leak Rate Test (ILRT) boundary. Valves HV-42-1(2)47A, B, C, and D are open to perform the ILRT. Therefore, the instrument tubing is subject to the pressure required by the ILRT (44 psig).

3.6.4 Licensee's Basis for Requesting Relief

The licensee requests relief pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the proposed alternatives will provide an acceptable level of quality and safety. The pressurizing medium in the drywell and drywell pressure instrumentation is nitrogen gas and the pressure is less than one psig during normal operation. A VT-2 examination looking for a nitrogen gas leak with less than one psig driving pressure would be inconclusive.

In addition, there are no test taps on the subject instrument tubing and plant modifications that would be required in order to perform the ASME Code, Section XI, pressure tests, resulting in a hardship. Adding another pressure test once every inspection period to satisfy ASME Code, Section XI requirements is also a hardship in that it would present a redundant testing situation resulting in additional radiation exposure to examination personnel without a compensating increase in the level of quality and safety. The proposed alternative involves the performance of channel checks of the remote pressure indicators to verify drywell pressure instrumentation operability every 12 hours in accordance with the plant TSs. Additionally, the use of 10 CFR 50 Appendix J, Option B, Integrated Leak Rate Testing provides adequate assurance of structural integrity of the tubing and components, and therefore an acceptable level of quality and safety.

3.6.5 NRC Staff Evaluation of Relief Request I3R-09

Table IWD-2500-1, Examination Category D-B requires a visual examination, VT-2, of the pressure retaining boundary during a system leakage test in accordance with IWD-5221. These examinations are to be performed each inspection period. IWD-5221 specifies that the system leakage test be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy TS surveillance requirements).

The normal operating pressure of the drywell and drywell pressure instrumentation is less than one psig. The licensee has proposed monitoring integrity of the system using channel checks of the drywell pressure instrumentation, which are required to be performed every 12 hours in accordance with technical specifications. A significant tubing leak would result in an improper reading on a pressure indicator and the situation would then be identified and corrected. In addition, the licensee has stated that the tubing and components are included in the integrated leak rate test boundary and are pressurized to 44 psig during the test. In a November 8, 2007 letter, in response to a request for additional information, the licensee discussed the extension that was previously requested to change the interval of the ILRT from 10 years to 15 years. If the ILRT extension request is approved, the licensee plans to conduct the ILRT in 2012 for Unit 1 and 2011 for Unit 2. This schedule may change, but the licensee stated the ILRT will be completed in the interval. In the November 8, 2007, letter, the licensee confirmed there are no tests that affect the internal pressure of these components other than the ILRT.

A system leakage test with less than one psig differential pressure would be inconclusive. Monitoring pressure indication every 12 hours will provide the opportunity to discover and correct any large leaks in the tubing and components. Pressurizing the system to 44 psig during the ILRT will ensure that any significant leaks are discovered and corrected at the time of the ILRT.

As the licensee has stated that the ILRT will be performed during the interval, the NRC staff finds that the alternative proposed provides an acceptable level of quality and safety.

3.7 Relief Request I3R-10, revision 0

3.7.1 Component for Which Relief is Requested

Containment Atmospheric Control Tubing to Suppression Pool Pressure and Level Instrumentation Outboard of SV-57-1(2)01

3.7.2 Code Requirements

The 2001 edition through 2003 addenda of the ASME Code, Section XI, paragraph IWD-2500, states that components shall be examined and pressure tested as specified in Table IWD-2500-1. Table IWD-2500-1, Examination Category D-B, Item Number D2.10, requires performance of VT-2 visual examination during system leakage tests. The licensee requested relief from performance of system leakage tests and VT-2 visual examination requirements specified in Table IWD 2500-1.

3.7.3 Licensee's Proposed Alternative

LGS TSs require monitoring suppression pool pressure every 12 hours to verify proper pressure. Additionally, TSs require channel checks every 24 hours to verify operability of the suppression pool level indicators. This is performed by verifying proper level readings. A significant tubing leak will give an improper reading, and will be corrected and retested. The tubing and components are also included in the ILRT boundary. Valves SV-57-101 and SV-57-201 are open to perform the ILRT. Therefore, the instrument tubing is subject to the pressure required by the ILRT (44 psig).

3.7.4 Licensee's Basis for Requesting Relief

The licensee requests relief pursuant to 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternatives will provide an acceptable level of quality and safety. The LGS Suppression Pool pressure is less than one psig during normal operation. The pressurizing medium in the Suppression Pool and level instrumentation is nitrogen gas. A visual (VT-2) examination looking for a nitrogen gas leak with less than one psig driving pressure would be inconclusive.

In addition, there are no test taps on the subject instrument tubing and plant modifications would be required in order to perform the ASME Code, Section XI, pressure tests resulting in a hardship. An additional pressure test once every inspection period to satisfy ASME Code, Section XI requirements is also a hardship in that they would present a redundant testing situation that would result in additional radiation exposure to examination personnel without a compensating increase in the level of quality and safety. The proposed alternative to perform channel checks of the remote pressure indicators to verify drywell pressure instrumentation operability every 12 hours in accordance with the plant TSs, and the use of 10 CFR 50 Appendix J, Option B Integrated Leak Rate Testing provides adequate assurance of structural integrity of the tubing and components, and therefore an acceptable level of quality and safety.

3.7.5 NRC Staff Evaluation of Relief Request I3R-10

ASME Code, Section XI, Table IWD-2500-1, Examination Category D-B, requires a visual examination, VT-2, of the pressure retaining boundary during a system leakage test in accordance with IWD-5221. These examinations are to be performed each inspection period. IWD-5221 specifies that the system leakage test be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements).

The normal operating pressure of the suppression pool is less than one psig. The licensee has proposed monitoring integrity of the system by monitoring suppression pool pressure, which is required to be done every 12 hours in accordance with technical specifications, and additionally, through channel checks to verify operability of the suppression pool level indicators, which is required by technical specifications every 24 hours. This is performed by verifying proper level readings. A significant tubing leak would result in an improper reading, and the situation would then be identified and corrected. In addition, the licensee has stated that the tubing and components are included in the integrated leak rate test boundary and are pressurized to 44 psig during the test. In a November 8, 2007, letter, in response to a request for additional information, the licensee discussed the extension that was previously requested to change the interval of the ILRT from 10 years to 15 years. If the ILRT extension request is approved, the licensee plans to conduct the ILRT in 2012 for Unit 1 and 2011 for Unit 2. This schedule is subject to change, but the licensee stated the ILRT will be completed in the interval. In the November 8, 2007 letter, the licensee confirmed there are no tests that affect the internal pressure of these components other than the ILRT.

A system leakage test with less than a one psig differential pressure will have a very low probability of finding small leaks. Monitoring suppression pool pressure indication every 12 hours and performing channel checks of the suppression pool level indicators every 24 hours will provide the opportunity to discover and correct any large leaks in the tubing and components. Pressurizing the system to 44 psig during the ILRT will ensure that any significant leaks are discovered and corrected at the time of the ILRT. As the licensee has stated, the ILRT will be performed during the interval. The NRC staff finds that the alternative proposed provides an acceptable level of quality and safety.

3.8 Relief Request I3R-11, revision 0

3.8.1 Component for Which Relief is Requested

Post-Loss-of-Coolant Accident (LOCA) Recombiner piping and H2/O2 sampling lines for Unit 1(2) as detailed below:

Post LOCA Recombiner piping HBB-1(2)28 and HBB-1(2)27 between and including "A" Recombiner and valves HV-57-1(2)61 and HV-57-1(2)62. HBB-1(2)26 and HBB-1(2)24 between and including "B" Recombiner and valves HV-57-1(2)63 and HV-57-1(2)64.

Hydrogen/oxygen sampling lines HCB-1(2)16 and HCB-1(2)17, between connections on the Combustible Gas Analyzer Package 1(2)0-S205, and valves SV-57-1(2)59, SV-57-1(2)41, SV-57-1(2)42 and SV-57-1(2)47B, SV-57-1(2)43, SV-57-1(2)44 and SV-57-1(2)46B, and SV-57-1(2)45 (HCB-1(2)17). HCB-1(2)16 and HCB-1(2)17, between connections on the Combustible Gas Analyzer Package 1(2)0-S206, and valves SV-57-1(2)84 and SV-57-1(2)46A, SV-57-1(2)86 and SV-57-1(2)47A, SV-57-1(2)95, SV-57-1(2)90 and 57-1(2)090, and SV-57-1(2)85 (HCB-1(2)17).

3.8.2 Code Requirements

The 2001 edition through 2003 addenda of the ASME Code, Section XI, paragraph IWC-2500, states that components shall be examined and pressure tested as specified in Table IWC-2500-1. Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires performance of VT-2 visual examination during system leakage tests. The licensee requested relief from performance of system leakage tests and VT-2 visual examination requirements specified in Table IWC-2500-1.

3.8.3 Licensee's Proposed Alternative

System Contaminated Pipe Inspections (CPIs) are performed once per refueling outage on post-Loss-of-Coolant-Accident recombiner piping. During CPI testing associated with the Leak Reduction Program (Updated Final Safety Analysis Report (UFSAR) section 6.2.8), this piping is pressurized to 44 psig. CPIs for this system are performed similar to 10 CFR 50, Appendix J, Local Leak Rate Testing, and as such, offer the following advantages over system pressure tests:

A. CPIs are performed more frequently than periodic system leakage tests;

B. CPIs have the ability to quantify leakage that is not feasible with a visual (VT-2) examination on this air filled piping; and

C. CPIs conservatively include through-valve leakage that would not be identified in a visual (VT-2) examination.

3.8.4 Licensee's Basis for Requesting Relief

The licensee requests relief pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the proposed alternatives will provide an acceptable level of quality and safety. The pressurizing medium in these lines is essentially nitrogen gas and the pressure is less than one psig during normal operation or the lines are isolated. A VT-2 examination looking for a nitrogen gas leak with less than one psig driving pressure would be inconclusive.

In addition, for the hydrogen/oxygen sampling lines, the combustible gas analyzer continuously samples containment. A tubing leak will cause improper (high) readings that would be identified, corrected and retested.

The ASME Code, Section XI, IWC-5210(b)(2) allows for gas tests which permit location and detection of through-wall leakage. System CPIs will be utilized to meet the ASME Code, Section XI, IWC-5000 pressure testing requirements and will be maintained and controlled independent of the ASME Code, Section XI program. In the event the CPI fails to meet its acceptance criteria, further testing would be performed to determine the location of the leaks and appropriate corrective maintenance. An appropriate retest would then be performed.

3.8.5 NRC Staff Evaluation of Relief Request I3R-11

ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H requires a visual examination, VT-2, of the pressure retaining boundary during a system leakage test in accordance with ASME Code, Section XI, IWC-5220. These examinations are to be performed each inspection period. IWC-5220 specifies that the system leakage test be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements).

The hydrogen recombiner and combustible gas analyzer piping is either isolated or pressurized to less than one psig during normal plant operation. In lieu of the required ASME Code, Section XI, VT-2 examination during system leakage testing, the licensee has proposed testing the system under the Leak Reduction Program using CPIs. The Leak Reduction Program is described in the LGS UFSAR section 6.2.8. Within this section, the UFSAR states that gas systems will be bubble leak tested with a zero leakage acceptance criteria or leakage will be quantified by means of a pressure decay or helium test. The UFSAR also states that components whose leakage contributes significantly to the total leak rate or increases substantially between tests will be repaired to maintain total leakage as-low-as-practical. During these CPI tests, the piping is pressurized to 44 psig. The CPIs are performed once per refueling outage, which is more frequent than the leakage tests required by Table IWC-2500-1. As noted above, the CPIs have the ability to quantify leakage and conservatively include through-valve leakage, which would not be identified during a VT-2 examination.

A system leakage test with less than a one psig differential pressure would be inconclusive and ineffective at finding leaks. Since the CPIs pressurize the piping to 44 psig, performing CPIs at a frequency of once per refueling outage will provide an acceptable level of quality and safety to that of a VT-2 visual examination during a system leakage test as required by ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H.

3.9 Relief Request I3R-12, revision 0

3.9.1 Component for Which Relief is Requested

Class 2 Primary Containment Atmospheric Control piping, as follows:

Hydrogen/oxygen sample lines HCB-1(2)16, between and including containment penetrations X-28A and X-28B and valves SV-57-1(2)42, SV-57-1(2)43, SV-57-1(2)44 and SV-57-1(2)95.

Drywell low flow nitrogen makeup line HCB-1(2)16, between and including containment penetration X-62 and valves HV-57-1(2)16 and SV-57-1(2)59.

Hydrogen/oxygen sample lines HCB-1(2)16, between and including containment penetrations X-221A and valves SV-57-1(2)41 and SV-57-1(2)84.

Nitrogen purge line HBB-1(2)25, between and including valves HV-57-1(2)09, HV-57-1(2)21 and HV-57-1(2)31.

Drywell air purge line HBB-1(2)24, between and including valves HV-57-1(2)23 and HV-57-1(2)35. Suppression pool air purge line HBB-1(2)26, between and including valves HV-57-1(2)24 and HV-57-1(2)47.

Drywell purge to standby gas treatment line HBB-1(2)27, between and including valves HV-57-1(2)14 and HV-57-1(2)15, and line HCB-1(2)17, between and including connection to line HBB-127 and valve SV-57-145.

Suppression pool low flow nitrogen makeup line HCB-1(2)16, between and including containment penetration X-220A, valve SV-57-1(2)90 and connection to drywell low flow nitrogen makeup line HCB-1(2)16 to valve 57-1(2)068B.

Hydrogen/oxygen sample line HCB-1(2)16, between and including containment penetration X-221B and valves SV-57-1(2)86 and HV-55-126.

Drywell purge exhaust bypass line HBB-1(2)27, between and including valves 57-1807(2815) and HV-57-1(2)17.

Suppression pool purge exhaust bypass line HBB-1(2)28, between and including valves 57-1810(2818) and HV-57-1(2)18.

Suppression pool purge air exhaust lines HBB-1(2)28 and HCB-1(2)17, between and including valves HV-57-1(2)04, HV-57-1(2)12 and SV-57-1(2)85.

3.9.2 Code Requirements

The 2001 edition through 2003 addenda of the ASME Code, Section XI, paragraph IWC-2500, states that components shall be examined and pressure tested as specified in Table IWC-2500-1. Table IWC-2500-1, Examination Category C-H, Item Number C7.10 requires performance of VT-2 visual examination during system leakage tests. The licensee requested relief from performance of system leakage tests and VT-2 visual examination requirements specified in Table IWC-2500-1.

3.9.3 Licensee's Proposed Alternative

10 CFR 50 Appendix J, Option B, Local Leak Rate Testing (LLRT) is currently performed once per refueling outage. During LLRTs, the subject piping is pressurized to 44 psig, a substantially higher pressure than that developed during a periodic system functional test. As such, the LLRT offers the following advantages over system pressure tests:

A. LLRTs are performed more frequently than periodic system leakage tests;

B. LLRTs have the ability to quantify leakage that is not feasible with a visual (VT-2) examination on this essentially gas-filled piping; and

C. LLRTs conservatively include through-valve leakage that would not be identified in a visual (VT-2) examination

3.9.4 Licensee's Basis for Requesting Relief

The licensee requests relief pursuant to 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternatives will provide an acceptable level of quality and safety. During normal plant operation, this piping is either isolated or less than one psig (normal containment pressure). The pressurizing medium is essentially nitrogen gas. A visual (VT-2) examination looking for a nitrogen gas leak with less than one psig driving pressure would be inconclusive.

IWC-5210(b)(2) allows for gas tests which permit location and detection of through-wall leakage. 10 CFR 50 Appendix J, Option B, Local Leak Rate Testing will be utilized to meet the ASME Code, Section XI, IWC-5000 pressure testing requirements and will be maintained and controlled independent of the ASME Code, Section XI, program. In the event the LLRT fails to meets its acceptance criteria, further testing would be performed to determine the location of the leaks and appropriate corrective maintenance. An appropriate retest would be performed.

3.9.5 NRC Staff Evaluation of Relief Request I3R-12

ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H requires a visual examination, VT-2, of the pressure retaining boundary during a system leakage test in accordance with ASME Code, Section XI, IWC-5220. These examinations are to be performed each inspection period. IWC-5220 specifies that the system leakage test be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements).

During normal plant operation, the piping addressed in this relief request is either isolated or pressurized to less than one psig (normal containment pressure). As an alternate to the testing requirements of ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H, the licensee proposes to use 10 CFR 50 Appendix J, Option B, Local Leak Rate Testing (LLRTs), which is currently performed once per refueling outage. Performance at a frequency of once per outage is more frequent than once per period as is specified in Table IWC-2500-1, Examination Category C-H. During LLRTs, the subject piping is pressurized to 44 psig, a higher pressure than that developed when the systems are in service performing their normal operating function. Leakage is quantified during the LLRTs, and compared to acceptance criteria. In the event the LLRT fails to meets its acceptance criteria, further testing would be performed to determine the location of the leaks and appropriate corrective maintenance.An appropriate retest would be performed.

Since the LLRTs are conducted at 44 psig, use of the LLRTs to detect leakage at a frequency of every refueling outage or at least once per ISI inspection period provides an equivalent level of quality and safety to that of a VT-2 visual examination during a system leakage test as required by ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H.

4.0 CONCLUSION

The NRC staff has reviewed the licensee's submittal and concludes that the ASME Code requirements are impractical for the welds, and components listed in revision 0 of RR Nos. I3R-06, I3R-07, and I3R-08, for the LGS, Units 1 and 2 Third 10-year ISI interval. Furthermore, the NRC staff concludes that the licensee’s proposed VT-2 visual examinations, the plant’s leakage and radiation monitoring system, and/or volumetric examinations of the subject welds and components provide reasonable assurance of the structural integrity of the subject welds and components. Therefore, Relief Request Nos. I3R-06, I3R-07, and I3R-08 are granted, pursuant to 10 CFR 50.55a(g)(6)(i), for the for LGS, Units 1 and 2 Third 10-year ISI interval which began on February 1, 2007 and is scheduled to conclude on January 31, 2017. The NRC staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) for these requests is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject requests for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The NRC staff has reviewed the licensee’s submittal and determined that the proposed alternatives to the requirements of the ASME Code, Section XI, regarding revision 0 of Relief Request Nos I3R-02, I3R-05, I3R-09, I3R-10 and I3R-11, will provide an acceptable level of quality and safety. Therefore, the proposed alternatives for these requests are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for LGS Units 1 and 2, for the Third 10-year ISI interval, which began on February 1, 2007 and is scheduled to conclude on January 31, 2017. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

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