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10 CFR 50.55a

RS-09-072
December 30, 2009

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Relief Requests Associated with the Third Inservice Inspection Interval

In accordance with 10 CFR 50.55a, "Codes and standards," paragraphs (a)(3)(i), (a)(3)(ii), and (g)(5)(iii), Exelon Generation Company, LLC (EGC), hereby requests NRC approval of the attached relief requests associated with the Third Inservice Inspection (ISI) Interval for Clinton Power Station (CPS), Unit 1. The third interval of the CPS, Unit 1, ISI program will comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2004 Edition. The latest edition and addenda of the code incorporated by reference in 10CFR50.55a(b)(2) of the regulation is the 2004 Edition.

Proposed Relief Request No. I3R-01 requests approval for alternate risk-informed selection and examination criteria for examination Category B-F, B-J, C-F-1, and C-F-2 pressure retaining piping welds. Proposed Relief Request No. I3R-02 requests approval of alternative requirements for nozzle-to-vessel weld and inner radius examinations. Proposed Relief Request No. I3R-03 requests relief from certain pressure testing requirements for the reactor pressure vessel head flange seal leak detection system. Proposed Relief Request No. I3R-04 requests approval of an alternative to performance of VT-2 visual inspections for Instrument Air system piping. Proposed Relief Request No. I3R-05 requests relief from the requirement to perform visual inspections on high pressure core spray, low pressure core spray, and residual heat removal pump casings due to the fact that the pump casings are inaccessible. The bases for these relief requests are provided in Attachments 1, 2, 3, 4, and 5, respectively. Relief Requests similar or identical to I3R-01, I3R-02, and I3R-4 have previously been approved for use at CPS.

EGC requests approval of these requests by December 30, 2010, to support implementation of the third 10-year ISI interval.

There are no regulatory commitments contained within this letter.

Should you have any questions concerning this letter, please contact Mr. Mitchel A. Mathews at (630) 657-2819.

Sincerely,

Jeffrey L. Hansen

Jeffrey L. Hansen
Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1. 10 CFR 50.55a Request Number I3R-01
2. 10 CFR 50.55a Request Number I3R-02
3. 10 CFR 50.55a Request Number I3R-03
4. 10 CFR 50.55a Request Number I3R-04
5. 10 CFR 50.55a Request Number I3R-05

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1 and 2
Reference:	Table IWB-2500-1, Table IWC-2500-1
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, B5.20, B9.11, B9.21, B9.31, B9.32, B9.40, C5.11, C5.51, and C5.81
Description:	Alternate Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	Pressure Retaining Piping

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The code of record for the third ten-year Inservice Inspection Program interval at Clinton Power Station (CPS) is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition.

3.0 APPLICABLE CODE REQUIREMENT:

Table IWB-2500-1, Examination Category B-F, requires volumetric and surface examinations on all welds for Item Number B5.10 and surface examinations for all welds for Item Number B5.20.

Table IWB-2500-1, Examination Category B-J, requires volumetric and surface examinations on a sample of welds for Item Numbers B9.11 and B9.31 and surface examinations on a sample of welds for Item Numbers B9.21, B9.32, and B9.40. The weld population selected for inspection includes the following:

1. All terminal ends in each pipe or branch run connected to vessels.

2. All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:

   a. primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.

   b. cumulative usage factor *U* of 0.4.

3. All dissimilar metal welds not covered under Examination Category B-F.

4. Additional piping welds so that the total number of circumferential butt welds, branch connections, or socket welds selected for examination equals 25% of the circumferential butt welds, branch connection, or socket welds in the reactor coolant piping system. This total does not include welds exempted by Paragraph IWB-1220.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and surface examinations on a sample of welds for Item Numbers C5.11 and C5.51 and surface examinations on a sample of welds for Item Number C5.81. The weld population selected for inspection includes the following:

1. Welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or of all carbon and low alloy steel welds (Examination Category C-F-2) not exempted by Paragraph IWC-1220. (Some welds not exempted by Paragraph IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and 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 ISI Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or carbon and low alloy welds (Examination Category C-F-2) in each system;

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

    c. within each system, examinations shall be distributed between piping sizes prorated to the degree practicable.

4.0 <u>REASON FOR REQUEST:</u>

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative utilizing Reference 1 along with two enhancements from Reference 4 will provide an acceptable level of quality and safety.

As stated in "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" (i.e., Reference 2):

> *"The staff concludes that the proposed RISI Program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10CFR50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."*

The initial Clinton Power Station (CPS) Risk-Informed Inservice Inspection (RISI) program was submitted during the First Period of the Second Inspection Interval. This initial RISI Program was developed in accordance with EPRI TR-112657, Revision B-A, as supplemented by Code Case N-578-1. The program was approved for use by the NRC via a Safety Evaluation as transmitted to Exelon (Reference 5).

The transition from the 1989 Edition to the 2004 Edition of ASME Section XI for Clinton Power Station's Third Inspection Interval does not impact the currently approved Risk-Informed ISI evaluation methods and process used in the Second Inspection Interval, and the requirements of the new Code Edition/Addenda will be implemented as detailed in the CPS ISI Program Plan.

The Risk Impact Assessment completed as part of the original baseline RISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RISI methodology. For the Third Interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RISI methodology and evaluation will be maintained for the new interval. The original methodology of the calculation has not changed, and the change in risk was simply re-assessed using the initial 1989 ASME Section XI program prior to RISI and the new element selection for the Third Interval RISI Program. This same process has been maintained in each revision to the CPS RISI assessment that has been performed to date.

The actual "evaluation and ranking" procedure including the Consequence Evaluation and Degradation Mechanism Assessment processes of the currently approved (Reference 5) RISI Program remain unchanged and are continually applied to maintain the Risk Categorization and Element Selection methods of EPRI TR-112657, Revision B-A. These portions of the RISI Program have been and will continue to be reevaluated and revised as major revisions of the site Probabilistic Risk Assessment (PRA) occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, and Element Selection steps encompass the complete program process applied under the CPS RISI Program.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

The proposed alternative originally implemented in the risk informed in-service inspection plan for Clinton Power Station (Reference 3), along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10CFR50.55a(a)(3)(i). This original program along with these same two enhancements is currently approved for Clinton Power Station's Second Inspection Interval as documented in Reference 5.

The Third Inspection Interval RISI Program will be a continuation of the current application and will continue to be a living program as described in the Reason For Request section of this relief request. No changes to the evaluation methodology as currently implemented under EPRI TR-112657, Revision B-A, are required as part of this interval update. The following two enhancements will continue to be implemented.

a. In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RISI Selected Examinations" of EPRI TR-112657, CPS will utilize the requirements of Paragraph -2430, "Additional Examinations" contained in Code

Case N-578-1 (Reference 4). The alternative criteria for additional examinations contained in Code Case N-578-1 provide a more refined methodology for implementing necessary additional examinations. The reason for this selection is that the guidance discussed in EPRI TR-112657 includes requirements for additional examinations at a high level, based on service conditions, degradation mechanisms, and the performance of evaluations to determine the scope of additional examinations, whereas ASME Code Case N-578-1 provides more specific and clearer guidance regarding the requirements for additional examinations that is structured similar to the guidance provided in ASME Section XI, Paragraphs IWB-2430 and IWC-2430. Additionally, similar to the current requirements of ASME Section XI, CPS intends to perform additional examinations that are required due to the identification of flaws or relevant conditions exceeding the acceptance standards, during the outage the flaws are identified.

b. To supplement the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods" of EPRI TR-112657, CPS will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" contained in Code Case N-578-1 (Reference 4). To implement Note 10 of this table, paragraphs and figures from the 2004 Edition, No Addenda of ASME Section XI (Clinton Power Station's Code of record for the Third Interval) will be utilized which parallel those referenced in the Code Case for the 1989 Edition. Table 1 of Code Case N-578-1 will be used as it provides a detailed breakdown for examination method and categorization of parts to be examined. Additionally, Section 4 of EPRI TR-112657 states "Application of RISI uses NDE techniques that are designed to be effective for specific degradation mechanisms and examination locations." Section 4 also identifies methods of examination for each degradation mechanism with the primary method being ultrasonic testing (UT) techniques. However, EPRI TR-112657 does not identify the examination volumes for components without a degradation mechanism. In addition, EPRI TR-112657 does not specify examination volumes and methods for socket welds. CPS has requested to use the examination methods from Code Case N-578-1 instead of the methods from EPRI TR-112657. 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.

CPS uses UT techniques for RISI volumetric examinations.

For the components addressed by the RISI Program, ASME Section XI focuses primarily on weld examinations. Risk Informed examination volumes also include portions of piping and fitting base materials that are susceptible to particular degradation mechanisms.

The ASME Section XI, Mandatory Appendix I, "Ultrasonic Examinations," specifies that UT examination procedures, equipment, and personnel used to detect and size flaws in piping welds shall be qualified by performance demonstration in accordance with ASME Section XI Appendix VIII, "Performance Demonstration for Ultrasonic Examination

Systems." The RISI Program complies with Appendix VIII for weld examinations. In cases where the examination requirements cannot be met, CPS will submit a request for relief in accordance with 10CFR50.55a, "Codes and standards."

The examination methods are designed to be effective for specific degradation mechanisms and examination locations. The volumetric scanning will be in both axial and circumferential directions to detect the flaws in these orientations.

Additionally, all CPS dissimilar metals (DM) welds, as characterized in ASME Section XI, Article 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 RISI categories, and appropriate elements were selected per the category requirements for examination during the Second Inspection Interval.

Piping welds, including DM welds in vessel nozzles, 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 RISI Program population. They are contained in the CPS Intergranular Stress Corrosion Cracking (IGSCC) Augmented Inspection Program (2.2.4) and are subject to the inspection requirements of BWRVIP-75-A "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules". Furthermore, all piping welds and welds, including DM welds in vessel nozzles classified as Category A (resistant material) per BWRVIP-75-A are included in the RISI Program.

The CPS RISI Program, as developed in accordance with EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, and 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by the EPRI TR-112657 as supplemented by Code Case N-578-1.

Enclosure 1 contains a summary of the Regulatory Guide 1.200, Revision 1, evaluation performed on Revision CL06C of the CPS PRA model and the impact of the identified gaps on the technical adequacy of the CPS PRA model to support this RISI application.

In addition to this risk-informed evaluation, selection, and examination procedure, all ASME Section XI piping components, regardless of risk classification, will continue to receive Code required pressure testing as part of the current ASME Section XI program. VT-2 visual examinations are within the ASME pressure boundary and are examined as part of the system leakage tests required by ASME Section XI. These examinations are scheduled in accordance with the CPS pressure-testing program, which remains unaffected by the RISI Program.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the Third Ten-Year Inspection Interval for Clinton Power Station.

7.0 PRECEDENTS:

Similar relief requests have been approved for:

CPS Second Inspection Interval Relief Request 4208 was authorized per SER dated April 8, 2002. The Third Inspection Interval Relief Request utilizes an identical RISI methodology as was previously approved.

Peach Bottom Atomic Power Station Fourth Inspection Interval Relief Request I4R-44 was authorized per SER dated February 26, 2009.

8.0 REFERENCES:

1. Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999

2. Letter from W. H. Bateman (NRC) to G. L. Vine (EPRI) "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)," dated October 28, 1999

3. Initial Risk-Informed Inservice Inspection Evaluation, Revision 0 - Clinton Power Station, dated October 15, 2001 (Letter RS-01-219 from K. A. Ainger (AmerGen) to the NRC, Clinton Power Station Second Interval Inservice Inspection Program - Relief Request 4208, "Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds Risk-Informed Inservice Inspection Program," dated October 15, 2001)

4. American Society of Mechanical Engineers (ASME) Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B"

5. Letter from A. J. Mendiola, (NRC) to J. L. Skolds (Exelon) "Clinton Power Station, Unit 1 - Risk-Informed Inservice Inspection Program, Relief Request 4208 (TAC No. MB53211)," dated April 8, 2002

Summary Statement of Clinton PRA Model Capability for Use in Risk-Informed Licensing Actions

Introduction

Exelon Generation Company (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Clinton PRA.

PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC Nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities for equipment that can have a significant impact on the PRA model are updated approximately every four years.

In addition to these activities, EGC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for EGC Nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks

(corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. The most recent update of the Clinton PRA model (designated as the 2006C model) was completed in March 2007.

PRA Self Assessment and Peer Review

Several assessments of technical capability have been made and continue to be planned for the Clinton PRA model. A chronological list of the assessments performed includes the following:

- An independent PRA peer review was conducted under the auspices of the BWR Owners' Group (BWROG) in 2000, following the Industry PRA Peer Review process [1]. This peer review included an assessment of the PRA model maintenance and update process.
- A self assessment analysis was previously performed against Addenda B of the ASME PRA Standard (ASME RA-Sb-2005, [4]) and the draft of Revision 1 Regulatory Guide 1.200 (DG-1161) to support scoping/planning for the CPS PRA 2006 update project.
- During 2005 and 2006 the CPS PRA model results were evaluated in the BWROG PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process.
- The Clinton 2006 PRA self assessment was revised in 3Q09 to address consistency with Regulatory Guide 1.200 Revision 1 [6] in preparation for the CPS 2009 PRA peer review.
- A current industry peer review of the Clinton PRA is scheduled for the fourth quarter of 2009.

A summary of the disposition of the BWROG PRA Peer Review facts and observations (F&Os) for the Clinton PRA models was documented as part of the statement of PRA capability for MSPI. All of the significance level "A" F&Os have been resolved and 90 of the 92 significance level "B" F&Os have been resolved. The remaining two open significance level "B" F&Os are not significant for the current model, as noted in Table 1.

A self assessment of the 2003 CPS PRA was performed in support of the CPS 2006 PRA Update. This Gap Analysis was performed using Addenda B of the ASME PRA Standard (ASME RA-Sb-2005) and the draft of Revision 1 Regulatory Guide 1.200 (DG-1161). Potential gaps to Capability Category II of the Standard were identified and used to plan the Clinton 2006 PRA Update.

The Clinton 2006 PRA self assessment was revised in 3Q09 in preparation for the CPS 2009 peer review to address consistency with Regulatory Guide 1.200 Revision 1 [6], including the NRC positions stated in Appendix A of [6] and the clarifications in [5], and to identify which gaps were closed following completion of the CPS 2006 PRA update and the CPS PRA 2009 internal flooding update. Identified gaps have been considered with particular focus on technical elements important to the risk-informed inservice inspection relief request.

A summary of this assessment of the current open items, including the partially resolved items, relative to the RI-ISI relief request is provided in attached Table 2. The remaining gaps, including any new items that may be identified in the planned industry peer review, will be reviewed for consideration during future model updates. The currently identified items are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. These items are or are being documented in the PRA Updating Requirements Evaluation (URE) database so that they can be tracked and their potential impacts accounted for in applications where appropriate. In addition, plant changes made since the last PRA update have been reviewed and determined to not have a significant PRA impact. These items are also documented in UREs for consideration in future PRA updates, as appropriate.

General Conclusion Regarding PRA Capability

The Clinton PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection

In the risk-informed inservice inspection (RI-ISI) program at Clinton, the EPRI Risk-informed ISI methodology [7] is used to define alternative inservice inspection requirements. Plant-specific PRA-derived risk significance information is used during the RI-ISI plan development to support the consequence assessment, risk ranking, element selection and risk impact steps.

The importance of PRA consequence results, and therefore the scope of PRA technical capability, is tempered by three fundamental components of the EPRI methodology.

First, PRA consequence results are binned into one of three conditional core damage probability (CCDP) and conditional large early release probability (CLERP) ranges before any welds are chosen for RI-ISI inspection as illustrated below. Broad ranges are used to define these bins so that the impact of uncertainty is minimized and only substantial PRA changes would be expected to have an impact on the consequence ranking results.

Consequence Results Binning Groups		
Consequence Category	**CCDP Range**	**CLERP Range**
High	CCDP > 1E-4	CLERP > 1E-5
Medium	1E-6 < CCDP ≤ 1E-4	1E-7 < CLERP ≤ 1E-5
Low	CCDP ≤ 1E-6	CLERP ≤ 1E-7

The risk importance of a weld is therefore not tied directly to a specific PRA result. Instead, it depends only on the range in which the PRA result falls. As a consequence, any PRA modeling uncertainties would be mitigated by the wide binning provided in the methodology. Additionally, conservatism in the binning process (e.g., as would typically be introduced through PRA attributes meeting ASME PRA Standard Capability Category I versus II) will tend to result in a larger inspection population.

Secondly, the impacts of particular PRA consequence results are further dampened by the joint consideration of the weld failure potential via a non-PRA-dependent damage mechanism assessment. The results of the consequence assessment and the damage mechanism assessment are combined to determine the risk ranking of each pipe segment (and ultimately each element) according to the EPRI Risk Matrix. The Risk Matrix, which equally takes both assessments into consideration, is reproduced below.

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	**CONSEQUENCES OF PIPE RUPTURE** IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	**LOW**	**MEDIUM**	**HIGH**
HIGH FLOW ACCELERATED CORROSION	**LOW** **Category 7**	**MEDIUM** **Category 5**	**HIGH** **Category 3**	**HIGH** **Category 1**
MEDIUM OTHER DEGRADATION MECHANISMS	**LOW** **Category 7**	**LOW** **Category 6**	**MEDIUM** **Category 5**	**HIGH** **Category 2**
LOW NO DEGRADATION MECHANISMS	**LOW** **Category 7**	**LOW** **Category 7**	**LOW** **Category 6**	**MEDIUM** **Category 4**

Thirdly, the EPRI RI-ISI methodology uses an absolute risk ranking approach. As such, conservatism in either the consequence assessment or the failure potential assessment will result in a larger inspection population rather than masking other important components. That is, providing more realism into the PRA model (e.g. by meeting higher capability categories) most likely would result in a smaller inspection population.

These three facets of the methodology reduce the importance and influence of PRA on the final list of candidate welds.

The limited manner of PRA involvement in the RI-ISI process is also reflected in the risk-informed license application guidance provided in Regulatory Guide 1.174 [8].

Section 2.2.6 of Regulatory Guide 1.174 provides the following insight into PRA capability requirements for this type of application:

> *There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.*
>
> *An example is risk-informed inservice inspection (RI-ISI). In this application, risk significance was used as one criterion for selecting pipe segments to be periodically examined for cracking. During the staff review it became clear that a high level of emphasis on PRA technical acceptability was not necessary. Therefore, the staff review of plant-specific RI-ISI typically will include only a limited scope review of PRA technical acceptability.*

In addition to the above, it is noted that welds determined to be low risk significant are not eliminated from the ISI program on the basis of risk information. For example, the risk significance of a weld may fall from Medium Risk Ranking to Low Risk Ranking, resulting in it not being a candidate for inspection. However, it remains in the program, and if, in the future, the assessment of its ranking changes (either by damage mechanism or PRA risk) then it may again become a candidate for inspection. If it is discovered during the RI-ISI update process that a weld is now susceptible to flow-accelerated corrosion (FAC), inter-granular stress corrosion cracking (IGSCC), or microbiological induced cracking (MIC) in the absence of any other damage mechanism, then it is addressed in an "augmented" program where it is monitored for those special damage mechanisms. That occurs no matter what the Risk Ranking of the weld is determined to be.

<u>Conclusion Regarding PRA Capability for Risk-Informed ISI</u>

The Clinton PRA model continues to be suitable for use in the risk-informed inservice inspection application. This conclusion is based on:

- PRA maintenance and update processes in place,
- PRA technical capability evaluations that have been performed and are being planned, and
- RI-ISI process considerations, as noted above, that demonstrate the relatively limited sensitivity of the EPRI RI-ISI process to PRA attribute capability beyond ASME PRA Standard Capability Category I.

In support of the PRA analyses for the Clinton 10-year interval evaluation using the CL06C PRA model, the remaining gaps to the PRA standard have been reviewed to determine which, if any, would merit RI-ISI-specific sensitivity studies in the presentation of the application results. The result of this assessment concluded that no additional sensitivity studies are merited.

References

1. Boiling Water Reactors Owners' Group, *BWROG PSA Peer Review Certification Implementation Guidelines*, Revision 3, January 1997
2. American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME RA-S-2002, New York, New York, April 2002
3. U.S. Nuclear Regulatory Commission, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Draft Regulatory Guide DG-1122, November 2002
4. American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-Sb-2005, New York, New York, December 2005
5. U.S. Nuclear Regulatory Commission Memorandum to Michael T. Lesar from Farouk Eltawila, "Notice of Clarification to Revision 1 of Regulatory Guide 1.200," for publication as a Federal Register Notice, July 27, 2007
6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007
7. *Revised Risk-Informed Inservice Inspection Evaluation Procedure,* EPRI TR-112657, Revision B-A, December 1999
8. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 1, November 2002

Table 1: Impact of Open Significant PRA Peer Review Findings for the Clinton PRA Model

Peer Review Element	FACTS & OBSERVATIONS (F&Os)			Impact Assessment
	ID	Priority	Summary	
TH-8	TH-8-1	B	Additional plant specific room heat-up calculations (or enhancements to existing calculations) should be performed to support modeling assumptions regarding room-cooling requirements. Areas specifically identified are Control Room, Reactor Core Isolation Cooling (RCIC), Low Pressure Core Spray, Low Pressure Coolant Injection, and Switchgear rooms.	Non-Significant Impact: Primarily a documentation issue. The PRA already makes appropriate assumptions regarding the need for room cooling in the appropriate areas. No impact on RI-ISI application.
HR-6	HR-6-1	B	All pre-initiator Human Error Probabilities (HEPs) in the Clinton PSA model are based on screening estimates. For post-initiator screening HEPs with Risk Achievement Worths (RAWs) greater than 1.1, the HEPs were re-evaluated with more detailed calculations. For consistency sake, the pre-initiator HEP calculations should follow the same approach.	Non-significant impact: Pre-initiator HEPs contribute approximately 2% of CDF. Fine-tuning the HEPs for pre-initiators would be expected to reduce the relative importance of these events. No significant impact on RI-ISI application.

TABLE 2: Status of Identified Gaps to Capability Category II of the ASME PRA Standard

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
1	Review initiating event precursors in identifying the initiating events to be modeled. A rigorous explicit assessment of all the events in NUREG-1275 has not been performed.	IE-A7	Deferred: Explicit analysis of event precursors is judged not to provide significant insights to the CPS IE analysis, which includes initiating events known to be relevant to BWR-6 plants in general and CPS in particular. This type of activity is known to have been performed for another BWR plant (review of hundreds of events INPO SENs, SOERs, SERs, and NRC SECY letters on precursors) and no new initiating events were identified. It is expected that future industry studies will provide this generic assessment.	No Impact: Documentation item.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
2	Assumptions regarding loss of switchgear room cooling should be supported by room cooling calculation.	IE-C4 AS-B3 SC-B2 SC-C1 SC-C2 SY-A17 SY-A19 SY-A20 SY-B7 SY-B8	Deferred: Switchgear room cooling calculations have not been performed at this time but are being considered for a future update.	Non-Significant Impact: The PRA already makes appropriate assumptions regarding the need for room cooling and explicitly models room cooling in certain areas. Modeling cooling failures for switchgear might make the Shutdown Service Water (SX) system piping going to the SX cooler more important, but this is Class 3 piping, which is not in the scope of the RI-ISI program. SX failures already have high importance for DG cooling, Emergency Core Cooling System (ECCS) room cooling and Decay Heat Removal (DHR), and more extensive Switchgear heat removal modeling would not change this.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
3	The following should be considered in the pre-initiator HEP evaluation: 1) A list of the PRA systems to consider for test and maintenance actions 2) Rules for identifying and screening test and maintenance actions from the PRA 3) A list of procedures reviewed, the potential test and maintenance actions associated with the procedures, and the disposition of the action (screened or evaluated). 4) Identify T&M activities that require realignment of the system outside its normal operational or stand by status.	HR-A1, HR-A2 HR-A3, HR-C2 HR-C3	<u>Deferred</u>: The CPS PRA includes over 100 pre-initiator HEPs in the model, and the approach is believed to meet the intent of the identified SRs. Performing this task with a more rigorous review and documentation of test and maintenance procedures is judged not to have significant impact on the PRA model and results. The current methodology and documentation for identifying pre-initiator HEPs is judged adequate to support applications of the PRA. Any additional documentation enhancement would not result in increasing the number of pre-initiator HEPs included in the model or significantly impact their relative importances.	<u>No Impact</u>: This is primarily a Documentation item.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
4	Pre-initiator HEPs in the Clinton PRA model are based on screening estimates (URE 2001-084, peer review F&O HR-6-1), should not use screening values for dominant pre-initiator HEPs.	HR-B1, HR-B2 HR-D1, HR-D2 HR-D3, HR-D4	Deferred: Future updates of the CPS PRA will consider explicit/specific pre-initiator HEP calculations. The current calculations are based on representative procedures/practices for similar pre-initiator HEPs. The current estimates are generally higher error probabilities than would be obtained if various explicit recovery factors and testing frequencies were applied in specific HEP calculations for each pre-initiator. The impact on the model is non-significant, pre-initiator HEPs contribute approximately 2% to the CL06C CDF.	Non-significant impact: Pre-initiator HEPs contribute approximately 2% of CDF. Fine-tuning the HEPs for pre-initiators would be expected to reduce the relative importance of these events.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
5	Failure data development using surveillance test data should fulfill the requirements of DA-C10, and should be documented appropriately. Review surveillance test procedures and identify all failure modes that are fully tested by the procedures. Include data for the failure modes that are fully tested. The results of unplanned demands on equipment should also be accounted for.	DA-C10	Deferred: The maintenance rule data is used directly, but a confirmation that the data are collected exactly consistent with the requirements in the Standard has not been performed. Future updates of the CPS PRA will consider enhancement to the documentation and investigation of the plant failure data implied by this SR. This is judged to have a minimal impact on the unavailabilities and failure probabilities used in the model.	Non-significant impact: Any adjustment to failure data counts resulting from a rigorous review of testing procedures is judged to have a non-significant impact on CDF and LERF values. Not significant. The model is reasonably consistent with data from the plant MR database, which is adequate for RI-ISI application.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
6	As needed in maintenance unavailability determination, perform interviews of maintenance staff for equipment with incomplete or limited maintenance information and document appropriately.	DA-C12	Deferred: Future updates of the CPS PRA will consider performance of interviews of plant personnel to supplement maintenance unavailability estimates for equipment with limited maintenance information.	Non-significant impact: Any refinements to maintenance unavailabilities are judged to result in a negligible impact on CDF (i.e., the dominant maintenance terms, by far, with respect to CDF are trains with good maintenance information - ECCS trains, RCIC, Emergency Diesel Generators (EDGs), SX). The model is reasonably consistent with data from the plant Maintenance Rule (MR) database, which is adequate for RI-ISI application.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
7	The CPS internal flooding analysis and documentation should be updated to meet ASME Standard expectations.	SC-A6, SY-A4, IF Technical Element	An internal flooding update to the CPS CL06c model has recently been completed and will be available in the short-term for use in future applications of the PRA.	<u>Non-significant impact</u>: Internal flooding analyses do not impact RI-ISI calculations. For the RI-ISI analysis the Internal Flooding initiators are not used to represent the consequences from flooding events. Rather the impact of flooding from the RI-ISI consequence analysis is evaluated by tagging appropriate basic events from the non-flooding portions of the Internal Events PRA model. Therefore the fact the RI-ISI analysis does not use the results of the updated Internal Flooding analysis is not critical to the results of the RI-ISI analysis.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
8	Document significant basic events that contribute to the significant initiating events whose frequencies are quantified using fault tree methods.	QU-D5a	Deferred: This documentation aspect has not been incorporated into the CPS PRA notebooks. Initiating event fault trees are not linked into the accident sequence models. Documentation of the importance of failures in initiating event fault trees in the base PRA notebooks is a documentation enhancement.	No Impact: Documentation item. Although the overall importance of some basic events may not be directly obtained in the quantification results, it is possible to estimate these importances. However, initiators associated with this gap are not directly used in the RI-ISI analysis. Documenting the relative importance of basic events to CDF and LERF for these fault tree based initiators has no bearing on the conditional core damage (and large early release) probability calculations used in the RI-ISI analysis.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
9	The following enhancements to the documentation of the CPS PRA should be considered to comply with the documentation requirements in the Standard: • Provide a list of human actions and equipment failures (significant basic events) that cause accidents to be non-dominant. • Bases for the elimination of mutually exclusive events from the model need to be added. • Include cutsets segregated by accident sequence in the documentation.	QU-F2	Deferred: These recommendations are documentation enhancements for the base PRA and are maintained for consideration for future PRA updates.	No Impact: Documentation item.

Item	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
10	Several SRs associated with treatment of model uncertainty and related model assumptions have been recently redefined. NRC has issued [6] a clarification to its endorsement of the PRA Standard. NRC and EPRI are currently preparing guidance on an acceptable process for meeting these requirements.	QU-E1 QU-E2 QU-E4 QU-F4 IE-D3 AS-C3 SC-C3 SY-C3 HR-I3 DA-E3 IF-F3 LE-G4	The CPS 2006 PRA includes CDF and LERF parametric uncertainty analysis; consideration has been given to modeling uncertainty, however the approach used pre-dates NUREG-1855. It involves documenting how assumptions for the technical elements of a PRA can impact the risk results, and then from that performing selected quantitative sensitivity studies. These recently redefined SRs will be addressed during a future PRA model update using a process consistent with NUREG-1855.	To be determined once the new NRC/EPRI guidance is implemented. However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results, and, further, the current approach provides appropriate insights into important modeling assumptions that may be pertinent to applications.

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1
Component Numbers:	Nozzles N1, N2, N3, N5, N6, N7, N8, N9, and N16 (See Enclosure 1 for specific nozzle identification numbers)
Examination Category:	B-D
Item Number:	B3.90 and B3.100
Description:	Alternative to Table IWB-2500-1 (Inspection Program B)

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The code of record for the third ten-year Inservice Inspection Program interval at Clinton Power Station (CPS) is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition. Additionally, for ultrasonic examinations, ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 2001 Edition, is implemented as required (and modified) by 10CFR50.55a(b)(2)(xv) and 10CFR50.55a(b)(2)(xxiv).

3.0 APPLICABLE CODE REQUIREMENT:

Class 1 nozzle-to-vessel weld and nozzle inner radii examination requirements are given in Subsection IWB, Table IWB-2500-1, "Examination Category B-D Full Penetration Welds of Nozzles in Vessels – Inspection Program B," Item Numbers B3.90 and B3.100 respectively. The method of examination is volumetric. All nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles must be examined each interval. All of the nozzles identified in Enclosure 1 are full penetration welds.

4.0 REASON FOR REQUEST:

The identified ISI Class 1 nozzles are scheduled for examination for the upcoming inspection interval at CPS. The proposed alternative provides an acceptable level of quality and safety, and the reduction in scope could provide a dose savings of as much as 25 rem for the entire interval.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested from performing the required examinations on 100% of the identified nozzles. Alternatively, in accordance with Code Case N-702 (Reference 2), CPS proposes to examine a minimum of 25% of the nozzle inner radii and nozzle-to-vessel welds, including at least one nozzle from each system and nominal pipe size. For each of the identified nozzles, both the inner radius and the nozzle-to-shell weld would be examined. As a minimum, the following nozzles would be selected for examination: one of the two 20" recirculation outlet nozzles (i.e., N1); three of the ten 10" recirculation inlet nozzles (i.e., N2); one of the four 24" main steam nozzles (i.e., N3); one of the two 12" core spray nozzles (i.e., N5); one of the three 10" low pressure coolant injection nozzles (i.e., N6); one of the two 6" head spray nozzles (i.e., N7 and N8); one of the two 4" jet pump instrumentation nozzles (i.e., N9); and the vibration instrumentation nozzle (i.e., N16).

Code Case N-702 proposes that visual examination (i.e., VT-1) may be used in lieu of volumetric examination for the nozzle inner radii (i.e., Item B3.100). Note, however, that CPS is not currently using ASME Code Case N-648-1 on enhanced magnification visual examination and has no plans of using this Code Case in the future. CPS will continue to perform volumetric examinations of all required nozzle inner radii.

Basis for Use:

The Electric Power Research Institute (EPRI) Technical Report 1003557, "BWRVIP-108, BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," (Reference 1) provides the basis for Code Case N-702. The EPRI report found that failure probabilities due to a low temperature overpressure event at the nozzle blend radius region and nozzle-to-vessel shell weld are very low (i.e., $< 1 \times 10^{-6}$ for 40 years) with or without any inservice inspection.

On December 19, 2007, the NRC issued a Safety Evaluation (SE) approving the use of BWRVIP-108 as a basis for using Code Case N-702 (Reference 3). In Reference 3, Section 5.0, "Plant Specific Applicability," it states that licensees who plan to request relief from the ASME Section XI requirements for RPV nozzle-to-vessel shell welds and nozzle inner radius sections may reference the BWRVIP-108 report as the technical basis for the use of Code Case N-702 as an alternative. However, each licensee should demonstrate the plant-specific applicability of the BWRVIP-108 report to their units in the relief request by showing that the general and nozzle-specific criteria addressed below are satisfied:

(1) The maximum Reactor Pressure Vessel (RPV) heatup/cooldown rate is limited to less than 115 °F per hour.

(2) For the Recirculation Inlet Nozzles, the following criteria must be met:

   a. $(pr/t)/C_{RPV} < 1.15$

   b. $[p(ro^2+ri^2)/(ro^2-ri^2)]/C_{NOZZLE}<1.15$

(3) For the Recirculation Outlet Nozzles, the following criteria must be met:

   a. $(pr/t)/C_{RPV}<1.15$

   b. $[p(ro^2+ri^2)/(ro^2-ri^2)]/C_{NOZZLE}<1.15$

Demonstration of how CPS meets the NRC plant-specific applicability is provided in Enclosure 2. Based upon all RPV nozzle-to-vessel shell welds and nozzle inner radii sections meeting the NRC plant-specific criteria, Code Case N-702 is applicable to CPS.

Therefore, use of Code Case N-702 provides an acceptable level of quality and safety pursuant to 10CFR50.55a(a)(3)(i) for all RPV nozzle-to-vessel shell welds and nozzle inner radii sections.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the entire third 10-year inservice inspection interval for CPS, Unit 1.

7.0 PRECEDENTS:

Similar relief requests have been approved for:

a. A similar request was approved for use at Duane Arnold Energy Center on August 29, 2008 (i.e., Reference 4).

b. An identical request was approved for use at CPS during the station's second inservice inspection interval on August 24, 2009 (i.e., Reference 5).

8.0 REFERENCES:

1. EPRI Technical Report 1003557, "BWRVIP-108: BWR Vessel and Internals Project Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," dated October 2002
2. ASME Boiler and Pressure Vessel Code, Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1," dated February 20, 2004
3. Letter from Matthew A. Mitchell (NRR), to Rick Libra, BWRVIP Chairman, "Safety Evaluation of Proprietary EPRI Report, 'BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108),'" dated December 19, 2007
4. Letter from Lois James (NRR) to Richard L. Anderson (Duane Arnold Energy Center), "Duane Arnold Energy Center - Safety Evaluation for Request for Alternative to Reactor Pressure Vessel Nozzle to Vessel Weld and Inner Radius Examinations (TAC NO. MD8193)," dated August 29, 2008
5. Letter from S. J. Campbell (NRR) to C. G. Pardee (EGC), "Clinton Power Station, Unit No.1 -Proposed Alternative to 10 CFR 50.55a Examination Requirements for Reactor Pressure Vessel Weld Inspections (TAC No. ME0218)," dated August 24, 2009

IDENTIFICATION NUMBER	WELD DESCRIPTION	Code Category	ITEM NUMBER
N1A	20" Recirculation Outlet Nozzle N1A to Vessel Weld	B-D	B3.90
N1A-IRS	20" Recirculation Outlet Nozzle N1A Inner Radius	B-D	B3.100
N1B	20" Recirculation Outlet Nozzle N1B to Vessel Weld	B-D	B3.90
N1B-IRS	20" Recirculation Outlet Nozzle N1B Inner Radius	B-D	B3.100
N2A	10" Recirculation Inlet Nozzle N2A to Vessel Weld	B-D	B3.90
N2A-IRS	10" Recirculation Inlet Nozzle N2A Inner Radius	B-D	B3.100
N2B	10" Recirculation Inlet Nozzle N2B to Vessel Weld	B-D	B3.90
N2B-IRS	10" Recirculation Inlet Nozzle N2B Inner Radius	B-D	B3.100
N2C	10" Recirculation Inlet Nozzle N2C to Vessel Weld	B-D	B3.90
N2C-IRS	10" Recirculation Inlet Nozzle N2C Inner Radius	B-D	B3.100
N2D	10" Recirculation Inlet Nozzle N2D to Vessel Weld	B-D	B3.90
N2D-IRS	10" Recirculation Inlet Nozzle N2D Inner Radius	B-D	B3.100
N2E	10" Recirculation Inlet Nozzle N2E to Vessel Weld	B-D	B3.90
N2E-IRS	10" Recirculation Inlet Nozzle N2E Inner Radius	B-D	B3.100
N2F	10" Recirculation Inlet Nozzle N2F to Vessel Weld	B-D	B3.90
N2F-IRS	10" Recirculation Inlet Nozzle N2F Inner Radius	B-D	B3.100
N2G	10" Recirculation Inlet Nozzle N2G to Vessel Weld	B-D	B3.90
N2G-IRS	10" Recirculation Inlet Nozzle N2G Inner Radius	B-D	B3.100
N2H	10" Recirculation Inlet Nozzle N2H to Vessel Weld	B-D	B3.90
N2H-IRS	10" Recirculation Inlet Nozzle N2H Inner Radius	B-D	B3.100
N2J	10" Recirculation Inlet Nozzle N2J to Vessel Weld	B-D	B3.90
N2J-IRS	10" Recirculation Inlet Nozzle N2J Inner Radius	B-D	B3.100
N2K	10" Recirculation Inlet Nozzle N2K to Vessel Weld	B-D	B3.90
N2K-IRS	10" Recirculation Inlet Nozzle N2K Inner Radius	B-D	B3.100
N3A	24" Main Steam Nozzle N3A to Vessel Weld	B-D	B3.90
N3A-IRS	24" Main Steam Nozzle N3A Inner Radius	B-D	B3.100
N3B	24" Main Steam Nozzle N3B to Vessel Weld	B-D	B3.90
N3B-IRS	24" Main Steam Nozzle N3B Inner Radius	B-D	B3.100
N3C	24" Main Steam Nozzle N3C to Vessel Weld	B-D	B3.90
N3C-IRS	24" Main Steam Nozzle N3C Inner Radius	B-D	B3.100
N3D	24" Main Steam Nozzle N3D to Vessel Weld	B-D	B3.90
N3D-IRS	24" Main Steam Nozzle N3D Inner Radius	B-D	B3.100
N5A	12" Core Spray Nozzle N5A to Vessel Weld	B-D	B3.90
N5A-IRS	12" Core Spray Nozzle N5A Inner Radius	B-D	B3.100
N5B	12" Core Spray Nozzle N5B to Vessel Weld	B-D	B3.90
N5B-IRS	12" Core Spray Nozzle N5B Inner Radius	B-D	B3.100
N6A	10" Low Pressure Core Injection Nozzle N6A to Vessel Weld	B-D	B3.90
N6A-IRS	10" Low Pressure Core Injection Nozzle N6A Inner Radius	B-D	B3.100
N6B	10" Low Pressure Core Injection Nozzle N6B to Vessel Weld	B-D	B3.90

IDENTIFICATION NUMBER	WELD DESCRIPTION	Code Category	ITEM NUMBER
N6B-IRS	10" Low Pressure Core Injection Nozzle N6B Inner Radius	B-D	B3.100
N6C	10" Low Pressure Core Injection Nozzle N6C to Vessel Weld	B-D	B3.90
N6C-IRS	10" Low Pressure Core Injection Nozzle N6C Inner Radius	B-D	B3.100
N7	6" Top Head Spray Nozzle N7 to Vessel Weld	B-D	B3.90
N7-IRS	6" Top Head Spray Nozzle N7 Inner Radius	B-D	B3.100
N8	6" Top Head Spare Nozzle N8 to Vessel Weld	B-D	B3.90
N8-IRS	6" Top Head Spare Nozzle N8 Inner Radius	B-D	B3.100
N9A	4" Jet Pump Instrumentation Nozzle N9A to Vessel Weld	B-D	B3.90
N9A-IRS	4" Jet Pump Instrumentation Nozzle N9A Inner Radius	B-D	B3.100
N9B	4" Jet Pump Instrumentation Nozzle N9B to Vessel Weld	B-D	B3.90
N9B-IRS	4" Jet Pump Instrumentation Nozzle N9B Inner Radius	B-D	B3.100
N16	Vibration Instrumentation Nozzle to Vessel Weld	B-D	B3.90
N16-IRS	Vibration Instrumentation Nozzle Inner Radius	B-D	B3.100

1. The maximum Reactor Pressure Vessel (RPV) heatup/cooldown rate is limited to less than 115 °F/hour.

   This criterion is met by adherence to Clinton Power Station Technical Specification 3.4.11, "Reactor Coolant System Pressure/Temperature Limits," Surveillance Requirement 3.4.11.1 which requires verification that the Reactor Coolant System heatup and cooldown rates are limited to less than or equal to 100 °F in any one hour period and, less than or equal to 20 °F in any one hour period during RPV pressure testing.

2. For the Reactor Recirculation Inlet (N2) Nozzles, $(pr/t)/C_{RPV}$ must be less than 1.15, where:

   p = normal RPV pressure = 1025 psig
   r = RPV inner radius = 110.19 inches
   t = RPV wall thickness = 6.1 inches
   C_{RPV} = 19332

   Result: $(pr/t)/C_{RPV} = 0.96$

3. For the Reactor Recirculation Outlet (N1) Nozzles, $(pr/t)/C_{RPV}$ must be less than 1.15, where:

   p = normal RPV pressure = 1025 psig
   r = RPV inner radius = 110.19 inches
   t = RPV wall thickness = 6.1 inches
   C_{RPV} = 16171

   Result: $(pr/t)/C_{RPV} = 1.14$

4. For the Reactor Recirculation Inlet (N2) Nozzles $[p(ro^2+ri^2)/(ro^2-ri^2)]/C_{NOZZLE}$ must be less than 1.15, where:

   p = normal RPV pressure = 1025 psig
   ro = nozzle outlet radius = 11.69 inches
   ri = nozzle inner radius = 5.81 inches
   C_{NOZZLE} = 1637

   Result: $[p(ro^2+ri^2)/(ro^2-ri^2)]/C_{NOZZLE} = 1.04$

For the Reactor Recirculation Outlet (N1) Nozzles $[p(ro^2+ri^2)/(ro^2-ri^2)]/C_{NOZZLE}$ must be less than 1.15, where:

p = normal RPV pressure = 1025 psig
ro = nozzle outlet radius = 16.3125 inches
ri = nozzle inner radius = 9.0 inches
C_{NOZZLE} = 1977

Result: $[p(ro^2+ri^2)/(ro^2-ri^2)]/C_{NOZZLE} = 0.97$

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1, 2, and 3
Reference:	Table IWB-2500-1, IWB-5200 Table IWC-2500-1, IWC-5200 Table IWD-2500-1, IWD-5200
Examination Category:	B-P, C-H, and D-B
Item Number:	B15.10, C7.10, and D2.10
Description:	Pressure Testing the RPV Head Flange Seal Leak Detection System
Component Number:	RPV Head Flange Seal Leak Detection System
Drawing Number:	M05-1071, Sht. 1

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The code of record for the third ten-year Inservice Inspection Program interval at Clinton Power Station (CPS) is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition.

3.0 APPLICABLE CODE REQUIREMENT:

Table IWB-2500-1, Examination Category B-P, Item Number B15.10, requires all ISI Class 1 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with Paragraph IWB-5220. This pressure test is to be conducted prior to plant startup following each reactor refueling outage.

Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires all ISI Class 2 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with Paragraph IWC-5220. This pressure test is to be conducted once each inspection period.

Table IWD-2500-1, Examination Category D-B, Item Number D2.10, requires all ISI Class 3 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with Paragraph IWD-5220. This pressure test is to be conducted once each inspection period.

4.0 IMPRACTICALITY OF COMPLIANCE:

Pursuant to 10CFR50.55a(g)(5)(iii), relief is requested on the basis that pressure testing the RPV Flange Leak Detection Line is deemed impractical.

The Reactor Pressure Vessel (RPV) 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 (See Figure I3R-03.1). This line is required during plant operation and will indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in a High Pressure Alarm in the Main Control Room.

The configuration of this system precludes manual testing while the vessel head is removed. As Figure I3R-03.1 portrays, the configuration of the vessel tap, combined with the small size of the tap and the high test pressure requirement (approximately 1025 psig), prevents the tap from being temporarily plugged. Also, 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. Due to the groove that the O-ring sits in and the pin/wire clip assembly (See Figure I3R-03.2), pressurization in the opposite direction into the recessed cavity and retainer clips would likely damage the O-ring and thus result in further damage to the O-ring.

5.0 BURDEN CAUSED BY COMPLIANCE:

Pressure testing of this line during the 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 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 reset and re-tension the reactor vessel head. This is considered to impose an undue hardship and burden on CPS.

Based on the above, CPS requests relief from the ASME Section XI requirements for system leakage testing of the Reactor Vessel Head Flange Seal Leak Detection System.

6.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

A VT-2 visual examination on 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 during each refueling outage in accordance with the frequencies specified by Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1.

7.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the Third Ten-Year Inspection Interval for CPS, Unit 1.

8.0 PRECEDENTS:

Similar relief requests have been approved for:

Peach Bottom Atomic Power Station Fourth Interval Relief Request I4R-25 was granted per SER dated February 26, 2009

Limerick Generating Station Third Interval Relief Request I3R-08 was granted per SER dated March 11, 2008

LaSalle County Station Third Interval Relief Request I3R-08 was granted per SER dated January 30, 2008

Susquehanna Steam Electric Station Third Inspection Interval Relief Request 3RR-07 was granted per SER dated September 24, 2004.

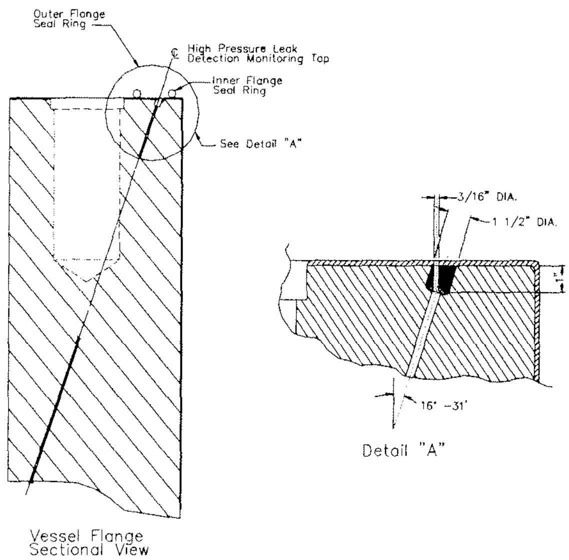

Figure I3R-03.1: Flange Seal Leak Detection Line Detail

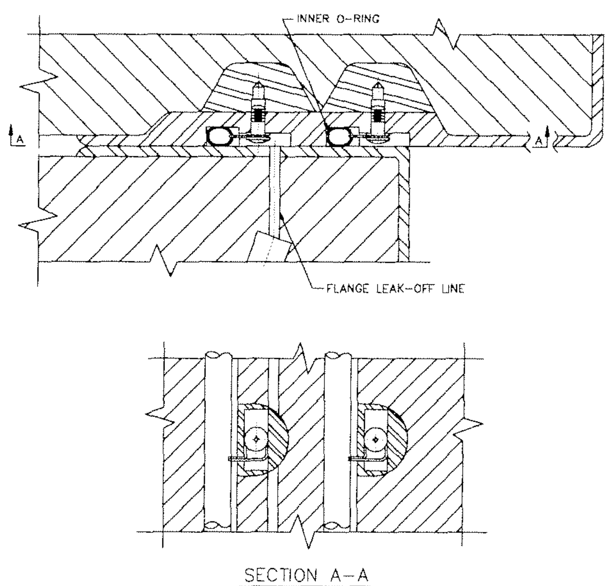

Figure I3R-03.2: O-Ring Configuration

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	2, 3
Component Numbers:	Multiple lines (See Note 1 below)
Examination Category:	C-H, D-B
Item Number:	C7.10, D2.10
Description:	Alternative to Performance of System Pressure Tests and VT-2 Visual Examination Requirements for all ISI Class 2 Instrument Air (IA) Piping and the ISI Class 3 IA Piping Supplying, all Safety Relief Valves (SRVs), and both Feedwater Containment Outboard Isolation Check Valves

Note 1: A more detailed description of the pressure testing boundary is identified below.

ISI Class 2 IA system piping and components between containment isolation valves 1IA012A/B and 1IA013A/B and check valves 1IA042A/B. This includes the following lines, valves, and components shown on Clinton Power Station (CPS) Piping and Instrumentation Diagram (P&ID) M05-1040 Sht. 7 not listed above.

- Lines 1IA71BA/BB-1, 1IA14GA/GB-1, 1IA95A/B-1, 1IA93AA/BA-3/4, and 1IA96AA/BA-3/4
- Valves 1IA131A/B, 1IA129A/B, and the blind flanges on lines IIA95A/B-1

ISI Class 3 IA system piping and components requiring inspection. This includes the following IA lines and valves supplying all 16 SRVs and both Feedwater containment outboard isolation check valves.

- P&ID M05-1040 Sht. 7 lines - 1IA79CA/CB-1, 1IA92AA/BA-3/4, 1IA102BA-1/2, 1IA103BA-1/2, 1IA71AA/AB-1, 1IA87A/B-1/2, 1IA125A/B-1/2, 1IA122A/B-1, 1IA88A/B-1/2, 1IA71CA/CB-1, 1IA71DA/EA/FA/GA-1/2, and 1IA71DB/EB/FB/GB/FC-1/2.
- P&ID M05-1040 Sheet 7 valves - 1IA075A/B, 1IA076A/B, 1IA130A/B, 1IA1170A/B, 0IA18MA/B, 1IA044A/B, 1IA1171A/B, 1IA1172A/B, 1IA096C/D, and 1IA097A/B. NOTE - Strainers 1IA26FA/FB are not Code components.
- P&ID M10-9002 Sheet 1 lines - 1IA71DA/DB/EA/EB/FA/FB/FC-1/2, 1IA85A/B/C/D/E/F/G-1/2, 1MS71CE/DE-1/2, 1MS72AE/BE-1/2, 1MS73BE/CE-1/2, 1MS74CE-1/2, 1MS71CG/DG-3/4, 1MS72AG/BG-3/4, 1MS73BG/CG-3/4, 1MS74CG-3/4, 1MS71CH/DH-1/2, 1MS72AH/BH-1/2, 1MS73BH/CH-1/2, 1MS74CH-1/2, 1MS71CF/DF-3/4, 1MS72AF/BF-3/4, 1MS73BF/CF-3/4, 1MS74CF-3/4, 1MS71CC/DC-2, 1MS72AC/BC-2, 1MS73BC/CC-2, 1MS74CC-2, 1MS71CJ/CK/DJ/DK-1 1/4, 1MS72AJ/AK/BJ/BK-1 1/4, 1MS73BJ/BK/CJ/CK-1 1/4, and 1MS74CJ/CK-1 1/4

- P&ID M10-9002 Sheet 1 valves - 1IA094A/B/C/D/E/F/G, 1B21-F039B/C/D/E/H/K/S, 1B21-F331C/D, 1B21-F332A/B, 1B21-F333B/C, 1B21-F334, valves 'G' on M10-9002 Sheet 1 and 1B21-F082B/C/D/E/H/K/S
- P&ID M10-9002 Sheet 1 accumulators - 1B21-A003B/C/D/E/H/K/S
- P&ID M10-9002 Sheet 2 lines - 1IA71GA/GB-1/2, 1IA86C/E-1/2, 1MS75AE/BE-1/2, 1MS76CE/DE-1/2, 1MS77AE/CE/DE-1/2, 1MS78BE/CE-1/2, 1MS75AC/BC-2, 1MS76CC/DC-2, 1MS77AC/CC/DC-2, 1MS78BC/CC-2, 1MS75AG/AH/BG/BH-1 1/4, 1MS76CG/CH/DG/DH-1 1/4, 1MS77AG/AH/CG/CH/DG/DH-1 1/4, and 1MS78BG/BH/CG/CH-1 1/4
- P&ID M10-9002 Sheet 2 valves - 1IA095C/E, 1B21-F036A/F/G/J/L/M/N/P/R and 1B21-F081A/F/G/J/L/M/N/P/R
- P&ID M10-9002 Sht. 2 accumulators - 1B21-A004A/F/G/J/L/M/N/P/R
- P&ID M10-9004 Sht. 8 lines - 1FW26BA/BB-1/2, 1FW27BA/BB-1/2, 1FW26CA/CB-2, and 1FW28AA/AB-3/4
- P&ID M10-9004 Sht. 8 valves - 1B21-F433A/B and 1B21-F492A/B
- P&ID M10-9004 Sht. 8 accumulators - 1B21-A300A/B

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The code of record for the third ten-year Inservice Inspection Program interval at CPS Inservice Inspection program is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition.

3.0 APPLICABLE CODE REQUIREMENT:

Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires all ISI Class 2 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with Paragraph IWC-5220. This pressure test is to be conducted once each inspection period.

Table IWD-2500-1, Examination Category D-B, Item Number D2.10, requires all ISI Class 3 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with Paragraph IWD-5220. This pressure test is to be conducted once each inspection period.

4.0 REASON FOR REQUEST:

Pursuant to 10CFR50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Performance of a VT-2 visual examination would require applying a leak detection solution to a large amount of piping and components, many of which are in elevated dose rate areas with limited access. VT-2 visual inspections would result in additional radiation exposure (i.e., estimated to be 2 rem) and industrial safety challenges without any added benefit in the level of quality and safety. These inspections would not be consistent with As Low As Reasonably Achievable (ALARA) practices.

Relief is requested from the performance of system pressure tests and VT-2 visual examination requirements specified in Tables IWC-2500-1 and IWD-2500-1 for all ISI Class 2 IA piping and the ISI Class 3 IA piping supplying all SRV's and both Feedwater containment outboard isolation check valves.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

As an alternative to the examination requirements of Tables IWC-2500-1 and IWD-2500-1, CPS will perform pressure decay testing on the ISI Class 2 and 3 IA piping supplying all 16 SRV's and both Feedwater containment outboard isolation check valves as required in surveillance procedure CPS 9061.11, "Instrument Air Check Valve Operability and Pipe Pressure Test."

Surveillance procedure CPS 9061.11, verifies the operability of SRV actuation capability and check valves in the IA supply lines to all 16 SRV's and both Feedwater containment outboard isolation check valves. This surveillance test is performed for each individual SRV and both Feedwater containment outboard isolation check valves as a requirement of the CPS Inservice Testing (IST) Program. One specific test this surveillance performs, is a pressure decay test of the SRV and Feedwater containment outboard isolation check valve accumulators, as well as associated piping and valves. The pressure decay test is performed by isolating and pressurizing these accumulators and associated piping to the nominal operating pressure. The decay in pressure is then monitored through calibrated pressure measuring instrumentation. If any pressure decay acceptance criterion (see Attachment 1) is exceeded, the surveillance identifies appropriate troubleshooting steps to perform, including soap-bubble application to locate leakage.

The pressure decay test performed as part of CPS 9061.11 identifies any degradation of the ISI Class 2 and 3 Automatic Depressurization System (ADS) supply piping and the SRV and Feedwater containment outboard isolation check valve accumulators and associated piping. The volume tested by this surveillance encompasses all piping and components requiring testing under ASME Section XI for these portions of the IA system. This surveillance is performed on a greater frequency than that required in Tables IWC-2500-1 or IWD-2500-1 and the test pressure is consistent with the pressure requirements of both tables. Thus, the testing performed during this surveillance will provide the same level of quality and safety as the pressure testing and VT-2 visual examination requirements of Tables IWC-2500-1 and IWD-2500-1.

The VT-2 visual examination described in Tables IWC-2500-1 and IWD-2500-1 and performed once per inspection period would not provide an increase in safety, system reliability, or structural integrity. In addition, performance of a VT-2 visual examination would require applying a leak detection solution to a large amount of piping and components, many of which are in elevated dose rate areas with limited access. VT-2 visual inspections would result in additional radiation exposure (estimated 2 Rem) and industrial safety challenges without any added benefit in the level of quality and safety. These inspections would not be consistent with radiation exposure practices of "As Low As Reasonably Achievable (ALARA)."

In summary, relief is requested from the performance of system pressure tests and VT-2 visual examination requirements specified in Tables IWC-2500-1 and IWD-2500-1 for the ISI Class 2 and 3 IA system piping and components identified in this request on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the Third Ten-Year Inspection Interval for CPS, Unit 1.

7.0 PRECEDENTS:

Similar relief requests have been approved for:

Clinton Power Station Second Inspection Interval Relief Request 4212, Rev. 1 was authorized per SER dated December 13, 2007. The Third Inspection Interval Relief Request utilizes an identical approach as was previously approved.

LaSalle County Station Second Inspection Interval Relief Requests PR-08 and PR-10 were authorized per SER dated June 28, 2002.

ENCLOSURE 1 TO ATTACHMENT 4
10 CFR 50.55a Request Number I3R-04
Acceptance Criteria from Clinton Power Station Procedure CPS-9061.11
(For Information Only)

Component	Leakage Criterion	Pressure Drop Test Duration	Comments
Accumulator Headers for all SRV's except 1B21-F051C and D	≤ 1.5 psig	≥ 108 minutes	
Accumulator Headers for 1B21-F051C and D	≤ 1.5 psig	≥ 31 minutes	Smaller volume than other SRV's.
Accumulator Headers for Feedwater Check Valve	≤ 1.5 psig	≥ 26 minutes	Smaller volume than SRV's.
ADS Supply Header to Accumulator Headers	≤ 22 psig	≥ 60 minutes	This inspection tests over 200 feet of piping and components.

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	2
Reference:	IWC-2500, Table IWC-2500-1
Examination Category:	C-G
Item Number:	C6.10
Description:	Examination of the ISI Class 2 High Pressure Core Spray, Low Pressure Core Spray, and Residual Heat Removal Pump Casing Welds
Component Number:	1E12-C001A, 1A RHR Pump Casing Welds 1E12-C001B, 1B RHR Pump Casing Welds 1E12-C001C, 1C RHR Pump Casing Welds 1E22-C001, HPCS Pump Casing Welds 1E21-C001, LPCS Pump Casing Welds
Drawing Number:	B-69, B-71, and B-73

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The code of record for the third ten-year Inservice Inspection Program interval at Clinton Power Station (CPS) Inservice Inspection program is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2004 Edition.

3.0 APPLICABLE CODE REQUIREMENT:

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.

Per Table IWC-2500-1, the multiple-component concept applies, and examinations are limited to either 100% of the welds of one of three Residual Heat Removal Pumps, one High Pressure Core Spray Pump, and one Low Pressure Core Spray Pump, or distributed among any of the pumps of that same group 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.

4.0 IMPRACTICALITY OF COMPLIANCE:

Pursuant to 10CFR50.55a(g)(5)(iii), relief is requested on the basis that conformance with these code requirements is impractical as conformance would require extensive structural modifications to these pumps.

CPS's three Residual Heat Removal Pumps (i.e., 1E12-C002A, 1E12-C002B, and 1E12-C002C), one High Pressure Core Spray Pump (i.e., 1E22-C001), and one Low Pressure Core Spray Pump (i.e., 1E21-C001) were originally designed where the pump casing welds were encased in concrete, thus making the welds inaccessible for inservice inspection.

Therefore, it is impractical for CPS to perform the surface examination of these welds without destruction of the concrete resulting in unnecessary engineering and installation costs and radiation exposure without a compensating increase in safety. Additionally, due to the design of the subject pumps, access to the affected welds can only be achieved through disassembly of the pump, removal of the pump internals, and the required surface examinations performed 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.

5.0 BURDEN CAUSED BY COMPLIANCE:

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the subject pumps. Based on this, the Code requirements are deemed impractical in accordance with 10CFR50.55a(g)(5)(iii).

6.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

In the event the subject welds become accessible upon disassembly of any one (1) 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 detailed in the Impracticality Of Compliance section of this relief request. The examination method will be determined by CPS based on radiation environment data at the time access is enabled. Additionally, a VT-2 visual examination during system pressure testing per Examination Category C-H will be performed once each period by examining the surrounding area (exposed areas around these components where the pump casing join/merge with the concrete) for evidence of leakage in accordance with Paragraph IWA-5241(b). These examinations will provide reasonable assurance of continued structural integrity of the piping systems.

7.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the Third Ten-Year Inspection Interval for CPS, Unit 1.

8.0 PRECEDENTS:

Similar relief requests have been approved for:

- LaSalle County Station Third Inspection Interval Relief Request I3R-03 was granted per SER dated January 30, 2008.
- Limerick Generating Station Third Inspection Interval Relief Request I3R-07 was granted per SER dated March 11, 2008.
- Susquehanna Steam Electric Station Third Inspection Interval Relief Request 3RR-02 was granted per SER dated February 1, 2005.