UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 22, 2010

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 – RELIEF REQUESTS I3R-01, I3R-02, I3R-03, I3R-04, AND I3R-05 ASSOCIATED WITH THE THIRD INSERVICE INSPECTION INTERVAL (TAC NOs. ME2987, ME2988, ME2989, ME2990, AND ME2991)

Dear Mr. Pacilio:

By letter dated December 30, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093650139), as supplemented by letter dated July 14, 2010 (ADAMS Accession No. ML101960011), Exelon Generation Company, LLC (EGC, the licensee), submitted Relief Requests (RRs) I3R-01, I3R-02, I3R-03, I3R-04, and I3R-05 to the Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Section XI requirements for the third 10-year inservice inspection (ISI) interval at Clinton Power Station (CPS), Unit No. 1.

Specifically, for RRs I3R-01 and I3R-02, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee requested to use the proposed alternatives on the basis that the alternatives provide an acceptable level of quality and safety. For RR I3R-04, pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee requested to use the proposed alternative on the basis that complying with current ASME OM Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. For RRs I3R-03 and RR I3R-05, pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief and use of alternative requirements on the basis that the code requirement is impractical. Note that the NRC staff evaluated RR I3R-03 pursuant to 10 CFR 50.55a(a)(3)(ii).

The NRC staff has determined that proposed alternatives RRs I3R-01 and I3R-02 provide an acceptable level of quality and safety. The NRC staff has determined that, with regards to RRs I3R-03 and I3R-04, the alternatives provide a reasonable assurance of structural integrity and compliance to the requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff has determined that, with regards to RR I3R-05, the alternative provides reasonable assurance of structural integrity and that granting this relief 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.

The NRC staff has reviewed the subject requests and concludes, as set forth in the enclosed safety evaluation that the licensee had adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(3)(i) for alternative RRs I3R-01 and I3R-02, as set forth in 10 CFR 50.55a(3)(ii) for alternative RRs I3R-03 and I3R-04, and as set forth in

10 CFR 50.55a(g)(6)(i) for alternative RR I3R-05. Therefore, the NRC authorizes alternative RR I3R-01, I3R-02, I3R-03, I3R-04, and I3R-05 at CPS for the third 10-year ISI interval, which begins on July 1, 2010, and ends on June 30, 2020.

The NRC staff's Safety Evaluation regarding RRs I3R-01, I3R-02, I3R-03, I3R-04, and I3R-05 is enclosed. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector. This completes the NRC staff's efforts on TAC Nos. ME2987, ME2988, ME2989, ME2990, and ME2991.

If you have any questions, please contact the Clinton Project Manager, Nicholas DiFrancesco, at 301-415-1115.

Sincerely,

/RA by E. Brown for/

Robert D. Carlson, Branch Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUESTS FOR RELIEF I3R-01, I3R-02, I3R-03, I3R-04, AND I3R-05

ASSOCIATED WITH THE THIRD INSERVICE INSPECTION INTERVAL

EXELON GENERATION COMPANY, LLC

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter dated December 30, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093650139), as supplemented by letter dated July 14, 2010 (ADAMS Accession No. ML101960011), Exelon Generation Company, LLC (EGC, the licensee), submitted Relief Requests (RRs) I3R-01, I3R-02, I3R-03, I3R-04, and I3R-05 to the Nuclear Regulatory Commission (NRC) for the use of alternatives to certain 2004 Edition American Society of Mechanical Engineers Code for Operation and Maintenance of Nuclear Power Plants (ASME Code), Section XI requirements for the third 10-year Inservice Inspection (ISI) interval at Clinton Power Station (CPS), Unit No. 1.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(g), "Inservice inspection requirements," specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(g)(6)(i) states that the Commission may grant such relief and may impose such alternative requirements as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest, given the consideration of the burden upon the licensee. The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (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. The licensee has submitted RRs I3R-01 and I3R-02 pursuant to 10 CFR 50.55a(a)(3)(i); and RR I3R-04 pursuant to 10 CFR 50.55a(a)(3)(ii).

The regulation at 10 CFR 50.55a(g)(5)(iii) states that if the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in 10 CFR 50.4, information to support the determinations. Pursuant to 10 CFR 50.55a(g)(6)(i), the Commission will evaluate

Enclosure

determinations under paragraph (g)(5) that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines are authorized by law and will not endanger life or property or the common defense and security and are 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. The licensee has submitted RRs I3R-03 and I3R-05 pursuant to 10 CFR 50.55a(g)(5)(iii) which the Commission may grant pursuant to 10 CFR 50.55a(g)(6)(i). Note: I3R-03 was reviewed by the NRC staff pursuant to the requirements of 10 CFR 50.55a(a)(3)(ii).

All risk-informed applications are assessed against Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." RG 1.174 states that a probabilistic risk assessment (PRA) used in risk-informed licensing action should be performed in a manner that is consistent with accepted practices. The NRC staff utilized RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," to determine whether the technical adequacy of the PRA used to support a submittal is consistent with accepted practices. The NRC staff also assessed the licensee's proposed RI-ISI program against the guidance in RG 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," and Standard Review Plan (SRP) 3.9.8, "Risk-Informed Inservice Inspection of Piping."

For all reactor pressure vessel (RPV) nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI requires 100 percent inspection during each 10-year ISI interval. However, ASME Code Case N-702 proposes an alternative which reduces the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radius areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. The NRC has approved the BWRVIP-108 report, which contains the technical basis supporting ASME Code Case N-702. The December 19, 2007, safety evaluation (SE) regarding the [Boiling Water Reactor Vessel and Internals Project] BWRVIP-108 report specified plant-specific requirements to be satisfied by applicants who propose to use ASME Code Case N-702.

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, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of 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 of the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The applicable code of record for the third ISI interval for CPS is the 2004 Edition (No Addendum) of the ASME Code, Section XI. The proposed alternative is sought for the third 10-year ISI interval which is scheduled to end on June 30, 2020.

The NRC's findings with respect to authorizing alternatives to the ASME Code are given below.

3.0 TECHNICAL EVALUATION

3.1 Relief Request I3R-01

3.1.1 Introduction

Relief Request I3R-01 requests approval of a risk-informed inservice inspection (RI-ISI) program and examination criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 pressure retaining piping welds as an alternative to the ASME Code, Section XI ISI requirements.

Specifically, pursuant to Title 10 of the 10 CFR 50.55a(a)(3)(i), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

3.1.2 Components for Which Relief is Requested

The scope of the RI-ISI program is limited to the ASME Code, Section XI, Class 1 and 2 piping welds (Categories B-F, B-J, C-F-1, and C-F-2 welds) only. The proposed alternative would be a continuation of the CPS RI-ISI program for the second 10-year ISI interval which was submitted to the NRC staff by letters dated October 15, and November 20, 2001, and February 7, 2002 (ADAMS Accession Nos. ML012950371, ML013620355, and ML020560371, respectively). The NRC staff authorized CPS, to implement a RI-ISI program during the second 10-year ISI interval by letter dated April 8, 2002 (ADAMS Accession No. ML020800820).

3.1.3 ASME Code Requirements

3.1.4 Licensee Proposed Alternative to Code and Basis for Use

The CPS RI-ISI program is an alternative pursuant to 10 CFR 50.55a(a)(3)(i). In the submittal, the licensee requests NRC authorization to extend the RI-ISI program, previously approved for use in the second ISI interval, to the third ISI interval at CPS. The program scope will be implemented as an alternative to the ASME Code, 2004 Edition, Section XI examination program for Class 1 Examination Categories B-F and B-J and Class 2 Examination Categories C-F-1 and C-F-2 piping welds. The proposed alternative is sought for the third 10-year ISI interval which began July 1, 2010, and scheduled to conclude June 30, 2020.

Pursuant to 10 CFR 50.55a(a)(3), relief is requested for the above-stated piping welds. The initial CPS RI-ISI Program was submitted during the second 10-year ISI interval. This initial RI-ISI program was developed in accordance with Electric Power Research Institute (EPRI) TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, Final Report," December 1999, as supplemented by ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B."

The program was approved for use by the NRC via SE as transmitted to the licensee by letter dated April 8, 2002. In its submittal dated December 30, 2009, the licensee states that the RI-ISI program has been updated and continues to meet EPRI TR-112657, Revision B-A, and Regulatory Guide 1.174 risk acceptance guidelines.

The licensee in submittal dated December 30, 2009, states that the third interval program will be a continuation of the current application which included the following two enhancements to the EPRI methodology:

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 Subarticle-2430, "Additional Examinations" contained in ASME Code Case N-578-1. In addition, CPS intends to perform additional examinations required due to the identification of flaws, which are determined to exceed the acceptance standards, during the current refueling outage prior to the units return to service.

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 ASME Code Case N-578-1. 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. The ultrasonic examination volume to be used based on degradation mechanism and component configuration will be the examination figures specified in Section 4 of EPRI TR-112657."

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.

The information provided by the licensee in support of the request has been evaluated and the basis for disposition is documented below.

3.1.5 NRC Staff's Evaluation of Proposed Alternative

The NRC staff has reviewed and evaluated the licensee's proposed RI-ISI program, including those portions related to the applicable methodology and processes, based on guidance and acceptance guidelines provided in RGs 1.174 and 1.178, in SRP 3.9.8, and in the EPRI-TR-112657, Revision B-A. An acceptable RI-ISI program plan is expected to meet the five key principles discussed in RGs 1.74 and 1.178, SRP 3.9.8, and the EPRI-TR, as stated below:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption.

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.

5. The impact of the proposed change should be monitored by using performance measurement strategies.

The first principle is met because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(a)(3)(i); 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 licensee stated that no changes to the evaluation methodology, as currently implemented under the EPRI-TR, are required as part of this interval update, that the methodology of the calculation of the risk impact assessment for the third 10-year ISI interval has not changed, and the calculation remains part of the living program. In letter dated July 14, 2010, the licensee stated that the augmented Service Water and Flow-Accelerated Corrosion programs remain unaffected by the RI-ISI program. The licensee also stated that the High-Energy Line Break Break Exclusion Region augmented inspection program was integrated into the RI-ISI during the second ISI interval in accordance with EPRI TR 1006937, "Extension of the EPRI Risk-Informed Inservice Inspection Methodology to Break Exclusion Region Programs," which is acceptable to the NRC staff. Because the methodology used to develop the RI-ISI program for the third 10-year ISI interval is unchanged from the methodology approved by the NRC staff for development of the RI-ISI program used in the second 10-year ISI interval and the Augmented Inspection Programs remain unchanged, the second and third principles are met.

The fourth principle, that any increase in CDF and risk are small and consistent with the Commission's Safety Goal Policy Statement, requires an estimate of the change in risk. The change in risk estimate is dependent on the location of inspections in the proposed ISI program compared to the location of inspections that would be performed using the requirements of ASME Code, Section XI. The licensee stated in its application that the original calculation methodology was used to re-assess this change in risk.

The fourth principle also requires demonstration of the technical adequacy of the PRA. As discussed in RGs 1.178 and 1.200, an acceptable change in risk evaluation (and risk-ranking evaluation used to identify the most risk significant locations) requires the use of a PRA of appropriate technical quality that models the as-built and as-operated plant. In the present relief request application and in the July 14, 2010, response to an NRC staff request for additional information (RAI), the licensee provided information on the technical adequacy of its PRA. The licensee reported that the Boiling Water Reactor Owners Group (BWROG) performed a PRA Peer Review in 2000; the licensee performed a self assessment in 2003 against ASME PRA Standard ASME RA-Sb-2005 and draft Revision 1 of RG 1.200; the BWROG performed cross-comparison studies, including the CPS PRA model, in 2005 and 2006, as part of the mitigating systems performance indicator process; the licensee revised its self-assessment in 2009 for consistency with Revision 1 of RG 1.200; and an industry peer review of the Clinton PRA was completed in 2009. In its application, the licensee provided an impact assessment of the open items from the BWROG PRA peer review and 2009 self-assessment. In response to an NRC staff RAI, the licensee also provided an impact assessment of new findings from the 2009 peer review. The licensee provided its evaluation of all identified gaps indicating that they are not significant to the RI-ISI application. The NRC staff finds the licensee has assessed the technical adequacy of its PRA using the appropriate version of RG 1.200 and the quality of the PRA is sufficient to support the proposed RI-ISI program.

The NRC staff has previously determined that it is not necessary to develop a new deterministic ASME program for each new 10-year interval but, instead, it is acceptable to compare the new proposed RI-ISI program with the last deterministic ASME program. The licensee states in their application that a new Risk Impact Analysis was performed, and the licensee's July 14, 2010, RAI response indicates that the revised program continues to satisfy the acceptance guidelines of RG 1.174 and EPRI TR-112657 when compared to the last deterministic Section XI inspection program. Thus, the NRC staff finds that the licensee's analysis provides assurance that the fourth key principle is met.

The fifth principle of risk-informed decision-making requires that the impact of the proposed change be monitored by using performance measurement strategies. As described in the submittals, the RI-ISI program is a living program that requires periodic updating and that, as a minimum, will include reviews of risk ranking of piping segments on an ASME period basis, and with major revisions of the site PRA. Thus, the NRC staff concludes 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 interval RI-ISI program plan; therefore, the proposed program for the third 10-year ISI interval is acceptable.

3.1.6 Conclusion

As set forth above, the NRC staff determines that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a, and is in compliance with the ASME Code's requirements. Therefore, the NRC staff authorizes RR I3R-01 at CPS for the third 10-year ISI interval. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by Authorized Nuclear Inservice Inspector.

3.2 <u>Relief Request I3R-02</u>

3.2.1 Introduction

The licensee submitted RR I3R-02 to use an alternative to ASME Code, Section XI inspection requirements regarding examination of certain RPV nozzle-to-vessel welds and nozzle inner radii at CPS.

The proposed alternative is in accordance with ASME Code Case N-702, "Alternative Requirements for BWR Nozzle Inner Radius and Nozzle-to-Shell Welds," without using the visual (VT-1) examination specified in the code case. The technical basis for ASME Code Case N-702 was documented in an EPRI report for the BWRVIP, "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 Inner Radii." The BWRVIP-108 report was approved by the NRC in a SE dated December 19, 2007.

The December 19, 2007, SE for the BWRVIP-108 report specified plant-specific requirements which must be met for applicants proposing to use this alternative. This submittal intended to demonstrate that the relevant CPS RPV nozzle-to-vessel welds and their inner radii meet these plant-specific requirements so that Relief Request I3R-02 can be approved.

3.2.2 Component(s) for which the Alternative is Requested

ASME Code Class:	1
Examination Category:	B-D, Full Penetration Welded Nozzles in Vessels
Item Number:	B3.90 (nozzle-to-vessel welds) and B3.100 (nozzle inner radius sections)
Description:	See Enclosure 1 to Attachment 2 (I3R-02)[1] to the licensee's December 30, 2009, letter, for the complete list of components covered by this alternative. [Note: The RPV feedwater nozzles and control rod drive (CRD) return line nozzles are not included.]

3.2.3 ASME Code Requirement for which the Alternative is Requested

The licensee requested an alternative to the following requirements of ASME Code, Section XI, 2004 Edition:

> ASME Code 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 Welded Nozzles in Vessels - Inspection Program B," Item Numbers B3.90, "Nozzle-to-Vessel Welds," and B3.100, "Nozzle Inside Radius Section," respectively. Volumetric examination is required each interval for all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles. All of the nozzle assemblies identified are made with full penetration welds.
>
> Additionally, for ultrasonic examinations, ASME [Code,] Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 2001 Edition, is implemented as required (and modified) by 10 CFR 50.55a(b)(2)(xv) and 10 CFR 50.55a(b)(2)(xxiv).

3.2.4 Licensee Proposed Alternative and Basis for Use

Licensee's Proposed Alternative to the ASME Code (as stated by the licensee):

> 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 ASME Code Case N-702, 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

[1] This refers to Attachment 2 to the I3R-02 enclosure to the licensee's submittal dated December 30, 2010 (ADAMS Accession No. ML093650139), which shows a complete list of applicable nozzles. This Attachment is not included in this SE.

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) .

[ASME] 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.

Licensee's Bases for Alternative:

The 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," provides the basis for ASME 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., $< 1 \times 10^{-3}$ for 40 years) with or without any ISI.

On December 19, 2007, the NRC issued a SE approving the use of BWRVIP-108 as a basis for using ASME Code Case N-702. In Section 5.0, "Plant Specific Applicability," of the NRC staff's December 19, 2007, SE, it states that licensees who plan to request relief from the ASME code, 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 ASME 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:

Criterion 1: the maximum RPV heatup/cooldown rate is less than 115° F/hour,

(1) Per CPS Technical Specification 3.4.11, the RPV heatup/cooldown rate is limited to less than or equal to 100° F in any 1 hour period. This criterion is met.

Criteria 2 and 3: for the recirculation inlet nozzles,

(2) $(pr/t)/C_{RPV} = 0.96 < 1.15$

(3) $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NOZZLE} = 1.04 < 1.15$

Criteria 4 and 5: for the recirculation outlet nozzles,

(4) $(pr/t)/C_{RPV} = 1.14 < 1.15$

(5) $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NOZZLE} = 0.97 < 1.15$

Demonstration of how CPS meets the NRC plant-specific applicability is provided above. Based upon all RPV nozzle-to-vessel shell welds and nozzle inner radii sections meeting the NRC plant-specific criteria, ASME Code Case N-702 is applicable to CPS. Therefore, use of ASME Code Case N-702 provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i) for all RPV nozzle-to-vessel shell welds and nozzle inner radii sections.

3.2.5 NRC Staff's Evaluation of Proposed Alternative

The December 19, 2007, SE for the BWRVIP-108 report, specified five plant-specific criteria that licensees must meet to demonstrate that the BWRVIP-108 report results apply to their plants. The five criteria are related to the driving force of the probabilistic fracture mechanics (PFM) analyses for the recirculation inlet and outlet nozzles. It was stated in the December 19, 2007, SE that the nozzle material fracture toughness-related reference temperature (RT_{NDT}) used in the PFM analyses was based on data from the entire fleet of BWR RPVs. Therefore, the BWRVIP-108 report PFM analyses are bounding with respect to fracture resistance, and only the driving force of the underlying PFM analyses needs to be evaluated. It was also stated in the December 19, 2007, SE that, except for the RPV heatup/cooldown rate, the plant-specific criteria are for the recirculation inlet and outlet nozzles only because the probabilities of failure, P(F|E)s, for other nozzles are an order of magnitude lower. The plant-specific heatup/cooldown rate that the NRC staff established in Criterion 1 regards the rate under the plant's normal operating condition, which is limiting. Events with excursions of heatup/cooldown rates exceeding 115° F/hour are considered as transients. According to the December 19, 2007, SE, the PFM results with a very severe low temperature overpressure transient is not limiting, largely because the event frequency for that transient is $1x10^{-3}$ as opposed to 1.0 for the normal operating condition.

The licensee provided in the submittal EGC's plant-specific data for the CPS RPV and its evaluation of the five driving force factors, or ratios, against the criteria established in the December 19, 2007, SE. The NRC staff verified the licensee's evaluation, which indicated that all criteria are satisfied. Considering that the driving force factor for the recirculation inlet and outlet nozzles is lower than the plant-specific criterion (1.15) and the P(F|E)s for other RPV nozzles are an order of magnitude lower than the recirculation inlet nozzles, the NRC staff concluded that the licensee's proposed alternative for all CPS RPV nozzles included in this application (see Section 3.2.4) provides an acceptable level of quality and safety. It should be noted that RPV feedwater nozzles and CRD return line nozzles are outside the scope of ASME Code Case N-702 and are also outside the scope of this application.

ASME Code Case N-702 permits a VT-1 visual examination in lieu of volumetric examination for Item Nos. B3.20 and B3.100 which includes the visual examination of the nozzle inner radius without performing a sensitivity demonstration of detecting a 1-mil width wire or crack. This is not consistent with the NRC position established in RG 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," regarding ASME Code Case N-648-1. However, since the licensee stated in the submittal that for the nozzle inner radius sections (Item No. B3.100) requiring examination, a volumetric examination will be performed, the inconsistency between ASME Code Case N-702 and the NRC position regarding VT-1 is not an issue in this application.

3.2.6 Conclusion

The NRC staff has reviewed the submittal regarding the licensee's evaluation of the five plant-specific criteria specified in the December 19, 2007, SE for the BWRVIP-108 report, which provides technical bases for use of ASME Code Case N-702, to examine RPV nozzle-to-vessel welds and nozzle inner radii at CPS. Based on the evaluation in Section 3.2.5, the NRC staff determined that the licensee's proposed alternative, pursuant to 10 CFR 50.55a(a)(3)(i), provides an acceptable level of quality and safety and applies to all requested CPS RPV nozzles, with the exception of the RPV feedwater nozzles and CRD return nozzles. It should be noted that the licensee's request did not include the VT-1 visual examination specified in ASME Code Case N-702, Item Nos. B3.20 and B3.100. The requested duration is the remainder of the third 10-year interval of the CPS ISI program.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

3.3 Relief Request I3R-03

3.3.1 Introduction

The licensee in RR I3R-03 pursuant to 10 CFR 50.55a(g)(5)(iii) requested relief from performing a system leakage test of the Reactor Vessel Head Flange Seal Leak Detection Piping at the ASME Code-required test pressure corresponding to nominal operating pressure during system operation.

The licensee has stated in the RR that the configuration of the vessel tap, combined with the small size of the tap and the test pressure requirement prevents the tap from being temporarily plugged. 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, pressurization in the opposite direction into the recessed cavity and retainer clips would likely damage the O-ring and thus, result in replacement of the O-ring.

The NRC staff has evaluated the licensee's proposed alternative in the RR pursuant to 10 CFR 50.55a(a)(3)(ii) that compliance to the Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.3.2 System/Component(s) for Which Relief is Requested

Reactor Vessel Head Flange Seal Leak Detection Piping

3.3.3 ASME Code Requirements

The 2004 Edition of ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H, Item Number C7.10 requires a system leakage test 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) [IWC-5221].

3.3.4 Licensee Proposed Alternative and Basis for Use

Request for Relief:

Relief is requested from performing the system leakage test at a pressure corresponding to nominal operating pressure during system operation. The licensee proposed an alternative pressure testing requirement in lieu of the system leakage test required under IWC-5221 for the Reactor Vessel Head Flange Seal Leak Detection Piping.

Licensee's Basis for Requesting Relief:

The Reactor Vessel Head Flange Seal Leak Detection Piping is separated from the reactor pressure boundary by one passive membrane, which is an O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange. This piping is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in the annunciation of a High Level alarm in the Control Room. Failure of the inner O-ring is the only condition under which this line is pressurized.

The configuration of this piping precludes system pressure testing while the vessel head is removed because the odd configuration of the vessel tap coupled with the high test pressure requirement prevents the tap in the flange from being temporarily plugged or connected to other piping. The opening in the flange is smooth-walled, making the effectiveness of a temporary seal very limited. Failure of this seal could possibly cause ejection of the device used for plugging or connecting to the vessel.

The configuration also precludes pressure testing with the vessel head installed because the seal prevents complete filling of the piping, which has no vent available. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips that are housed in recessed cavities in the flange face. If a pressure test was performed with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the inner O-ring that would tend to push it into the recessed cavities that house the retainer clips. The thin O-ring material would very likely be damaged by this inward force. Purposely failing the inner O-ring to perform the Code-required test would require procurement of 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.

Licensee's Proposed Alternative:

A VT-2 visual examination and the system leakage test will be performed on the Reactor Vessel Head Flange Seal Leak Detection Piping during flood-up of the refueling pool during a refueling outage. The hydrostatic head developed due to water above the vessel flange during flood-up will allow for the detection of any gross indications in the piping. This examination will be performed with the frequencies specified by Tables IWB-2500-1and IWC-2500-1.

3.3.5 NRC Staff's Evaluation of Proposed Alternative

The ASME Code, Section XI of Record requires that all Class 2 components undergo a system leakage test once each inspection period (40 months). In RR I3R-03, the licensee requested relief from performing a system leakage test of the Reactor Vessel Head Flange Seal Leak Detection Piping at the Code-required test pressure corresponding to the nominal operating pressure during system operation. The piping is located between the inner and the outer O-ring seals of the vessel flange, and is required during plant operation in order to detect failure of the inner flange seal O-ring. The design of this line makes the Code-required system leakage test difficult either with the vessel head in place or removed. The piping cannot be filled completely with water since it cannot be vented to remove entrapped air from the line either with the vessel head in place or removed due to its configuration. If a pressure test were to be performed with the head in place, the space between the inner and the outer O-ring seals would be pressurized. The test pressure would exert a net inward force on the inner O-ring that would tend to push it into the recessed cavities that house the retainer with the possibility of damaging the inner O-ring seal. The configuration of this piping also precludes system pressure testing while the vessel head is removed because the odd configuration of the vessel tap coupled with the high test pressure requirement prevents the tap in the flange from being temporarily plugged or connected to other piping. The opening in the flange is smooth walled, making the effectiveness of a temporary seal very limited. Failure of this seal could possibly cause ejection of the device used for plugging or connecting to the vessel.

If the licensee were to perform the system leakage test in accordance with the Code requirement by pressurizing the space between the inner and the outer O-ring seals, it will likely fail the inner O-ring and subsequently require replacement of the damaged O-ring with a new O-ring. This will result in loss of outage time and at the same time expose test crew to additional radiation in the process of de-tensioning and removal of the reactor vessel head, replacement of the inner O-ring and the installation of the reactor vessel head. This evolution would create extreme hardship to the licensee without a compensating increase in the level of quality and safety. The licensee, however, has proposed to perform a VT-2 visual examination of the Reactor Vessel Head Flange Seal Leak Detection Piping when the reactor cavity is flooded with water during a refueling outage. The NRC staff believes that the hydrostatic head developed due to water above the vessel flange during flood-up will allow for the detection of any gross inservice flaws if present in the subject piping and the proposed testing would provide reasonable assurance of structural integrity. Therefore, it is acceptable.

3.3.6 Conclusion

Based on staff's evaluation, a system leakage test of the Reactor Vessel Head Flange Seal Leak Detection Piping at the Code-required test pressure corresponding to the nominal operating pressure during system operation would cause hardship to the licensee without a compensating increase in the level of quality and safety. The licensee's proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in RR I3R-03 is authorized for the Third 10-year ISI interval of CPS.

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.

3.4 Relief Request I3R-04

3.4.1 Introduction

RR I3R-04 requested approval of an alternative to the requirement of the ASME Code, Section XI, in regard to pressure testing of all Class 2 and Class 3 instrument air (IA) piping supplying all safety relief valves and both feedwater containment outboard isolation check valves. The licensee proposed to perform a pressure decay test in lieu of the visual examination during pressure testing of each safety relief valve (SRV) and the feedwater isolation check valve accumulator including all associated piping. The performance of pressure decay test of each SRV and the feedwater isolation check valve accumulator along with associated piping will be conducted under the surveillance procedure CPS 9061.11 "Instrument Air Check Valve Operability and Pipe pressure Test" that verifies the operability of all 16 SRVs and both check valves in accordance with the CPS Inservice Test program.

The NRC staff has reviewed the licensee's proposed alternative I3R-04 pursuant to 10 CFR 50.55a(a)(3)(ii) that the alternative would provide reasonable assurance of structural integrity and compliance to Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.4.2 System/Component(s) for Which Relief is Requested

Class 2 IA piping and components between containment isolation valves.
Class 3 IA piping and components supplying to all SRVs and both feedwater containment outboard isolation check valves.

3.4.3 ASME Code Requirements

The 2004 Edition of the ASME Code, Section XI, paragraphs IWC-5200 and IWD-5200, state that pressure retaining components shall be tested at the frequency stated in, and visually examined by the methods specified in Table IWC-2500-1, Examination Category C-H and in Table IWD-2500-1, Examination Category D-B respectively. Tables IWC-2500-1 and IWD-2500-1 require performance of VT-2 visual examination during system pressure tests. This pressure test is required to be conducted once each inspection period.

Examination Categories:	Class 2 – C-H, All Pressure Retaining Components. Class 3 – D-B, All Pressure Retaining Components.
Item Numbers:	Class 2 – C7.10, Piping and Pressure Retaining Components. Class 3 – D2.10, Pressure Retaining Components.

3.4.4 Licensee's Proposed Alternative and Basis for Use

Relief is requested from performance of system leakage tests and VT-2 visual examination requirements specified in Tables IWC-2500-1 and IWD-2500-1 for all Class 2 and Class 3 IA pressure retaining components in accordance with Paragraphs IWC-5220 and IWD-5220 respectively.

Licensee's Basis for Requesting Relief:

Pursuant to 10 CFR 50.55a(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 surfaces of large amount of piping and components, many of which are located in elevated dose rate areas with limited access. Therefore, the VT-2 visual examination would expose test crew to additional radiation exposure (estimated 2 rems) and industrial safety challenges without a compensating increase in the level of quality and safety. Moreover, the VT-2 examination would not be consistent with "As Low As Reasonably Achievable" (ALARA) practices.

The surveillance procedure CPS 9061.11 verifies operability of the SRV actuation and functioning of check valves in the IA supply lines to all 16 SRVs and both feedwater containment outboard isolation check valves. The surveillance test is performed for each individual SRV and the feedwater containment outboard isolation check valve as a requirement of the CPS Inservice Test program. One specific test under this surveillance is the pressure decay test of the accumulators of the SRV and the feedwater containment outboard isolation check valves and the associated piping. 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 the acceptance criteria of the pressure decay test specified in the procedure is exceeded, the surveillance identifies appropriate troubleshooting methods to perform including application of soap solution to the surface of the component to detect any evidence of leakage.

The pressure decay test performed under CPS 9061.11 identifies any degradation of the Class 2 and 3 IA supply piping to the SRVs in the Automatic Depressurization System (ADS), and the feedwater containment outboard isolation check valve accumulators and associated piping. The volume tested by this surveillance encompasses all piping and components requiring testing under the ASME Code, Section XI for the portions of the IA system. The surveillance is also performed at a frequency greater than that required per Tables IWC-2500-1 and IWD-2500-1 and the test pressure is consistent with the pressure requirements of both tables. Thus, the tests performed under this surveillance will invariably ensure structural integrity of the components as that of pressure testing in accordance with Tables IWC-2500-1 and IWD-2500-1.

Licensee's Proposed Alternate Examination:

As an alternative to the VT-2 visual examination requirements of Tables IWC-2500-1 and IWD-2500-1, the CPS will perform a pressure decay test of all Class 2 and Class 3 IA piping and accumulators supplying to each of the 16 SRVs and the two feedwater containment outboard isolation check valves required under the CPS Inservice Test program as outlined in the surveillance procedure CPS 9061.11.

3.4.5 NRC Staff's Evaluation of Proposed Alternative

The ADS utilizes selected SRVs for depressurization of the reactor. Each of the SRVs utilized for automatic depressurization is equipped with an air accumulator and check valve arrangement. The accumulators assure that the valves can be held open following failure of the air supply to the accumulators.

The 2004 ASME Code, Section XI, requires a system leakage test of all pressure retaining components of the ADS and the SRV accumulators including the associated piping once every 40 months and a VT-2 visual examination during the system leakage test to detect evidence of leakage from the pressure retaining components. The Code further states that the contained fluid in the system which is air shall serve as the pressurizing medium. Therefore, soap solution is applied to the surface of the component to detect any evidence of leakage. As an alternative to the VT-2 visual examination requirements of the Code, the licensee proposes to take credit for the Technical Specifications surveillance performed which states that each ADS SRV shall be determined operable automatically and manually on a 24 month frequency. In addition, the Technical Specifications requires that these valves are to be surveillance tested in accordance with the Inservice Testing program during each refueling outage.

The licensee's surveillance is performed in accordance with procedure CPS 9061.11, "Instrument Air Check Valve Operability and Pipe Pressure Test," which verifies operability of the SRV IA supply system, actuator, solenoids, and the valve stroke distance. This procedure requires performance of a pressure decay test by isolating and pressurizing the ADS and the SRV accumulators including their associated piping to the nominal operating pressure and monitoring the pressure decay to ensure leak-tight integrity of the components. The decay in pressure of 1.5 psig within 60 to 108 minutes for larger volume components such as the accumulator headers for SRVs and 26 to 31 minutes for smaller volume components such as the accumulator header for the feedwater check valves as stated in the above surveillance procedure, is an acceptable limit beyond which the pressure boundary is subjected to investigation for location of any leakage.

The NRC staff considers this leakage criterion based on pressure decay to be an acceptable alternative to the Code-required VT-2 visual examination of the pressure boundary. Even the ASME Code, Section XI, allows rate of pressure loss as an alternative to the VT-2 visual examination during system leakage test of buried components that are isolable by means of valves. Nevertheless, assuming that the pressure decay provides an adequate assurance of a leak-tight integrity of the components during the surveillance, the proposed alternative offers further conservatism as reflected in the frequency of surveillance being once every 24 months as opposed to the Code-required VT-2 visual examination frequency of once every 40 months.

The NRC staff believes it to be unnecessary to conduct a VT-2 visual examination in accordance with Tables IWC-2500-1 or IWD-2500-1 for Class 2 or Class 3 components of the ASME Code, Section XI, over and above the surveillance required under Technical Specifications and the inservice tests to ensure leak-tight integrity of the pressure boundary of the ADS SRV accumulators and their associated piping. Therefore, the NRC staff finds that the licensee's proposed alternative would provide reasonable assurance of structural integrity while achieving ALARA goals and compliance to Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.4.6 Conclusion

The NRC staff concludes that the licensee's proposed alternative to perform pressure decay test for the ADS SRV accumulators and both feedwater containment outboard isolation check valves including associated piping every refueling outage in accordance with surveillance procedure CPS 9061.11, "Instrument Air Check Valve Operability and Pipe Pressure Test," would provide reasonable assurance of structural integrity and compliance to Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the CPS for the third 10-year inservice inspection interval.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

3.5 Relief Request I3R-05

3.5.1 Introduction

Relief request I3R-05 requests relief and use of alternative requirements for examination of Code Class 2 pumps on the basis that the code requirement is impractical. The licensee requested relief pursuant to 10 CFR 50.55a(g)(5)(iii). The NRC staff has evaluated the licensee's proposed alternative in the relief request pursuant to 10 CFR 50.55a(g)(6)(i).

3.5.2 Components for Which Relief is Requested

Code Class 2
Examination Category C-G Pressure Retaining Welds in Pumps
Item No. C6.10 Pump Casing Welds for the following pumps

Residual Heat Removal (RHR) Pumps 1E12-C001A, 1E12-C001B and 1E12-C001C
High-Pressure Core Spray (HPCS) Pump 1E22-C001
Low-Pressure Core Spray (LPCS) Pump 1E21-C001

3.5.3 ASME Code Requirements

ASME Code, Section XI, Table IWC-2500-1, Examination Category C-G, Item No. C6.10, requires 100 percent surface examination, as defined by Figure IWC 2500-8, of all pump casing welds. Where multiple pumps of similar design, size, function, and service exist in a system, only one of the multiple pumps is required to be examined. The examination may be performed from the inside or outside surface of the pump.

3.5.4 Licensee's Request for Alternative and Basis for Use

Proposed Alternative:

> In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the ASME Code – required surface examination of the casing welds on the Class 2 pumps identified above. 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 detailed in 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 [ASME Code, Section XI,] 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 ASME Code, Section XI, IWA-5241(b).

Basis for Impracticality:

The CPS residual heat removal pumps, high pressure core spray pumps, and low pressure core spray pumps 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.

3.5.5 NRC Staff's Evaluation of Proposed Alternative

The ASME Code requires surface examination of all casing welds in the subject ASME Code, Section XI, Class 2 pumps. However, the subject pumps are encased in concrete which makes the pressure retaining welds inaccessible from the outside surface. The inside surface of these welds may be partially accessible during disassembly, however, the pumps are not regularly disassembled for maintenance. To gain access for outside surface examination, it would be necessary to destroy the concrete encasing the pumps. Alternatively, the pumps would require disassembly solely for the purpose of examining the inside surface of the casing welds. Either of these options would create a significant burden on the licensee, therefore making the ASME Code-required surface examinations impractical to perform at CPS.

The licensee has proposed to perform the ASME Code-required surface examinations, or VT-1 visual examinations, from the inside of the pumps, if any of the subject pumps are disassembled during routine maintenance activities. Additionally, the accessible areas of these pumps (where the pump body merges with the surrounding concrete) are subject to a VT-2 visual examination during system leakage tests. The VT-2 visual examinations should provide an indication if significant leakage due to degradation of the pump body is experienced.

The licensee has shown that it is impractical to meet the ASME Code-required 100 percent surface examination coverage for the subject pump casing welds due to the encasement of these components in concrete. Based on the licensee's proposed alternative to examine the welds from the inside surface, if disassembled during maintenance activities, along with the VT-2 visual examinations performed during ASME Code-required pressure testing, it is concluded that there is reasonable assurance of structural integrity of the subject Class 2 pumps.

3.5.6 Conclusion

The NRC staff concludes that complying with the ASME Code requirement for surface examination of the RHR, HPCS and LPCS pumps at CPS is impractical. Furthermore, based on the above, the NRC staff determines that the proposed alternative described in RR I3R-05 provides reasonable assurance of structural integrity of the subject components. The NRC staff has determined that granting this relief 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. Therefore, the NRC grants relief and use of the proposed alternative in accordance with 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI at CPS.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: E. Andruszkiewicz
N. DiFrancesco
S. T. Ghosh
K. Hoffman
P. Patnaik

Date: December 22, 2010

10 CFR 50.55a(g)(6)(i) for alternative RR I3R-05. Therefore, the NRC authorizes alternative RR I3R-01, I3R-02, I3R-03, I3R-04, and I3R-05 at CPS for the third 10-year ISI interval, which begins on July 1, 2010, and ends on June 30, 2020.

The NRC staff's Safety Evaluation regarding RRs I3R-01, I3R-02, I3R-03, I3R-04, and I3R-05 is enclosed. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector. This completes the NRC staff's efforts on TAC Nos. ME2987, ME2988, ME2989, ME2990, and ME2991.

If you have any questions, please contact the Clinton Project Manager, Nicholas DiFrancesco, at 301-415-1115.

Sincerely,

/RA by E. Brown for/

Robert D. Carlson, Branch Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure:
Safety Evaluation

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NAME	NDiFrancesco	CNorton	THarris	TLupold
DATE	12/02/10	12/21/10	12/14/10	10/17/10
OFFICE	APLA/BC	CSGB/BC	CVIB/BC	LPL3-2/BC
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