SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION OF THE SECOND TEN YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NOS. 95-020, 95-030, AND 95-040 FOR FLORIDA POWER AND LIGHT COMPANY CRYSTAL RIVER NUCLEAR PLANT, UNIT 3 DOCKET NUMBER: 50-302

1.0 INTRODUCTION

The Technical Specifications for Crystal River Nuclear Plant, Unit 3 state that the inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).
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 difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice 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 the components. The regulations require that inservice examination of components and system pressure tests conducted during the first ten-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the Crystal River Nuclear Plant, Unit 3 second 10-year inservice inspection (ISI) interval is the 1983 Edition through the Summer 1983 Addenda Code, except that the extent of examination for Class 1, Examination Category B-J, and Class 2, Examination Category C-F and C-G welds in the Residual Heat Removal (RHR), Emergency Core Cooling (ECC), and Containment Heat Removal (CHR) systems has been determined by the requirements of the 1974 Edition through summer 1975 Addenda (74S75) as permitted and required by 10CFR50.55a(b). The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, 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. In a letter dated September 22, 1995, Florida Power and Light Company submitted to the NRC Requests for Relief (RR) Nos. 95-010, 95-020, 95-030, and 95-040 for the Crystal River Nuclear Plant, Unit 3. This SE addresses only RR Nos. 95-020, 95-030, and 95-040. RR 95-010 will be addressed later.

2.0 EVALUATION AND CONCLUSIONS

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the information provided by the licensee in support of its Second Ten-Year Interval Inservice Inspection Program Plan, RR Nos. 95-020, 95-030, and 95-040 for the Crystal River Nuclear Plant, Unit 3.

Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the Technical Letter Report attached. For RR No. 95-020, the staff has concluded that compliance with the Code requirements would result in hardship or unusual difficulty without a compensating increase level of quality and safety. The licensee's proposed alternative to use Code Case N-498-1 in lieu of the Code requirements provides reasonable assurance of operational readiness and safety. Therefore, the licensee's alternative contained in RR No. 95-020 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii), provided that all requirements of Code Case N-498-1 are satisfied. This relief is authorized until such time as the Code Case is published in a future revision of Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement this Code Case, the licensee is required to follow all provisions in Code Case N-498-1, with limitations issued in Regulatory Guide 1.147, if any.

For RR No. 95-030, the staff concludes that requiring the licensee to perform the Code-required examination of the reactor pressure vessel transition piece-to-bottom head weld with the examination device currently being deployed for in-vessel examinations, will result in a burden without a compensating increase in quality and safety, because there is a potential for damage to occur if the examination device impacts the in-vessel components. Therefore, the licensee's proposed alternative contained in RR No. 95-030 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second interval only, on a one-time basis.

The licensee request to use the 1989 Edition, Section XI, Appendix I, Supplement 3 regarding examination of Class 1 and 2 ferritic piping with diameters greater than 20 inches is authorized in accordance with 10 CFR 50.55a(g)(4)(iv) provided all of the provisions of Appendix I, Article I-2000, Supplement 3, are met.

Enclosure 2

TECHNICAL LETTER REPORT
ON THE SECOND 10-YEAR INSERVICE INSPECTION INTERVAL
REQUESTS FOR RELIEF 95-020, 95-030 AND 95-040
FOR
CRYSTAL RIVER NUCLEAR PLANT, UNIT 3
FLORIDA POWER CORPORATION
DOCKET NUMBER: 50-313

1.0 INTRODUCTION

By letter dated September 22, 1995, Florida Power Corporation submitted Requests for Relief 95-010, 95-020, 95-030, and 95-040. In a letter dated February 16, 1996, the licensee resubmitted Request for Relief 95-030 (Supplement 1) that provided additional information in support of the request for relief. By letter dated March 12, 1996, the licensee submitted Supplement 2, to provide further information in support of Request for Relief 95-030. Provided with this Technical Letter Report are the evaluations of Request for Relief 95-020, 95-030, and 95-040. Request for Relief 95-010 will be evaluated in a separate report. The Idaho National Engineering Laboratory (INEL) staff has evaluated the subject requests for relief in the following section.

2.0 EVALUATION

The Code of record for the Crystal River Nuclear Plant, Unit 3, second 10-year inservice inspection (ISI) interval, that began March, 1987, is the 1983 Edition through the Summer 1983 Addenda, of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI* of the Code, except that the extent of examination for Class 1, Examination Category B-J, and Class 2, Examination Category C-F and C-G welds in the Residual Heat Removal (RHR), Emergency core Cooling (ECC), and Containment Heat Removal (CHR) systems has been determined by the requirements of the 1974 Edition through summer 1975 Addenda (74S75) as permitted and required by 10CFR50.55a(b). The information provided by the licensee in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below.

A. Request for Relief 95-020, Examination Categories D-A, D-B, and D-C, Items D1.10, D2.10, and D3.10, System Hydrostatic Tests of Class 3 Systems

Code Requirement: Section XI, Table IWD-2500-1, Examination Categories D-A, D-B, and D-C, Items D1.10, D2.10, and D3.10 require a system hydrostatic test in accordance with IWD-5223 for Class 3 systems each 10-year interval. Paragraph IWD-5223(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure, P_{sv}, for systems with design temperatures of 200°F or less, and at least 1.25 times the system pressure, P_{sv}, for systems with design temperatures above 200°F.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required 10-year hydrostatic tests for portions of Class 3 piping in Main Steam, Auxiliary Steam, Emergency Feedwater, Condensate, Makeup and Purification, Domestic Water, Nuclear Services & Decay Heat Sea Water, Spent Fuel, and Chilled Water systems.

Licensee's Basis for Requesting Relief (as stated):

"Crystal River Unit 3 is currently in the third period of the second inservice inspection interval. As such, hydrostatic testing of most ASME Class III systems has previously been completed. For the remaining portions of ASME Class III systems listed in I(a)[1] above, relief is requested from the periodic hydrostatic tests of class III Pressure Retaining Components required by Table IWD-2500-1. Code Case N-498-1 outlines the basis for relief to be the following:

(1) A system inservice or functional pressure test shall be conducted at or near the end of each inspection interval or during the same inspection period of each inspection interval of Inspection Program B.

(2) The boundary subject to test pressurization during the system inservice or functional pressure test shall extend to all Class 3 components included in those portions of systems

[1]Refers to the Class 3 systems in the licensee's request for relief.

required to operate or support the safety system function up to and including the first normally closed valve, including a safety or relief valve, or valve capable of automatic closure when the safety function is required.

(3) Prior to performing the VT-2 visual examination, the system shall be pressurized to nominal operating pressure for at least 4 hours for insulated systems and 10 minutes for noninsulated systems. The system shall be maintained at nominal operating pressure during performance of the VT-2 visual examination."

(4) The VT-2 visual examination shall include all components identified in (2) above.

(5) Test instrumentation requirements of IWA-5260 are not applicable.

"Substitution of an inservice or functional pressure test for the required hydrostatic tests in accordance with Code Case N-498-1 will ensure the continued integrity of ASME Class III Pressure Retaining Components while minimizing dose to inspection personnel and reducing the quantity of radioactive water produced at Crystal River Unit 3".

Licensee's Proposed Alternative Examination (as stated):

"A system inservice test or system functional test will be performed in accordance with the guidance in Code Case N-498-1."

Evaluation: The licensee requested relief from the Code-required hydrostatic tests for the subject Code Class 3 systems. ASME Section XI Code Case N-498-1, "*Alternative Rules for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems*" dated May 11, 1994, provides alternatives to the Code required 10-year hydrostatic tests for Class 3 systems. Code Case N-498, "*Alternative Rules for 10-Year System Hydrostatic Testing for Class 1 and 2 Systems*," has previously been approved for use. The rules for Code Classes 1 and 2 in N-498-1 are unchanged from N-498. The NRC staff found Code Case N-498 acceptable because the alternative provides adequate assurance of operational readiness.

A significant amount of effort may be necessary (depending on system, plant configuration, Code class, etc.) to temporarily remove or disable Code safety and/or relief valves to meet test pressure requirements. The safety assurance provided by the enhanced leakage gained from a slight increase in system pressure during a hydrostatic test may be offs [illegible] gated by the following factors: having to gag or remove code sa[illegible] and/or relief valves, placing the system (and thus the plant) [illegible] an off-normal state, erecting temporary supports in steam lines, possible extension of refueling outages, and resource requirements to set up testing with special equipment and gages.

Information prepared in conjunction with ASME Code Case N-498-1 notes that the system hydrostatic test is not a test of the structural integrity of the system but rather an enhanced leakage test, as indicated in a paper by S. H. Bush and R. R. Maccary, *"Development of In-Service Inspection Safety Philosophy for U.S.A. Nuclear Power Plants,"* ASME, 1971. Piping components are designed for a number of loadings that are postulated to occur under the various modes of plant operation. However, hydrostatic testing only subjects the piping components to a small increase in pressure over the design pressure and therefore does not present a significant challenge to pressure boundary integrity since piping dead weight, thermal expansion, and seismic loads, which may present far greater challenges to the structural integrity of a system than fluid pressure, are not part of the loading imposed during a hydrostatic test. Accordingly, hydrostatic pressure testing is primarily regarded as a means to enhance leak detection during the examination of components under pressure, rather than as a measure to determine the structural integrity of the components.

Revision N-498-1 specifies requirements for Class 3 that are identical to those for Class 2 components. In lieu of 10-year hydrostatic pressure testing at or near the end of the 10-year interval, Code Case N-498-1 requires a VT-2 visual examination at nominal operating pressure and temperature in conjunction with a system leakage test in accordance with paragraph IWA-5000 of the

1992 Edition of Section XI.

Giving consideration to the limited amount of increased assurance provided by the higher pressures associated with a hydrostatic test versus the pressures for the system leakage test, and the hardship associated with performing the ASME Code required hydrostatic test, the INEL staff finds that compliance with the Section XI hydrostatic testing requirements results in hardship for the licensee without a compensating increase in the level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii), provided that all requirements of Code Case N-498-1 are satisfied. This relief should be authorized until such time as the Code Case is published in a future revision of Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement this Code Case, the licensee should be required to follow all provisions in Code Case N-498-1, with limitations issued in Regulatory Guide 1.147, if any.

B. Request for Relief 95-030, Examination Category B-A, Item B1.21, Reactor Pressure Vessel Circumferential Head Weld

Requirement: Examination Category B-A, Item B1.21, requires 100% volumetric examination of the accessible length of one circumferential head weld as defined by Figure IWB-2500-3.

Licensee's Code Relief Request: The licensee requested relief from performing the examination of the reactor pressure vessel transition piece-to-bottom head weld, ISI Exam Number B1.2.2.

Licensee's Basis for Requesting Relief (as stated):

"The subject weld is the reactor vessel transition-piece-to-bottom-head weld. This weld is located below the beltline region and is not subject to the majority of the neutron flux escaping from the core. An evaluation of neutron embrittlement as a potential damage mechanism and other potential damage mechanisms associated with this weld is included in the attached response to Question No. 2A[2] of Reference B (see Attachment 1). The evaluation concludes that service-induced degradation of the transition-piece-to-bottom-head weld as a result of corrosion, fatigue, nuclear, or thermal embrittlement mechanisms is extremely unlikely.

"The weld has been visually and ultrasonically inspected once during preservice inspection (essentially 100% coverage). The volumetric examination method utilized during the preservice inspection was Manual Contact Ultrasonic. During this examination the weld received a 360 degree scan with the exception of those areas where physical interference prevented examination with the manual transducer. A review of the data sheets for this examination revealed that there were no reportable or recordable indications detected.

"During Refueling Outage 5 (May 1985), the weld was partially inspected (approximately 5%). This inspection was performed per the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition with Addenda through Summer 1975 and Regulatory Guide 1.150. The extent of this examination was acceptable since the 1974 Edition of ASME Section XI, Table IWB-2500, Category B-A only required the examination of 5% of this weld. The examination was performed using the ARIS II remote scanner, a device that utilized immersion ultrasonic techniques. The examination revealed no baseline indications, no reportable indications and no recordable indications.

"Although use of the immersion method allowed the weld to be inspected with inspection equipment at a distance of 20 inches away from the weld, access for examination of this weld was severely limited by the flow stabilizers, the core support lugs, and the incore instrumentation nozzles.

"Since the last inservice inspection was performed, improvements in volumetric examination methods have shown the contact examination method to be much more accurate and reliable than the immersion method. As a result, equipment designed to use the immersion method has been abandoned and modern reactor vessel inspection equipment has been designed to utilize the contact examination method.

"Crystal River Unit 3 is currently into the second interval of operation. During the last outage of this interval (started February 16, 1996), volumetric examination of reactor vessel welds will be performed using modern automated reactor vessel inspection equipment. Equipment developed for use of the contact method of examination requires more physical access to the surface of the weld than the immersion method previously used. Therefore, the surface

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[2]Attachments and responses to questions are not included with this evaluation.

of the weld which could be successfully examined using this equipment will be less than that achieved during the first interval.

"The response to RAI Question No. 2E, addresses the obstructions located on, and adjacent to this weld which creates an area in which it is very difficult to maneuver the ultrasonic transducer manipulator. The response provides a detailed access study which includes information about FPC's concern for potential impact to the incore instrument nozzles while examining this weld. The nozzles are small and manufactured to close tolerances. If an inadvertent collision were to occur, the nozzles could be severely damaged. A damaged nozzle could prevent the reinsertion of a incore instrument or could require a critical-path in-vessel repair.

"FPC response to RAI Question No. 2D addresses the radiation dose potential associated with the examination of this lower reactor vessel weld. The response explains how the current estimated dose for the inspection of the weld (minimal) could potentially increase due to damage of the robotic manipulator due to an impact with the interferences surrounding this weld (One inch protrusions which are remnants of flow stabilizers, guide lugs and incore instrument guide tubes).

"Access to the subject weld from the vessel exterior presents considerable problems. The region underneath the reactor vessel (below the subject weld) is limited and gained only by passage through a small tunnel leading from outside the biological shield wall to the area directly below the reactor vessel and inside the reactor vessel support skirt. Due to the congested nature of the area (52 incore guide tubes that penetrate the lower head, the lower head insulation and its support structure), a technician would have to access the area and perform a manual examination of the weld. ALARA concerns would prohibit the consideration of such manual external inspection of the weld.

"A survey of three ISI service organizations which provide reactor pressure vessel examination services (Framatome, Westinghouse/WesDyne, and Southwest Research) resulted in the determination that no alternate equipment is available which would provide significantly better access to the weld. Our survey also revealed that access limitations for inspecting this weld are common to other PWRs, particularly those with a B&W reactor vessel. The survey also confirmed, based on previous industry experience with limited exams performed on equivalent welds during PWR reactor vessel examinations, that the probability of identifying a service induced flaw in the subject weld would be low.

"Accordingly, due to the hardship imposed in implementing this requirement Florida Power Corporation requests relief from examination of the transition-piece-to-bottom-head weld based upon:

"The results of evaluating potential damage mechanisms for this weld which revealed a low probability of service-induced degradation due to corrosion, fatigue, nuclear, or thermal embrittlement.

"The small possibility of a significant flaw existing in the weld as demonstrated by the results of previous examinations of the transition-piece-to-bottom-head weld which identified no reportable or recordable indications.

"The lack of identification of any service-induced flaws in any of the reactor vessel welds.

"The severe access limitation addressed in this request and further demonstrated by the access study provided in the response to RAI Question No. 2E.

"The critical-path outage time and significant cost required to perform the examination.

"The industry-wide lack of inspection equipment that would provide better access for the inspection of the weld.

"The greater flaw tolerance of the weld as it is addressed in the response to RAI Question No. 2C.

"The potential increase in radiation dose due to possible damage/repair of the inspection tools during the examination of the weld.

"Should examination of the subject weld have to be performed, FPC estimates that the critical-path outage time for performing a limited examination using automated reactor vessel inspection equipment and the contact method would be a minimum of 12 hours, which is estimated to cost approximately $250,000.00."

Licensee's Proposed Alternative Examination (as stated):

"The accessible areas of the reactor vessel interior including the interior surfaces and welded attachments within and beyond the beltline region will receive the VT-1 and VT-3 visual examinations required by Section XI of the ASME Code. A VT-2 visual examination will be preformed on the exterior of the reactor vessel during the inservice leak test performed during start-up."

Evaluation: The Code provides examination requirements for reactor pressure vessel welds. In the case of the reactor pressure vessel transition piece-to-bottom head weld, (Item B1.21), a volumetric examination of the accessible portion of the subject weld is required. The Florida Power Corporation requested relief from performing the Code-required volumetric examination of the reactor pressure vessel transition piece-to-bottom head weld. The licensee has cited hardships associated with the examination of the subject weld that include minimal volumetric coverage (less than 5%), potential for damage to, and subsequent repair of incore instrumentation tubes, increased radiation dose associated with potential damage and subsequent repair to the examination device, and the minimal possibility of flaws existing in the subject weld.

The licensee contends that the requirement to perform a volumetric examination on the accessible portion of the subject weld results in a hardship. This is due in part to the currently available ultrasonic inspection technique being employed by the licensee. The scheduled reactor pressure vessel inservice examinations will be performed utilizing a contact ultrasonic examination technique. This technique requires that the scanning head be in contact with the vessel inside surface. As a result of the use of the contact technique, accessibility to maximize volumetric coverage becomes more critical relative to obstructions such as guide lugs, flow stabilizers, and incore instrumentation guide tubes. The licensee estimates that the volumetric coverage of the subject weld with the contact technique will be less than 5%

Crystal River Nuclear Plant, Unit 3, is currently in the last refueling outage of the second 10-year interval. The remote inspection device scheduled to perform the reactor pressure vessel examinations utilizes a contact ultrasonic technique. Considering (1) the licensee's concern for potential damage to in-vessel components that could occur when examining with the contact ultrasonic technique, (2) the minimal coverage obtainable on the subject weld, and (3) the fracture analysis[3] that supports the low potential for flaw initiation and larger allowable flaw size, based on lower overall stresses associated with the subject weld and material ductility (affected less by neutron embrittlement), the INEL staff concurs that performing the Code-required volumetric examination would result in a burden without providing a compensating increase in quality and safety. However, the licensee's concern over ultrasonic equipment that could cause in-vessel damage should be addressed with the examination vendor to minimize this potential for all future examinations.

The licensee has proposed to perform VT-1 and VT-3 visual examinations within and beyond the beltline region and a VT-2 visual examination on the exterior of the reactor vessel during the inservice leak test. The INEL staff believes that the proposed alternative examination in conjunction with the volumetric examination of other reactor pressure vessel welds will provide reasonable assurance of the continued structural integrity of this component. Therefore, in consideration of the licensee's concerns associated with the Code-required examination for the second interval, for Crystal River Nuclear Plant, Unit 3, the INEL staff believes that the licensee's request for relief should be authorized on a one-time basis only, pursuant to 10 CFR 50.55a(a)(3)(ii).

D. Request for Relief 95-040, Calibration Block Requirements

Code Requirement: IWA-2232(3)(b) Ultrasonic Examination, requires that ultrasonic examination of Class 1 and Class 2

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[3] *Flaw Acceptance Handbook for Crystal River Unit 3 Reactor Pressure Vessel and Nozzle Weld Inspections*, prepared by Structural Integrity Associates, Inc., Report No. SIR-95-145, Revision 0, dated February 1, 1996, and *Input to Items A and C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld at Crystal River Unit 3*, prepared by Structural Integrity Associates, Inc., Report No. SIR-96-016, Revision 0, dated February 1996.

ferritic steel piping systems shall be conducted in accordance with Appendix III.

Licensee's Code Relief Request: The licensee requested relief from complying with calibration block requirements for main loop welds in ferritic steel piping for the reactor coolant system.

Licensee's Basis for Requesting Relief (as stated):

"Article 5 of Section V of the 1980 and later editions of the ASME Boiler and Pressure Vessel code have authorized the use of flat calibration blocks with a correction factor as shown in Nonmandatory Appendix A for examination of piping greater than 20 inches in diameter.

"The use of flat calibration blocks for examination surfaces with diameters greater than 20 inches is now shown as an alternative in the recently approved 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code in Appendix I, Supplement 3 - Calibration Blocks For Examination of Parts With Curved Surfaces - which states in paragraphs (b)(3)(a):

(b) For calibration blocks for examination surfaces with diameters greater than 20 in., one of the following shall be applied.

(3) A flat calibration block may be used meeting the following requirements:

(a) the minimum radius to be examined shall be determined and the search unit contact area and frequency shall be selected so that the minimum radius is greater than the critical radius as determined by Appendix A of Article 5 of Section V."

Licensee's Proposed Alternative Examination (as stated):

"Examination of ASME Class 1 and 2 ferritic piping with diameters greater than 20 inches will be performed in accordance with the requirements of the 1989 Edition of the ASME Boiler and Pressure Vessel Code Section XI Appendix I, Supplement 3."

Evaluation: The Code requires that calibration blocks be of the same diameter, thickness, and material as the area to be examined. The licensee proposed, as an alternative to Code requirements, to implement calibration block requirements contained in the 1989 Edition of Section XI. Based on a review of the requirements of the 1989 Edition of Section XI, Appendix I, Article I-2000, *Examination Requirements*, Supplement 3, it has been determined that provisions for piping calibration blocks for curved surfaces allow the use of a flat plate when examining welds in piping greater than 20 inches in diameter. Since the 1989 Edition of ASME Section XI has been approved for use in the Code of Federal Regulations, 10 CFR 50.55a, and there are no related calibration block requirements associated with the use of flat blocks in lieu of curved blocks other than those contained in Supplement 3, it is recommended that the licensee's alternative be authorized in accordance with 10 CFR 50.55a(g)(4)(iv), provided that all of the provisions of Appendix I, Article I-2000, Supplement 3 are met.

3.0 CONCLUSION

The INEL staff has evaluated Requests for Relief 95-020, 95-030, and 95-040. Based on these evaluations, it is recommended that for Request for Relief 95-020, the licensee's proposed alternative to Code requirements, be authorized pursuant to 10 CFR 50.55a(a)(3)(ii), with the condition stated in the evaluation.

For Request for Relief 95-030, the INEL staff believes that requiring the licensee to perform the Code-required examination with the examination device currently being deployed for in-vessel examinations, will result in a burden without a compensating increase in quality and safety. This is based on the potential damage that may occur if the examination device impacts with in-vessel components. However, considering that the Code requires the examination of only the accessible portion of the subject weld, it is recommended that this relief be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second interval only, on a one-time basis.

For Request for Relief 95-040, there are no related calibration block requirements associated with the use of flat blocks in lieu of curved blocks other than those contained in Appendix I, Article I-2000, *Examination Requirements*, Supplement 3. Therefore it is recommended that, the use of Appendix I, Article I-2000, *Examination Requirements*, Supplement 3, of the 1989 Edition of Section XI, be authorized in accordance with 10 CFR 50.55a(g)(4)(iv), provided that all of the provisions of Appendix I, Article I-2000, Supplement 3, are met.