February 15, 2001

Mr. John K. Wood
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT 1 - SAFETY EVALUATION FOR INSERVICE EXAMINATION PROGRAM RELIEF REQUESTS IR-023, REVISION 2; IR-029, REVISION 2; IR-030, REVISION 1; IR-041, REVISION 0; IR-044, REVISION 0; AND IR-045, REVISION 0 (TAC NO. MA8689)

Dear Mr. Wood:

By letter dated April 17, 2000 (PY-CEI/NRR-2491L), and as supplemented by letters dated August 11, 2000 (PY-CEI/NRR-2515L), October 23, 2000 (PY-CEI/NRR-2519L), and January 31, 2001 (PY-CEI-NRR-2542L), FirstEnergy Nuclear Operating Company submitted requests for relief from the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, 1989 Edition, for the Inservice Examination Program for the Perry Nuclear Power Plant, Unit 1 (PNPP). All relief requests are for the second 10-year inspection interval.

Relief Request IR-023, Revision 2, requests relief from the requirements of ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition, Article IWF-5000, with regard to visual examination and functional testing of snubbers. Pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), the staff authorizes the proposed alternative use of an updated Section 6.4.1 of the PNPP Operational Requirements Manual (ORM) for snubber visual examination and functional testing, based on a finding that the proposed alternative provides an acceptable level of quality and safety for the second 10-year inspection interval.

Relief Request IR-029, Revision 2, proposes an alternative to the Code criteria for selection of welds for inservice examination. The staff concludes that the licensee has provided information to support the determination that the performance of Code-required examinations for the identified welds would result in a hardship without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the staff authorizes the proposed alternative identification of welds for the second 10-year interval for PNPP.

Relief Request IR-030, Revision 1, proposes permanent relief from (1) the augmented inspection requirements of the reactor pressure vessel (RPV) shell welds at the PNPP, as required by 10 CFR 50.55a(g)(6)(ii)(A)(2); and (2) the Ten-Year Interval Inservice Inspection Requirements for the circumferential welds as required by Inspection Category B-A, Inspection Item B1.11 to Table IWB-2500-1 of Article IWB to the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code. The staff concludes that the alternative assessments performed by the licensee provide an acceptable basis for permanently deferring the volumetric inspections of the circumferential welds in the PNPP RPV. Therefore, pursuant to 10 CFR

50.55a(a)(3)(i), the staff authorizes the proposed alternative based on a finding that it provides an acceptable level of quality and safety.

IR-041, Revision 0, proposes relief from the documentation requirements of subarticles IWA-4140, “Repair/Replacement Program and Plan,” and IWA-4910, “Reports and Records,” of the 1992 Edition, 1992 Addenda of ASME Section XI for repairs, replacements and modifications to Class MC (IWE) and CC (IWL) components. The staff concludes that the licensee’s documentation requirements will ensure the level of quality and safety of PNPP’s Repair/Replacement activities equivalent to the level that could be achieved by the implementation of the requirements of Subarticles IWA-4140 and IWA-4190 of the 1992 E&A. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the use of the licensee’s proposed alternatives on the basis that they provide an acceptable level of quality and safety for the second 10-year inspection interval.

IR-044, Revision 0, proposes relief from performing the Code-required surface examinations of the reactor pressure vessel closure nuts. Alternative examinations are proposed in accordance with Code Case N-627. The staff concludes that the proposed alternative requirement of Code Case N-627 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes use of Code Case N-627 for the second 10-year inservice inspection interval. Use of Code Case N-627 is authorized for the duration of the second 10-year inservice inspection interval until such time as the code case is approved on a generic basis by the Nuclear Regulatory Commission (NRC).

IR-045, Revision 0, proposes relief from having to perform partial examinations from the flange face in order to defer the RPV shell-to-flange and head-to-flange welds to the end of the inspection interval. The alternative scheduling is proposed in accordance with the provisions of Code Case N-623. The staff concludes that the Code requirements will result in hardship without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the staff authorizes use of Code Case N-623 for the second 10-year inservice inspection interval. Use of Code Case N-623 is authorized for the duration of the second 10-year inservice inspection interval at PNPP until such time as the code case is approved on a generic basis by the NRC.

The staff’s safety evaluation is enclosed.

Sincerely,

/RA/
Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosure: As stated

cc w/encl: See next page

50.55a(a)(3)(i), the staff authorizes the proposed alternative based on a finding that it provides an acceptable level of quality and safety.

IR-041, Revision 0, proposes relief from the documentation requirements of subarticles IWA-4140, “Repair/Replacement Program and Plan,” and IWA-4910, “Reports and Records,” of the 1992 Edition, 1992 Addenda of ASME Section XI for repairs, replacements and modifications to Class MC (IWE) and CC (IWL) components. The staff concludes that the licensee’s documentation requirements will ensure the level of quality and safety of PNPP’s Repair/Replacement activities equivalent to the level that could be achieved by the implementation of the requirements of Subarticles IWA-4140 and IWA-4190 of the 1992 E&A. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the use of the licensee’s proposed alternatives on the basis that they provide an acceptable level of quality and safety for the second 10-year inspection interval.

IR-044, Revision 0, proposes relief from performing the Code-required surface examinations of the reactor pressure vessel closure nuts. Alternative examinations are proposed in accordance with Code Case N-627. The staff concludes that the proposed alternative requirement of Code Case N-627 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes use of Code Case N-627 for the second 10-year inservice inspection interval. Use of Code Case N-627 is authorized for the duration of the second 10-year inservice inspection interval until such time as the code case is approved on a generic basis by the Nuclear Regulatory Commission (NRC).

IR-045, Revision 0, proposes relief from having to perform partial examinations from the flange face in order to defer the RPV shell-to-flange and head-to-flange welds to the end of the inspection interval. The alternative scheduling is proposed in accordance with the provisions of Code Case N-623. The staff concludes that the Code requirements will result in hardship without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the staff authorizes use of Code Case N-623 for the second 10-year inservice inspection interval. Use of Code Case N-623 is authorized for the duration of the second 10-year inservice inspection interval at PNPP until such time as the code case is approved on a generic basis by the NRC.

The staff’s safety evaluation is enclosed.

Sincerely,

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440
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<u>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION</u>

<u>FOR THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION</u>

<u>RELIEF REQUESTS ON ASME CODE, SECTION XI,</u>

<u>PERRY NUCLEAR POWER PLANT, UNIT 1</u>

<u>DOCKET NO. 50-440</u>

<u>RELIEF REQUEST NO. IR-023, REVISION 2</u>
<u>RELIEF REQUEST NO. IR-029, REVISION 2</u>
<u>RELIEF REQUEST NO. IR-044, REVISION 0</u>
<u>RELIEF REQUEST NO. IR-045, REVISION 0</u>

<u>INTRODUCTION</u>

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 (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 difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the 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 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Code, Section XI, for the Perry Nuclear Power Plant (PNPP), Unit 1, second 10-year inservice inspection (ISI) interval is the 1989 Edition.

By letter dated April 17, 2000, First Energy Nuclear Operating Company (the licensee) for PNPP, submitted requests for relief from the requirements of the ASME Code, Section XI, 1989 Edition, for the second 10-year inservice inspection interval of PNPP.

- IR-023, Revision 2, requests relief from the requirements of ASME Code Section XI, 1989 Edition, Article IWF-5000, with regard to visual examination and functional testing

of snubbers. Article IWF-5000 references first Addenda to ASME/ANSI OM-1987, Part 4 (OMa-4).

- IR-029, Revision 2, proposes an alternative to the Code criteria for selection of welds for inservice examination.

- IR-044, Revision 0, requests relief from Code-required surface examination of reactor vessel closure nuts with an alternative VT-1visual examination in accordance with the requirement Code Case N-627, "VT-1 Visual Examination in Lieu of Surface Examination for RPV Closure Nuts."

- IR-045, Revision 0, requests relief from the requirement of the Code concerning deferral of reactor vessel head-to-flange and shell-to-flange welds to the end of inspection interval by proposing alternative scheduling in accordance with the provisions of Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel."

The staff has evaluated the licensee's proposed alternatives pursuant to 10 CFR 50.55a(a)(3) for the second 10-year inservice inspection interval at PNPP.

RELIEF REQUEST NO. IR-023, REVISION 2

Identification of Components

All safety-related hydraulic and mechanical snubbers.

Code Requirement

ASME Code, Section XI, 1989 Edition, Subarticle IWF-5000, Inservice Inspection Requirements for Snubbers, states that preservice examinations and tests of snubbers (IWF-5200), inservice examinations and tests of snubbers (IWF-5300), and examination and tests of snubber repairs and replacements (IWF-5400) shall be in accordance with the first Addenda to ASME/ANSI OM-1987, Part 4. Furthermore, 10 CFR 50.55a(b)(2)(viii) specifies that the ASME/ANSI OM Part 4 Edition and Addenda to be used shall be the OMa-1988 Addenda to the OM-1987 Edition.

Licensee's Requested Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested from performing preservice examinations and tests of snubbers, inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements in accordance with ASME/ANSI OM Part 4 of the OMa-1988 Addenda to the OM-1987 Edition. The relief is requested for PNPP's second 10-year inspection interval.

Licensee's Proposed Alternative (As Submitted)

Preservice and inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements in accordance with the technical requirements within section

6.4.1 of the Operational Requirements Manual (ORM). The functional testing requirements therein, or, in the case of the second sample plan, proposed to be therein, are as follows:

At least once per refueling interval, a representative sample of snubbers shall be tested using one of the following three plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The snubber functional test period may start ninety days prior to a scheduled refueling outage and shall be completed prior to the end of the scheduled refueling outage. The Nuclear Regulatory Commission shall be notified in writing pursuant to 10 CFR Part 50.4 of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented.

1) At least 10 percent of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria an additional 5 percent of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 6.4.1-1 of the ORM. “C” is the total number of snubbers of a type found not meeting the acceptance requirements. The cumulative number of snubbers of a type tested is denoted by “N”. At the end of each day’s testing, the new values of “N” and “C” (previous day’s total plus current day’s increments) shall be plotted on Figure 6.4.1-1. If at any time the point plotted falls on or below the “Accept” line, testing of snubbers of that type may be terminated. When the point plotted lies in the “Continue Testing” region, additional snubbers of that type shall be tested until the point falls in the “Accept” region, or all the snubbers of that type have been tested. Testing equipment failures during functional testing may invalidate that day’s testing and allow that day’s testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of the equipment failure are retested; or

3) An initial representative sample of 55 snubbers of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where “C” is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an “Accept” line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the “Accept” line, testing of that type of snubber may be terminated. If the point plotted falls above the “Accept” line, testing must continue until the point falls on or below the “Accept” line or all the snubbers of that type have been tested. The representative sample selected for the function test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional but shall not be included in the sample plan, and failure of this functional test shall not be the sole cause for increasing the sample size under the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be

reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

Staff Evaluation

The licensee stated in its letter of April 17, 2000, that PNPP performs preservice and inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements, in accordance with the technical requirements contained in Section 6.4.1 of the PNPP Operational Requirements Manual (ORM). The requirements were originally relocated from Technical Specification 4.7.4 upon PNPP's incorporation of the Improved Technical Specifications (NUREG-1434). The licensee stated that the basic technical requirements within ORM 6.4.1 for examination and testing of snubbers are essentially the same as those within ASME/ANSI OM Part 4 of the OMa-1988 Addenda to the OM-1987 Edition. Additionally, the ORM provides requirements for the inspection of snubbers following transient events and a snubber service life replacement program.

It is noted that Revision 0 of this relief request, which requested similar relief for the snubber activities for PNPP's first 10-year inspection interval, was approved by the NRC in a safety evaluation dated September 7, 1990. Revision 1 of this relief request, which was resubmitted for the start of PNPP's second 10-year inspection interval, was also approved by the NRC in a safety evaluation dated November 22, 1999.

In its safety evaluation of November 22, 1999, the staff concluded the following three methods of snubber functional test, as specified in the PNPP ORM, to be acceptable:

(1) Functionally test 10 percent of a type of snubber with an additional 5 percent tested for each functional testing failure,

(2) Functionally test a sample size and determine sample acceptance or rejection using Figure 6.4.1-1 of the ORM, or

(3) Functionally test a representative sample size and determine sample acceptance or rejection using the equation, $N = 55(1 + C/2)$, as stated in ORM 6.4.1, where "C" is the number of snubbers found to not meet the functional test acceptance criteria, and "N" is the total number of snubbers tested.

Revision 2 of the PNPP relief request IR-023 involves a proposed update of the second ORM testing plan option. The staff reviewed the proposed ORM update and found it to be acceptable for the PNPP's second 10-year interval for the inservice inspection (ISI) program because it is consistent with the industry standard "37 testing sample plan," as specified in "ASME OM Code for Operation and Maintenance of Nuclear Power Plants," 1995 Edition, up to the 1996 Addenda, which is the Code of reference approved in 10 CFR 50.55a(b)(3)(v) (reference: 64 FR 51388, September 22, 1999).

The licensee stated that, upon implementation of the proposed ORM update, it will perform preservice and inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements, in accordance with the technical requirements within Section 6.4.1 of the ORM. At least once per refueling interval, a representative sample of snubbers shall be tested using one of the above three (3) plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The proposed alternative program provides reasonable assurance of demonstrating the

operational readiness of the snubbers and is, at least, equivalent to the requirements of Article IWF-5000. Therefore, the staff finds the proposed alternative program provides an acceptable level of quality and safety.

Conclusion

Based on the information provided by the licensee, the staff concludes that the licensee's proposed alternative provides an adequate justification for an alternative from the requirements of ASME Code 1989 Edition, Section XI, Article IWF-5000 (which references first Addenda to OM-1987, Part 4), with regard to visual examination and functional testing of PNPP snubbers. The staff determined that the proposed alternative use of the PNPP's updated ORM for snubber activities would provide an acceptable level of quality and safety because it is consistent with the ASME OM Code for Operation and Maintenance of Nuclear Power Plants. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for the second 10-year interval of the PNPP ISI program.

Principal Contributor

A. Lee

RELIEF REQUEST No. IR-029, REVISION 2

Identification of Components

ASME Code Class 1 piping welds 4 inches nominal pipe size and greater in examination Category B-J, Section XI.

Code Requirement

ASME Code, Section XI, 1989 Edition, Table IWB-2500-1, Category B-J requires 100 percent surface and volumetric examination of 25 percent of the circumferential butt welds and their intersecting longitudinal welds. In accordance with Category B-J, Note 1 of the Table, the examinations are to include all terminal ends, joints where the seismic and operational load stress levels exceed primary plus secondary stress intensity range of 2.4Sm (i.e., "high stress" location) or a cumulative usage factor U of 0.4 (i.e., "high fatigue" location), all dissimilar metal welds, and additional welds (if necessary) such that the total number of circumferential butt welds equals 25 percent.

Licensee's Requested Relief

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested from selecting the welds in accordance with Category B-J, Note 1, when structural interferences make such selection impractical. The relief is requested for PNPP's second 10-year inspection interval.

Alternate Examination

Welds of the same size and similar configuration, but that are not "high stress" welds, will be examined in place of the obstructed welds to maintain the 25 percent selection requirement.

Licensee's Basis for Relief (As Submitted)

The welds identified in the relief request are "high stress" welds, but examination is impractical as they are in radiation areas and are encased in jet impingement shields. The jet shields are elbow or tee-shaped structural steel enclosures around Reactor Recirculation (RR) System and Main Steam (MS) System piping welds. The smallest of the RR jet shields weighs over 1600 lbs and is assembled with 48 bolts. Each of the MS jet shields weighs over 2240 lbs and is assembled with 180 bolts. The bolting for all of the jet shields is high strength, one time use, bolting that must be torqued to 10,000-16,000 ft-lbs. Disassembly for inspection and re-assembly of these jet shields would be a labor-intensive effort with over 100 man-hours each. General area dose rates for the MS jet shield locations range from 20-50 mr/hr and contact dose rates for the RR piping beneath the RR jet shields range from 200-400 mr/hr. Therefore, removal of any of the jet shields would require significant dose expenditure.

The structural integrity of the piping pressure boundary was demonstrated during construction by meeting the requirements of the ASME Code Section III, and additionally by meeting the requirements of ASME Section XI during preservice inspections. The subject RR and MS welds were examined (prior to installation of the jet shields) in accordance with the appropriate Code requirements; weld techniques and welders were qualified in accordance with Code requirements, and materials were purchased and traced in accordance with the appropriate Code and NRC requirements and guidelines. There were no reportable indications during preservice inspection. Additionally, the MS jet shields were removed in the first inspection interval (at considerable dose and monetary cost), the welds received inservice examinations, and they were found to be free of reportable indications.

The pressure boundary passed the required preservice hydrostatic and first interval inservice pressure tests, and has operated for a total of about 3,220 equivalent full power days from November 1987 through the end of 1999 without leakage indication attributable to the subject welds.

Complete examinations meeting the requirements of the ASME Code Section XI have been performed on similar "high stress" welds within the RR and MS Systems where jet shields are not present or are easily removed, with satisfactory results. These welds are subject to the same operating and environmental conditions as the obstructed welds.

Other RR and MS welds of the same size and configuration, but that are not "high stress" welds will be examined in place of the obstructed welds. In accordance with ASME Research White Paper, "Risk-Based Alternative Selection Process for Inservice Inspection of LWR Nuclear Power Plant Components," (Library of Congress Catalogue Number 94-71660) a recent industry survey, which included 50 nuclear units representing 733 cumulative years of operation, found that there is no apparent relationship between the type of welds selected for inspection (i.e., high design stress/fatigue welds versus low stress/fatigue welds) and the detection of flaws.

Design, procurement and operational provisions against nil ductile failure of the subject welds remain as described in the PNPP USAR.

Revision 1 of this relief request, which requested the same relief for welds of the same Examination Category for PNPP's second 10-year inspection interval, was approved by the NRC (reference TAC No. MA3437, dated 11/22/99).

Revision 2 adds two additional welds to the scope of the relief. In preparation for an upcoming refueling outage, it was identified that the two welds are also within jet shields.

In summary, because of the dose burden, acceptable initial condition, successful Code hydrotest and operating experience without related leakage indications, the satisfactory examination of identical welds, the substitution of welds of similar size and configuration, protection against brittle failure, and the previous approval of the same relief, it is concluded that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Evaluation

The Code details specific selection criteria for welds to be examined. These criteria require examination of certain “high stress” welds. The “high stress” welds identified in the licensee’s table are in radiation areas and are encased in jet impingement shields. The jet shields are elbow or tee-shaped structural steel enclosures around RR and MS system piping welds. The smallest of the RR jet shields weighs over 1600 lbs and is assembled with 48 bolts. Each of the MS jet shields weighs over 2240 lbs and is assembled with 180 bolts. The bolting for all of the jet shields is high strength, one time use bolting that must be torqued to 10k-16k ft-lbs. Disassembly for inspection and re-assembly of these jet shields would be a labor-intensive effort with over 100 man-hours each. General area dose rates for the MS jet shield locations range from 20-50 mr/hr and contact dose rates for the RR piping beneath the RR jet shields range from 200-400 mr/hr. Disassembly and reassembly of the jet shields to allow examination of the subject welds would involve significant radiation doses to workers and would constitute a hardship on the licensee. Therefore, performance of the surface and volumetric examination of the obstructed welds (selected according to Code criteria) would result in a hardship on the licensee.

The licensee proposes to substitute the examinations of welds of the same size and similar configuration that are not “high stress” welds, in place of the obstructed welds required by the Code. The specific welds and their descriptions are included in the table to their relief request. The staff believes that flaws are also likely to be found in other welds of the same size and configuration that are beyond the scope of the code-defined “high stress” welds. The licensee will examine a sample population of welds to maintain the 25 percent selection requirement of the Code. Therefore, the sample size and the proposed examination of similar welds will provide reasonable assurance of detection of degradation in the welds. Inasmuch as the alternative provides reasonable assurance of structural integrity, compliance with the code (involving significant dose expenditure) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Conclusion

The selection of welds of the same size and configuration as an alternative to the stress-based criteria of the Code for examination in the subject systems will provide a reasonable assurance of detection of degradation in the welds when the sample population of welds examined remain the same. Compliance with the applicable 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 second 10-year ISI inspection interval of PNPP.

Principal Contributor

P. Patnaik

RELIEF REQUEST No. IR-044, REVISION 0

Identification of Components

Reactor Pressure Vessel (RPV) Closure Nuts (72 total)

Code Requirement

ASME Code, Section XI, 1989 Edition, Table IWB-2500-1, Examination Category B-G-1, Item B6.10, requires a surface examination of RPV closure nuts.

Relief Requested

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested from performing the Code-required surface examinations.

Alternate Examination

The alternative requirements of Code Case N-627, “VT-1 Visual Examination in lieu of Surface Examination for RPV Closure Nuts”, Section XI, Division 1, will be used. That is, VT-1 visual examinations will be used in lieu of the surface examination required by Table IWB-2500-1, Examination Category B-G-1, Item B6.10.

Licensee’s Basis for Relief (As Submitted)

Use of Code Case N-627 will allow VT-1 visual examinations of the RPV closure nuts as an alternative to performing surface examination. Within the industry, there have been no failures of RPV closure nuts. Furthermore, ASME Section XI Subcommittee determined that for the intended purpose of the RPV closure nut examinations, a VT-1 examination could replace the surface examination. This was incorporated into the 1989 Addenda of ASME Section XI and is unchanged through the current Edition and Addenda. Within 10 CFR 50.55a, the NRC has endorsed the 1995 Edition ASME Code, Section XI including the 1996 Addenda (refer to 64 FR 51395) without any limitations on the use of the Table IWB-2500-1, Category B-G-1 requirements.

Evaluation

The staff has evaluated potential degradation of RPV closure nuts under different loading conditions in an aggressive environment. The primary degradation mechanisms leading to failure of nuts are corrosion, cracking, wear, and thread damage. These degradation mechanisms tend to initiate on the surface of the nut, and therefore, surface examination is required by the ASME Code, Section XI. However, detection of degradation can also be made by VT-1 visual examination in accordance with the ASME Code, Section V. Consequently, the alternative requirement of VT-1 visual examination of RPV closure nuts was first incorporated into the 1989 Addenda and later into subsequent editions of the ASME Code, Section XI. Code

Case N-627, "VT-1 Visual Examination in Lieu of Surface Examination for RPV Closure Nuts" was approved by ASME on May 7, 1999. The staff has evaluated the alternative requirement of VT-1 visual examination in lieu of surface examination of the RPV closure head nuts for PNPP, and has determined that the proposed alternative would provide an acceptable level of quality and safety since VT-1 visual examination will effectively detect any primary degradation mechanism such as corrosion, cracking, or wear in the RPV closure nuts and provide assurance of structural integrity.

Conclusion

The staff concludes that the proposed alternative requirement of VT-1 visual examination in lieu of surface examination of RPV closure head nuts for PNPP, in accordance with Code Case N-627 will provide an acceptable level of quality and safety and, therefore, the alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year inservice inspection interval of PNPP. Use of Code Case N-627 is authorized for the duration of the second 10-year inservice inspection interval at PNPP until such time as the code case is approved on a generic basis by the NRC. At that time, if the licensee intends to continue to implement this code case, the licensee must follow all provisions in Code Case N-627, with limitations as addressed by the NRC, if any.

Principal Contributor

P. Patnaik

RELIEF REQUEST NO. IR-045, REVISION 0

Identification of Components

Reactor Pressure Vessel (RPV) Shell-to-Flange and Head-to-Flange Welds (PNPP Mark Numbers 1B13-AE and 1B13-AG)

Code Requirements

ASME Code, Section XI, 1989 Edition, Table IWB-2500-1, Examination Category B-A, Items B1.30 and B1.40, allow deferral of examinations to the end of the inspection interval only if partial examinations are conducted from the flange face in the first inspection period.

Relief Requested

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested from having to perform partial examinations from the flange face in order to defer the RPV shell-to-flange and head-to-flange welds to the end of the inspection interval. The alternative scheduling is proposed in accordance with the provisions of Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel", Section XI, Division 1.

Alternative Examination

The alternative requirements of Code Case N-623 will be used. The Code Case states that inspection of shell-to-flange and head-to-flange welds of a reactor vessel may be deferred to

the end of the interval without conducting partial exams from the flange face provided the following conditions are met:

- No welded repair/replacement activities have ever been performed on the shell-to-flange or head-to-flange weld.

- Neither the shell-to-flange weld nor head-to-flange weld contains identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420(b).

- The vessel is not in the first inspection interval.

PNPP meets all the above conditions.

Licensee's Basis for Relief (As Submitted)

Background:

The 1975 Edition of ASME Section XI only allowed RPV shell welds to be deferred to the end of the interval. Since that time, the industry has built an extensive experience base with the examination of RPV shell, head-to-shell, flange-to-shell, and nozzle-to-shell welds with no unacceptable examination results. Typically, RPV weld examinations, many of which are performed with automated UT equipment, are the most costly inservice inspection (ISI) examinations from both a financial and a dose perspective. As such, and considering the failure-free industry examination experience, ASME Subcommittee XI has taken progressive steps to allow deferral of different welds. In 1978, ASME Section XI was changed to allow deferral of bottom head welds. In 1988, it was changed to allow deferral of bottom and top head welds. In 1993, Code Case N-521 (endorsed by NRC in RG 1.147) was issued to allow deferral of nozzle-to-shell welds. All of these steps were made to allow licensees to perform the examinations in the most efficient manner (i.e., least expense and dose), which is all at one time at the end of the interval. As the partial flange-to-shell and head-to-flange welds are typically manual examinations with less impact than the other RPV weld examinations, Subcommittee XI did not take steps to allow their deferral until the issuance of Code Case N-623 in 1999.

Hardship:

Elimination of the flange weld partial examination as a condition for deferral will allow the flange welds to be deferred to the end of the interval and examined only once. This would coincide with the requirements for all the remaining RPV shell and head welds. In case of the shell-to-flange weld, the partial examinations are performed from the flange surface following disassembly or prior to re-assembly, while the reactor cavity is drained down. Thus, in the typical outage where critical path is through the refuel floor, these examinations become a critical path activity. Duration to complete the partial shell-to-flange weld examinations is approximately 4 hours for 2 NDE technicians. Dose rates at the RPV flange with the cavity drained range from 50-300 mr/hr. Therefore, elimination of the shell-to-flange weld partial examinations would save approximately 4 hours of critical path time and a cumulative dose of up to 2.4 REM.

Evaluation

The licensee has requested relief from the requirement of the ASME Code, Section XI, 1989 Edition to perform partial examination from the flange face in order to defer the examination of RPV shell-to-flange and head-to-flange welds to the end of the second 10-year inspection interval. A partial examination of each weld normally performed by manual ultrasonics would consume critical path time during an outage and would cause personnel radiation exposure. However, the deferral of head-to-flange and shell-to-flange weld examinations to the end of the interval will result in ALARA benefits to the licensee with performance of one-time mechanized ultrasonic examination of these welds in conjunction with similar examination of the other RPV welds that are also conducted at the end of the inspection interval.

Code Case N-623, approved by ASME on February 26, 1999, allows deferral of these welds to the end of the inspection interval without conducting partial examinations from the flange face under the following conditions:

- No welded repair/replacement activities have ever been performed on the shell-to-flange or head-to-flange weld.

- Neither the shell-to-flange weld nor head-to-flange weld contains identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420(b).

- The vessel is not in the first inspection interval.

The above conditions lead to the conclusion that there are no flaws present in the component which can grow to a critical size within an inspection interval to cause a component failure. Hence, the conditions of Code Case N-623, with regard to deferral of inspections of the subject welds to the end of the inspection interval without conducting a partial examination from the RPV flange face in the first inspection period of the interval, would provide reasonable assurance of structural integrity. Given that the proposed alternative would provide reasonable assurance of structural integrity, the staff has determined that compliance to the Code requirement would result in a hardship or unusual difficulty to the licensee due to increased personnel radiation exposure without a compensating increase in the level of quality and safety.

Conclusion

The staff concludes that the Code requirement to perform partial examination of head-to-flange and shell-to-flange welds of PNPP reactor vessel for deferral of inspections to the end of the inspection interval will result in hardship (increased radiation exposure to personnel) or unusual difficulty without a compensating increase in the level of quality and safety. The alternative provisions of Code Case N-623 provide reasonable assurance of structural integrity. Therefore, use of Code Case N-623 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) at PNPP. Use of Code Case N-623 is authorized for the second 10-year inservice inspection interval at PNPP until such time as the code case is approved on a generic basis by the NRC. At that time, if the licensee intends to continue to implement this code case, the licensee must follow all provisions in Code Case N-623, with limitations as addressed by the NRC, if any.

Principal Contributor

P. Patnaik

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

RELIEF REQUESTS ON ASME CODE, SECTION XI,

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

RELIEF REQUEST NO. IR-030, REVISION 1

INTRODUCTION

Section 50.55a(g)(6)(ii)(A) to Title 10 of the *Code of Federal Regulations* [10 CFR 50.55a(g)(6)(ii)(A)] requires nuclear licensees to augment their inspection programs by implementing once, as part of the inservice inspection interval (ISI) that is in effect on September 8, 1992, examinations of reactor pressure vessel (RPV) shell welds, as specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in the Reactor Vessel," to Table IWB-2500 in Subsection IWB of the 1989 Edition of Section XI, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. However, paragraph (g)(6)(ii)(A)(*5*) to 10 CFR 50.55a [i.e., 10 CFR 50.55a(g)(6)(ii)(A)(*5*)] allows licensees to propose alternatives to the augmented inspection requirements when the licensee determines that it is unable to completely satisfy the augmented inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A), and if the proposed alternatives provide an acceptable level of quality and safety in lieu of complying with the requirements of the rule.

By letter dated April 17, 2000, FirstEnergy Nuclear Operating Company (henceforth FENOC, or the licensee) requested approval of an alternative examination program to the augmented and inservice inspection requirements for circumferential shell welds in the RPV of the PNPP. In their letter of April 17, 2000, FENOC informed the staff that, pursuant to the requirements of 10 CFR 50.55a(g)(6)(ii)(A)(*5*) and the alternative program provisions of 10 CFR 50.55a(a)(3)(i), it was seeking to permanently defer the augmented inspections and all subsequent required inservice inspections of the circumferential shell welds in the PNPP RPV. By letter dated January 31, 2001, FENOC informed the staff that, since it could completely satisfy the augmented inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A), it could not use the provisions of 10 CFR 50.55a(g)(6)(ii)(A)(*5*) as part of the bases for permanently deferring the augmented inspections of the circumferential shell welds in the PNPP RPV. Instead, FENOC clarified that pursuant to 10 CFR 50.55a(a)(3)(i), it would like to use the probabilistic fracture mechanics analysis provided in their letter of April 17, 2000, as an alternative to support permanently deferring the augmented inspections required by 10 CFR 50.55a(g)(6)(ii)(A) for the circumferential shell welds, and permanently deferring all subsequent inservice inspections of the circumferential shell welds as required by 10 CFR 50.55a(g)(4) and ASME Code Section XI Table IWB-2500-1 for Examination Category B-A, Inspection Item B1.11.

Finally, the licensee's letter of January 31, 2001, clarified that Revision 0 to IR-030, granted by the Nuclear Regulatory Commission (NRC) on September 18, 1997, was only for temporary relief whereas Revision 1 to IR-030 is for permanent relief.

EVALUATION

Applicable Requirements

- 10 CFR 50.55a requires that all inservice examinations and system pressure tests conducted during the first 10-year interval and subsequent intervals on ASME Code Class 1, 2, and 3 components must comply with the requirements in the latest edition and addenda of Section XI incorporated by reference in 10 CFR 50.55a(b) on the date twelve months prior to the start of the 10-year interval. For PNPP, the applicable edition of Section XI for the current I0-year ISI interval is the 1980 Edition, as modified through to the Winter 1981 Addenda of the edition.

- Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class 1, 2 and 3 components must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the ASME Code (Section XI) to the extent practical within the limitations of design, geometry and materials of construction of the components.[1] As written in Title 10, *Code of Federal Regulations*, 10 CFR 50.55a(g)(4), in part, requires licensees to invoke inservice inspection requirements of ASME Section XI Table IWB-2500-1 for ASME Code Class 1 components.

- Section 10 CFR 50.55a(g)(6)(ii)(A)(*2*) requires that all licensees augment their reactor vessel examination by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Inspection Item B1.10, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," Table IWB-2500-1 to Section XI . . ."

- 10 CFR 50.55a(g)(6)(ii)(A)(*2*) also requires that the augmented examinations of the reactor pressure shell welds cover essentially 100 percent of the RPV shell welds. Both Examination Category B-A and 10 CFR 50.55a(g)(6)(11)(A)(*2*) define "essentially 100-percent" examination as covering 90 percent or more of the examination volume of each weld. The schedule for implementation of the augmented inspection is dependent upon the number of months remaining in the 10-year ISI interval that was in effect on September 8, 1992.

- Inspection Item Group B1.10 of Table IWB-2500-1 to Section XI of the ASME Code covers ISI requirements for volumetric examinations of RPV circumferential shell welds (Inspection Item B1.11) and longitudinal shell welds (Inspection Item B1.12).

Basis for Licensing Action Request

- 10 CFR 50.55a(a)(3)(i) indicates that licensees may use proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), and (g) to 10 CFR 50.55a when authorized

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1 Except for design and access provisions and preservice inspection requirements.

by the Director of the Office of Nuclear Reactor Regulations, and if the proposed alternative are determined to provide an acceptable level of quality and safety in lieu of actually complying with the requirements.

Applicability of the Request

FENOC has identified that the alternative examination program is applicable to the following RPV shell components:

- Weld 1B13-AA - “Lower Head to Number 1 Shell Ring Circumferential Seam”
- Weld 1B13-AB - “Number 1 Shell Ring to Number 2 Shell Ring Circumferential Seam”
- Weld 1B13-AC - “Number 2 Shell Ring to Number 3 Shell Ring Circumferential Seam”
- Weld 1B13-AD - “Number 3 Shell Ring to Number 4 Shell Ring Circumferential Seam”

Licensing Action Request and Proposed Alternative Program

Pursuant to the alternative provisions of 10 CFR 50.55a(a)(3)(i), FENOC is seeking to use the results of a probabilistic fracture mechanics evaluation as an alternative basis for permanently deferring any further volumetric examinations of PNPP circumferential RPV shell welds, as covered under the scope of the following requirements:

- the augmented volumetric inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A)(*2*)

- the 10-Year Inservice Inspection Interval requirements of 10 CFR 50.55a(g)(4), and hence the ISI requirements of Inspection Item B1.11 of Examination Category B-A in Table IWB-2500-1 to Section XI of the ASME Code.

FENOC’s probabilistic failure analysis has been calculated in accordance with the guidelines of Boiling Water Reactor Vessel and Internals Project (BWRVIP) Topical Report BWRVIP-05, “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” (Reference 1) and the staff’s acceptance criteria stated in the probabilistic fracture mechanics assessment (PFMA) (i.e., in the staff’s SE) on BWRVIP-05 dated July 28, 1998 (Reference 2), and forms the basis for justifying a permanent deferral of the required augmented and inservice volumetric examinations of the circumferential shell welds in the PNPP RPV. This is consistent with the guidelines of Generic Letter 98-05, which was issued on November 10, 1998, and summarized the staff’s position on the contents of relief requests submitted under the BWRVIP-05 guidelines.

Bases for Submitting Requests to Permanently Defer Volumetric Examinations of Circumferential RPV Shell Welds

Topical Report BWRVIP-05 provides the technical basis for permanently deferring the augmented inspections of circumferential welds in the RPV shells of boiling water reactors (BWRs). In the report, the BWRVIP concluded that the probabilities of failure for BWR RPV circumferential shell welds are orders of magnitude lower than that for the longitudinal shell welds. To assess the BWRVIP safety assessment, the NRC conducted an independent risk-informed, PFMA of the analysis presented in the BWRVIP-05 document.[2] In the staff's

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2 The staff's PFMA of BWRVIP-05 is documented in a letter dated July 28, 1998, to Mr. Carl Terry, Chairman of the BWRVIP.

assessment, the staff conservatively calculated the probability that a RPV shell weld would catastrophically fail during the licensed operating term for a BWR nuclear plant. In the assessment, the NRC used the FAVOR Code to perform the PFMA. The staff calculates the final failure probability for a RPV shell weld as the product of frequency for the critical (limiting) transient event and the conditional failure probability for the weld using the limiting conditions from that event.

For the analysis, the staff identified that a cold overpressure event in a foreign reactor was the limiting pressure and temperature event for BWR RPVs. By the staff's calculations, the staff estimated that the probability for the occurrence of the limiting over-pressurization transient was $1X10^{-3}$ per reactor year. The staff then determined the conditional probabilities of failure for longitudinal and circumferential welds in ABB-Combustion Engineering (CE), Chicago Bridge and Iron Works (CB&I), and Babcock and Wilcox (B&W) fabricated vessels using the pressures and temperatures from the limiting event. The conditional failure probabilities for vessel welds were calculated as a function of a nil ductility reference temperature (Mean RT_{ndt} value) for the welds.[3]

Table 2.6-4 of the staff's PFMA identifies the conditional failure probabilities for the bounding reference cases for longitudinal and circumferential welds in CB&I, CE and B&W fabricated vessels. The materials and neutron radiation parameters used by the staff in calculating the conditional probability failures for the reference cases were also identified in Table 2.6-4 of the staff's PFMA. According to Table 2.6-4, B&W fabricated vessels were determined to have the highest conditional probability of failure for circumferentially oriented flaws ($8.17x10^{-5}$ per reactor year). For circumferentially oriented flaws in circumferential shell welds fabricated by CB&I, the conditional probability of failure were somewhat lower ($1.0x10^{-6}$ per reactor year as calculated by the BWRVIP; $2x10^{-7}$ per reactor year as calculated by the NRC). The corresponding mean RT_{ndt} value used to calculate the conditional probability of failure for the CB&I reference case was 44.5 °F. Using this data, the staff calculated the best-estimate failure probability for CB&I fabricated circumferential welds to be $2x10^{-10}$ per reactor year.[4]

The staff considers that when the adjusted reference temperature (RT_{ndt}) value for a RPV shell weld is less than the upper bound RT_{ndt} value for its correspond limiting weld reference case study (as specified in Table 2.6-4 of the PFMA), the shell weld is considered to have less embrittlement than the corresponding weld in the case study, and therefore to have a conditional probability of failure less than or equal to that calculated for the reference case study.

<u>Assessment of FENOC 's Probabilistic Failure Analysis</u>

The PNPP RPV is a Chicago Bridge and Iron Works (CB&I) fabricated vessel. To evaluate FENOC's analysis, the staff confirmed that the fluence factors, chemistry factors, ΔRT_{ndt} values, margin terms, and RT_{ndt} values were calculated in accordance with the guidelines of Regulatory Guide 1.99, Revision 2, and that the copper and nickel contents listed for the circumferential

3 The key parameters In the analysis for calculating the Mean RT_{ndt} values are the initial RT_{ndt} value for the weld, the end- of- license mean neutron fluence, the mean chemistry (percent copper and nickel) of the welds. The methods for calculating the Mean RT_{ndt} values are consistent with the methods in Regulatory Guide 1.99, Revision 2.

4 This value is the product of the conditional probability of failure for the CB&I reference case (2.0E-7 per reactor year) and the estimated frequency for the limiting event (1E-3 per reactor year).

welds were consistent with the values listed in the CEOG Task Report CE-NPSD 1039, Revision 2. In its submittal of April 17, 2000, FENOC evaluated the basis for permanently deferring the required augmented and ISI volumetric inspections of circumferential welds 1B13-AB and 1B13-AC based on FENOC's determination of the peak end-of-license fluence, mean chemistry values, and initial RT_{NDT} values for the weld materials. PNPP design documents indicate that circumferential weld 1B13-AB is 6 inches below the bottom of the active fuel region and that circumferential weld 1B13-AC is 16 inches above the top of the active fuel region. The fast neutron flux at elevations below and above the active fuel region falls off very rapidly. For the purposes of evaluating the welds for the relief request, FENOC determined the peak end-of-license neutron fluence values (E > 1.0 MeV) for the welds by using a conservative extrapolation of the calculated (r,z) and (r,θ) end-of-license neutron fluence distributions for the active fuel region; this is a very conservative practice. FENOC's calculations were performed using the two dimensional code DORT (Ref. 3), and the approximations and the cross sections used in the analytical estimate are those recommended by the staff in the Draft Regulatory Guide DG-1053, and are therefore acceptable. These methods resulted in end-of-license fluence values of 0.19×10^{19} n/cm^2 and 0.29×10^{19} n/cm^2 for probabilistic fracture mechanics evaluations of circumferential welds 1B13-AB and 1B13-AC, respectively. In this case the circumferential welds are considered to be within beltline region of the vessel because the peak end-of-license neutron fluences for the welds are greater than 1.0×10^{17} n/cm^2. In contrast, the peak end-of-license neutron fluence value used for analysis of the limiting circumferential weld in the CB&I reference case is 0.51×10^{19} n/cm^2.

As previously stated, the staff considers that when the adjusted reference temperature (RT_{ndt}) value for a RPV shell weld is less than the upper bound RT_{ndt} value for its corresponding limiting weld reference case study (as specified in Table 2.6-4 of the PFMA), the shell weld is considered to have less embrittlement than the corresponding weld in the case study, and therefore to have a conditional probability of failure less than or equal to that calculated for the reference case study. In this case, the peak end-of-license neutron fluence values for the welds resulted in corresponding reference temperature shifts (ΔRT_{ndt} values) of 22.8 °F and 35.7 °F, and in upper bound reference temperatures (upper bound RT_{ndt} values) of 11.4 °F and 25.6 °F, respectively. In contrast, the corresponding ΔRT_{ndt} and upper bound RT_{ndt} values for the CB&I reference case are 109.5 °F and 100.5 °F, respectively. The attached Table 1 to this safety evaluation further illustrates the comparison of the upper bound RT_{ndt} calculations for circumferential welds 1B13-AB and 1B13-AC to that for circumferential weld evaluated for the CB&I reference case. Since the RT_{ndt} values for the circumferential welds are bounded by the corresponding upper bound RT_{ndt} value for the CB&I reference case, the staff therefore concludes that the conditions probabilities of failure for the circumferential welds in the PNPP RPVs should be less than that calculated by the staff (2.0×10^{-7} per reactor year) for the corresponding CB&I reference case, and that FENOC has provided sufficient assurance that the degree of projected embrittlement of the circumferential welds in the beltline of the PNPP RPV are also bounded by that assessed for the CB&I reference case. Based on this analysis, the staff concludes that the assessment of the circumferential welds in the beltline of the PNPP RPV is consistent with the staff's analysis in SECY-98-219, and that FENOC's alternative probabilistic fracture mechanics program is an acceptable basis to provide assurance of structural integrity for permanently deferring the volumetric examinations of these circumferential RPV shell welds. Thus, the program will provide an acceptable level of quality and safety.

Assessment of FENOC's Operational and Procedural Controls in Support of the FENOC Probabilistic Failure Analysis

In its final safety evaluation (Ref. 2) on Topical Report BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," the staff identified non-design basis events which should have been considered in the BWRVIP-05 evaluation. In particular, the staff concluded that the potential for and consequences of cold over-pressure transients should have been considered in the BWRVIP-05 analysis. FENOC has assessed the systems that could lead to a cold over-pressurization of the PNPP reactor pressure vessel (RPV). These include the reactor core isolation cooling (RCIC), high pressure core spray (HPCS), feedwater, standby liquid control (SLC), control rod drive (CRD), and reactor water cleanup (RWCU) systems.

PNPP has two high pressure make-up systems. The RCIC system is driven by a steam turbine. During cold shutdown conditions, there is no steam available for operation of the system. Therefore, the RCIC cannot contribute to an over-pressurization event during cold shutdown. The other high pressure make-up system is the HPCS system. The HPCS injection valve is closed on reactor vessel high water level. The reactor vessel water level instruments which provide this signal are calibrated for hot, pressurized conditions. During mode 4 operation, these instruments would normally give a false high level signal which would prevent the HPCS injection valve from opening on a HPCS initiation. Therefore, it is unlikely that HPCS initiation would result in an over-pressurization event.

The PNPP feedwater system is comprised of three sets of pumps:

- Four motor-driven reactor feedwater booster pumps
- Two turbine-driven reactor feedwater pumps (RFPTs)
- One motor-driven feedwater pump (MFP)

The discharge pressure of the feedwater booster pumps is too low to contribute to a postulated RPV cold over-pressure transient. The RFPTs are driven by steam turbine, and during cold shutdown conditions, there is no steam available for operation of these pumps. Therefore, the RFPTs cannot contribute to an over-pressurization event during cold shutdown. The primary function of the 20 percent capacity MFP is to serve as an automatic source of feedwater following a loss of an operating RFPT to prevent an RPV low level scram or to prevent the actuation of the RCIC system following the loss of both RFPTs. There are redundant RPV level controls in place that would isolate the reactor and trip the MFP prior to the RPV becoming solid. In addition, several operator errors would have to occur to result in a cold over-pressure transient via the MFP.

There are no automatic starts associated with the SLC system. SLC injection requires operator action to manually start the system from the control room or from the local test station. In addition, in the event of manual initiation during shutdown, the SLC injection rate of approximately 41 gpm would allow operators sufficient time to control reactor pressure.

During normal cold shutdown conditions, RPV level and pressure are normally controlled through a feed and bleed process using the CRD and RWCU systems. Plant procedures are in place to respond to any unanticipated rise in reactor water level. In addition, the CRD system typically injects water into the reactor at a rate of less than 60 gpm, which would allow operators sufficient time to respond, significantly reducing the possibility of an event that would result in violation of pressure-temperature limits.

The CRD system and the RWCU system are also used to control RPV level and pressure during pressure testing of the RPV. During pressurization and performance of an RPV pressure test, the rate of pressure increase is limited to less than 50 psig per minute, and procedural controls are in place to reduce pressure to less than 700 psig if the temperature nears the limits of the temperature vs. pressure curve when at test pressure. These practices minimize the likelihood of exceeding the pressure-temperature limits during the test.

Operators are trained in methods of controlling water level within specified limits, in addition to responding to abnormal water level conditions during shutdown. Procedures and controls for reactor temperature, level, and pressure are in place to minimize the potential for RPV cold over-pressurization events. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification pressure-temperature limits.

On the basis of the evaluation of high pressure injection sources, operator training and established plant-specific procedures, FENOC determined that appropriate controls are in place to minimize the potential for RPV cold over-pressurization events. The information provided regarding the PNPP high pressure injection systems, operator training, and plant-specific procedures provides assurance of RPV weld structural integrity and thus an acceptable level of quality and safety to support approval of the alternative examination request. The staff concludes that a non-design basis cold over-pressure transient is unlikely to occur at PNPP.

CONCLUSIONS

The staff has determined that FENOC has performed acceptable alternative probabilistic fracture mechanics assessments of circumferential welds in the PNPP RPV. The staff has also determined that FENOC's operational and procedural controls provide sufficient assurance that it is unlikely that a non-design basis cold over-pressure transient will occur at PNPP, and that the FENOC's information regarding the PNPP high pressure injection systems, operator training, and plant-specific procedures provide a sufficient basis to support approval of the alternative examination request. With respect to the alternative examination program proposed by FENOC, the staff concludes that the probabilistic failure analysis of the circumferential welds in the PNPP RPV shell, when taken in conjunction with FENOC's operational and procedural controls to prevent over-pressurization events, provides an acceptable level of quality and safety in lieu of actually performing the both required augmented volumetric inspections of the circumferential welds themselves, as required by 10 CFR 50.55a(g)(6)(ii)(A), and all other volumetric examinations required by 10 CFR 50.55a(g)(4), and hence by 10-year ISI Interval requirements of ASME Boiler and Pressure Vessel Code, Section XI, Examination Category B-A, Inspection Item B1.11. The staff therefore concludes that the proposed alternatives to 50.55a(g)(6)(ii)(A) and 50.55a(g)(4) are authorized pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), in that FENOC may permanently defer conducting volumetric examinations of the circumferential welds in the PNPP RPV for the remaining time in the operating license.

REFERENCES

1. EPRI Topical Report No. TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," September 28,1995.

2. Letter from G. C. Lainas, Acting Director, Division of Engineering, Office of Nuclear Reactor Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Power Company, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," July 28, 1998.

3. Radiation Safety Information Computational Center (RSICC) Computer Code Collection (CCC) - 543: "TORT-DORT, Two- and Three-Dimensional Discrete Ordinates Transport, Version 2.8.14," January 1994.

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Table 1

Probabilistic Fracture Mechanics Assessments for Circumferential Shell Welds in the Perry Nuclear Power Plant (PNPP) Reactor Pressure Vessel

Parameter	CB&I Probabilistic Fracture Mechanics Reference Case Criteria (1)	Probabilistic Fracture Mechanics Assessments for Circumferential Shell Weld 1B13-AB (2)		Probabilistic Fracture Mechanics Assessments for Circumferential Shell Weld 1B13-AC (2)	
		NRC Calculation	FENOC Calculation	NRC Calculation	FENOC Calculation
Neutron Fluence (n/cm^2)	5.1×10^{18}	1.9×10^{18}	1.9×10^{18}	2.9×10^{18}	2.9×10^{18}
Initial RT_{NDT} (°F)	-65	-20.0	-20.0	-60.0	-60.0
Chemistry Factor	134.1	41.0	41.0	54.0	54.0
Copper Content (Wt.-%)	0.100	0.030	0.030	0.040	0.040
Nickel Content (Wt.-%)	0.990	0.810	0.810	0.970	0.970
ΔRT_{NDT} (°F)	109.5	22.8	22.8	35.7	35.7
Margin Term (°F)	56.0	22.8	22.8	35.7	35.7
Mean Adjusted Reference Temperature (°F)	44.5	2.8	2.8	-24.3	-24.3
Upper Bound Adjusted Reference Temperature (°F)	100.5	25.6	25.6	11.4	11.4

Notes:
(1) The evaluation criteria listed here are for the Chicago Bridge and Iron Works reference case for circumferential RPV welds, as copied from Table 2.6-4 of the staff's final safety evaluation on Topical Report BWRVIP-05, dated July 28, 1998. These criteria will be used be the staff as the licensing basis for permanently deferring the volumetric examinations of the circumferential welds in the PNPP RPV.
(2) The adjusted reference temperatures calculated by the staff for the PNPP RPV were in agreement with those calculated by FENOC for the RPV. The values calculated by the staff are consistent with the methodology for calculating adjusted reference temperatures in Regulatory Guide 1.99, Revision 2 (May 1988). The values calculated by the staff will be used as the licensing basis for permanently deferring the volumetric examinations of the circumferential welds in the PNPP RPV.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

RELIEF REQUESTS ON ASME CODE, SECTION XI,

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

RELIEF REQUEST No. IR-041, REVISION 0

INTRODUCTION

By letter dated April 17, 2000, the licensee requested relief from some of the ASME Code, Section XI requirements.

Pursuant to 10 CFR 50.55a(b) and (g), inservice inspection of containment must meet the requirements of the 1992 Edition, 1992 Addenda (1992 E&A) of ASME Code, Section XI, Subsections IWE and IWL. Pursuant to 10 CFR 50.55a(g)(6)(ii)(B), the first period containment examinations must be completed by September 9, 2001. Alternatives to the requirements of 10 CFR 50.55a(g) may be authorized under 10 CFR 50.55a(a)(3), if (i) the proposed alternative provides an acceptable level of quality and safety, or (ii) compliance with the specific requirement of the Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The following evaluation addresses the merits of relief request IR-041 related to the documentation requirements associated with repair and replacement requirements of 1992 E&A of Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code (the Code). As a result of discussion with the NRC staff, the licensee revised RR IR-041 in its entirety by letter dated August 11, 2000.

Identification of Components

All Class MC pressure retaining components.

Code Requirements

ASME Section XI, 1992 Edition, 1992 Addenda, Subsections IWE and IWL

Licensee's Requested Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee has requested relief from the documentation requirements of subarticles IWA-4140, "Repair/Replacement Program and Plan," and IWA-4910, "Reports and Records," of the 1992 Edition, 1992 Addenda of ASME Section XI for repairs, replacements and modifications to Class MC (IWE) and CC (IWL) components.

Licensee's Proposed Alternative (As Submitted)

For Class MC(IWE) and CC(IWL) components, PNPP's repair/replacement program will comply with all the requirements of the 1992 Edition, 1992 Addenda of ASME Section XI except the documentation requirements of subarticles IWA-4140 and IWA-4910. In their place the documentation requirements of subarticles IWA-4800 and IWA-7520 of the 1989 Edition will be used for repairs and replacements to the containment vessel.

Basis for Relief

In accordance with 10 CFR 50.55a(g)(4)(ii), PNPP's In-Service Inspection (ISI) and Repair/Replacement (R/R) programs were recently updated to the 1989 Edition of Section XI for the PNPP's second 10-year inspection interval.

To simplify the process, the R/R documentation requirements for all components within the scope of Section XI were updated to the requirements of the 1989 Edition with no Addenda. The documentation requirements are given in subarticles IWA-4800, "Records" (for Repairs), and IWA-7520, "Reports and Records" (for Replacements). They are essentially the same as those from the 1983 Edition, Summer 1984 Addenda, which was the reference code for PNPP's previous inspection interval, and they did not require the generation of any new procedures or record types. However, the documentation requirements in the 1992 Edition, 1992 Addenda are different. They are given in subarticles IWA-4140, "Repair/Replacement Program and Plan," which describes activities that are to be documented in a formal R/R Plan, and in IWA-4910, "Reports and Records." There was a major restructuring of the R/R rules by the 1992 Edition, 1992 Addenda, with R/Rs all consolidated under IWA-4000. Most of the changes in the 1992 Edition, 1992 Addenda were clarifications of the previous requirements, not new requirements. An exception was a new requirement within subarticle IWA-4140 to have a formal R/R Plan for each R/R activity. R/R Plans increase the documentation requirements associated with R/R activities. Also, Subarticle IWA-4910 added a list of "as applicable" R/R record requirements, rather than simply providing reference to IWA-6000, "Records and Reports," as IWA-4800 did in the 1989 Edition.

Note that PNPP holds a National Board Nuclear Repair (NR) Certificate of Authorization and PNPP's R/R program currently meets the intent of the IWA-4140 and 4910 requirements through the NR Manual and various supporting instructions. However, the requirements are documented in various quality records (e.g., Corrective Action documents, Design Change Packages, etc.) rather than being documented and/or summarized in a formal R/R Plan. If applied to the IWE and IWL components, the IWA-4140 and IWA-4910 requirements would require the revision and/or development of new procedures and record types. As R/R Plans are not required by subarticles IWA-4800 or IWA-7520 of the 1989 Edition, such plans are not required for IWB, IWC, IWD, and IWF examinations. Therefore, for consistency, PNPP proposes to use the documentation requirements of IWA-4800 and IWA-7520 of the 1989 Edition for all components that are within the scope of Section XI until PNPP updates to a later version of the code in accordance with 10 CFR 50.55a(g)(4)(ii). In this manner, the documentation requirements for Class MC (IWE) and CC (IWL) components will be no different than those for Class 1, 2, and 3 components. It will eliminate the need to create new R/R Plan records and the administrative burden associated with the processing and maintenance of those records.

Based on the previous discussion, relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). Compliance with the repair and replacement documentation requirements of the 1989 Edition will provide an acceptable level of quality and safety.

Staff Evaluation

By letter dated April 17, 2000, the licensee requested relief from the requirements of IWA-4140 of the 1992 E&A of the Code pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that incorporating these requirements in the PNPP's repair and replacement (R/R) procedures would result in hardship without a compensating increase in the level of quality and safety. Based on subsequent discussions with the staff, the licensee's letter of August 11, 2000, revised its request to seek instead approval of an alternative under 10 CFR 50.55a(a)(3)(i), and provided a table of comparison between the requirements of IWA 4140 and IWA 4190 of the 1992 E&A of the Code and those of the PNPP's procedures based on the 1989 Edition of the Code.

The table provided by the licensee in its letter of August 11, 2000, is attached as Table 1 to this evaluation. Table 1 shows that the significant requirements of Subarticles IWA-4140 and IWA-4190 of the 1992 E&A are covered by PNPP's program and procedures. Moreover, the licensee emphasizes: "The PNPP holds a National Board Nuclear Repair (NR) Certificate of Authorization for repairs and replacements to Class 1, 2, 3, and MC components. This Certificate of Authorization is based on PNPP's repair and replacement program, which is written to the requirements of the 1989 Edition with no Addenda of ASME Section XI for PNPP's second 10-year inspection interval." Because PNPP has a certified program that addresses significant requirements in the Code, the proposed alternative provides assurance that repaired and replaced components will be controlled by a program and procedures that ensure component integrity. Therefore, the staff finds that the licensee's proposed alternative will provide an acceptable level of quality and safety.

Conclusion

Based on its review of Table 1, and the fact that the licensee holds NR certification for repair and replacement activities, the staff concludes that the licensee's documentation requirements will provide assurance of an acceptable level of quality and safety for Class MC and Class CC components of the PNPP containment. Therefore, the staff authorizes the proposed alternative, pursuant to 10 CFR 50.55a(a)(3)(i), for the second 10-year ISI inspection interval at PNPP.

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The following table provides a comparison of the requirements in IWA-4140 and IWA-4910 (1992 Edition & Addenda) against the equivalent requirements in PNPP's Repair/Replacement (R/R) program, the National Board Nuclear Repair (NR) Manual and supporting procedures.

1992 Edition with 1992 Addenda Requirements	Perry Nuclear Power Plant (PNPP) Program/Procedure(s) that Meet the Intent of the 1992 Edition with 1992 Addenda Requirements
<u>IWA-4140 Comparison</u> (a) Repairs and replacement of items shall be completed in accordance with the Repair/Replacement (R/R) Program. The program is a document or set of documents that defines the managerial and administrative control for completion of repairs or replacement of items.	NR Manual statement of policy states in part that PNPP recognizes the need for a formal and comprehensive Quality Assurance (QA) Program for R/Rs, and modification of nuclear code-stamped items in accordance with the requirements of American Society of Mechanical Engineers (ASME), Section XI, National Board Inspection Code (NBIC), RA-2300, and 10CFR50, for the "NR" Certificate of Authorization.
(b) As part of the R/R Program, repairs and replacement of items shall be made in accordance with R/R Plans that include the essential requirements for completion of the repair or replacement. A R/R Plan shall identify the following:	R/Rs are made in accordance with the NR Manual and the supporting site procedures and instructions.
(1) Applicable code edition, addenda, and code cases of Section XI;	Plant Administrative Procedure (PAP) 1001, "Inservice Examination Program," Section 6.1, "Compliance Requirements," details the code edition, addenda and code cases of Section XI that are used for Section XI activities. Accordingly, they are documented within the applicable Non-Conforming Condition (NCC) Condition Report or Design Change documents.
(2) Construction code edition, addenda, and code cases used to construct the item being repaired or replaced;	Nuclear Engineering Instruction (NEI) 0356, "ASME Design Documents For Pressure-Retaining Equipment, Vessels, And Piping" addresses construction code requirements. It is written to the 1974 edition of the ASME Boiler and Pressure Vessel (B&PV) Code, however, design requirements of specific components may be to a different code of record. In those instances, equivalent requirements based on the component code of record shall be used.

1992 Edition with 1992 Addenda Requirements	Perry Nuclear Power Plant (PNPP) Program/Procedure(s) that Meet the Intent of the 1992 Edition with 1992 Addenda Requirements
IWA-4140 Comparison (3) Construction code edition, addenda, and code cases applicable to the repair or replacement;	Same (1) and (2) on previous page.
(4) For a repair, description of the flaw and the nondestructive examination method used to detect the flaw;	The IWA-4140 requirements are the same as IWA-4130 (a) (1), 1989 Edition, which is the code of reference for our current program. PAP-1608, "Condition Report Process," Attachment 12, requires ASME failures to be evaluated per IWA-4130. Also, see (6) below for control of all ASME work.
(5) For a repair, the flaw removal method, method of measurement of the cavity created by removing the flaw, and requirements for reference points during and after the repair.	The IWA-4140 requirements are the same as IWA-4130 (a) (2), 1989 Edition, which is the code of reference for our current program. Also, see (4) above.
(6) Description of the work to be performed on the item;	NR Manual, Section 9, states in part that the Work Order (WO) is the controlling document for all R/Rs in accordance with the NR Manual (a Warehouse Job Ticket may be used in place of a WO for repairs/replacements on work performed outside of operating plant areas). The WO shall contain the following: Weld History Record, Weld Process Sheets, drawings, procedures, instructions and test procedures. The Authorized Nuclear Inservice Inspector (ANII) reviews the WO prior to ASME work and after the completion of ASME work.
(7) Applicable weld procedure, heat treatment, nondestructive examination, tests, and material requirements;	The IWA-4140 requirements are the same as IWA-4130 (a) (3), 1989 Edition, which is the code of reference for our current program. Also, see (4) and (6) above. Section 7 of the NR Manual describes the controls established to assure that purchased materials and items are in compliance with the requirements of the code and Purchase Order.

1992 Edition with 1992 Addenda Requirements	Perry Nuclear Power Plant (PNPP) Program/Procedure(s) that Meet the Intent of the 1992 Edition with 1992 Addenda Requirements
IWA-4140 Comparison (8) Applicable examination, test, and acceptance criteria to be used to verify acceptability;	NR Manual, Section 11, contains the Test Control requirements. This section requires that all testing activities shall be performed in accordance with written instructions which address acceptance criteria, including those specified in the code, design documents and procurement documents. Non Destructive Examination (NDE) requirements are listed on the Weld History Record, and the NDE acceptance criteria are listed in the associated Nuclear Quality Instruction (NQI). Pressure testing requirements following a R/R are identified in Technical Administrative Instruction (TAI) 1106-2, “Section XI Pressure Testing Program – Repair And Replacement.”
(9) Intended life of the repair or the item to be used for replacement when less than the remainder of the design life of the item;	This is included as part of IWA-4130 (a) (4), 1989 Edition, which is the code of reference for our current program. Also, see (4) on previous page.
(10) For replacement, whether application of the ASME code symbol stamp is required in accordance with IWA-4920;	No comparison necessary as IWA-4920 neither requires nor prohibits stamping for installation.
(11) Documentation in accordance with IWA-4900 and IWA-6000.	This is met in part by IWA-6000, and IWA-7520 (1989 Edition) of our current program. The evaluation report for repair would be documented on an NCC Condition Report per PAP-1608 or a Design Change document. Form NIS-2 is addressed in Section 13 of the NR Manual and also by NQI-1741, “Preparing And Processing NIS-2/NR-1 And Form R Reports.”
(c) The Repair Replacement Program, plans, and evaluations required by IWA-4150 shall be subject to review by enforcement and regulatory authorities having jurisdiction at the plant site.	No comparison required as paragraph does not contain any requirements.

1992 Edition with 1992 Addenda Requirements	Perry Nuclear Power Plant (PNPP) Program/Procedure(s) that Meet the Intent of the 1992 Edition with 1992 Addenda Requirements
<u>IWA-4910 Comparison</u> (c) The reports and records required by IWA-4130 and IWA-6000 shall be completed for all repairs and replacements.	See (11) on previous page.
(b) The following reports and records shall, to the extent required by the construction code and this Article, be maintained by the Owner, as applicable: (1) Certified Design Specification (2) Certified Design Report (3) Design Report (4) Overpressure Protection Report (5) Manufacturers Data Report (6) Material Certification (7) Evaluation Report required by IWA-4150	These IWA-4910 requirements are the same as IWA-7520 (a), items 1 through 7, 1989 Edition, which is the code of reference for our current program. The identified records are identified as Quality Assurance records within procedures PAP-0231, "Configuration Management Program," or PAP-0309, "Configuration Change Processes," and their supporting instructions. As such, they are maintained in accordance with PAP-1701, "Records Management Program," which meets Quality Assurance record keeping requirements of 10CFR50, Appendix B.
(c) Revisions to existing reports, records, and specifications may be shown as an amendment, or as a supplement, and attached to the original record or report to provide an up to date record of the repair or replacement.	This IWA-4910 requirement is the same as IWA-7520 (b), 1989 Edition, which is the code of reference for our current program. The documents discussed are safety related documents for which revision controls are in place under PNPP's 10CFR50, Appendix B, Quality Assurance and Configuration Management Control programs.
(d) Form NIS-2 shall be completed for all replacements	This IWA-4910 requirement is the same as IWA-7520 (a) (8), 1989 Edition, which is the code of reference for our current program. Section 13 of the NR Manual requires the completion of NR-1 and NIS-2 reports. Details for preparing the NIS-2/NR-1 reports are given in NQI-1741.