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July 26, 1999

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555-0001

Subject: Request for Approval of Alternate to ASME Code Requirements

Pursuant to 10 CFR 50.55a(a)(3), Consolidated Edison Company of New York, Inc. (Con Edison) hereby submits a request for approval of two (2) alternatives to the ASME Boiler & Pressure Vessel Code Section XI, Subsection IWE requirements for performing containment examinations. The proposed alternatives are contained in the Attachment, summarized as follows:

1. Relief Request No. 43 proposes an alternative to the ASME B&PV Code, Section XI, Subsection IWE requirement to perform a visual VT-3 examination of seals and gaskets on electrical penetrations, airlocks, and hatches. The proposed alternative is based upon the existing requirements for leakage testing per 10 CFR 50, Appendix J.

2. Relief Request No. 48 proposes an alternative to the ASME B&PV Code, Section XI, Subsection IWE requirement to perform successive examination of flaws, degradation, or repairs following the determination that a component is acceptable for continued service. The proposed alternative is to conduct repair and augmented examination activities in accordance with IWA-4000.

These basic relief requests were developed by EPRI and were submitted to the NRC for use at Davis-Besse. NRC approval was documented in NRC Letter to Davis- Besse dated June 30, 1998, TAC # MA0414.

These proposed alternatives if approved, would be utilized while conducting the necessary containment inspections during the 2000 refueling outage, currently scheduled to commence April 2000. NRC authorization of the proposed ASME

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Code alternatives is requested by December 31, 1999. No new regulatory commitments are being made by Con Edison in this correspondence.

Should you or your staff have any questions regarding this matter, please contact Mr. John McCann, Manager, Nuclear Safety & Licensing.

Very truly yours,

Attachments

C: Mr. Hubert J. Miller
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ATTACHMENT

Relief Request Nos. 43 and 48

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
July 1999

RELIEF REQUEST NUMBER 43

COMPONENT IDENTIFICATION

Code Class:	MC
References:	IWE-2500 Table 1
Examination Category:	E-D
Item Number:	E5.10 & E5.20
Description:	Alternate Examination for Seals and Gaskets of Pressure Retaining Components

CODE REQUIREMENT

IWE-2500, Table IWE-2500-1 requires seals and gaskets on airlocks, hatches, and other devices to be visually examined VT-3 once each interval to assure containment leak tight integrity.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements of this section would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Testing of seals and gaskets in accordance with 10 CFR 50 Appendix J will provide adequate assurance of the leak-tight integrity of the seals and gaskets. In addition to the conditions cited below, the penetrations are pressurized and monitored by the Weld Channel Penetration Pressurization System when the reactor is above cold shutdown.

The penetrations discussed below contain seals and gaskets:

Electrical Penetrations

The electrical penetration system consists of 60 electrical penetrations. Fifty-nine of these penetrations are manufactured by three vendors and are described below, while the sixtieth penetration is a spare sleeve that does not include seals.

The Crouse-Hinds and Westinghouse types are identical in design. The design of this type of electrical penetration utilizes a single canister that is sealed at both ends by a combination of metal and ceramic seals. Epoxy layers on both ends provide a physical support for the conductors within the penetration canister. All of the Westinghouse and Crouse-Hinds penetrations are welded to the sleeve inside containment. The entire canister is continuously pressurized by the Weld Channel Penetration Pressurization System and monitored for any leakage when the reactor is above cold shutdown.

RELIEF REQUEST NUMBER 43

The Conax penetrations are of a modular design consisting of a stainless steel header and 18 independently mounted conductor feed through modules. The header plate and the individual feed through modules are the pressure retaining boundary. This type of penetration does not have a sealed canister. The conductor modules are threaded into the header plate and the header plate is welded to the sleeve, which goes through the containment wall. Leakage monitoring of the Conax penetrations is accomplished by interconnecting ports, machined in the header plate, to each conductor feed through module. A small hole is provided on each conductor feed through module, stainless steel tubular housing, to allow the feed through module to be pressurized when the header plate is pressurized. Metal compression fittings (swagelok type) are used for mounting the conductor feed through modules to the header plate in a double seal manner. The individual conductors passing through the feed through modules are surrounded by polysulfone and are sealed (swaged) at each end of the feed through housing. Weld channel rings are used to create a double weld seal between the header plate and the containment sleeve. All weld joints necessary to maintain containment integrity are continuously monitored for leaks by the Weld Channel Penetration Pressurization System when the reactor is above cold shutdown.

These seals and gaskets cannot be inspected without disassembly of the penetration to gain access to the seals and gaskets. These seals and gaskets are pressurized and monitored for leakage during plant operation. To disassemble a satisfactory seal or gasket for examination provides no increase in quality or safety.

The Weld Channel Penetration Pressurization System provides pressurized air or nitrogen to the penetrations such that if a leak were to develop, a release would not occur since each penetration is double seal welded and pressurized to maintain a positive pressure which is higher than the anticipated containment accident pressure. This system also serves the spare sleeve.

Containment Equipment and Personnel Hatches

The equipment hatch is fabricated from welded steel and furnished with a double-gasketed flange and a bolted, dished door. Provisions are made to continuously pressurize the space between the double-gaskets of the door flanges and the weld channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double-gasket space prior to opening the door.

The personnel hatches are double doors with double-gaskets, mechanically latched, welded assemblies. The personnel hatch doors are interlocked to prevent both from being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. These hatches also contain other gaskets and seals such as handwheel shaft seals, electrical penetrations, blank flanges, and equalizing pressure connections. Disassembly is required to gain access to these gaskets and seals.

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The equipment hatch and personnel hatches are included in the preventive maintenance program and receive scheduled maintenance every 24 months or each refueling outage. The maintenance includes visual examination of the door sealing and gasketed surfaces.

Seals and gaskets receive a 10 CFR 50 Appendix J, Type B test. As noted in 10 CFR 50 Appendix J, the purpose of Type B testing is to measure leakage of containment or penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. Examination of seals and gaskets requires that joints, which are proven adequate through Appendix J testing, be disassembled. For electrical penetrations, this would involve a pre-maintenance Appendix J test, de-termination of cables at electrical penetrations if enough cable slack is not available, disassembly of the joint, removal and examination of the seals and gaskets, re-assembly of the joint, re-termination of the cables if necessary, post maintenance testing of the cables, and a post maintenance Appendix J test of the penetration. The work required for the containment hatches would be similar except for the de-termination, re-termination, and testing of cables. This imposes the risk that equipment could be damaged. The 1992 Edition, 1993 Addenda, of ASME Section XI recognizes that disassembly of joints to perform these examinations is not warranted. Note 1 in Examination Category E-D was modified in the 1995 Edition of ASME Section XI to state that sealed or gasket connections need not be disassembled solely for performance of examinations. However, without disassembly, most of the surface of the seals and gaskets would be inaccessible.

For those penetrations that are routinely disassembled, a Type B test is required upon final assembly and prior to start-up. Since the Type B test will assure the leak tight integrity of primary containment, the performance of the visual examination would not increase the level of safety or quality.

Seals and gaskets are not part of the containment pressure boundary under current Code rules (NE-1220 {b}). When the airlocks and hatches containing these materials are tested in accordance with 10 CFR 50, Appendix J, degradation of the seal or gasket material would be revealed by an increase in the leakage rate. Corrective measures would be applied and the component re-tested. Repair or replacement of seals and gaskets is not subject to Code (1992 Edition, 1992 Addenda) rules in accordance with Paragraph IWA-4111(b)(5) of ASME Section XI.

The Weld Channel Penetration Pressurization System provides a means of determining leak tightness of containment, including access airlock seals on a continual basis when the plant is above cold shutdown conditions.

The visual examination of seals and gaskets in accordance with IWE-2500, Table IWE-2500-1 is a burden without any compensating increase in the level of safety or quality.

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The requirement to examine seals and gaskets has been removed in the rewrite of Subsection IWE of ASME Section XI, 1998 Edition.

EPRI proposed alternate #1 was approved by the NRC for use at Davis-Besse (Reference NRC Letter to Davis-Besse dated June 30,1998 TAC # MA0414) as RR-E1.

PROPOSED ALTERNATIVE PROVISIONS

The leak-tightness of seals and gaskets will be tested in accordance with 10 CFR 50, Appendix J. The 10 CFR 50, Appendix J, Type B testing is performed at least once each inspection interval.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the Third Inspection Interval, July 1, 1994 through June 30, 2004. The interval has been extended to May 18, 2005 as discussed in Con Edison to USNRC letter dated April 9, 1999.

JUSTIFICATION FOR RELIEF

The Weld Channel Penetration Pressurization System is within the scope of the Maintenance Rule and monitors system leakage on a continual basis when the reactor is above cold shutdown.

The penetration seals and gaskets, while not part of the containment pressure boundary, are functionally included in the Appendix J testing. The functionality of the containment, penetration seals and gaskets, (including those of electrical penetrations) is verified during the Type B testing as required by 10 CFR 50, Appendix J.

The ASME committee has recognized the problems associated with the examination of seals and gaskets, and this requirement has been removed as shown in Subsection IWE of ASME Section XI, 1998 Edition.

EPRI proposed alternate #1 was previously approved by the NRC for use at Davis-Besse (Reference NRC Letter to Davis-Besse dated June 30, 1998, TAC # MA0414).

RELIEF REQUEST NUMBER 48

COMPONENT IDENTIFICATION

Code Class:	MC
References:	IWE-2420(b) & (c)
Examination Category:	NA
Item Number:	NA
Description:	Successive Examination Following Repair

CODE REQUIREMENT

Paragraphs IWE-2420(b) and IWE-2420(c) of the 1992 Edition, 1992 Addenda of ASME Section XI requires that when component examination results require evaluation of flaws, areas of degradation, or repairs in accordance with Article IWE-3000, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period listed in the schedule of the inspection program of Paragraph IWE-2411 or Paragraph IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C.

BASIS FOR RELIEF

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii). Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

10 CFR 50.55a was amended in the Federal Register (61FR41303) to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. The purpose of a repair is to restore the component to an acceptable condition for continued service in accordance with the acceptance standards of Article IWE-3000. Paragraph IWA-4150 requires the owner to conduct an evaluation of the suitability of the repair including consideration of the cause of failure.

If the repair has restored the component to an acceptable condition, successive examinations are not warranted. If the repair was not suitable, then the repair does not meet code requirements and the component is not acceptable for continued service. Neither Paragraph IWB-2420(b), Paragraph IWC-2420(b), nor Paragraph IWD-2420(b) requires a repair to be subject to successive examination requirements. Furthermore, if the repair area is subject to accelerated degradation, it would still require augmented examination in accordance with Table IWE-2500-1, Examination Category E-C. The successive examination of repairs in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) constitutes a burden without a compensating increase in quality or safety.

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Repair is not included in Paragraphs IWE-2420(b) and IWE-2420(c) in the 1997 Addenda of the Section XI Code. This addenda was been approved by the ASME Main Committee.

In their resolution to public comment #3.3, the NRC stated, "The purpose of IWE-2420(b) is to manage components found to be acceptable for continued service (meaning no repair or replacement at this time) as an Examination Category E-C component... If the component had been repaired or replaced, then the more frequent examination would not be needed."

EPRI proposed alternate #7 was approved by the NRC for use at Davis-Besse (Reference NRC Letter to Davis-Besse dated June 30, 1998 TAC # MA0414) as RR-E6.

PROPOSED ALTERNATIVE PROVISIONS

Successive examinations in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) are not required for repairs made in accordance with Article IWA-4000.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the Third Inspection Interval, July 1, 1994 thru June 30, 2004. The interval has been extended to May 18, 2005 as discussed in Con Edison to USNRC letter dated April 9, 1999.

JUSTIFICATION FOR RELIEF

1. In SECY 96-080, "Issuance of Final Amendments to 10 CFR Section 50.55a to Incorporate by Reference the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, Division 1, Subsection IWE and Subsection IWL", dated April 17,1996, response to comment # 3.3 states "The purpose of IWE-2420(b) is to manage components found to be acceptable for continued service (meaning no repair or replacement at this time) as an Examination Category E-C component... If the component had been repaired or replaced, then the more frequent examination would not be needed."

2. Repair is not included in Paragraphs IWE-2420(b) and IWE-2420(c) in the 1997 Addenda of the Section XI Code.

3. The NRC approved a similar relief request, RR-E6 for use at Davis-Besse (Reference NRC Letter to Davis-Besse dated June 30, 1998 TAC # MA0414).

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December 21, 1999

Re: Indian Point Unit No. 2
Docket No. 50-247

See Reports

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

Subject: 10 CFR §50.59(b) Report for Indian Point Unit No. 2

Pursuant to 10 CFR §50.59(b)(2), enclosed please find a report of the changes, tests and experiments conducted at Indian Point Unit No. 2 during the period June 22, 1997 to June 21, 1999. The changes set forth in the report represent the changes made to the facility as defined in the Indian Point Unit No. 2 Final Safety Analysis Report pursuant to 10 CFR §50.59(b)(1).

Should you or your staff have any questions regarding this matter please contact Mr. John McCann, Manager, Nuclear Safety and Licensing.

Very Truly Yours,

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PDR ADOCK 05000247

9912610316

cc: Mr. Hubert J. Miller
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