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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE INSERVICE INSPECTION PROGRAM REQUESTS FOR RELIEF

COMMONWEALTH EDISON COMPANY

ZION NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-295 AND 50-304

1.0 INTRODUCTION

The Technical Specifications for Zion Nuclear Power Station, Units 1 and 2, state that the inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the staff, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first ten-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the Zion Nuclear Power Station, Units 1 and 2, second 10-year inservice inspection (ISI) Interval is the 1980 Edition, through winter 1981 Addenda. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

Pursuant to 10 CFR 50.55a(g)(5)(iii), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is impractical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement. The Commission will evaluate determinations under paragraph

10 CFR 50.55a(g)(5) that Code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements that it determines to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed.

In a letter dated January 21, 1994, Commonwealth Edison Company (CECo or the licensee) submitted Volumetric Examination Requests for Relief Nos. IWB-14 and IWC-6.

2.0 EVALUATION

The Code of record for Zion Nuclear Power Station, Units 1 and 2, second 10-year ISI interval is ASME Section XI, 1980 Edition through winter 1981 Addenda. The information provided by the licensee in support of the requests for relief from Code requirements has been evaluated and is documented below.

A. Request for Relief No. IWB-14, Examination Category B-D, Item B3.140, Steam Generator Primary Nozzle Inside Corner Radii

Code Requirement: Table IWB-2500-1, Examination Category B-D, Item B3.140 requires a volumetric examination of all steam generator primary nozzle inside corner radii, as defined by Figure IWB-2500-7(d).

Licensee's Code Request for Relief: The licensee requested relief from performing the Code required volumetric examinations of the steam generator primary nozzle inside corner radii at Zion Nuclear Power Station, Units 1 and 2.

Licensee's Basis for Requesting Relief: The licensee stated:

"Relief is requested from performing volumetric examinations on the steam generator primary nozzle inside corner radii of all four steam generators on the basis that compliance with the Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of plant quality and safety.

Prior to the fall of 1993, it was widely believed in the industry that it was not possible to perform examinations on PWR Steam Generator Primary Nozzle inside corner radii. The steam generator primary nozzles contain an inherent geometric constraint which made the performance of a meaningful volumetric examination using conventional ultrasonic techniques difficult. A lack of symmetry between the inner and outer radii of curvature and the large thickness of the vessel head made it difficult to determine the effective examination angle and to verify that the Code required volume was achieved. In addition, the difficulty in interpreting the UT data due to clad roll, the scattering effect of the cast material on ultrasonic beams, and rough surface finish would reduce the effectiveness of the exam.

The difficulty in performing this examination is well known in the industry. Relief from performing a volumetric exam was granted for Zion Station's first 10-year interval and was requested for the second 10-year interval. Relief was originally requested in relief request IWB-2 and was reviewed by the NRC in the Safety Evaluation Report (SER) dated February 11, 1986. This relief request was granted provided that a volumetric examination was performed to the extent practical.

In fall 1993, Commonwealth Edison contracted Westinghouse to perform three-Dimensional ultrasonic modeling of the steam generator primary nozzles. Three-Dimensional modeling of the nozzles provides understanding of sound beam behavior in the nozzles, assists in determination of the optimal sound beam angle, and provides input for the design of calibration blocks. A procedure was developed from the results of the three-Dimensional modeling effort. In addition, a full scale mock-up with EDM notches the size of the Code allowable flaw that were placed at the boundaries of the examination volume as well as the radii corner was fabricated to validate the procedure technique.

The result of these efforts was successful since it was demonstrated that the examination procedure was able to detect all of the notches in the mockup. The examination was performed on the 1C Steam Generator hot and cold leg nozzles and essentially full examination coverage was achieved and no flaw induced indications were found.

Relief is requested from performing exams on the remaining steam generator primary inlet and outlet nozzles due to the high radiation exposure that will be received by plant personnel. The total radiation exposure to personnel to prepare and examine the primary nozzles of one steam generator was 2R. It was also estimated that an additional 1R of exposure was received to shield and decontaminate the area to facilitate preparation and inspection. It is estimated that a total of 9R of radiation exposure would be expended to decontaminate, shield, build scaffold, remove insulation, prepare the surface, and inspect the remaining nozzles of the other three steam generators. Similar dose rates are expected for Unit 2.

The Steam Generator primary hot leg and cold leg nozzles do not experience thermal stratification or a high thermal gradient during operation and therefore are not highly susceptible to thermally induced fatigue cracking."

<u>Licensee's Proposed Alternative Examination:</u> The licensee proposed to perform ultrasonic examinations on one hot leg and one cold leg primary nozzle and VT-3 visual examinations on the inner radii surfaces from the inside of all four steam generators. The licensee stated:

"Ultrasonic examinations on one steam generator will serve to provide a reasonable sample from which to assess the integrity of the steam generator primary nozzles. The visual exams will ensure that gross flaws are not present in the remaininq nozzle inner radii. Cracking of PWR steam generator primary nozzle inner radii have not been a problem in the industry.

This relief request will apply to Zion Station's Second Ten-Year Interval Program only. For the Third Ten-Year Interval, Zion Station will inspect the Steam Generator Primary Nozzles at the Code required frequency."

Evaluation: The staff reviewed the information provided by the licensee concerning the volumetric examinations of the steam generator primary nozzles. The total radiation exposure to decontaminate, shield, build scaffolding, remove insulation, prepare the surface, and complete the ultrasonic examinations of the nozzles for which the licensee is requesting relief is estimated at 18 Rem. The additional data obtained from these inspections do not provide a compensatory increase in the level of quality and safety to justify the hazards of personnel radiation exposure received to obtain the data.

As an alternative, the licensee has proposed to perform ultrasonic examinations of one cold leg and one hot leg primary nozzle and a VT-3 visual examination of the inner radii surfaces from the inside of all four steam generators.

The staff has concluded that the licensee's proposed alternative examinations provide a reasonable sample from which to assess the integrity of the steam generator primary nozzles and that they will ensure that gross flaws are not present in the nozzle inner radii.

The staff has evaluated the information provided by the licensee in support of its Volumetric Examination Request for Relief No. IWB-14. Based on the information submitted, the staff has concluded that the licensee's alternative examinations are acceptable. The alternative examinations contained in Volumetric Examination Relief Request No. IWB-14 are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) due to the hardship or unusual difficulty that would be encountered without a compensating increase in the level of quality and safety if the licensee performed the Code required volumetric examinations. This relief is authorized provided the licensee performs: 1) the proposed ultrasonic examinations on the hot leg and cold leg primary nozzles of one steam generator per unit and the visual examinations on the inner radii surfaces from the inside of all primary nozzles of all four steam generators on each unit before the end of the second ten-year inspection interval; and 2) the proposed ultrasonic examinations on the hot and cold leg primary nozzles of a different steam generator during the first refueling outage of each unit in the third ten-year inspection interval. Thereafter, the Code required inspection schedule for these nozzles will be followed.

B. Request for Relief No. IWC-6, Examination Category C-A, Item C1.20, Volumetric Examination of Regenerative Heat Exchanger Head to Shell Welds

Code Requirement: Table IWC-2500-1, Examination Category C-A, Item C1.20 requires volumetric examinations of essentially 100% of shell circumferential welds at gross structural discontinuities, the head-to-shell circumferential welds, and shell-to-tube sheet welds in accordance with Figure IWC-2520-1 during each inspection interval. For multiple vessels with similar design, size, and service, the required examinations may be limited to one vessel or distributed among the vessels.

Licensee's Code Relief Request: The licensee requested relief from performing the Code required volumetric examinations of one of the regenerative heat exchanger head-to-shell welds at Zion Nuclear Power Station, Units 1 and 2.

Licensee's Basis for Requesting Relief: The licensee stated:

"Relief is requested from performing examinations on one head to shell weld (welds 2, 4, and 6 as shown in Figure CWE-2-1150) of the Regenerative Heat Exchanger on the basis that compliance with the Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of plant quality and safety.

The Regenerative Heat Exchanger is located in a small cubicle beneath removable blocks inside Containment. The Regenerative Heat Exchanger consists of three vessels connected by 3" NPS piping as shown in Figure CWE-2-1150. The shell side of the heat exchanger contains letdown flow and the tube side of the heat exchanger contains charging flow. The purpose of the heat exchanger is to transfer heat from letdown flow (as an initial measure of cooling prior to demineralization) to charging flow in order to heat up charging flow prior to injection into the Reactor Coolant Loops.

In the case where multiple vessels exist, Section XI allows the examination scope to be limited to one vessel or distributed among the vessels. Liquid penetrant and visual (VT-1) exams were performed on Unit 1 Welds 1, 7, and 8 (as shown in Figure CWE-2-1150) in accordance with relief request IWC-5 which was improved [approved] by the NRC in the SER dated February 11, 1986. No recordable indications were found.

Welds 2, 4, and 6 are similar with respect to material composition, thermal gradient, and flow rates. Welds 1, 3, and 5 are similar to welds 2, 4, and 6 with respect to material composition bt [but] differ from welds 2, 4, and 6 with respect to thermal gradients and flow rates.

Relief is requested from performing examinations on welds 2, 4, or 6 due to the high radiation fields in the area (reference HP survey dated

December 17, 1993). In addition, access to weld 2 is limited by the concrete ceiling above that portion of the vessel and is not exposed when the removable blocks are removed. The close and tight proximity of the components in the area prohibit the rotation of personnel and will result in high radiation exposures to a few individuals. Welds 4 and 6, which are similar to weld 2 with respect to vessel configuration and material composition, are in even higher radiation fields. No indications were found in Unit 1 Weld 1 which is similar to weld 2 with respect to material composition and experiences a higher thermal gradient but lower flow rate than Weld 2 (reference figure 5a-3 for temperature and flow estimates for each shall [shell] and accompanying attachments for assumption and calculations).

In general, the temperature gradients and flow rates across the Regenerative Heat Exchanger nozzles are not extreme. Therefore, the Regenerative Heat Exchanger is not highly susceptible to thermal fatigue. The vessel materials used in the Regenerative Heat Exchanger have good operating histories in PWR water environments.

Shielding is not practical since the high radiation fields would result in high radiation exposure to the shielders resulting in no net savings in radiation exposure. The tight proximity of the vessels and related piping make it difficult to install shielding which adds to the time spent in the area and to the total radiation exposure. In addition, there are no connections available that would enable the station to chemically flush the best exchanger in an effort to lower the dose rates.

The total radiation exposure estimate to build scaffold, remove insulation, prepare welds and perform examinations on any of the affected welds is 4.5R. Due to the restricted access in the cubicle, this exposure will be distributed among very few individuals. Due to the nature of the activities, the potential exists that one or more individuals would receive a dose greater than 1R. Similar doses are expected for Unit 2 based on current radiation surveys.

The normal radiation exposure allowed to site personnel is 300 mR/day. Extensions greater than 300 mR/day may be granted as necessary in unusual circumstances.

Any through wall leak of the heat exchanger will be detected by the RCS leak rate monitors. In the event that leakage is detected, the Regenerative Heat Exchanger could be easily isolated and alternate paths of letdown and charging could easily be established with minimal effect on plant safety.

The data obtained from this inspection does not provide a compensatory increase in quality and safety to justify the hazards of personnel radiation exposure received to obtain the data."

<u>Licensee's Proposed Alternative Examination:</u> The licensee proposed to perform liquid penetrant and VT-1 visual examinations of welds 1, 7, and 8 in accordance with relief request IWC-5 from the SER dated February 11, 1986. In addition, a VT-2 visual examination of the regenerative heat exchanger would be performed. The licensee stated:

"Liquid Penetrant and Visual exams will assure the detection of any surface flaw. VT-2 examinations will be conducted on the Regenerative Heat Exchanger. Regenerative Heat Exchanger cracking has not been a problem in the industry."

<u>Evaluation:</u> In an NRC SE dated February 11, 1986, the NRC granted relief to perform liquid penetrant and visual exams in lieu of volumetric examinations of the shell circumferential welds. The staff has reviewed the recent information provided by the licensee concerning the examinations of the regenerative heat exchanger head-to-shell welds. The total radiation exposure to build scaffolding, remove insulation, prepare the welds, and perform the ultrasonic examinations of the welds on both Units 1 and 2 is estimated at 9 Rem. Due to the inaccessibility of the area, rotation of personnel is very difficult and the exposure would probably be acquired by a very low number of individuals. The potential exists that an individual could receive more than 1 Rem. The additional data obtained from these inspections do not provide a compensatory increase in the level of quality and safety to justify the hazards of personnel radiation exposure received to obtain the data.

As an alternative, the licensee has proposed to perform liquid penetrant and VT-1 visual examinations on welds 1, 7, and 8 and a VT-2 visual examination of the regenerative heat exchangers and has requested relief from performing these examinations on weld 2, 4, or 6.

The staff has concluded that the licensee's proposed alternative examinations provide a reasonable sample from which to assess the integrity of the regenerative heat exchanger welds and that they will ensure that gross flaws are not present in the regenerative heat exchanger.

The staff has evaluated the information provided by the licensee in support of its Request for Relief No. IWC-6. Based on the information submitted, the staff has concluded that the licensee's alternative examinations are acceptable. The alternative examinations contained in Volumetric Examination Relief Request No. IWC-6 are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) due to the hardship or difficulty that would be encountered without a compensating increase in the level of quality and safety if the licensee performed the examinations approved in its February 11, 1986, evaluation of relief request IWC-5. This relief is authorized provided the licensee performs the proposed liquid penetrant and VT-1 visual examinations on welds 1, 7, and 8 and a VT-2 visual examination of the regenerative heat exchangers of both Units.

3.0 CONCLUSIONS

The staff has evaluated the information provided by the licensee in support of its Volumetric Examination Requests for Relief Nos. IWB-14 and IWC-6. Based on the information submitted, the staff has concluded that the licensee's alternative examinations are acceptable. The alternative examinations contained in Volumetric Examination Requests for Relief Nos. IWB-14 and IWC-6 are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) due to the hardship or difficulty that would be encountered without a compensating increase in the level of quality and safety if the licensee performed the Code required volumetric examinations. As noted above, these alternatives are authorized provided the licensee performs the proposed liquid penetrant and VT-1 visual examinations on welds 1, 7, and 8 and a VT-2 visual examination of the regenerative heat exchangers and the ultrasonic examinations on the hot leg and cold leg primary nozzles of one steam generator per Unit and the visual examinations on the inner radii surfaces from the inside of all primary nozzles of all four steam generators on each Unit.

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