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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Response to Request for Additional Information Regarding the Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan

References: (1) Letter from T. J. Tulon (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan," dated January 17, 2003

(2) Letter from T. J. Tulon (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan," dated February 7, 2003

In the referenced letters, Exelon Generation Company, LLC (EGC) requested relief from certain requirements of 10 CFR 50.55a, "Codes and standards," for the fourth interval inservice inspection program plan for Quad Cities Nuclear Power Station, Units 1 and 2. On June 13 and July 9, 2003, the NRC requested additional information related to Relief Requests I4R-01, I4R-02, and I4R-05. The attachment to this letter provides the requested information.

If you have any questions or require additional information, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing
Mid-West Regional Operating Group

Attachments:
1. Response to Request for Additional Information
2. Relief Request Number I4R-05, Revision 1

A047

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station

ATTACHMENT 1
Response to Request for Additional Information

Request 1 – Relief Request I4R-01

Please provide additional information on the type of geometric and material reflectors that prevent a meaningful volumetric examination of the standby liquid control nozzle. Information may be in the form of a mockup, or results from a volumetric examination of a similar construction using the same materials. Discussion should include why a best effort, one-sided examination from the shell side through the ferritic weldments cannot be done or a visual examination, VT-1, of the inner radius from the inside. Please provide previous relief request information on this weld along with NRC evaluation.

Response

Relief request I4R-01 was submitted to the NRC in Reference 5. This relief request includes Figure I4R-01.1, which shows the inside surface geometry of the Standby Liquid Control nozzle. The ultrasonic beam would need to travel through the full thickness of the vessel into a complex cladding/socket configuration. The long ultrasonic metal path and potential for multiple geometric reflectors preclude a meaningful ultrasonic examination. Ultrasonic examination of this configuration will not allow a technique to distinguish the difference between geometrical and flaw type signals.

In addition, a VT-1 examination cannot be performed because there is not sufficient access to the area of interest from the inside of the reactor since the nozzle is located below the core plate.

The NRC's evaluation of this relief for the third ten-year inservice inspection (ISI) interval is documented in Reference 6.

Request 2 – Relief Request I4R-02

Page 4 of 5 discusses two enhancements that were approved by the staff. For enhancement one, please describe in detail the sample expansion and time line for performing additional examinations. Secondly, please provide scenarios specific to your RI-ISI Program that show how your use of Code Case 578-1 is a "more refined methodology for implementing necessary additional examinations" as stated on page 3 of 5.

Page 4 of 5 states that enhancement two was approved by the staff under the third inspection interval. Furthermore, this approval is part of the basis used as providing an acceptable level of quality and safety. In reviewing the referenced safety evaluation, no discussions were found that assessed this enhancement. No references to Note 10 or Table 4-1 were found in the evaluation. Please provide the sections where this is evaluated, or provide the basis for this enhancement.

Response

Enhancement 1, Sample Expansion and Time Line for Performing Additional Examinations

EPRI TR-112657 (i.e., Reference 4) Section 3.6.6.2 states the following regarding additional examinations: "Additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be inspected on the segment or segments initially. If unacceptable flaws or relevant conditions are found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same service related root cause conditions or degradation mechanism."

Exelon Generation Company, LLC (EGC) intends to use the additional examination criterion outlined in Subarticle-2430 of Code Case N-578-1. The Code Case does not deviate from the EPRI TR-112657 regarding additional examinations. Rather, Code Case N-578-1 enhances the EPRI TR by providing additional details. Specifically, for High and Medium Risk category piping structural elements (i.e., Risk Group Categories 1 through 5 as defined in Table I-8 of N-578-1), EGC will use the following criteria.

Examinations performed that reveal flaws or relevant conditions exceeding the applicable acceptance standards shall be extended to include additional examinations. The additional examinations shall include piping structural elements with the same postulated failure mode and the same or higher failure potential.

(1) The number of additional elements shall be the number of piping structural elements with the same postulated failure mode originally scheduled for that fuel cycle.

(2) The scope of the additional examinations may be limited to those high safety significant piping structural elements (i.e., Risk Group Categories 1 through 5) within systems, whose material and service conditions are determined by an evaluation to have the same postulated failure mode as the piping structural element that contained the original flaw or relevant condition.

If the additional required examinations reveal flaws or relevant conditions exceeding the referenced acceptance standards, the examination shall be further extended to include additional examinations.

(1) These examinations shall include all remaining piping elements whose postulated failure modes are the same as the piping structural elements originally examined.

(2) The required additional examinations will be performed during the same outage that the relevant condition was detected.

No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root/probable cause conditions.

For the inspection period following the period in which the original examination discovering the flaw or relevant condition was completed, the examinations shall be performed as originally scheduled.

Enhancement 2, Supplement Requirements Listed in Table 4-1 of EPRI TR-112657

EGC plans to supplement the requirements listed in Table 4-1, "Summary of Degradation - Specific Inspection Requirements and Examination Methods" of EPRI TR-112657 with the provisions listed in Table 1 of Code Case N-578-1 specifically as described below.

(1) The categorization of parts to be examined under the Risk Informed Inservice Inspection (RI-ISI) Program from Table 1 of Code Case N-578-1 (i.e., Examination Category R-A designation and corresponding Item Numbers) will be used in conjunction with Table 4-1 of EPRI TR-112657. Additionally, the provisions of Table 1 of Code Case N-578-1 for Piping Elements Not Subject to a Damage Mechanism (i.e., item numbers, parts examined, examination methods, acceptance standards, examination extent and frequency) will be used. However, the Examination Requirements/Figure No. for Piping Elements Not Subject to a Damage Mechanism will be consistent with the examination requirements that are applicable to those elements subject to Thermal Fatigue as described in Table 4-1 of EPRI TR-112657. The use of Code Case N-578-1 provisions identified in this paragraph does not constitute a deviation from those of EPRI TR-112657 since these provisions are not addressed in Table 4-1 of EPRI TR-112657, Revision B-A.

(2) Code Case N-578-1 Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" provides guidance for the examination method applicable to socket welds and does not deviate from Table 4-1 of EPRI TR-112657. Specifically, N-578-1 allows a VT-2 examination of socket welds to be performed each refuel outage in lieu of a volumetric or surface examination, regardless of the degradation mechanism. The VT-2 examination method is a more meaningful examination method considering the nature of flaw propagation and the socket weld configuration.

The NRC's evaluation of the Quad Cities Nuclear Power Station (QCNPS) Third ISI Interval relief request regarding RI-ISI (i.e., Reference 1) was reviewed and there are no direct references to Note 10 or Table 4-1. However, both Code Case N-578-1 and EPRI TR-112657 are referenced in Reference 1. The Note 10 reference refers to Table 1 "Examination Categories" of Code Case N-578-1 and for the fourth interval, EGC intends to utilize paragraphs and figures from the 1995 Edition with the 1996 Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, which parallel those referenced in the 1989 Edition. The Table 4-1 reference refers to Table 4-1 of EPRI TR-112657 and EGC intends to supplement the table as described above.

Request 3 – Relief Request I4R-02

Have any welds that were selected for inspection in the Risk-Informed Inservice Inspection (RI-ISI) program that was approved by the staff in Reference 1 been removed from the population of welds that will be inspected during the fourth ten-year interval? If so, why was [were] the weld[s] removed from the population of welds to be inspected?

Response

No, currently the RI-ISI Program welds are the same as those selected during the third ten-year ISI interval. Any changes to the welds selected will be subject to the requirements of Reference 4.

Request 4 – Relief Request I4R-02

Have any welds that were not selected for inspection in the RI-ISI program that was approved by the staff in Reference 1 been selected for inspection during the fourth ten-year interval? If so, why was [were] the weld[s] added to the population of welds to be inspected?

Response

As stated above, currently the RI-ISI Program welds are the same as those selected during the third ten-year ISI interval.

Request 5 – Relief Request I4R-02

The relief request includes the following paragraph:

> "The Risk Impact Assessment completed as part of the original baseline RI-ISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RI-ISI methodology. For the Fourth Interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RI-ISI methodology and evaluation will be maintained for the new interval. As such, the initial screening of the risk impact assessment is not a part of the living program process and is not required to be continually updated."

The staff does not concur with the implication that, if there is no change in methodology, the change in risk assessment is not part of the living process. RG 1.178, SRP 3.9.8, and the EPRI Topical report (Refs. 2, 3, and 4) require an evaluation of the change in risk arising from the proposed change in the ISI program. Please provide a discussion on the potential change in risk between the RI-ISI program proposed for implementation in the fourth interval and the ASME Section XI requirements from which relief was granted in Reference 1. If inspections were discontinued or relocated between the third and the fourth interval RI-ISI programs, please provide an estimate of the change in risk.

Response

As described above, there have been no changes to the RI-ISI Program weld selection that would impact the risk assessment. The probabilistic risk assessment (PRA) model for the fourth ten-year interval was identical to the one utilized for the third ten-year interval regarding RI-ISI weld selections. Therefore, there has been no change in risk between the RI-ISI Program proposed for the fourth ten-year interval and the ASME Section XI requirements from which relief was granted during the third ten-year interval in Reference 1.

The QCNPS PRA model was updated in late 2002 and a project is underway to compare the current RI-ISI selection against the updated PRA model. This project is scheduled for completion in 2004 during the first period of the fourth ten-year interval. There are two QCNPS Unit 2 refueling outages (i.e., February 2004 and February 2006) and one QCNPS Unit 1 refueling outage (i.e., March 2005) within the first period. Therefore, any additional weld examinations, resulting from the updated PRA model, can be completed prior to the end of the first period (March 2006).

EGC will continue to maintain the delta risk assessment between the current RI-ISI fourth interval program and the ASME Code Section XI requirements from which relief was granted in Reference 1 consistent with the methodology described in Reference 4.

Request 6 – Relief Request I4R-05

The basis states that the alternative provides an acceptable level of quality and safety, but the majority of the discussion focuses on the difficulty and impracticality of performing the examination by summarizing at the end of your discussion by stating: "It is impractical to purposely fail the inner O-ring in order to perform a test." Please demonstrate why the alternative provides an acceptable level of quality and safety. Secondly, has this configuration been tested in the past, and if so, was it successful or unsuccessful? If it was not tested in the past, has relief previously been granted, and if so, please provide previous relief request information on this weld along with NRC evaluation.

Response

The configuration described in relief request I4R-05 has not been tested in the past. Previous relief was granted for the third ten-year ISI interval in Reference 6. Reference 6 concluded that the system pressure test required by Section XI for the subject Class 2 line is impractical because of the possibility of damage to the O-ring seals. As such, relief for the third ten-year ISI interval was granted in accordance with 10 CFR 50.55a(g)(6)(i). Attachment 2 provides I4R-05, Revision 1, which revises I4R-05 to request relief in accordance with 10 CFR 50.55a(g)(5)(iii).

References

1. Letter from A. J. Mendiola (U. S. Nuclear Regulatory Commission) to O. D. Kingsley (Exelon Generation Company), "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Request for Relief for Risk Informed Inservice Inspection Program Plan - Relief Request CR-33," dated February 5, 2002

2. U. S. Nuclear Regulatory Commission, "An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping," Regulatory Guide 1.178, dated September 1998

3. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Trial Use for the Review of Risk-Informed Inservice Inspection of Piping," NUREG-0800, SRP Chapter 3.9.8, dated May 1998

4. Electric Power Research Institute, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," EPRI TR-112657, Revision B-A, dated January 2000

5. Letter from T. J. Tulon (Exelon Generation Company) to U. S. Nuclear Regulatory Commission, "Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan," dated February 7, 2003

6. Letter from R. A. Capra (U. S. Nuclear Regulatory Commission) to D. L. Farrar (Commonwealth Edison Company), "Evaluation of the Third Ten-Year Interval Inspection Program Plan and Associated Requests for Relief for Quad Cities Nuclear Power Station, Units 1 and 2 (TAC Nos. M85764 and M85765)," dated September 15, 1995

ATTACHMENT 2

Relief Request Number I4R-05, Revision 1

RELIEF REQUEST NUMBER: I4R-05
(Page 1 of 5)

COMPONENT IDENTIFICATION

Code Class:	2
Reference:	Table IWC-2500-1
Examination Category:	C-H
Description:	Exemption From Pressure Testing Reactor Pressure Vessel Head Flange Seal Leak Detection System.
Component Number:	Flange Seal Leak Detection Line Pressure Retaining Components.

CODE REQUIREMENTS

Table IWC-2500-1 requires a Visual VT-2 examination to be performed during a system leakage test.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that compliance with the specified Code requirement has been determined to be impractical.

The Reactor Pressure Vessel Head Flange Leak Detection Line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange (See Figure I4R-05.2). This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in the annunciation of a High Level Alarm in the control room. On this annunciation, control room operators would quantify the leakage rate from the O-ring and then isolate the leak detection line from the drywell sump by closing the AO 1(2)-220-51 valve (see Figure I4R-05.1). This action is taken in order to prevent steam cutting of the O-ring and the vessel flange. Failure of the inner O-ring is the only condition under which this line is pressurized.

The configuration of this system precludes manual testing while the vessel head is removed because the odd configuration of the vessel tap (See I4R-05.2), combined with the small size of the tap and the high test pressure requirement (1000 psig minimum), prevents the tap in the flange from being temporarily plugged. The opening in the flange is only 3/16 of an inch in diameter and is smooth walled making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.

RELIEF REQUEST NUMBER: I4R-05

BASIS FOR RELIEF (Continued)

A pneumatic test performed with the head installed is precluded due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips spaced 15° apart. The retainer clips are contained in a recessed cavity in the top head (see Figure I4R-05.3). If a pressure test was performed with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is only .050" thick with a silver plating thickness of .004" to .006" and could very likely be damaged by this deformation into the recessed areas on the top head.

In addition to the problems associated with the O-ring design that preclude this testing it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber. This would make the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 1000 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 1000 psig.

System leakage testing of this line is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. It is extremely impractical to purposely fail the inner O-ring in order to perform a test.

Based on the above, QCNPS requests relief from the ASME Section XI requirements for system leakage testing of the Reactor Pressure Vessel Head Flange Seal Leak Detection System.

PROPOSED ALTERNATE EXAMINATION

A VT-2 visual examination will be performed on the line during vessel flood-up during a refueling outage. The static head developed due to the water above the vessel flange during flood-up will allow for the detection of any gross indications in the line. This examination will be performed with the frequency specified by Table IWC-2500-1 for a System Leakage Test (once each inspection period).

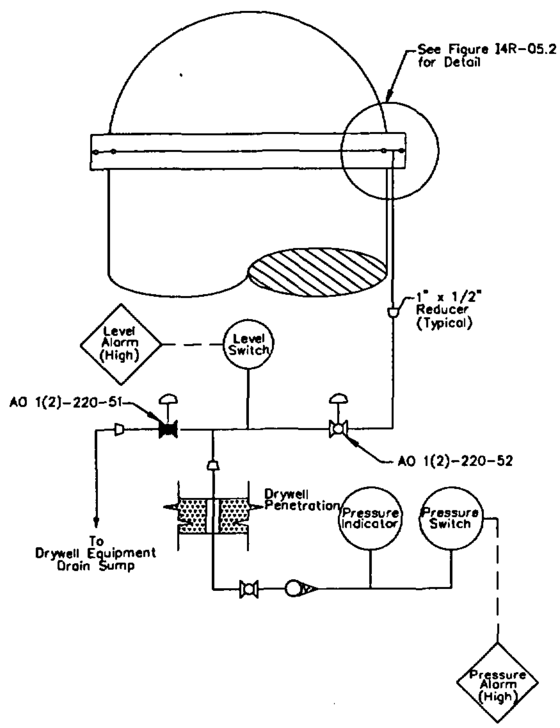
APPLICABLE TIME PERIOD

Relief is requested for the fourth ten-year inspection interval of the Inservice Inspection Program for QCNPS Units 1 and 2.

RELIEF REQUEST NUMBER: I4R-05
(Page 3 of 5)

FIGURE I4R-05.1

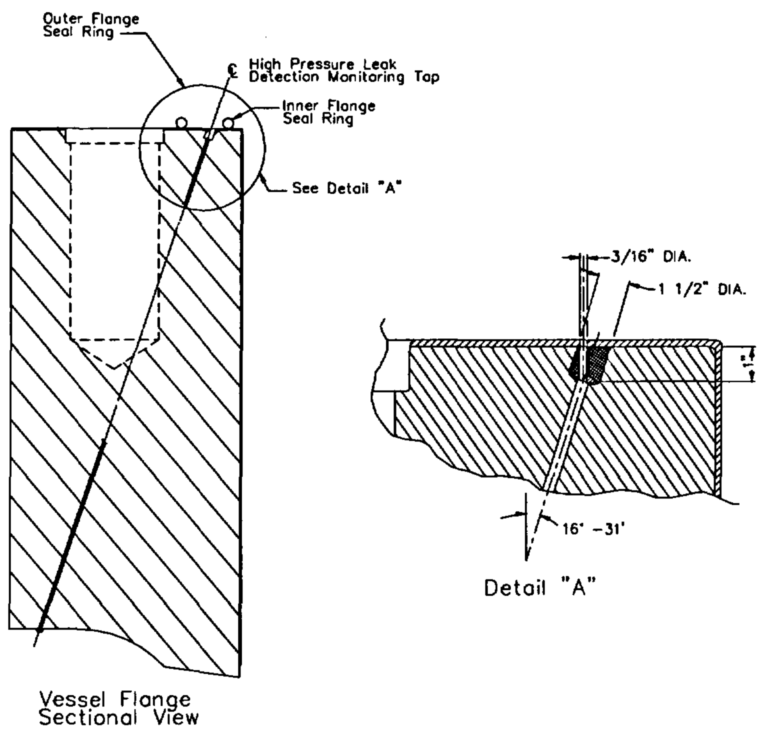
HEAD FLANGE SEAL LEAK DETECTION SCHEMATIC



RELIEF REQUEST NUMBER: I4R-05
(Page 4 of 5)

FIGURE I4R-05.2

FLANGE SEAL LEAK DETECTION LINE DETAIL



RELIEF REQUEST NUMBER: I4R-05

FIGURE I4R-05.3

O-RING CONFIGURATION