UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GPU NUCLEAR CORPORATION

NUREG-0619 ROUTINE INSPECTION CRITERIA

FOR THE FEEDWATER AND CONTROL ROD DRIVE RETURN LINE NOZZLES

OYSTER CREEK NUCLEAR G NERATING STA. N

DOCKET NO. 50-219

1.0 INTRODUCTION

In a letter dated April 8, 1992 (Reference 1), GPU Nuclear Corporation (GPUN/licensee) plans to revise the inspection intervals for these nozzles, as follows:

1. Perform ultrasonic testing (UT) inspections of the Feedwater (FW) and Control Rod Drive Return Line (CRDRL) nozzles once each inservice inspection interval in accordance with the ASME Boiler and Pressure Vessel Code Section XI.

2. Eliminate future NUREG-0619 routine pressure and temperature (PT) examinations. Internal PT examinati ns would only be performed if flaws, which would compromise nozzle integrity, e detected.

3. Reschedule the FW nozzle UT inspection from the 14R refueling outage (scheduled for January 1993) to the 15R refueling outage (scheduled for October 1994)

The NRC staff has completed the review of the licensee's April 8, 1992 letter and reached the conclusion that the licensee's proposal, as described above, is not acceptable as explained in this safety evaluation (SE). During its evaluation, the staff reviewed the correspondence between the NRC and the licensee since 1980 pertaining to this issue. The licensee's plan appears to be in conflict with notarized commitments transmitted by the licensee in letters dated April 9, 1981 (Reference 2) and August 23, 1981 (Reference 3).

2.0 BACKGROUND

The NRC staff published NUREG-0619 (Reference 4) and transmitted the report to licensees and applicants for implementation as the resolution of the generic activity A-10 on BWR feedwater and CRDRL nozzle cracking. The report addressed a variety of potential solutions to the generic problem, which included a combination of the removal of the stainless steel cladding, acceptable thermal sleeve designs, modification of plant systems, operating procedural changes, bypass leakage detection systems and periodic inservice inspections. NUREG-0619 did not contain requirements for a permanent solution but provided interim guidance for plant-specific action.

In the case of the FW nozzles, NUREG-0619 contains a Table 2 "Routine Inspection Intervals" that describes frequencies of UT, visual examinations and routine PT as a function of refue ing cycles or startup/shutdown cycles. This table became the primary reference for implementation of corrective action. Table 2 of NUREG-0619 states that the routine PT inspection is required even if the UT and leak test results are satisfactory. The routine PT was intended to determine whether the thermal sleeve design was effective and to confirm that excessive crack growth has not occurred. For the FW nozzle, the NRC staff intended to limit crack growth to less than 1 inch during 40 years of operation. (Reference 4)

For Oyster Creek Nuclear Generating Station (OCNGS), the licensee responded to NUREG-0619 with References 2, 3 and 5. The existing condition of the OCNGS feedwater nozzles dates from remedial action in the 1977 refueling outage when the stainless steel cladding (nominal thickness 0.219") and a thin layer of base metal (approximately 0.188") was removed from the feedwater nozzle blend radius and bore region. Prior to clad removal, internal dye penetrant examinations detected 54 unacceptable flaw indications as follows:

A nozzle - 36 indications 1/2" - 4" long
B nozzle - 3 indications 1" - 2 1/2" long
C nozzle - 4 indications 2" - 12" long with numerous branches
D nozzle - 11 indications 1" - 12" long

After clad removal, the machined surfaces were dye penetrant inspected again with the following results:

A nozzle - 5 indications 1/2" - 1 1/2" long
B nozzle - No indications
C nozzle - 4 indications 1/2" - 3" long
D nozzle - 3 indications 1/4" - 1" long

These flaws were removed by hand with pencil grinders, blended to a 3:1 slope, polished with flapper wheels and photographed. The deepest grindout, after clad removal, was 7/32" in the D nozzle. The original feedwater sparger was replaced with an improved design that contained a single piston ring seal at the thermal sleeve and a flow baffle at the nozzle face. In the 1977 refueling outage, the CRDRL thermal sleeve was removed and a dye penetrant examination was performed on the inside diameter of the nozzle. No indication of cracking was observed. In 1981, the licensee committed to perform an internal PT of both the feedwater nozzle and the CRDRL nozzle during the 13R outage.

The licensee has a full-scale mockup for both FW and CRDRL nozzles. The ASME Section XI inspection interval ended after the 12R outage (1988). To perform the examinations required by the ASME Code, machined notches were introduced into the mockup ranging in size from 0.030 to .468 inch in depth, with grindout areas to simulate the 1977 repair configuration. GPUN determined that a flaw with a depth of as much as 0.172" can escape detection (Reference 6). During the 12R outage, an automated ultrasonic examination was performed on the welds, inner radii and inner bore of all four feedwater nozzles using the phased-array UT technique. No indications were determined to be reportable. Between the previous feedwater nozzle inspection (7R) and the 12R outage, OCNGS experienced 73 startup/shutdown cycles. The total number of cycles accumulated up to the 12R outage is 157.

OCNGS did not have an on-line leakage monitoring system until a Thermal Transient Monitoring System (TTMS), a modified version of the EPRI "Fatigue-Pro," was installed during the 12R outage. The TTMS must trend the surface temperature of the nozzle over a long period of time. An increasing departure from the trend may be evidence of increased leakage past the thermal sleeve. A leakage increase detection threshold equal to 2.0 gpm is the theoretical sensitivity of the leak detection capability of the TTMS (Reference 7). NUREG-0619 indicates that leakage in excess of about 0.5 gpm can be detrimental to the nozzle.

The NRC staff evaluated the phased array technique, as described in References 6 and 7, and published an SE dated November 14, 1990 (Reference 9). The NRC staff concluded that the licensee's proposed FW and CRDRL nozzle UT inspection was acceptable for the 13R outage provided:

1. Any surface indication detected by the phased array system and not proven to be geometric in nature will require that a liquid penetrant examination be performed that meets the requirements of Section XI.

2. The phased array system should demonstrate the capability to detect thermal fatigue cracks that are 0.172 inch in depth. The demonstration need not be a blind demonstration. As an example, if data is available from other tests (such as PISC II or past inservice inspection examinations), this data could be used to illustrate crack detection capability.

During the 13R outage, the licensee used the phased array system to examine all five nozzles and detected no reportable indications (References 1 and 8).

3.0 STAFF EVALUATION

The NRC staff evaluated information contained in the licensee's letter dated April 8, 1992, and determined that the licensee's proposal to revise the NUREG-0619 inspection criteria is not acceptable. During the review, the staff considered the documents in the References and the summary of information in the Background. The staff considers NUREG-0619 and the SE (Reference 9) as the applicable regulatory criteria. The staff considers the licensee's letters dated April 9, 1981 and August 23, 1981, (References 2 and 3) as the outstanding commitment because the licensee has not demonstrated the size crack that could be detected reliably during the 13R UT examination of the nozzles. The staff will evaluate each of the licensee's plans separately.

Licensee Plan 1:

GPU Nuclear plans to perform UT inspections of the FW and CRDRL nozzles once each inservice inspection interval in accordance with the ASME Boiler and Pressure Vessel Code Section XI.

Staff Evaluation of Plan 1:

1. The inspection frequency depends upon the dimensions of flaws that could exist and the potential for flaw growth during the interval between inspections. The licensee has not demonstrated the size of the flaws that could be detected reliably during the 13R UT.

2. OCNGS has experienced more thermal cycles than defined by Table 2 of NUREG-0619 for a routine PT examination. As the thermal cycles continue to accumulate, the licensee plans to increase the interval between scheduled inspections when compared with Table 2. Normally, a reduction in the interval would be expected to account for increased operation.

3. The licensee's plan does not define the action that will occur as a function of thermal cycles and/or degradation of the piston ring seal.

4. The Technical Specifications and 10 CFR 50.55a(g)(4) already require the ASME Section XI examinations. Therefore, the licensee's plan eliminates the augmented inservice inspection addressed by NUREG-0619.

Licensee Plan 2:

GPU Nuclear plans to eliminate future NUREG-0619 routine PT examinations. Internal PT examinations would only be performed if flaws, which would compromise nozzle integrity, are detected.

Staff Evaluation of Plan 2

1. The objective of the routine PT examination in NUREG-0619 was to determine the effectiveness of the thermal sleeve and to limit the potential flaw growth. The internal PT examination is more sensitive than that performed by the phased array UT system used by the licensee in the 12R and 13R outages.

2. The licensee has not provided documentation to show the actual dimensions of flaws that could exist in each feedwater and CRDRL nozzle.

3. Limiting the internal PT examination to cases that would "compromise nozzle integrity" involves monitoring flaw growth in a reactor vessel. The licensee has not provided documentation to show that a conservative inspection interval, i.e., the ASME Section XI requirement, has been selected considering the combined effects of the thermal sleeve design, the bypass leakage monitoring system, and operating conditions.

Licensee Plan 3

GPU Nuclear plans to reschedule the FW nozzle UT inspection from the 14R refueling outage (scheduled for January 1993) to the 15R refueling outage (scheduled for October 1994).

Staff Evaluation of Plan 3

The licensee's plan 1 and 3 are not consistent with regard to the inspection interval.

4.0 CONCLUSION

NUREG-0619 was transmitted to licensees and applicants for implementation as the resolution of generic activity A-10. The OCNGS had a combination of design, materials of construction, and operating procedures that caused thermal fatigue cracks in all four reactor pressure vessel feedwater inlet nozzles, as described in the Background and References. During the 7R outage (1977), the licensee performed a liquid penetrant examination, removed these cracks by grinding, and installed an improved FW sparger. The overall conditions that caused the thermal fatigue cracking are still present. The remedial action completed in 1977 only changed the frequency and severity of the phenomenon. In 1981, the licensee docketed notarized commitments regarding NUREG-0619 (References 2 and 3).

In a letter dated November 20, 1985, the licensee provided an assessment of its actions to mitigate thermal fatigue cracking. Based on acceptance criteria used during the 1977 repair, small flaws could have been inadvertently left in the FW nozzles. Based on the licensee's fracture mechanics analyses, these assumed flaws could have propagated to a depth in the order of 0.1 inches. To complete the ASME Section XI examinations required by 10 CFR 50.55a(g)(4), the licensee performed an ultrasonic test of all four FW nozzles using the phased array technique during the 12R outage. No flaw indications were reported.

To address the commitments related to the routine PT in the 13R outage, the licensee requested an evaluation of the phased array system. The staff observed a demonstration of the UT technique with machined notches to simulate flaws and published an SE in November 1990 (Reference 9) that included two conditions. During the 13R, the licensee performed an examination on all five nozzles, which detected no reportable indications. The current review determined that the conditions in the 1990 SE have not been completed and the NRC staff considers the examinations performed during the 13R an open issue.

The NRC staff evaluated the licensee's letter dated April 8, 1992, and considered the documents in the References. The NRC staff concluded that the licensee's plan is not acceptable. The licensee's proposal to revert back to the ASME Section XI inspection interval, in effect, eliminates the augmented ISI. The licensee has not presented a technical basis that demonstrates that thermal fatigue has been eliminated in the locations identified in NUREG-0619. The licensee has not documented the theoretical capability of the phased array UT performed during the 12R and 13R outages.

The background information shows that the cracking rate and severity were different for each feedwater nozzle. The licensee has not demonstrated that its installed leakage detection system is capable of identifying bypass leakage in each feedwater nozzle around the entire bottom 180° segment of each pipe. The technical justification submitted by the licensee does not explain the reason that the proposed inspection interval is appropriate for the clad CRDRL nozzle without controls on the number of thermal cycles and the operating parameters.

The NRC staff concludes that the licensee's proposal is not acceptable to eliminate future NUREG-0619 routine PT examinations and perform internal PT examinations only if flaws which would compromise nozzle integrity are detected. The licensee's plan is based on the concept of monitoring flaw growth in the reactor pressure vessel. When NUREG-0619 was implemented, the radiation exposure to perform a routine PT examination was well known from the numerous repairs. When the licensee committed in 1981 to perform a PT examination during the 13R outage, the anticipated surface examination was several factors more sensitive than the phased array UT system that was actually used.

To resolve this issue the NRC staff concludes that the licensee should provide the same type of information submitted in its letter dated November 20, 1985 to justify the substitution of UT for PT examinations. The actual dimensions of the flaws that could exist in each of the nozzles must be determined from the data obtained in the 13R outage. The postulated flaw growth rate should be determined based on the effectiveness of the thermal sleeve and plant operating conditions. The licensee should consider the following factors:

1. The phased array technique is dependent upon the contours of external scanning surface, the internal geometry of the nozzles and the orientation of the thermal fatigue flaws. The actual dimensions of the postulated flaw (sizing) is not the same as the theoretical capability of the method for detection.

2. The actual dimensions of the postulated flaw should be different for each nozzle and each examination zone. If the licensee elects to conservatively envelope the actual dimensions, the selection method should account for the grindouts in the specific FW nozzle

and the cladding in the CRDRL nozzle. The UT data obtained in 13R for those nozzles should demonstrate sufficient sensitivity to detect and size the postulated flaws in the controlling location of the nozzle.

3. The fracture mechanics evaluation to determine a conservative inspection interval should describe the methodology and assumptions as a function of operating conditions.

5.0 REFERENCES

1. GPU Nuclear Corporation letter 5000-92-3089, dated April 8, 1992, Subject: Revision to NUREG-0619 Routine Inspection Criteria for Feedwater and Control Rod Drive Return Line Nozzles.

2. Jersey Central Power & Light Company (JCP&L) letter dated April 9, 1981, Subject: NUREG-0619, NUDOCS Accession Number 8104150160.

3. JCP&L letter dated August 23, 1981, Subject: Control Rod Drive Nozzle Inspection, NUDOCS Accession Number 81083190164.

4. NUREG-0619 "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," November 1980.

5. GPU Nuclear Corporation letter dated May 22, 1981, Subject: NUREG-0619.

6. GPU Nuclear Corporation letter dated July 12, 1990, Subject: Feedwater Nozzle & Control Rod Drive Return Nozzle Examinations.

7. GPU Nuclear Corporation letter dated January 18, 1990, Subject: Feedwater Nozzle Examination.

8. GPU Nuclear Corporation letter dated April 18, 1991, Subject: Control Rod Drive Return Line Nozzle 13R Inspection.

9. NRC Safety Evaluation dated November 14, 1990.

10. GPU Nuclear Corporation letter dated November 20, 1985, Subject: Cycle 11 Refueling Outage Feedwater Nozzle Internal Inspection Deferment

Principal Contributor: Martin Hum

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