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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RESPONSE TO GENERIC LETTER 94-03

OYSTER CREEK NUCLEAR GENERATING STATION

GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

1.0 BACKGROUND

The core shroud in a Boiling Water Reactor (BWR) is a stainless steel cylindrical component within the reactor pressure vessel (RPV) that surrounds the reactor core. The core shroud serves as a partition between feedwater in the reactor vessel's downcomer annulus region and the cooling water flowing up through the reactor core. In addition, the core shroud provides a refloodable volume for safe shutdown cooling and laterally supports the fuel assemblies to maintain control rod insertion geometry during operational transients and accidents.

In 1990, crack indications were observed at core shroud welds located in the beltline region of a foreign BWR. This reactor had completed approximately 190 months of power operation before discovery of the cracks. As a result of this discovery, General Electric Company (GE), the reactor vendor, issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications," on October 3, 1990, to all owners of GE BWRs. The RICSIL summarized the cracking found in the foreign reactor and recommended that at the next refueling outage plants with high-carbon-type 304 stainless steel shrouds perform a visual examination of the accessible areas of the seam welds and associated heat-affected zone (HAZ) on the inside and outside surfaces of the shroud.

Subsequently, a number of domestic BWR licensees performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic inspection experience. Of the inspections performed to date, significant cracking was reported at several plants. The combined industry experience from these plants indicates that both axial and circumferential cracking can occur in the core shrouds of GE designed BWRs.

On July 25, 1994 the NRC issued Generic Letter (GL) 94-03 "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors" to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in their core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

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- inspect their core shrouds no later than the next scheduled refueling outage

- perform a safety analysis supporting continued operation of the facility until the inspections are conducted

- develop an inspection plan which addresses inspections of all shroud welds, and which delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject

- develop plans for evaluation and/or repair of the core shroud

- work closely with the BWROG on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking

GPU Nuclear Corporation (GPUN) responded to GL 94-03 on August 24, 1994 (Reference 1). GPU supplemented its response to GL 94-03 with additional information on September 28, 1994, October 25, 1994, November 1, 1994, and November 3, 1994 (References 2, 3, 4 and 5). Section 2.0 of this Safety Evaluation (SE) gives the staff's assessment of GPUN's response to GL 94-03.

2.0 EVALUATION

GPUN inspected the Oyster Creek Nuclear Generating Station (OCNGS) core shroud during the unit's Fall 1994 refueling outage (RFO), the 15R Outage which commenced on September 9, 1994.

2.1 Susceptibility of the OCNGS Core Shroud

The core shroud cracks which are the subject of GL 94-03, result from intergranular stress corrosion cracking (IGSCC) which is most often associated with sensitized material near the component welds. IGSCC is a time-dependent phenomena requiring a susceptible material, a corrosive environment, and a tensile stress within the material.

Industry experience has shown that austenitic stainless steels with low carbon content are less susceptible to IGSCC than stainless steels with higher carbon content. BWR core shrouds are constructed from either type 304 or 304L stainless steel. Type 304L stainless steel has a lower carbon content that type 304 stainless steel. During the shroud fabrication process when the sections of the core shroud are welded together, the heating of the material adjacent to the weld metal sensitizes the material. Sensitization involves carbon diffusion out of solution forming carbides at grain boundaries upon moderate heating. The formation of carbides at the grain boundaries depletes the chromium in the adjacent material. Since the corrosion resistance of stainless steel is provided by the presence of chromium in the material, the area adjacent to the grain boundary depleted of chromium is thereby susceptible to corrosion. Increased material resistance to IGSCC will result if the carbon content is kept below 0.035%, as specified for type 304L grade material.

Currently available inspection data indicate that shrouds fabricated with forged ring segments are more resistant to IGSCC than rings constructed from welded plate sections. The current understanding for this difference is related to the surface condition resulting from the two shroud fabrication processes. Welded shroud rings are constructed by welding together arcs machined from rolled plate. This process exposes the short transverse direction in the material to the reactor coolant. Elongated grains and stringers in the material exposed to the reactor coolant environment are believed to accelerate the initiation of IGSCC.

Water chemistry also plays an important role in regard to IGSCC susceptibility. Industry experience has shown that plants which have operated with a history of high reactor coolant conductivity have been more susceptible to IGSCC than plants which have operated with lower conductivities[1]. Furthermore, industry experience has shown that reactor coolant systems (RCSs) which have been operated at highly positive, electro-chemical potentials (ECPs) have been more susceptible to IGSCC than RCSs that have been operated at more negative ECPs[2]. The industry has made a considerable effort to improve water chemistry at nuclear facilities over the past 10 years. Industry initiatives have included the introduction of hydrogen water chemistry as a means of lowering ECPs (i.e., making the ECPs more negative) in the RCS. The effectiveness of hydrogen water chemistry in reducing the susceptibility of core shrouds to IGSCC initiation has not been fully evaluated; however, its effectiveness in reducing IGSCC in recirculation system piping has been demonstrated.

Welding processes can introduce high residual stresses in the material at the weld joint. The high stresses result from thermal contraction of the weld metal during cooling. A higher residual tensile weld stress will increase the material's susceptibility to IGSCC. Although weld stresses are not easily quantified, previous investigation into weld stresses indicate that tensile stresses on the weld surface may be as high as the yield stress of the

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[1]Conductivity is a measure of the anionic and cationic content of liquids. As a reference, the conductivity of pure water is ~0.05 μs/cm. Reactor coolants with conductivities below 0.20 μs/cm are considered to be relatively ion free; reactor coolants with conductivities above 0.30 μs/cm are considered to have a relatively high ion content.

[2]The electrochemical potential (ECP) is a measure of a material's susceptibility to corrosion. In the absence of an externally applied current, and therefore, for reactor internals in the RCS, the electrochemical potential is equal to the open circuit potential of the material. Industry experience has shown that crack growth rates in reactor internals are low when the ECP $\leq$ ~-0.230 volts.

material. The stress decreases to compressive levels in the center of the welded section.

The OCNGS core shroud is considered to be highly susceptible to IGSCC and its susceptibility ranking is considered to be relatively high in comparison with the distribution of susceptibility rankings among domestic BWRs. The OCNGS plant specific susceptibility factors are summarized below:

(i) The top flange ring, top guide support ring, core support plate ring and shroud shells are each constructed from multiple cut and rolled Type 304 stainless steel plates that were welded together. The carbon contents of the plates in the range of 0.042% - 0.064%.

(ii) Weld residual stress levels are considered to be high based on weld shrinkage estimates.

(iii) OCNGS had initially been operated with a high ionic content reactor coolant. The initial 5 year average coolant conductivity for OCNGS was 0.526 μS/cm, which is considerably higher than the average value reported among U.S. BWRs (where the conductivities range from ~0.123 μS/cm to 0.718 μS/cm, and average ~ 0.340 μS/cm for the 36 U.S. domestic BWRs)

(iv) OCNGS has operated for 15.5 cumulative years at full power, which is above the median for U.S. BWRs (range is 3.7 years - 17.8 years, with a median of 10.8 years)

The BWRVIP has determined that the OCNGS shroud is highly susceptible to IGSCC, and has rated the OCNGS shroud as a Category "C" shroud (Reference 5). The staff finds that the BWRVIP assessment of the OCNGS shroud is acceptable. Considering the above plant-specific susceptibility factors as well as the industry-wide inspection experiences and the uncertainties in the residual stress profile resulting from fabrication, the staff concludes that significant cracking of the OCNGS core shroud cannot be ruled out.

2.2 OCNGS Core Shroud Inspections

GPUN performed limited VT-1 examinations of the OCNGS core shroud and the conical supports supporting shroud welds H7 and H8 during RFO 14R in the Fall of 1992. The camera-to-objective distance of these examinations varied from 4 inches to 18 inches in length. The RFO 14R examinations are summarized in Appendix 1 to this SE. The results of the RFO 14R shroud and conical support examinations were negative for relevant indications.

The licensee indicated in its response to GL 94-03 that RFO 15R inspections of the OCNGS core shroud would involve UT inspections of accessible areas on shroud welds H1, H2, H4, H5 and H6a, and enhanced visual examinations of welds H3, H6b and H9. The licensee inspected the OCNGS core shroud during RFO 15R, which commenced on September 9, 1994. The licensee's RFO 15R examinations are summarized in Appendix 2 to this SE. The examinations of the OCNGS core shroud indicated that significant cracking was evident in the H4 weld. GPUN concluded, after completing the UT examinations of the H4 weld, that they would proceed with implementing a modification of the OCNGS core shroud. The staff's assessment of the shroud modification design is provided in Section 2.3 of this SE.

2.3 OCNGS Core Shroud Modification Design

GPUN submitted its design for the OCNGS core shroud modification on October 25, 1994 (Reference 4), as superseded by the November 1, 1994 submittal (Reference 5). The OCNGS core shroud modification involves installation of a series of tie-rod assemblies symmetrically around the shroud. These tie-rod assemblies are designed to restrict vertical and lateral motion of the shroud assuming that all circumferential welds in the core shroud fail coincident with a design basis event. The staff issued its SE regarding the OCNGS core shroud modification on November 25, 1994 (Reference 6). The staff concluded that the shroud modification design selected by GPUN provides an acceptable alternative load carrying capability for OCNGS core shroud. The staff therefore concluded that the modification design was acceptable for implementation.

In its review, however, the staff noted that the scope of the licensee's modification submittal did not include any criteria for augmented, non-destructive examinations (i.e., augmented inservice inspections) of the repair assemblies during subsequent refueling outages, or the results of corrosion testing to support the use of hard rolled XM-19 material in the tie rod assembly design. In a letter to Mr. Robert A. Pinelli, Chairman of the BWR Owners Group, on August 31, 1994 (Reference 7), the staff stated that it considers modifications (i.e., repair options) of core shrouds to be alternatives to Section XI of the ASME Code, which fall under the scope of 10 CFR 50.55a. The staff has therefore taken the position that licensees implementing shroud modifications/repairs would be required to augment their Inservice Inspection (ISI) Programs to include examination of the modification/repair designs and appropriate portions of the core shrouds. This position is stated in Section 2.2.7 of the staff's generic SE on core shroud modifications (Reference 8). Therefore, in its SE of November 25, 1994 (Reference 6), the staff required GPUN to submit its augmented ISI scope for inspection of the gussets and tie rod assemblies during subsequent refueling outages. The staff also required GPUN to submit the results of corrosion tests which were to be performed on mockups of tie-rod assembly components fabricated from hard rolled XM-19 material (Ref. 6).

3.0 CONCLUSIONS

As a result of the core shroud inspections, and in particular of result of the examination of the H4 weld, GPUN has installed a pre-emptive modification of the OCNGS core shroud in lieu of full shroud inspections. The staff has reviewed the licensee's proposed core shroud modification design, and has found it acceptable for implementation at the OCNGS. The licensee's core shroud modification justifies operation of OCNGS during subsequent cycles.

4.0 OUTSTANDING ISSUES

There are no outstanding issues or staff comments in regard to the licensee's response to GL 94-03. Per the staff's SE of November 25, 1994, GPUN is required to submit its augmented inspection program for inspection of the tie-rod assemblies during subsequent refueling outages, and the results of corrosion testing on hard rolled XM-19 materials within 6 months of the plant re-start.

5.0 REFERENCES

1. GPU Nuclear Letter No. C321-94-2133 from Mr. R. W. Keaten, Vice President and Director of Technical Functions, to the U.S. Nuclear Regulatory Commission, submitting the response "Oyster Creek Nuclear Generating Station, Docket No. 50-219, NRC Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," dated August 24, 1994.

2. GPU Nuclear Letter No. C321-94-2157 from Mr. R. W. Keaten, Vice President and Director of Technical Functions, to the U.S. Nuclear Regulatory Commission, submitting the "Core Shroud - 15R Outage Contingency Repair," dated September 28,1994.

3. GPU Nuclear Letter No. C321-94-2185 from Mr. R. W. Keaten, Vice President and Director of Technical Functions, to the U.S. Nuclear Regulatory Commission, submitting the "Oyster Creek Nuclear Generating Station, Docket No. 50-219, Facility Operating License No. DPR-16, Core Shroud - 15R Outage Contingency Enhancement," Part 1, dated October 25, 1994.

4. GPU Nuclear Letter No. C321-94-2287 from Mr. R. W. Keaten, Vice President and Director of Technical Functions, to the U.S. Nuclear Regulatory Commission, submitting the "Oyster Creek Nuclear Generating Station, Docket No. 50-219, Facility Operating License No. DPR-16, Core Shroud - 15R Outage Contingency Enhancement," Part 2, dated November 1, 1994.

5. GPU Nuclear Letter No. C321-94-2171 from Mr. R. W. Keaten, Vice President and Director of Technical Functions, to the U.S. Nuclear Regulatory Commission, submitting the "Oyster Creek Nuclear Generating Station, Docket No. 50-219, Inspection Results for Oyster Creek Core Shroud," dated November 3, 1994.

6. Letter from the NRC to Mr. John J. Barton, Vice President and Director, GPU Nuclear Corporation, submitting the staff's "Safety Evaluation Regarding the Oyster Creek Core Shroud Repair (TAC No. M90104)," dated November 25, 1994.

7. Letter from Ashok C. Thadani, Associate Director for Inspection and Technical Assessment, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, to Robert A. Pinelli, Chairman, BWR Owners Group, dated August 31, 1994, submitting the staff's position on the "Boiling Water Reactor (BWR) Core Shroud Repair Design Criteria."

8. Letter from Brian W. Sheron, Director, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Bruce McLeod, Chairman, BWRVIP Technical Subcommittee on Repair, dated September 29, 1994, submitting the staff's "Safety Evaluation on Boiling Water Reactor (BWR) Core Shroud Design Criteria."

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Appendix 1 - Outage 14R Enhance VT-1 Examinations

Weld	Location	Portion Inspected [1]
Circ. Weld H5	From the O.D.	360° around
Lower Vertical Weld	@ Azimuth 80° from I.D.	Not Indicated
Circ. Weld H5	From the I.D.	Not Indicated
Lower Vertical Weld	From the O.D.	Not Indicated
Upper Vertical Weld	@ Azimuth 260° from O.D.	Not Indicated
Conical Sup. Weld to H7	From the O.D.	From Bracket 29 to Bracket 32
Conical Sup. Weld to H9	From the O.D.	From Bracket 29 to Bracket 32

1. No significant cracking was discovered as a result of the shroud examinations which were performed dur'ng the 14R refueling outage.

Appendix 2 - Outage 15R Core Shroud Examinations Results

WELD I.D.	INSPECTION METHOD	TOTAL AREA INSPECTED	INSPECTION RESULTS
H1	UT-OD Tracker	82" or 14% of circumference	No relevant indications
H2	UT-OD Tracker	162" or 27% of circumference	6 cracks, ID - for a total length of 19.33"; one crack, OD - 8.86"
H3	Enhanced VT-1	107" from ID: from 0° - 15" from 90° - 24" from ~180° - 22" from ~260° - 46"	2 cracks: 1 crack @ ~90° - 6" long 1 crack @ ~260° - 28" long
H3	UT - suction cup from ID	65" or 12% of circumference	No crack indications; UT scans revealed a lack of fusion for the UT length scanned
H4	UT-OD Tracker	276.5" or 49% of circumference	21 cracks indicated: Up. Side Weld, ID - ~63" T Lw. Side Weld, OD - ~138" T Lw. Side Weld, ID - ~173" T
H5	UT-OD Tracker	237" or 42% of circumference	No relevant indications
H6a	UT-OD Tracker and Suction Cup from OD	250" or 45% of circumference	9 surface indications recorded
H6b	UT-OD Tracker and Suction Cup from OD	217" or 31% of circumference	2 surface indications recorded
H9	Visual Exam.	224" Total: four equally spaced (4 X 56")	No relevant indications
Brackets	Visual Exam.	30 brackets	one linear indication in bracket #2