Attachment 1

SAFETY EVALUATION OF THE RESPONSE TO GL 94-03

FOR THE FITZPATRICK NUCLEAR STATION

NEW YORK POWER AUTHORITY (NYPA)

DOCKET NO. 50-333

TAC NO. M90092

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 an overseas 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 overseas 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 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:

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

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- 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

The New York Power Authority (NYPA) responded to GL 94-03 on August 24, 1994 (Reference 1), and supplemented the response on October 18, 1994 (Reference 2) and November 30, 1994 (Reference 3). Section 2.0 of this Safety Evaluation Report (SER) gives the staff's assessment of NYPA's response to GL 94-03.

2.0 EVALUATION

The NYPA originally scheduled inspection of the FitzPatrick (henceforth abbreviated FP) core shroud for the unit's Winter 1994-1995 refueling outage (RFO), which commenced on November 29, 1994. NYPA indicated that it intends to perform a pre-emptive modification of the FP core shroud in lieu of comprehensive core shroud inspections (Reference 4).

2.1 Susceptibility of the FP 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 ten 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 material. The stress decreases to compressive levels in the center of the welded section.

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

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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 us/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.

(i) The top flange ring, top guide support ring and core support plate ring are each constructed from six arc segments that were cut from rolled Type 304 stainless steel plates, with carbon contents in the range of 0.036% - 0.078%.

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

(iii) FP had initially been operated with a high ionic content reactor coolant. The initial five year average coolant conductivity for FP was 0.718 μS/cm, which is considered the highest 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) FP has operated for 12.8 cumulative years at full power, which is slightly 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 FP shroud is highly susceptible to IGSCC, and has rated the FP shroud as a Category "C" shroud (Reference 5). The staff finds that the BWRVIP assessment of the FP 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 FP core shroud cannot be ruled out.

2.2 Shroud Modification Design Part 1: Revised Core Shroud Inspection Scope

The licensee had originally indicated in their response to GL 94-03 that it that inspections of the FP core shroud would involve 100% UT inspections of all accessible areas on shroud welds H1 - H5, UT and enhanced VT-1 examinations of welds H6a and H6b, and enhanced VT-1 inspections of welds H7, H8 and H9. On October 21, 1994, NYPA informed the staff that it will perform a pre-emptive modification of the FP shroud in lieu of a comprehensive shroud inspection (Reference 4). The licensee has informed the staff that the inspection scope for the FP core shroud will be revised to one which supports implementation of the core shroud modification design. The revised inspection scope will involve, as a minimum, the following inspections:

- VT inspection of the portions of the welds joining the gusset plates used in the repair design to the jet pump support plate and the reactor pressure vessel, respectively,
- UT inspection of at least one of the vertical seam welds below the H4 circumferential weld,
- Inspection of the H3, H6a and H6b welds to gauge the extent of cracking in the shroud support rings.

The staff issued its SER regarding the FP core shroud repair design on January 5, 1994 (Reference 6). The scope of the staff's review covered both the licensee's alternative inspection scope for the FP core shroud and the licensee's scope for design of the FP core shroud modification. The licensee's reduced inspection scope is justified on the basis that the core shroud modification is designed to carry the structural loadings of the shroud in lieu of the shroud's circumferential welds. The licensee's revised inspection scope is in accordance with the recommendations of the guidelines established by the BWRVIP Task Group on Repairs (References 7 and 8).

2.3 Shroud Modification Design Part 2: Staff Evaluation of the FP Shroud Modification Design

The FP 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 concluded that a structural modification of the FP core shroud was acceptable in lieu of implementing comprehensive core shroud inspections. The staff also concluded that the modification design was acceptable for implementation, and that the licensee's modified scope for pre-modification and post-modification examinations of the core shroud and the shroud repair assemblies was in accordance with the "BWR Core Shroud Repair Design Criteria" (References 7 and 8). The staff issued its SER on the "BWR Core Shroud Repair Design Criteria" on September 29, 1994 (Reference 9).

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 10), 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 SER on core shroud modifications (Reference 9). Therefore, in its SER of December 29, 1994 (Reference 6), the staff required NYPA to submit its augmented ISI scope for inspection of the gussets and tie rod assemblies during subsequent refueling outages. The staff also required NYPA 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

NYPA has informed the staff that it intends to perform a pre-emptive modification of the Fitzpatrick 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 James A. Fitzpatrick Nuclear Power Plant. With the exception of the shroud inspections listed in Section 2.2 of this SER, the licensee's core shroud modification will serve as an alternative to comprehensive core shroud examinations.

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 reporting requirements of GL 94-03, NYPA will be required to submit the results of the shroud inspections performed during the Winter 1994 RFO to support implementation of the core shroud modification. It should be noted that currently the industry is having a difficult time performing more comprehensive inspections of lower shroud welds due to NDE equipment accessibility problems. The staff urges licensees to work with the EPRI NDE Center in order to develop improved tooling for inspections of shroud welds which are highly obstructed. Should improved inspections techniques become available, the staff recommendation is for licensee's to re-inspect the lower shroud welds at the earliest opportunity.

5.0 REFERENCES

1. Letter from William A. Josiger, Acting Executive Vice President, Nuclear Generation, New York Power Authority, to the U.S. Nuclear Regulatory Commission, dated August 24, 1994, submitting the "James A. Fitzpatrick Nuclear Power Plant, Docket No. 50-333, Response to Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs."

2. Letter from William J. Cahill, Jr., Executive Vice President and Chief Nuclear Officer, Nuclear Generation, New York Power Authority, submitting the "Supplemental Response to Generic Letter 94-03, 'Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs,'" dated October 18, 1994.

3. Letter from William J. Cahill, Jr., Executive Vice President and Chief Nuclear Officer, Nuclear Generation, New York Power Authority, submitting the "Revision to Core Shroud Safety Assessment Report," dated November 30, 1994.

4. Letter from the New York Power Authority to the U.S. Nuclear Regulatory Commission submitting "Request for NRC Approval of the Fitzpatrick Core Shroud Repair," dated October 21, 1994.

5. Submittal letter from Carl Terry, Executive Chairman of the BWRVIP Assessment Subcommittee, to the NRC, submitting the "BWR Core Shroud Inspection and Evaluation Guidelines," dated September 2, 1994.

6. Letter from Nick Conicella, Acting Project Manager, James A. Fitzpatrick Nuclear Power Plant, to Mr. William J. Cahill, Jr., Executive Vice President - Nuclear Generation, New York Power Authority, submitting the staff's "Safety Evaluation Regarding the Core Shroud Repair for the James A. FitzPatrick Nuclear Power Plant (TAC No. M90964)," dated January 5, 1994.

7. Letter from Carl D. Terry, Executive Chairman, Assessment Committee, Boiling Water Reactor Vessel and Internals Project, to the U.S. Nuclear Regulatory Commission, dated August 18, 1994, submitting the "BWR Core Shroud Repair Design Criteria," dated August 18, 1994.

8. Letter from J. T. Beckham, Chairman, BWR Vessel and Internals Project, to the U.S. Nuclear Regulatory Commission, dated September 13, 1994, submitting the "Response to NRC Request for Additional Information (RAI) Regarding Boiling Water Reactor (BWR) Core Shroud Repair Design Criteria," and Including "Revision 1 of the Core Shroud Repair Design Criteria."

9. 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."

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

Attachment 2

January 19, 1995

Docket No.: 50-333

Mr. William J. Cahill, Jr.
Executive Vice President Nuclear Generation
and Chief Nuclear Officer
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

SUBJECT: GENERIC LETTER (GL) 94-03, "INTERGRANULAR STRESS CORROSION CRACKING OF CORE SHROUDS IN BWRs," NEW YORK POWER AUTHORITY, FITZPATRICK NUCLEAR STATION, TAC NO.(s) M90092

Dear Mr. C 'll:

By letter dated August 23, 1994, the New York Power Authority (NYPA) provided its response to Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs," for the James A. FitzPatrick Nuclear Power Plant. The NRC staff requested in GL 94-03 that licensee's take the following actions with respect to their core shrouds: 1) inspect their core shrouds in their BWR plants no later than the next refueling outage; 2) perform materials related and plant specific consequence safety analyses with respect to their core shrouds; 3) develop core shroud inspection plans which address inspection of all core shroud welds and which takes into account the latest available inspection technology; 4) develop plans for evaluation and/or repair of their core shrouds; and 5) work closely with the BWR Owners Group with respect to addressing intergranular stress corrosion cracking of BWR internals.

The NRC staff required that licensee's submit, under oath or affirmation, the following information in response to GL 94-03 within 30 days of the date of issuance: 1) a schedule for inspection of their core shrouds; 2) a safety analysis, including a plant specific safety analysis as appropriate, which supports continued operation of the facility until inspections are conducted; 3) a drawing(s) of the core shroud configurations; and 4) a history of shroud inspections completed to date. The NRC staff also required that licensee's submit, under oath or affirmation, no later than 3 months prior to performing their core shroud inspections, their scope for inspection of their core shrouds and their plans for evaluating and/or repairing their core shrouds based on their inspection results. The NRC staff further required licensee's to submit, under oath or affirmation their core shroud inspection results within 30 days of completing their shroud examinations.

Enclosed with this letter is the staff's Safety Evaluation Report (SER) in regard to NYPA's response to GL 94-03. In regard to the information that was requested to be submitted within 30 days of the date of issuance of the GL, the staff has determined that NYPA has provided the operational, fabrication and materials related information requested in regard to operation and design of the James A. FitzPatrick Nuclear Power Plant. The staff has concluded in its SER that it cannot preclude the occurrence of a 360° throughwall crack in the James A. FitzPatrick core shroud based on its review of the submitted operational, fabrication and chemistry histories.

The staff has determined that NYPA has implemented a core shroud modification during the James A. FitzPatrick Winter 1994-1995 Refueling Outage. The staff has accepted the core shroud modification in lieu of comprehensive core shroud inspections. The staff issued its SER on the James A. FitzPatrick Core Shroud Modification Design on January 5, 1994. In its SER, the staff concluded that the NYPA's core shroud modification will serve to maintain the structural integrity of the core shroud during power operation, transient and accident conditions, and that the modification justifies operation of the James A. FitzPatrick Nuclear Power Plant during subsequent operating cycles. NYPA is required to submit within 30 days of their completion any results of core shroud inspections which were performed in support of implementing the James A. FitzPatrick core shroud modification. NYPA is also required to submit within six months of plant startup the results of corrosion testing and their proposed revisions to their inservice inspection program that were requested in support of implementation of the core shroud modification. GL 94-03 is part of the staff's continued program and efforts to evaluate the structural integrity of safety-related reactor vessel internals in boiling water reactors.

Sincerely,

C. Eugene Carpenter, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment 3

SALP INPUT

FACILITY NAME: James A. FitzPatrick

DOCKET NUMBER: 50-333

SUMMARY OF REVIEW ACTIVITIES:

This review evaluates New York Power Authority's (NYPA, the licensee) response in regard to the staff's issuance of Generic Letter 94-03, " Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs," which was issued to domestic BWR licensees on July 25, 1994.

ENGINEERING AND TECHNICAL SUPPORT:

NYPA has recently completed its winter 1994-1994 refueling outage for the James A. FitzPatrick reactor. The licensee made a conservative decision to implement a pre-emptive modification of the James A. FitzPatrick core shroud in lieu of comprehensive shroud inspections. The licensee's proposed modification method was reviewed and approved by the staff on January 5, 1995 (See SER, "Safety Evaluation Regarding the Core Shroud Repair for the James A. FitzPatrick Nuclear Power Plant (TAC No. M90964)," dated Jan. 5, 1995). The licensee has been very cooperative in providing pertinent information about the James A. FitzPatrick shroud modification to the staff.