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New York Power Authority

August 24, 1994
JPN-94-043

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Subject: 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

References: 1. NRC letter, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors (Generic Letter 94-03)," dated July 25, 1994.

2. BWROG-94-097 "Revision 1 of BWROG Shroud Document," dated August 5, 1994, transmitting GENE-523-A107P, "BWR Shroud Cracking Generic Safety Assessment" Revision 1, August 1994.

Dear Sir:

NRC Generic Letter 94-03 (Reference 1) requested licensees to inspect the core shroud in their BWR plants no later than the next scheduled refueling outage and perform an evaluation or repair as appropriate based on the results of the inspection. In addition, licensees were requested to perform a safety analysis supporting continued operation until inspections can be completed. This letter provides the Authority's response for the FitzPatrick plant, which is summarized in Attachment I.

The Authority will inspect the FitzPatrick core shroud during the upcoming refueling outage which is currently scheduled to begin November 29, 1994. Because cracking has been detected in the core shroud of plants similar in design to FitzPatrick, the Authority is developing a structural repair to the shroud which will be installed if necessary during this same outage. The inspection and repair plans are being developed consistent with the draft BWROG generic criteria. If necessary, these plans will be revised to conform to the final BWROG generic criteria following their submittal to the NRC, which is expected in the near future.

To demonstrate the safety of continued operation until the upcoming refueling outage, the Authority performed the plant specific assessment provided with this letter. This assessment is based upon the following analyses: a plant specific evaluation of the upper three welds performed in November, 1993; an evaluation of the applicability of the BWROG generic safety assessment prepared in July, 1994 (Reference 2); and, a plant specific probabilistic risk assessment of operation with undetected cracks in the core shroud. The evaluation also includes a plant specific evaluation of the factors that can affect susceptibility to shroud cracking.

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The probabilistic risk assessment concluded that the combined core damage frequency for all five initiating events considered (main steam line break, recirculation line break, design basis earthquake and seismic induced steam line and recirculation line breaks) is 6.3×10^{-7}/year (assuming operation with 360° through-wall cracking). This frequency is lower than both the NRC safety goal and the Fitzpatrick individual plant evaluation (IPE) internal events core damage frequency.

General Electric is preparing a plant specific evaluation to more fully quantify the impact of shroud cracking on certain design basis events and to assess operation during coastdown or other reduced power conditions. This evaluation will be provided to the NRC by October 17, 1994. Until then, the generic assessment in combination with the November, 1993 evaluation forms the basis for continued operation.

The generic and plant specific safety assessments, together with completed operator training to recognize and cope with core shroud cracking events, assure the safety of plant operation until the 1994 refuel outage.

Specific responses to the individual action items requested in Generic Letter 94-03 are contained in Attachments I through VI to this letter.

If you have any questions, please contact J. A. Gray, Jr.

Very truly yours,

W A Josiger

William A. Josiger
Acting Executive Vice President
Nuclear Generation

STATE OF NEW YORK
COUNTY OF WESTCHESTER
Subscribed and sworn to before me

this 24th day of August, 1994.

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4891272
Qualified in Westchester County
Commission Expires Jan. 27, 1996

Attachments:

I - Summary Response to GL 94-03 Reporting Requirements
II - Safety analysis supporting continued operation of FitzPatrick
III - Inspection Plan
IV - Repair Plan
V - Probabilistic Risk Assessment
VI - References
VII - Drawings of the core shroud configuration

cc: Regional Administrator
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COMMITMENTS/ACTION ITEMS

SUBJECT/ISSUE: Generic Letter 94-03 IGSCC of Core Shrouds in BWRs

REFERENCE LETTER: JPN-94-043

Commitment Number	Commitment/Action Item	Due Date
JPN-94-043-01	Provide plant specific quantitative evaluation of response to core shroud displacement to the NRC.	10/17/93
JPN-94-043-02	Provide the results of the inspection of the core shroud welds to the NRC within 30 days from completion of the inspection. (GL-94-03 reporting requirement 3).	30 days after inspection completion
JPN-94-043-03	If necessary, revise core shroud inspection plan to conform to BWROG generic inspection criteria document when the BWROG document is approved.	
JPN-94-043-04	If necessary, revise core shroud repair plan to conform to BWROG generic repair criteria document when the BWROG document is approved.	
JPN-94-043-05	If necessary, revise guidance to licensed operators when BWROG Emergency Procedures are approed by the BWROG.	

ATTACHMENT I to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

SUMMARY RESPONSE TO REPORTING REQUIREMENTS

New York Power Authority
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-33

ATTACHMENT I to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs

Summary Response to Reporting Requirements

Requirement

1. (a) - Schedule for inspection of the core shroud.

The Authority will inspect the FitzPatrick plant core shroud during the next refueling outage, currently scheduled to begin November 29, 1994.

1. (b) - Safety analysis, including a plant specific safety assessment, as appropriate, supporting continued operation of the facility until inspections are conducted.

See Attachment II.

1. (c) - Drawings of the core shroud configuration showing details of the core shroud geometry (e.g., support configurations for the lower core support plate and the top guide, weld locations and configurations).

See Attachment VII.

1. (d) - A history of shroud inspections for the plant should be provided addressing date, scope, methods and results, if applicable.

No shroud inspections have been performed since the plant was placed into operation.

2. - No later than 3 months prior to performing the core shroud inspections (If the inspections are scheduled to begin in less than 3 months from the receipt of this letter, the licensee should contact their NRC project manager to establish a schedule for providing the following information):

2. (a) - The inspection plan requested in item 3 of Requested Actions.

The current inspection plan is provided as Attachment III. The Authority is working with the BWROG in the development of a generic inspection criteria document. Guidance from the generic document will be incorporated into the Authority plan when the generic document is approved and submitted to the NRC.

2. (b) - Plans for evaluation and/or repair of the core shroud based on the inspection results.

A description of the repair, which is currently in the design and procurement stage, is provided as Attachment IV. The Authority is working with the BWROG in the development of a generic shroud repair criteria. The plan meets the current draft BWROG generic shroud repair criteria and will be modified as necessary to conform to the final version of the BWROG document.

3. - Within 30 days from the completion of the inspection, provide the results of the inspection.

The Authority will provide the inspection results to the NRC within 30 days of receipt of the results from the vendor.

ATTACHMENT II to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

PLANT SPECIFIC SAFETY ANALYSIS SUPPORTING CONTINUED OPERATION

Requirement 1. (b)

New York Power Authority
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

PLANT SPECIFIC SAFETY ANALYSIS

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SECTION I

BASIS FOR CONTINUED OPERATION

I. BASIS FOR CONTINUED OPERATION

A. Summary Basis for Continued Operation

The shroud is formed from highly ductile steel with high toughness properties, even accounting for any effect of neutron flux exposure. The applied loading on the shroud is relatively low. The combination of high ductility and low stress create shroud characteristics which are extremely flaw tolerant. Based on the BWROG analysis, the FitzPatrick shroud is expected to maintain margins against failure as specified in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (code) during the three months of operation which remain until the start of the scheduled refueling outage November 29, 1994. This evaluation also considered risks that were not considered as part of the original licensing basis.

The separation of the top guide assembly from the shroud is an extremely unlikely event. The more likely but still improbable scenario is a through-wall crack. If a significant through-wall crack exists, it would lead to some flow through the crack which would be detected during normal operation using available instrumentation and a normal shutdown would be conducted.

Finally, in the unlikely event of a design basis accident or seismic event with undetected 360 degree circumferential cracking beyond 90% of the shroud thickness, safe reactor shutdown can be achieved. Even applying very conservative assumptions, safe reactor shutdown is achieved automatically and adequate core cooling is provided, with manual backup available using the guidelines contained in the existing Emergency Operating Procedures.

The Authority has contracted with General Electric (GE) to perform additional plant specific analysis of the main steam line and recirculation line break loads and core shroud movement under postulated accident loads assuming 360° through-wall cracking of welds. This analysis will also assess operation during coast down or other reduced power conditions including the potential for decreased detectability for cracking. The Authority will provide the results to the NRC by October 17, 1994. The Authority currently estimates the submittal date to be October 17, 1994. The BWROG is considering an additional contract with GE to a perform additional analyses, which will be more specific by plant type, on a generic basis. The Authority will consider participating in that BWROG analysis, in place of the plant specific analysis, if it is approved by the BWROG.

Based on assessment of the applicability of the BWROG generic reports (References 1 and 2), the Authority concludes that there is no undue risk to the public during continued operation until November 29, 1994. This conclusion considered the uncertainties identified in the BWROG generic reports, and our conservative risk assessment which concludes that a cracked shroud will satisfy ASME code margins against weld failure for three additional months. Satisfying the ASME code margins against failure, coupled with the absence of flow or power anomalies characteristic of through-wall cracking during full power operation, provides reasonable assurance that the core shroud is and will remain intact, even under postulated licensing basis, and beyond licensing basis accident conditions. Therefore, the plant can safely continue operation without undue risk to the public health and safety.

B. History

In October, 1990, General Electric (GE) Rapid Information Communication Services Information Letter (RICSIL) 54 "Core Shroud Crack Indications" described the cracking found in an overseas reactor of a similar design (BWR4) and using the same material (high carbon 304 stainless steel) for shroud fabrication as the shroud in the FitzPatrick plant. In July 1993, GE issued revision 1 to RICSIL 54 providing information on cracks found in the Brunswick Unit 1 shroud which is also of the same material and design as the shroud installed in the FitzPatrick plant. This was followed in September 1993 by NRC Information Notice 93-79 (Reference 3) summarizing findings of cracking in the upper region of the shroud at the Brunswick plant and advising licensees to consider actions to avoid similar problems.

The Authority recognized that the nearly identical design, material, and fabrication of the FitzPatrick and Brunswick shrouds indicate a strong potential for finding similar cracks in the FitzPatrick shroud. In November 1993, the Authority made the decision to design a repair for the core shroud for installation during the next scheduled refueling outage in November, 1994. Design and procurement activities which could be started were initiated approximately one year in advance of the scheduled outage.

To provide assurance that continued operation of the plant would not present a significantly increased risk to the public health and safety during the balance of the current operating cycle (13 months at the time), the Authority performed a plant specific assessment (Reference 4) of the safety significance to the FitzPatrick plant of potential circumferential cracks in the top guide support ring welds of the core shroud assembly. These were the welds in which cracking had been observed at other BWR 4 plants. The assessment evaluated the upper region of the shroud (welds H1, H2, & H3) which bounded the cracks found in other plants at that time. The assessment concluded that the observed cracking at other plants did not indicate a need to inspect the FitzPatrick shroud prior to the refueling outage scheduled to start in November 1994.

In June and July of 1994, the NRC issued Information Notices 94-42 and 94-42 Supplement 1, (References 5 and 6) informing licensees of 360 degree crack indications in the lower region of the core shroud in two BWRs. Although the Authority had already completed a plant specific study (Reference 4) for the safety consequences of cracking in the upper region of the core shroud, the Authority joined with other utilities in the BWROG to assess the safety consequences of potential cracking in the lower region of the core shroud, and to develop generic inspection and repair criteria.

No detailed inspections to detect cracking of the FitzPatrick core shroud welds have been performed since initial pre-operational assembly. The Authority has concluded that the BWROG generic safety assessment (Reference 1) of the consequences of cracking in the lower region of the core justifies continued safe operation of the FitzPatrick plant during the balance of the current operating cycle.

C. Probabilistic Risk Assessment

A plant specific risk assessment is provided in Attachment V. Conclusions based on that assessment are:

- A realistic assessment using recent NUREG/CR-4407(Reference 7) recirculation/main steam line break frequencies shows the risk to the public health and safety from core damage to be 1.25×10^{-7} /year (for the initiating events of concern combined with a 360° through-wall shroud crack. Because the Fitzpatrick refueling outage is scheduled to begin November 29, 1994, the recirculation/main steam line break frequency for the remaining 90 days of operation is only 1/4 of the frequency per year. As the result, a lower cumulative core damage frequency is predicted to be 3.15×10^{-8} /quarter. This low probability provides a reasonable basis for continued operation through November 29, 1994.

  The conclusions below represent more conservative assessments of the core damage frequency, for the initiating events of concern, when combined with a 360° shroud through-wall crack. These conclusions assume older, (and more conservative) frequencies for recirculation /main steam line breaks. However, even the core damage frequencies provided by these assessments provide an acceptable basis for continued operation.

- All five initiating events examined have predicted core damage frequencies lower than the NRC safety goal of 1.0×10^{-4} /year, and lower than the Fitzpatrick IPE internal events core damage frequency of 1.92×10^{-6} /year. Therefore, the cumulative core damage frequency of any individual postulated event with a 360° core shroud through wall crack does not endanger the public health and safety.

- The combined core damage frequency (1.27×10^{-6} /year) for all five initiating events is lower than both the NRC safety goal and Fitzpatrick IPE internal events core damage frequency. Therefore, the cumulative core damage frequency of postulated events with a 360° core shroud through-wall crack does not endanger the public health and safety.

- The core damage frequency for a postulated main steam line break with a 360° core shroud through-wall crack is predicted to be 5.05×10^{-7} /year.

- The core damage frequency for a postulated recirculation line break with a 360° core shroud through-wall crack is predicted to be 7.34×10^{-7} /year.

- Without a concurrent MSLB or RLB, the design basis earthquake potential for shroud displacement is small and is not expected to adversely effect core cooling, control rod insertion, and SLCS performance. Therefore, earthquakes larger than the design basis [hence a lower initiating event frequency ($\approx 1.0 \times 10^{-7}$ /year)] or multiple random systems failure must occur before core damage results. As a result, shroud-induced core damage for a design basis earthquake is a non-dominant event.

- The core damage frequency for a postulated design basis earthquake induced main steam line break with a 360° shroud through-wall crack sequences is predicted to be 1.96×10^{-8} /year.

- The core damage frequency for a postulated design basis earthquake induced recirculation line break with a 360° shroud through-wall crack sequences is predicted to be 1.13×10^{-8} /year.

D. Inspection Programs

The integrity of the main steam line and recirculation line piping has been assured by the three Authority inspection and mitigation programs for intergranular stress corrosion cracking (IGSCC), in service inspection (ISI), and erosion/corrosion (E/C) described below. The information provided by these programs provides assurance that pipe integrity has been maintained. Assurance of this integrity significantly reduces the possibility of occurrence of the two design basis accidents, main steam line and recirculation line breaks, thereby reducing the probability for accident condition loadings on the shroud which could lead to shroud separation.

1. Intergranular Stress Corrosion (IGSC)

The Authority initiated an aggressive inspection program as required by Generic Letter 88-01 (Reference 8) and Technical Specification 4.6.F to detect pipe cracking. Previously detected cracking has been repaired, or evaluated, in accordance with the requirements of Generic Letter 88-01. In addition, the hydrogen water chemistry program initiated at the FitzPatrick plant in 1989 is expected to reduce the potential for crack initiation and subsequent crack growth in piping. During the 1992 refuel outage no new cracking was found during the IGSCC inspection program. The IGSCC inspection program is documented in Engineering Report JAF-RPT-MULT-01120 (Reference 9).

An NRC letter (Reference 10) documented the NRC review of the FitzPatrick IGSCC program. The NRC found the cumulative effects of weld overlay installation on the recirculation system were acceptable and met the intent of the requirements of Generic Letter 88-01, Supplement 1. (Reference 8).

2. In Service Inspection (ISI)

No cracking has been found during piping inspections of the main steam and feedwater welds from the reactor vessel to the outboard isolation valves. These inspections were performed as part of the ISI program in compliance with Technical Specification 4.6.F.

3. Erosion / Corrosion (E/C)

The inspections of the extraction steam piping, conducted for the Generic Letter 89-08 (Reference 11) required E/C program, have shown no degradation sufficient to cause failure of piping. The main steam system has been modeled in the EPRI Checkmate E/C computer program. Inspections are planned during the 1994-95 refueling outage. The Checkmate analysis shows that the main steam system is not susceptible to E/C. The extraction steam is more susceptible to E/C. Extensive inspections of this system have not found any major degradation.

4. Pipe Break Probabilities

Information published by the NRC has shown that the probability oi pipe breaks is significantly less than originally predicted during the plant design phase. This probability is discussed in detail in Reference 12.

A more recent paper (Reference 13) discusses the history of pipe break probabilities and provides a synopsis of EPRI, NRC and industry research. For example, NUREG/CR-4792 (Reference 14) states the probability of a double ended guillotine break (DEGB) ranges from 1.0E-12 to 3.8E-12 per reactor year for main steam and feedwater piping. The estimated DEGB probability resulting from IGSCC was estimated as 1.0E-3 assuming operating BWRs took no mitigating action. The probability did not include input from the inspection program developed by EPRI/BWROG in the 1980's. For example, FitzPatrick has used weld overlay, induction heat stress improvement and hydrogen water chemistry to mitigate IGSCC. Since 1985, inspection personnel have been qualified through the EPRI/BWROG training program at the EPRI NDE center. This training program improves the ability of inspection personnel to detect and size IGSCC cracks.

Also discussed in the report (Reference 13) is a recent study by EPRI (TR-102266) (Reference 15) which determined a DEGB probability of approximately 3.26E-5/reactor year. Using the NUREG/CR4407 (Reference 7) method shows a probability of 7.51E-6/reactor year.

E. <u>Shroud Wall Code Safety Margin</u>

The Authority prepared an engineering assessment (Reference 4) in November, 1993 to assess the safety significance of circumferential crack indications in the heat affected zone of the top guide support ring welds of the core shroud assembly, similar to those observed in other BWR-4 plants. Appropriate details from this report are presented in Section III of this safety assessment. In addition, engineering calculations were performed in August 1994, to support this response to Generic Letter 94-03.

The shroud is made of a highly ductile material with high toughness properties, even after accounting for any effect of neutron flux exposure. The applied loading on the shroud is relatively low. The combination of high ductility and low stress create shroud characteristics which are extremely flaw tolerant.

The Authority performed an engineering calculation in accordance with the ASME Code. The calculations determined the safety factor for resistance of the shell elements to longitudinal stresses due to normal and faulted internal-to-external pressure differentials to be not less than 20 to 1. The calculations demonstrate that the shroud structural integrity will be maintained, during normal operation, for 360° circumferential cracking with crack depths of up to an average of 95% of shroud thickness. Seismic and thermal loading were not considered in this calculation. However, seismic design evaluations for the core shroud repair will be evaluated for applicability when they are completed.

None of the cracks observed in other BWR-4 plants, if they presently existed in the FitzPatrick shroud, would represent a threat to safe operation of the FitzPatrick plant during the next three months of scheduled operation.

F. Operator Detection and Response to 360^{o} Through-the-Wall Cracking

Even if circumferential through-the-wall cracking did occur, plant operators have received specific training to detect and respond to this cracking during normal operation using existing instrumentation. Normal plant shutdown would be achieved. Appropriate procedures and lesson plans have been revised to address the potential of the occurrence of through-wall cracking. The EOPs provide the necessary guidance to respond to shroud crack events to protect the public health and safety. The BWROG Emergency Procedures Committee has recently prepared draft guidance for "Operator Actions to Detect Potential Shroud Cracks During Normal Operation." This guidance will be reviewed for applicability and provided to the operators, incorporated into procedure revisions, and used in operator training plans as appropriate when it has been approved by the BWROG.

1. Procedure Revision

Abnormal operating procedure (AOP-32) "Unexplained Reactivity Changes," contains guidance given in the supplement to General Electric (GE) SIL 462 to drive control rods in, prior to reducing flow, where indications of shroud support plate manway failures are present. The symptoms of 360^{o} through-wall shroud cracking are expected to be similar to manway failures and the actions required by AOP-32 are an appropriate response to shroud crack events.

2. Operator Training

Operator training lesson plans have been revised. System description lesson plan (SDLP-02a), "Reactor Pressure Vessel and Internals," was revised to include information related to SIL 462 for Failure of Core Shroud Support Plate Manways, including attendant indications and required actions. Training was also revised to address SIL 572, "Core Shroud Cracking in BWRs," including specific reference to the Brunswick inspection and required actions in AOP-32.

A computer based video animation of the onset and progression of core shroud cracking was developed for use in conjunction with the training materials. Continuing training was provided for all licensed operators using AOP-32, SDLP-02a, and the video animation in the first cycle of licensed operator requalification training in 1994 (01/03/94 - 02/11/94).

3. Emergency Procedure Guidelines (EPGs) and Emergency Operating Procedures (EOPs)

The Emergency Procedure Guidelines (EPGs) are the basis for plant specific Emergency Operating Procedures (EOPs). The EOPs are symptomatic in that they identify actions to respond to detected symptoms and do not require diagnosis of the event by the operator. They address a very wide range of events, both less severe and more severe than the design basis accidents.

The worst postulated event related to shroud cracking could result in separation and disengagement of the top guide from the fuel assemblies, which is further postulated to prevent a full scram: therefore the limiting event is a large steam line break with failure to completely insert the control rods.

The EOPs provide instructions for reactor pressure, water level, and power control, as well as control of key primary containment parameters. Actions specified in the EOPs for reactor power control are to (1) insert control rods using a variety of methods, and (2) initiate the standby liquid control system (SLCS) before suppression pool temperature increases to the allowable boron injection initiation temperature (BIIT) limit. The postulated event would clearly lead to SLCS injection within a very few minutes, resulting in safe shutdown.

For the postulated event in which shroud cracking occurs at or above weld H-4, water level can be controlled and a large compliment of water injection systems would be available. Separation of the shroud above the top of the fuel channels would not prevent maintaining the core in a flooded condition. Even if the core spray system were damaged by the shroud or top guide displacement, some core spray flow would be expected and any one Residual Heat Removal pump, in the low pressure coolant injection mode, would be sufficient to provide adequate makeup. EOPs require water level to be maintained at specific values. Thus, there would not be dilution of the stand-by liquid control boron solution by loss of vessel inventory through the break. The reactor would be shutdown and cooled down by following the guidance contained in the EOPs.

SECTION II

CONDITIONS WHICH INFLUENCE THE PROBABILITY OF CRACKING

AND

RATE OF CRACK GROWTH

A. Fabrication History

1. Fabrication Description

2. Material and Carbon Content

3. Susceptibility to Cracking

B. Reactor Coolant Chemistry - Conductivity and Hydrogen Addition

C. Neutron Flux Exposure - Fluence

II. Conditions Which Influence the Probability of Cracking and Rate of Crack Growth

(Generic Letter 94-03 requested licensee action 2.a.)

A. Fabrication History

1. Fabrication Description:

The core shroud and the location of the welds are illustrated in Attachment VI. The Fitzpatrick core shroud was fabricated by Sun Shipbuilding in 1970 and 1971. The core shroud separates the core region from the downcomer annulus which contains the jet pumps, and assures that feedwater flow is directed down the downcomer annulus, through the jet pumps, through the lower plenum, and up through the core.

The shroud is a series of shells and rings welded together to form a hollow cylindrical structure. The cylinder shape is 1.5 inches thick and has a nominal diameter of 14½ feet and height of 22¼ feet with a slightly tapered conical bottom piece below weld H-6B. The core shroud is constructed of plate, welded to segmented support rings. The various support rings present above and below the top guide and the core support plate were fabricated from arcs cut from hot rolled solution annealed plate, welded into a ring configuration and machined to size. This process results in the short traverse "end grain" of the plate being exposed to reactor coolant.

The top flange ring subassembly was fabricated from six arc segments, and welded end to end to form the rough ring assembly. The end grain surfaces of the rolled segments in the completed assembly are exposed to the reactor coolant environment.

The top guide ring was fabricated from six arc segments that were cut from plate materials. The arc segments were welded end to end to form the ring assembly. The end grain surfaces of the rolled segments are exposed to reactor coolant at the vertical surfaces of the ring, which are susceptible to crack initiation. The top guide flange supports the top guide which provides lateral support for the fuel assemblies and assures that the core geometry is maintained to allow for control rod insertion.

The core support plate ring was fabricated from six arc segments that were cut from plate materials. The arc segments were welded end to end to form the ring assembly. The end grain surfaces of the rolled segments are exposed to reactor coolant at the vertical surfaces of the ring. This ring forms a ledge that supports the core shroud cylinder and the lower core support plate.

The shell subassemblies were all made of two 1.5 inch plate sections that were rolled to curvature and joined by vertical welds. The plates sections were solution annealed prior to welding.

The bottom of the core shroud is welded to an Inconel 600 ledge at the bottom of the downcomer annulus. The jet pumps sit on this ledge. The shroud is further supported by 2 inch thick gusset plates welded to the top side of the support ledge and to the reactor vessel wall.

The steam separators and the dryers are mechanically attached to the core shroud at the shroud top flange. The core spray header for the core spray system is contained within and supported by the core shroud, and the connecting pipe enters through the vertical portion of the core shroud above the top guide support ring.

2. Material and Carbon Content

Attachment VI, Figure 1 is labeled to identify the segments of the core shroud and the "GEC" cross references to the certified material test reports provided by the material suppliers with the fabrication drawings.

Table I list the carbon content of weld wire from the certified material test reports provided in the reactor vessel fabrication records. Based on quantity, applicable procedures and welder qualifications it is believed that this material was used in the fabrication of the shroud.

Table I
Weld Wire Carbon Content

Type	Heat Number	Lot Number	% Carbon	% Ferrite
308	04697	2194920	0.054	7.8
308	04697	2194291	0.054	7.8
308	04697	2194292	0.053	6.4
308	04697	2194293	0.054	6.6
308	04697	2194294	0.054	6.8
308	04697	2194295	0.054	6.8

The core shroud was fabricated from type 304 ASTM A240 stainless steel with carbon contents ranging from 0.036 to 0.078 weight percent and 308 weld wire with 0.054 carbon content and greater than 5 % ferrite. Table II provides material carbon content from the certified material test reports for the plates used to fabricate the shroud.

Table II
Core Shroud Plate Material Carbon Content

Ring or Heat Number	Carbon Content
Top flange ring GEC-6, six segmented plates welded.	
F69767	0.056
F59549	0.061
F79866	0.078
Top Section upper barrel GEC-4, two plates rolled and welded.	
F79825	0.069
F79742	0.058
Top Guide Support Ring GEC-7, 6 segmented plates welded.	
F79793	0.063
F79804	0.064
Top Section lower barrel GEC-5, two plates rolled and welded.	
F59569	0.049
F79740	0.060
Middle section GEC-11, two plates rolled and welded.	
F79760	0.046
F79687	0.036
Lower Section upper barrel GEC-12, two plates rolled and welded.	
F79740	0.060
F79742	0.058
Core Support Ring GEC-13, six segmented plates welded.	
F59561	0.061
F69755	0.063
Lower section lower barrel GEC 14, three plates rolled and welded.	
F79760	0.065
F79574	0.047
F59545	0.050

The reported chemistry analyses show that all the material is within the ASME material specification requirements for plate and weld wire. Mechanical properties other than Brinell hardness were not reported. The shroud is a not a pressure retaining component. Therefore, test reports did not have to comply with ASME Section III, and mechanical tests were not reported.

3. Susceptibility to Cracking

Numerous factors contribute to the susceptibility of the shroud to stress corrosion cracking. Inspection of other shrouds, especially Brunswick 1 & 2, provided information on extent of cracking. This information can be evaluated against some of the factors for which information is available, such as material carbon content and environmental factors including neutron fluence and water chemistry. Other factors such as degree of cold work and weld residual stress may have a very significant influence on susceptibility, but quantitative information on these are not available. Some correlation between similar manufacturers can be made. This correlation provides some indication as to the material condition expected to be found in the FitzPatrick shroud.

It has been hypothesized that, if there is a relatively deep layer of surface cold work with high residual stresses, exposure of the "end grain" in and near the weld heat affected zone creates a potentially higher IGSCC susceptibility. The three cases identified in the BWROG report where cracks extended 360° around the circumference occurred at the top guide or core plate support rings in plants where rings were fabricated from plate. An analysis of boat samples from the Brunswick-1, Dresden-3, and Quad Cities, shows a large variation in cold work depth from approximately 0.010 to 0.060 inches; a distribution of cold work depth by plant is not provided.

The two plants having 360° cracks at the core plate - Dresden-3 and Quad Cities - are dissimilar to FitzPatrick. They are BWR-3 designs with 251" diameter reactor vessels (instead of 218"). The shrouds were fabricated by a different manufacturer. The plants similar to FitzPatrick in size of reactor vessel (218" diameter) and BWR design model (4) are Brunswick I and II. Brunswick I had a 360° crack indication, but at the H3 weld (compared to the lower level cracks in the Dresden and Quad Cities plants). The Brunswick I core support ring weld cracks are less than 30" in length. Further, Brunswick 2, which has been in operation longer than Brunswick 1 and has experienced high conductivity (similar to FitzPatrick), did <u>not</u> exhibit cracks longer than 30" at either the top guide or core plate ring, and had no 360° cracks at any location. Brunswick 2 did have 100-200" cracks in the mid-cylinder.

B. <u>Reactor Coolant Conductivity and Hydrogen Addition (Chemistry)</u>

The Authority compared the generic BWROG assessments to the FitzPatrick water chemistry. This section provides a plant specific evaluation of the FitzPatrick chemistry programs as they relate to the generic report and core shroud cracking phenomena. Implementation of hydrogen water chemistry (HWC), at the current target injection level of (12-14 SCFM), is expected to reduce the crack growth rate (CGR) near the top of the shroud when compared to normal water chemistry (NWC). Greater reductions in CGR are expected near the core bottom.

There is a correlation between poor early plant water chemistry and assumed electrochemical potential; and crack initiation and growth. IGSCC in recirculation piping and shroud head bolts discussed in the BWROG generic assessment have both been observed at the FitzPatrick plant. IGSCC cracks have also been found in an internal core spray pipe, dry tubes, shroud head bolts, and jet pump beam bolts. During the first five cycles, the water chemistry at the FitzPatrick plant significantly exceeded current EPRI and BWR chemistry guidelines. The FitzPatrick fuel warranty data for average reactor water conductivity while at power was

calculated to be 0.631 μS/cm which is less than the 0.718 μS/cm reported in Table 2-3 of the BWROG report. (This does not change the susceptibility grouping of FitzPatrick described in the BWROG generic assessment.) However, most of the poor chemistry was during the time period from 8/79 to 3/81. Beginning with the end of cycle 4, the average reactor water conductivity during operation has been maintained within the EPRI and BWR Chemistry Guidelines of ≤0.2 μS/cm with the last two cycles averaging closer to 0.1 μS/cm.

Hydrogen water chemistry has been in use since January 1989 to protect the recirculation system piping from IGSCC. The hydrogen flow rate has been gradually increased from 10.8 SCFM to 13.6 SCFM (Feedwater $[H_2]$= 0.35 to 0.4 ppm) over the years. The following table shows measured electro chemical potential (ECP) values at different locations in the primary system with normal water chemistry (NWC) and hydrogen water chemistry (HWC) at flow rates of 13-13.5 SCFM.

Measured ECP vs. Standard Hydrogen Electrode (SHE) in NWC and HWC

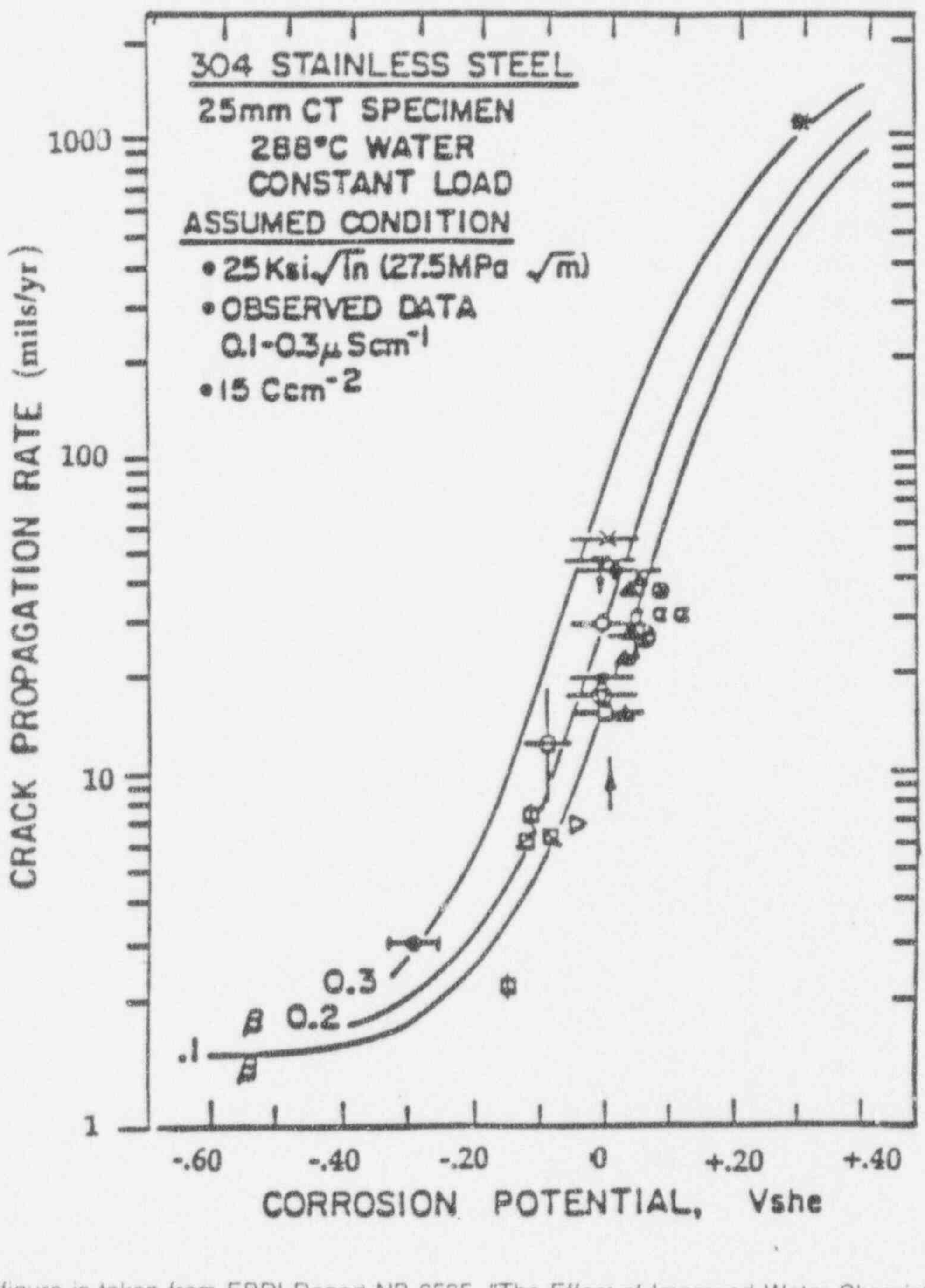
Sample Location	NWC (0 SCFM H_2) (VOLTS)	HWC (13-13.5 SCFM) (VOLTS)
Recirculation Sample Autoclave (CAV)	0.05	-0.45
Recirculation Flange (In-pipe)	0.16	-0.05 to -0.10
Core Support Plate[1]	0.15	0. 05
Top Guide[1]	0.20	0.10

1 These measurements were taken from an in-situ stress corrosion monitoring system installed in May 1990, consisting of five ECP sensors located in a modified LPRM.

Figure 1, taken from Reference 16, shows crack growth rates at conductivity ranges from 0.1μS/cm to 0.3μS/cm with varying ECP's. Based on this data and the measured ECP values in the core, even at hydrogen addition rates of 12-13 SCFM with low (0.1μS/cm) conductivity reactor water, there is a definite reduction in crack growth rates as compared to normal water chemistry at 0.2μS/cm from 300 mils/yr to <100 mils/yr. The exact ECP cannot be measured at each shroud weld. However, it is reasonable to expect that the 100 mv reduction in the ECP for 3 cycles when combined with average conductivity values of 0.2 μS/cm for 7 cycles, will have reduced the crack growth rates of existing cracks which may have been initiated during the first 5 cycles of high conductivity water chemistry.

The current reactor water chemistry is conductivity < 0.1 μS/cm, Cl ~1 ppb, SO_4 ~4-7 ppb and ECP's in the Crack Arrest Verification (CAV) system of ~-250 mV during the balance of the cycle.

Crack Growth Rate of Sensitized Type 304 Stainless Steel as a Function of Corrosion Potential



This figure is taken from EPRI Report NP-6585, "The Effect of Improved Water Chemistry and Corrosion Cracking of BWR Piping", dated December 1989.

FIGURE 1

C. <u>Neutron Flux Exposure - Fluence</u>

The fast neutron fluence (E >1 MeV) is one of the factors influencing stress corrosion cracking (SCC). A threshold value of 3-5E+20 nvt is estimated, in accordance with BWR Shroud Cracking Generic Safety Assessment (Reference 1), for Irradiation Assisted SCC (IASCC). GE estimated the peak fluence to be about 7.5E+20 nvt based on past fluence analysis work undertaken for the FitzPatrick plant.

In January 1994, the Authority initiated a plant-specific analyses to evaluate the shroud fluence and refine the GE estimates. This project is still in progress and is not expected to be complete until December 1994. Detailed models of the FitzPatrick core, shroud and vessel are being studied to determine analytically the fast fluence by employing historic cycle-specific information. Results generated at the Authority, based on the reactor power levels of the current fuel cycle (Cycle-11), have shown that for 11.79 years of full power operation:

1. the average fluence at the plane corresponding to the H-4 weld is about 3.8E+20 nvt along the inner surface of the shroud,

2. a peak fluence value of about 7.4E+20 nvt is estimated at the azimuth corresponding to the core edge closest to the shroud (i.e., 45 degree azimuth based on a 1/8th core symmetry),

3. the fluence along the azimuth corresponding to the maximum distance between the core and the shroud (0 degree azimuth) is about a factor of 2 lower than the average (< 2E+20 nvt), and

4. an additional reduction factor of about 2 is estimated between the inner and outer surfaces of the shroud.

The above results are in good agreement with the BWROG estimates provided in the generic report.

The fast fluence is one of the parameters influencing SCC. Peak fluence values are slightly higher than the estimated threshold for IASCC. Continued plant operation through the end of the current fuel cycle, and planned outage, is expected to increase the fluence by less than 3 percent at the peak location.

SECTION III

ASSESSMENT OF SHROUD RESPONSE

and

PERFORMANCE OF PLANT SAFETY FEATURES

A. Plant Specific Analysis Prepared for Welds H1, H2, and H3

B. Analysis for Shroud Response including Welds H4 through H8

C. Applicability of the BWROG Generic Reports

III. <u>Assessment of Shroud Response and Performance of Plant Safety Features</u>

(Generic Letter 94-03 requested licensee action 2. b.)

A. Plant Specific Analysis Prepared for Welds H1, H2, and H3

In November 1993, the Authority prepared a FitzPatrick plant specific assessment (Reference 4) for the implications of shroud cracking at the H1 through H3 welds. This scope was based upon the most significant cracking known to exist at that time. Indications on the H1 through H3 welds had been detected at the Brunswick Unit 1 plant. Because of the similarities in the design, construction, and operation of the FitzPatrick and Brunswick plants, the same extent of cracking observed at Brunswick was assumed to be present at FitzPatrick. The most significant difference, between Brunswick Unit 1 and FitzPatrick, is that FitzPatrick has operated with Hydrogen Water Chemistry for about 2 years longer than Brunswick Unit 1. For these reasons, it is appropriate and conservative to apply the Brunswick Unit 1 flaw indications on the Fitzpatrick H1, H2, and H3 welds for structural analysis. The purpose of that assessment was to determine if the FitzPatrick plant could safely operate until the next refueling outage, now scheduled for November 29, 1994, when inspections and repairs (if necessary) could be performed. Significant excerpts from this assessment follow:

"<u>SUMMARY AND CONCLUSIONS</u>

Crack indications observed in other BWR plants top guide support ring welds do not represent a threat to the safe operation of the FitzPatrick plant:

The combination of highly ductile material and low applied stresses make the shroud extremely flaw tolerant. A postulated 360 degree circumferential crack, with crack depths of up to 90% of the shroud thickness can be tolerated while maintaining the structural integrity of the shroud for normal and postulated accident conditions. Even with only 10% thickness remaining, the ASME Code safety margins are maintained.

If flaws existed in the H1, H2, H3 welds at start-up from the last refuel outage equivalent to those currently seen at Brunswick Unit 1, margins well in excess of the FSAR design basis will be maintained until shutdown for the next refueling outage.

The separation of the top guide assembly from the shroud is an extremely unlikely event. The more likely but still improbable scenario is a through-wall crack. If a significant through-wall crack exists, it would lead to some flow through the crack which would be detected during normal operation using available instrumentation and a normal shutdown would be conducted.

Finally, in the unlikely event of a design basis accident or seismic event with undetected 360 degree circumferential cracking beyond 90% of the shroud thickness, safe reactor shutdown can be achieved. Even applying very conservative assumptions, safe reactor shutdown is achieved automatically and adequate core cooling is provided, with manual backup available using the guidelines contained in the existing Emergency Operating Procedures.

SHROUD STRUCTURAL EVALUATION

Crack indications have been observed in various locations of the Brunswick Unit 1 core shroud. The circumferential crack indications located in the inside surface of the heat-affected zone of the top guide support ring horizontal weld are of the most potential significance because they appear to extend 360 degrees around the circumference of the shroud. The vertical welds in this region are relatively short (on the order of a couple of inches) compared to the length of the horizontal welds. The expected crack lengths from vertical welds would be much smaller and would not propagate to the extent that would allow full separation of the shroud. Therefore circumferential weld cracks are analyzed and the short vertical welds are not of concern.

Effect of Cracks on Shroud Strength

While the extent of the crack indications that have been reported at the Brunswick plant is significant, there remains sufficient structural strength in the components to meet their intended function. The shroud and support ring are made of ductile material with high toughness properties even after accounting for any effects due to neutron fluence. The applied loading on the shroud is mainly from the differential pressure during normal operation and the transient differential pressure increase due to design basis accident and seismic loads. The applied load during normal, high power (> 80%) operation is in the upward direction. The accident and seismic loads are generally small and well within the remaining structural integrity of the shroud.

The combination of high ductility and low applied stresses make the shroud extremely flaw tolerant. In fact, through-wall cracking of over 50% of the shroud circumference can be tolerated while maintaining ASME Code allowable design safety factors. The allowable circumferential flaw size is 95 inches for each 90° sector of the shroud. If 360° circumferential cracking is postulated, an allowable flaw size of up to an average of 90% of the thickness can be tolerated with sufficient remaining industry-accepted Code margins. Even if the crack depth is greater than 90% of the shroud thickness (up to the "critical flaw size", based on safety factors of 1.0), the full design basis and seismic loads can be accommodated.

Potential for Further Structural Degradation

Even if relatively deep cracks did occur, it is important to consider the non-uniformity of the crack growth around the circumference. Because of differences in sensitization, fluence, cold work and weld residual stresses around the circumference, uniform crack growth at different crack locations is not expected. This means that any further crack growth will not be uniform and the growth rate will be higher at some locations than others. Even if the growth continues until it is through-wall, this would only occur at specific locations (similar to the leak before break scenario in piping). Under the core internal pressure load, this would lead to a crack opening and leakage from the core. Leakage due to significant cracking will relieve the differential pressure loading and retard the subsequent crack growth rate. While the exact amount of leakage is difficult to predict, the fact remains that if leakage occurs (especially when the remaining ligament is small) it will eventually lead to detection.

In summary, the low stresses and high material ductility make postulation of a 360° crack leading to shroud separation extremely unrealistic.

JAF SHROUD STRUCTURAL ANALYSIS

A structural margin assessment for FitzPatrick was performed assuming that similar flaws found at Brunswick Unit 1 existed in FitzPatrick prior to start-up in the current fuel cycle and incorporated an allowance for crack growth. Fitzpatrick is currently operating under hydrogen water chemistry (HWC) and any crack growth occurring will be less than that expected with normal water chemistry (NWC). Thus, analysis using NWC crack growth rate will be conservative.

Structural analyses were performed on the H1, H2, and H3 welds to determine the safety margins for comparison with required safety margins specified in the FSAR. The analyses assumed FitzPatrick has similar crack indications at the beginning of the current fuel cycle as those observed in Brunswick Unit 1. Crack growth was added to determine the safety margins at the end of the current fuel cycle.

The structural analysis consisted of two steps: the determination of the axial stress magnitudes in the shroud and the calculation of safety margins.

Since all three welds have a fluence level much less than 8E20 n/cm^2, the material is not subject to embrittlement. Therefore, a limited load approach was used in conducting the structural evaluation as allowed by ASME Section XI Article IWB-3640.

Applied Loads and Calculated Stresses

The applied loads on the shroud consist of internal differential pressure, weight and seismic affects. Since the weight loading produces only a compressive stress which would retard crack growth, it was not included in calculating the total stress. The seismic loads consist of a horizontal shear force at the top of the shroud and an overturning bending moment. The shear force produces a shear stress of insignificant magnitude. The bending moment stress varies as a function of its vertical distance from the top of the shroud.

The magnitude of the applied loads was obtained from the reactor vessel seismic analysis and system information reports. The operating basis earthquake (OBE) loads (for the upset condition) were assumed to be 75% of the design basis earthquake (DBE) loads. Since the limited load approach was used, the stresses were calculated using strength of material formulas. The seismic stresses at weld H1 were conservatively assumed to be the same as those at weld H2. A nominal shroud thickness of 1.5 inches was used in the calculation. Table 1 shows the calculated stress magnitudes and also the stresses at the three welds.

For the purpose of the structural margin calculation, the nominal total stresses shown in Table 1 were treated as membrane stresses and the resulting axial load was determined. This load was divided by the area at a weld which is not cracked to determine the section stress. This stress was compared with the material flow stress to calculate safety margin. This approach is consistent with the ASME Code procedure for austenitic materials.

Evaluation of H3 Weld

Based on ultrasonic test results at Brunswick Unit 1, a 360° indication of uniform initial depth equal to 1.4 inches was assumed for FitzPatrick. The effective shroud thickness associated with the potential growth of the indication at the H3 weld is 2.25 inches (1.5 inches plus 0.75 inch leg of fillet weld). Based on crack growth rate of 5E-5 in/hour, the increase in the crack depth at the end of the fuel cycle is 0.467 inch. Thus, the postulated crack depth at the end of the current fuel cycle would be (1.4 + 0.467) or 1.867 inches. This assumed crack growth is conservative since it is based on NWC operation, whereas the FitzPatrick unit is currently operating with HWC. The remaining ligament considering crack growth is then (2.25 - 1.4 - 0.467) or 0.383 inch giving a load bearing area of 215.1 in^2. The net section stress based on this load bearing area was calculated as 4.34 ksi for the upset condition and 7.46 ksi for the faulted condition. The safety factors were determined by dividing the flow stress (3 Sm = 50.7 ksi) by the net section stress. The calculated safety factor values are shown in Table 2.

Additional structural area is contributed by the radial weld in the ring and alignment tabs which are not cracked. The inspection results of Brunswick indicated that four out of six radial welds are not cracked. Similarly, seven out of eight tabs are not cracked. The structural area contributed by these two elements was determined to be 21.75 in^2. Adding this load bearing area to that already calculated in the preceding paragraph gave a total load bearing area of 236.85 in^2. The net section stress based on this total area was calculated as 3.94 ksi for the upset condition and 6.77 ksi for the faulted condition. The corresponding values of safety factors are somewhat higher and are also shown in Table 2.

The upset and faulted conditions were judged to provide the limiting safety factors when compared to the required margins. Therefore, safety factors for only these operating conditions are reported in Table 1. A review of Table 2 indicates that even in the presence of this postulated cracking, including allowance for crack growth and ignoring the effects of HWC, large safety margins are present when compared to those required by the FSAR.

Evaluation of H2 Weld

A similar approach (assuming 360 degree circumferential cracking) was used at this weld. Based on the UT results from Brunswick Unit 1, an average crack depth of 0.5 inch was assumed. With a postulated crack growth, through end of current operating cycle, of 0.467 inch, the end of cycle crack depth would be 0.967 inch. At this location of the observed cracking, the section thickness is 2.25 inches (1.5 inches nominal thickness plus 0.75 inch from fillet weld leg). Thus, the remaining ligament is then 1.28 inches, giving a net section area of 751.04 in^2. The corresponding net section stress for the upset and faulted conditions are 1.22 ksi and 2.17 ksi, respectively. The calculated safety margins are shown in Table 2.

Evaluation of H1 Weld

Based on UT results from Brunswick Unit 1, the assumed indication depth at this weld is also 0.5 inch for 360 degrees. Since the assumed indication depth and the stress level at this weld are the same as those at the H2 weld, the safety factor as calculated for the H2 weld also applies to H1.

Table 1

H1, H2, H3 Applied Stresses (ksi)

Weld	Upset + OBE	Faulted + DBE
H3	0.81	1.08
H2	0.69	0.92
H1	0.51	0.68

Table 2

Calculated Vs. Required Safety Factors Comparison

Weld	Operating Condition	Calculated SF	Required SF
H3	Upset	11.7[1]	2.25
	Faulted	6.8[1]	1.125
	Upset	12.9[2]	2.25
	Faulted	7.5[2]	1.125
H2 & H1	Upset	41.7	2.25
	Faulted	23.4	1.125

Notes: 1. Without area contributed by radial welds and tabs
2. Including area contributed by radial welds and tabs

ANTICIPATED OPERATIONAL EVENTS

The previous section demonstrates that through-wall cracking or separation of the top guide from the shroud is improbable, but should it occur, it would be detectable during normal operation. Assuming there is no indication of shroud leakage, this section discusses anticipated operational scenarios that could increase shroud loads above those experienced during normal operations: pressure regulator failure-open; recirculation flow control failure increasing maximum flow, or inadvertent actuation of the Automatic Depressurization System.

Pressure Regulator Failure-open

This postulated Safety Analysis Report (SAR) event involves a failure in the EHC pressure controls, such that the turbine control valve and the turbine bypass valves are opened as far as the Maximum Combined Flow Limiter (MCFL) allows. FitzPatrick turbine by-pass capacity is 25% of rated main steam flow capacity, thus, the worst case involves an inadvertent increase of the steam flow to about 130% of rated steam flow (5% to second stage MSR reheat and other steam loads). A depressurization and cool down occurs which is isolated by the Main Steam Line Isolation Valve (MSIV) closure. This steam flow increase is small enough that the increased force on the shroud head (correlate to about 50% above the normal pressure drop) is within the load capability of the shroud.

Recirculation Flow Control Failure

This postulated event involves recirculation control failure that causes both recirculation pumps to increase to maximum flow. In this event, the pressure drop could change from a part-load condition to the high maximum flow condition over a time period of about 30 seconds, but it should not significantly exceed the pressure drop expected for normal full power, operating conditions. Normal operating procedures are considered sufficient to minimize the consequences of this potential transient, and the force on the shroud head is within the shroud capability.

Inadvertent Actuation of ADS

Inadvertent actuation of Automatic Depressurization System (ADS) valves is another postulated event that could put an increased load on the upper shroud. The maximum steam flow and depressurization rate are significantly smaller than for the postulated main steam line break accident, causing a short-term increase in steam flow of about 55% of rated capacity. The increase in pressure drop across the shroud head resulting from opening of the ADS valves would occur for about one second, before the turbine control valves would begin to close thus reducing the effect of the change in load. This is also a very low probability event, considered to be in the ASME emergency category in the vessel thermal duty design. The effect of this event is also within the shroud capability.

DESIGN BASIS ACCIDENTS

Although the combined probability of an accident when a severe undetected crack exists is very low, such a postulated event is addressed in this section.

The main steam line break accident imposes the largest potential lifting loads on the shroud head. The main steam line break inside primary containment is the postulated worst case and results in the highest pressure drop across the shroud. Liquid breaks (e.g., recirculation line breaks) do not impose large pressure drops across the shroud, and in fact the shroud pressure drop decreases from its initial value.

Main Steam Line Break

During this event, the reactor is rapidly depressurized as a result of a postulated instantaneous, double-ended break of one of the main steam lines within primary containment. A substantial pressure difference would develop across the shroud as fluid flow is drawn from the core region toward the break. If a sufficient pressure drop across the top guide support ring weld area is created, and sufficient cracking exists, it is postulated that this added differential pressure might cause separation of the shroud leading to an upward displacement of this structure and the fuel support top guide. The amount of lifting and the potential effect of these postulated occurrences on emergency operation are described below.

One of the key considerations of this postulated accident case is the ability to insert the control rods before or during the postulated accident. Specifically, sufficient lifting of the top guide prior to control rod insertion could cause reorientation of the fuel bundles and thus impede the insertion of control rods. Therefore, the timing of the top guide loads and control rod insertion is pertinent to the evaluation of this event.

The shroud head pressure drop characteristics calculated for the instantaneous, double-ended steam line break accident were evaluated by General Electric for a typical BWR.

The initial shroud head pressure drop loading is a result of the decompression wave which reduces system pressure overall, but would increase differential pressure across the shroud in the short term. This pressure loading increase is short-lived (less than two seconds) and decreases to below normal steady state loads. Even if the remaining shroud ligament is very small, the structural integrity of the shroud will remain intact for this postulated limiting event. If it is further postulated that the pulse causes the shroud to separate, the top guide assembly would lift. The flow diversion created by any separation reduces the upward lifting forces. For a realistic scenario, the top guide assembly would lift only a few inches, and the fuel channels would remain engaged.

Scram is initiated during the main steam line break accident by high drywell pressure. The drywell pressure passes the scram set point almost instantaneously, so the only delays in the rod insertion come from the sensors, the Reactor Protection System, and rod motion. Control rods that are partially inserted as part of normal operation are already in position to initiate shutdown and maintain fuel bundle orientation. During the postulated steam line break scenario, the insertion of all control rods will occur. Even if the upper shroud breaks free, control rod motion will be started before the upper shroud assembly and top guide lift enough

to clear the tops of the fuel bundles. Insertion of fully withdrawn control rods to 5% of stroke will occur by 0.9 second, early enough even for this degraded scenario for the control rods to be moving up between the bundles before any significant change in bundle alignment could take place. The remainder of the insertion is likely because blade insertion occurs while the fuel is still properly oriented even if the scenario lifts the top guide the postulated few inches. Reactor shutdown would be complete with all control rods inserted.

In the very unlikely case that scram may not be complete, the Standby Liquid Control System is available to provide shutdown capability, as discussed in the next section on potential operator actions.

Movement of the upper shroud assembly (in the very unlikely case that it occurs) could affect the core spray system if it damages the core spray line connection. However, injection of core spray flow outside the shroud or even total unavailability of the core spray system during the main steam line break will not prevent core cooling because the flow from any one Emergency Core Cooling System (ECCS) pump (e.g., low pressure coolant injection) is sufficient to restore and maintain reactor water level.

The main steam line break has also been evaluated for radiological release consequences in the FSAR. For a main steam line break to cause significant fission product release, the fuel cladding must be damaged. This would only result if the upper shroud assembly completely detaches, lifts sufficiently (over 14 inches) to disengage the top guide from the tops of the fuel bundles, and then lowers on the now reoriented fuel bundles. This is considered to be a non-credible event.

The uplift force from a main steam line break is sufficient to disengage the upper shroud assembly from the tops of the fuel bundles only if the shroud cracking is virtually complete (i.e. 360°, through wall) at the time of the steam line break. If significant cracking pre-existed, it would be detected and operators would shut down the plant. If preexisting cracks were not significant enough to be detected, then a considerable fraction of the uplift force would be dissipated by the energy requirements necessary to complete the crack that would fail the shroud. The flow path created by any separation reduces the upward lifting forces. Insufficient force would remain to cause the top guide to lift above the top of the fuel bundles. The upper shroud assembly would settle down to nearly its original position.

Although this could cause some superficial damage to the fuel channels, it would not be expected to damage any of the fuel cladding. Therefore, the radiological consequences of a main steam line break remains unchanged from that analyzed in the FSAR.

Recirculation Line Break

For the design basis recirculation line break, the differential pressure across the upper shroud decreases from the initial value as the reactor depressurizes, and thus the top guide assembly remains attached. Therefore, the recirculation line break analysis results are unchanged.

OPERATOR ACTIONS

The Emergency Procedure Guidelines (EPGs) are the basis for plant specific Emergency Operating Procedures (EOPs). The EOPs are symptomatic in that they respond to detected symptoms and do not require diagnosis of the event by the operator. They address a very wide range of events, both less severe and more severe than the design basis accidents.

The worst postulated event above would result in separation and disengagement of the top guide from the fuel channels, which is further postulated to prevent a full scram: therefore the limiting event is a large steam line break with failure to completely insert the control rods.

The EOPs provide instructions for reactor pressure, water level, and power control, as well as control of key primary containment parameters. Actions specified in the EOPs for reactor power control are to (1) insert control rods using a variety of methods, and (2) initiate the standby liquid control system (SLCS) before suppression pool temperature increases to the allowable boron injection initiation temperature (BIIT). The postulated event would clearly lead to SLCS injection within a very few minutes, resulting in safe shutdown.

Water level can be controlled easily for the postulated event because the break is high in the vessel and a large compliment of water injection systems would be available. Separation of the shroud above the top of the fuel channels would not prevent maintaining the core in a flooded condition. Even if the core spray system were damaged by the shroud or top guide displacement, some core spray flow would be expected and any one Residual Heat Removal pump, in the low pressure coolant injection mode, would be sufficient to provide adequate makeup. EOPs require water level to be maintained at specific values. Thus, there would not be dilution of the stand-by liquid control boron solution by loss of vessel inventory out the break. The reactor would be shutdown and cooled down by following the EOP procedures.

CONCLUSION

This evaluation considers the safety significance of circumferential cracks identified in the heat affected zone of the top guide support ring welds of the Brunswick Unit 1 Nuclear power plant. This report concludes that there is reasonable assurance that the observed phenomena does not represent a threat to the safe operation of the FitzPatrick plant. Therefore JAF can safely operate for the remainder of the current operating cycle in accordance with the GE SIL 572 Rev 1, shroud inspections and any necessary corrective actions will be performed during the next scheduled refuel outage."

B. Analysis for Shroud Response including Welds H4 through H8

Even though cracking has been detected on additional welds below H3 at other BWRs subsequent to the preparation of the above evaluation, to the extent that cracking could occur on welds H1 through H3 the conclusions reached are still valid. To fully assess the implications of possible cracking in shroud welds below H3, the Authority has contracted with General Electric to perform an additional FitzPatrick plant specific assessment of the design basis accidents. The scope of work for this assessment is as follows:

1. A FitzPatrick-specific assessment of the shroud response to the structural loadings resulting from design basis events (e.g., steam line break, recirculation line break) assuming through thickness cracking. The assessment will include results from the TRACG analysis - specifically, the pressure variation with time for a steam line break and the associated shroud displacement for the assumed through thickness cracking. The recirculation break assessment will use the analysis done for the VIP and predict the results for FitzPatrick using a potential flow model and the VIP TRACG results for the recirculation break. This will provide lateral loads on the shroud (asymmetric loads) resulting from the recirculation break.

2. An assessment of the ability of plant safety features to perform their function considering the shroud response to structural loadings (e. g., control rod insertion, ECCS injection.

3. Provide a report documenting the analysis results. The report will include among other things, the following results:

   - Calculated pressures for the steam line break (e.g., annulus, core plate, shroud head).

   - Shroud vertical displacements for assumed through thickness cracking at each weld.

   - Asymmetric loads for the recirculation break (e.g., force-time history curve).

   - Safety assessment for both the steam line break and the recirculation break.

   - Description of input parameters used in the analysis.

   - Assessment for operation at reduced power levels or during coast down including potential for decreased detectability of cracking or significant change in results.

This assessment cannot be completed within the time frame necessary to fulfill the reporting requirements of Generic Letter 94-03. A copy of this assessment will be submitted to the NRC by October 17, 1994.

C. Applicability of the BWROG Generic Reports

The Authority has participated in the BWROG effort to develop a generic safety assessment to address cracking in all susceptible weld in the core shroud. This report (Reference 1), previously provided to the NRC by the BWROG, is generally applicable to FitzPatrick in its assessment of the safety implications of shroud cracking. This generic assessment in combination with the November, 1993 evaluation described earlier and the information provided in this letter form the basis for continued operation until the FitzPatrick specific assessment is completed.

The BWROG generic report (Reference 1) considers the factors associated with the likelihood of developing shroud cracks. As described elsewhere in this submittal, the FitzPatrick plant is considered susceptible to developing shroud cracks. Section 5.7 of the BWROG generic report (Reference 1) provides the basis for continued operation for the grouping of plants including FitzPatrick. It states:

> ***"5.7 304 SHROUDS WITH WELDED-PLATE RINGS AND HIGHEST CONDUCTIVITY***
>
> *Inspected Plants:*
>
> *Brunswick 1: range from none to total of 360° cracking in various welds*
> *Brunswick 2: range from none to total of 73° cracking in various welds (H3 repaired, not inspected)*
> *Dresden 3: range from 1° to total of 360° cracking in various welds*
> *Quad Cities 1: range from none to total of 360° cracking in various welds*
>
> *Plants Planning Future Inspections:*
>
> *FitzPatrick, Hatch 1, Nine Mile Point 1, Oyster Creek, Pilgrim, Quad Cities 2*
>
> *Basis for Continued Operation:*
>
> *This grouping of plants has inspection results with 360° cracking to various depths. Structural margin is maintained with only 5-10% of the wall thickness in the remaining ligament. The PLEDGE model shows substantial reduction in crack growth rate with low conductivity, which all plants maintain. A PLEDGE calculation using bounding assumptions for conductivity and operating time shows that structural margin would be maintained for this group of plants for at least one more cycle of additional operation at current conductivity levels. However, the uncertainty in the residual stress assumptions lessen the certainty of this conclusion. In summary, development of 360°, > 90% deep cracks is unlikely, but cannot be ruled out.*

Even allowing for deep, 360° cracking, the very low probability of a LOCA scenario occurring in the next 3-4 months makes the overall likelihood of extensive cracking and a LOCA so low that there is a good basis for continued operation until the fall 1994 inspection results are available.

The group of plants for which 360° cracking cannot be ruled out consists of some BWR/2, 3 and 4 plants. Even if the extremely unlikely case of a LOCA with 360° through-wall cracking were assumed to occur at a plant in this group, evaluation of the subsequent conditions demonstrates that safe shutdown is achieved for all conceivable LOCA or LOCA plus seismic scenarios. This statement applies to the BWR/3 and BWR/4 plants even when considering the most bounding and conservative criteria for the LOCA. For BWR/2s, the safe shutdown statement applies as well, taking into consideration the high likelihood of detecting H8 weld failure during normal operation, and allowing for realistic evaluations of internal loads from the main steam line event.

Recommendation:

All plants continue operation through the fall of 1994. Some of the plants planning fall refueling outages should do at least some UT inspection to determine the extent and depth of cracking. Plants with 1995 refueling outages should develop planning options that take into account the results of the fall 1994 inspections of plants in this grouping."

Since the Authority is planning inspections and repair of the FitzPatrick shroud during the November 1994 refueling outage, which is three months away, this recommendation for continued operation is applicable.

Appendix A of the BWROG generic evaluation (Reference 1) discusses the safety consequences associated with 360° through-wall cracking for normal, transient and faulted conditions. This evaluation was performed by the BWROG with the intent to encompass the entire fleet of GE BWRs. There are no known aspects of the design, construction, or operation of FitzPatrick which are not bounded in the generic evaluation. Because these evaluations were performed generically, the Authority cannot verify that these evaluations absolutely bound the behavior of the FitzPatrick plant under all conditions. The Authority believes that any potential deviations in the behavior of FitzPatrick from that described in the evaluation are minor and that the conclusions reached are generally applicable to FitzPatrick.

As stated earlier, the Authority has contracted with General Electric to perform a plant-specific analyses for the design basis events (main steam line and recirculation line breaks) to assess the safety implications of having undetected 360° through-wall cracking. This will be submitted to the NRC by October 17, 1994.

ATTACHMENT III to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

INSPECTION PLAN

Requirement 2. (a)

New York Power Authority
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

CORE SHROUD INSPECTION PLAN

(Reporting Requirement 2. (a) - The inspection plan requested in item 3 of Requested Actions).

The current inspection plan is provided below. The Authority is working with the BWROG in the development of a generic inspection criteria document. Guidance from the generic document will be incorporated into the Authority plan when the generic document is approved by the BWROG and submitted to the NRC.

A. Core Shroud Inspection Description

Inspection of the core shroud will be a combination of visual (VT) and ultrasonic (UT) techniques. The combined UT and VT inspections will not provide 100% inspection of all shroud welds. The inspection sample is sufficiently representative to determine whether or not a repair is required. If the Authority determines that the repair will be installed, the Authority may suspend further inspection activities. Additional inspections will be considered based upon access availability to the reactor vessel considering repair activities and impact on outage schedule.

Inspections of the core shroud will be conducted as follows:

Welds H1, H2, H3, and H4
100% UT conducted, where accessible, from the shroud outside diameter. This inspection will examine approximately 80% of the weld length.

Weld H5
UT examination of eight (8) selected one foot lengths at various locations on the shroud outside diameter, typically at intersections of circumferential and vertical welds where residual stresses would be greatest and most likely to exhibit cracking. These locations are based upon access to the shroud avoiding interferences caused by the jet pumps. This inspection will examine approximately 17% of the total weld length.

Welds H6A and H6B
UT examination of two (2) selected one foot lengths on the shroud outside diameter of each weld at the 0° and 180° azimuths locations. These locations are based upon access to the shroud avoiding interferences caused by the jet pumps. This inspection will examine approximately 4% of the total weld length. Between 25% and 45% of the weld will be subject to VT examination as accessible at vessel azimuths other than at 0° and 180°.

Welds H7 and H8
VT inspection on the shroud outside diameter at vessel azimuths of 0° and 180°.

Weld H9 (Shroud Support Plate to Reactor Vessel
10% to 14% VT inspection (3-4 inches on both sides of the ten gusset plates to be used for the proposed shroud repair). This is the intersection of two welds which will have high residual stress and a higher probability for cracking.

Shroud Gusset Plates
Up to 100% VT inspections as accessible of welds attaching ten shroud gusset plates (side attachment to reactor vessel and bottom attachment to shroud support plate). The ten gusset plates are those selected as the bottom attachment location for the proposed repair.

Note: The H8 and H9 welds consist of an inconel 600 plate with Inconel 182 weld metal. These welds were stress relieved with the reactor vessel during fabrication. These welds have not been reported cracked to date.

B. Description of Examination Techniques

A remote visual VT-1 examination will be performed (as defined by ASME Code Section XI, Subsection IWA General Requirements, Sub-article IWA-2210, Paragraph IWA-2211) by qualified Level II and Level III personnel. The video system consists of an underwater camera, lighting, and recording device. The remote visual system will be set up such that it is capable of resolving a 1-mil wire on the inspection surface with the camera placed at distance equal to or greater than that at which the inspection will be performed. Video tapes of the entire examination will be made with the resolution consistent with the remote VT-1 qualification. Prior to any examination activities, the Level II and Level III examiners will receive training and be made familiar with the geometry of the shroud, particularly with those areas scheduled for examination. NYPA is participating with GE in the development of the standards for visual inspections of the core shroud.

Ultrasonic inspections will also be performed. In general, ultrasonic crack detection will be performed using a 45 degree shear wave technique. Sizing will be performed by using shear or refracted longitudinal wave techniques. Sizing of indications will follow EPRI approved sizing techniques.

C. Core Shroud Flaw Evaluation and Screening Criteria

The evaluation and screening criteria (Reference 17) was prepared for the Authority by General Electric. This criteria is consistent with criteria developed by the BWROG (Reference 18). Evaluation of flaw indications will assess whether operation can be justified without installing a repair. However, the Authority may elect to install the repair with crack indications within the screening criteria to reduce the need for future inspection activities.

1. Determination of the Effective Flaw Length

The effective flaw lengths are based in the ASME Code, Section XI proximity criteria as presented in Sub-article IWA-3300. When two indications are close to each other, rules are established to combine them based on proximity. The rules are summarized below.

If two surface indications are in the same plane (perpendicular distance between flaw planes is <3 in.) and are within two times the depth of the deepest indication, then the two indications must be considered as one indication. Combining the crack growth and proximity criteria, the flaws are assumed to be close enough to be considered as one continuous flaw if the ligament is less than (2 x 0.467 + 2 x shroud thickness). For a shroud thickness of 1.5 in., this bounding ligament is 3.93 in. The addition of the 0.93 is to include one cycle crack growth at the other (non-adjacent) end of each flaw. If the ligament is greater than 3.93 in., the effective flaw length, L_{eff} is determined by adding the projected tip growth to each end of the flaw. A similar approach is used to combine flaws when a circumferential flaw is close to an axial flaw.

After the circumferential and axial flaws have been combined per the above criteria, a map of the effective flaws in the shroud can be made, and the effective flaw length can be use for subsequent fracture mechanics analysis.

2. Structural Analysis

The structural analysis for the indications uses two methods; linear elastic fracture mechanics (LEFM) and limit load analysis to determine the allowable flaw sizes in the shroud. Since the limit load is concerned with the gross failure of the section, the allowable flaw length based in this approach may be used for comparison with the sum of the lengths of all flaws at a cross section. The LEFM approach considers the flaw tip fracture toughness and thus, the allowable flaw length may be used for comparison with the largest effective flaw length at a cross-section.

A additional consideration that applies only to the fracture mechanics analysis is the independence of flaws. Even when the two flaws are separated by a ligament that exceed 3.93 in. they may not be totally independent of each other. That is, the flaw tip stress intensity factor may be affected by the presence of the adjacent flaw. There will be no interaction between the two indications if the tips are at least 0.75(L1 + L2) apart. Thus, if the separation distance between two indications is less than 0.75(L1 + L2), for the purpose of fracture analysis, the equivalent flaw length is the sum of the two individual flaws.

3. Screening Criteria

Allowable through-wall flaw sizes were determined using both fracture mechanics and limit load techniques. The allowable through-wall flaws are based on many conservative assumptions and are intended for use only in the screening criteria. More detailed analysis can be performed to accept larger flaws (both through-wall or part through when measured flaw depths can be determined). However, since the intent of the screening criteria is to determine when additional evaluation or NDE characterization is needed, a conservative bounding approach is utilized.

The allowable through-wall flaws are:

- Circumferential Flaws
  - 95 in. using LEFM
  - 272 in. using Limit Load
- Axial Flaws
  - 57 in. using LEFM
  - 191 in. using Limit Load

For circumferential flaws, the fracture mechanics based limit for a single flaw is 95 in. This in itself is not sufficient since there could be several flaws (each less than 95 in.) in a circumferential plane that cumulatively add up to greater than 272 in. (the allowable flaw size based on limit load analysis). Thus, the cumulative flaw length should be less than 272 inches. While this fully assures the ASME code margins, an additional conservative assumption is included in the screening. **The cumulative flaw length cannot be more than 272 / 4 or 68 in. in any 90 degree sector of the shroud.** This is a conservative restriction that assures that long continuous flaws are not acceptable. With this provision, this criteria becomes more limiting than the fracture mechanics limit of 95 in.

4. Summary of Screening Criteria

The first step is to map the flaw indications observed. Next, the proximity rules are applied to the flaw map to develop effective flaw lengths. The results of the effective flaw lengths are also mapped. For circumferential flaws, all flaws are summed in any 90 degree sector using a template. The total flaw length in the 90° window must be less than 68 inches to meet the screening criteria. The next step is the LEFM based comparison using the interaction criteria. If the ligament between adjacent flaws, $S < 0.75\ (L1_{eff}+L2_{eff})$, then the length $L = L1_{eff}+L2_{eff}$ should be compared with the LEFM limit of 95 in.

ATTACHMENT IV to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

REPAIR PLAN

Requirement 2. (b)

New York Power Authority
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

Reporting Requirement 2. (b) Plans for evaluation and/or repair of the core shroud based on the inspection results.

Core Shroud Repair Design Description

The Authority has been working with the BWROG in the development of a generic shroud repair criteria. The design of the FitzPatrick repair meets the current draft BWROG generic shroud repair criteria and will be modified as necessary to agree with the final version. The repair is currently in the design and procurement stage.

The core shroud repair design consists of a series of stainless steel and XM-19 steel tie-rod assemblies which are installed in the shroud-vessel annulus, between attachment points near the top of the shroud (elevation about 395") and the shroud support plate (elevation about 114"). A conceptual design sketch is shown in Figure 2 in Attachment VI. These tie-rod assemblies provide redundant tensile (i.e., vertical) support to the cylindrical part of the core shroud, including its circumferential welds, for both vertical and overturning loads resulting from normal operation, transients, and design accident loads, including seismic and postulated pipe ruptures. This design protects against effects of potential cracking in all of the suspect stainless steel circumferential welds and any combination of welds from the top of the shroud to the bottom (i.e., welds H1 through H7).

The repair assembly design consists of the following main components:

- Ten tie-rod assemblies are planned, with the final number of tie-rods depending upon the detailed design and material choices. These tie-rods consist of an internal XM-19 spring rod within stainless steel pipe sections.

- Each tie-rod assembly is attached at the top end to an austenitic stainless steel bracket member which is supported by the shroud head flange and at the lower end to an existing shroud support plate gusset. The connection at the lower end is made using a special attachment which fits through a hole in a shroud support plate gusset. The hole in the 2-inch thick gusset plate will be made using a qualified EDM process. Each tie-rod assembly is pre-loaded lightly and secured in place to preclude any chance of vibration or becoming loose during operation.

- Each tie-rod assembly contains radial support members near the top mid section and the bottom which rotate and lock into position during installation. These radial supports resist seismic accident loads which could occur if the shroud circumferential welds are cracked through-wall, 360 degrees of the circumference.

- The tie-rods are spaced at approximately equal locations around the periphery of the shroud, between jet pumps. Special attention has been given to avoiding interferences with existing instrumentation.

Attachment IV
JPN-94-043
Response to Generic Letter 94-03
REPAIR PLAN

The proposed shroud repair has been designed to meet the following requirements:

- The ten tie-rod assemblies are designed to meet the load requirements including normal operating loads and differential pressures as well as accident loads associated with a postulated main steam line break, LOCA, and seismic events.

- Design and location of the tie-rod assemblies are such that a large margin against flow-induced vibration is provided. This is accomplished by use of the concentric rod and pipe assembly which increases damping.

- Size and location of tie-rod assemblies and supports are such that there is negligible effect on downcomer flow in the shroud-vessel annulus.

Important design features of the proposed repair approach are as follows:

- The tie-rod assemblies and other parts are all low carbon grade, solution-annealed austenitic stainless steel with no welding, thereby providing high resistance to IGSCC. Fabrication and installation processes will not increase the susceptibility of components to IGSCC.

- The tie-rods are readily removable, if required, with straightforward long handle tools.

- Installation of the tie-rod assemblies is straightforward, with minimum in-vessel work.

- The tie rod design eliminates the need for a time-consuming bolting preload operation in the vessel. The pre-loading of the tie-rod repair can be accomplished by straight-forward means (e.g., a dynamometer and chain-fall).

- All connections and bolting are positively locked using proven locking devices.

- Thermal expansion effects have been accommodated through appropriate sizing and spring constants of the tie-rod assemblies.

- The tie-rod assemblies are installed in the reactor vessel-shroud annulus and minimize interference with vessel and shroud inspections. They are readily removable if, for any reason, additional access is desired.

- As mentioned earlier, the arrangement of ten tie-rod assemblies provides structural redundancy sufficient to protect against 360 degree, through-wall cracks in any circumferential welds and any combination from the top to the bottom of the shroud.

- Installation of the repair is expected to provide a technical basis for substantially reducing the extent of future shroud inspections.

ATTACHMENT V to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

PLANT SPECIFIC PROBABILISTIC RISK ASSESSMENT

New York Power Authority
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333