JAMES A. FITZPATRICK NUCLEAR POWER PLANT

PROBABILISTIC RISK ASSESSMENT of the EFFECTS of POSTULATED EVENTS with 360^{O} SHROUD THROUGH-WALL CRACK

REACTOR ENGINEERING
NUCLEAR SYSTEMS ANALYSIS GROUP

Engineering Report: JAF-RPT-MISC-01788

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Section 1

INTRODUCTION

This probabilistic risk assessment (PRA) addresses the Nuclear Regulatory Commission's (NRC's) Generic Letter 94-03 [Reference 1] request to "perform a safety analysis supporting continued operation of the facility until inspections are conducted" for intergranular stress corrosion cracking of core shrouds in boiling water reactors.

To evaluate this concern for the James A. FitzPatrick Nuclear Power Plant (Fitzpatrick), the Reactor Engineering-Nuclear Systems Analysis Group (NSA) examined the potential occurrence of various initiating events with a 360° shroud through-wall crack. In particular, NSA:

- Determined the core damage frequency for each initiating event.
- Determined the cumulative core damage frequency of all postulated initiating events.
- Estimated the probability for significant shroud displacement, for each initiating event.
- Estimated the conditional probability for control rod insertion failure given shroud displacement for each initiating event.
- Estimated the conditional probability for standby liquid control system performance given shroud displacement and control rod insertion failure for each initiating event.
- Estimated the conditional probability for core spray system performance given shroud displacement for each initiating event.

Based on this evaluation, conclusions were drawn.

Section 2

ANALYSIS

2.1 Modeling Methodology

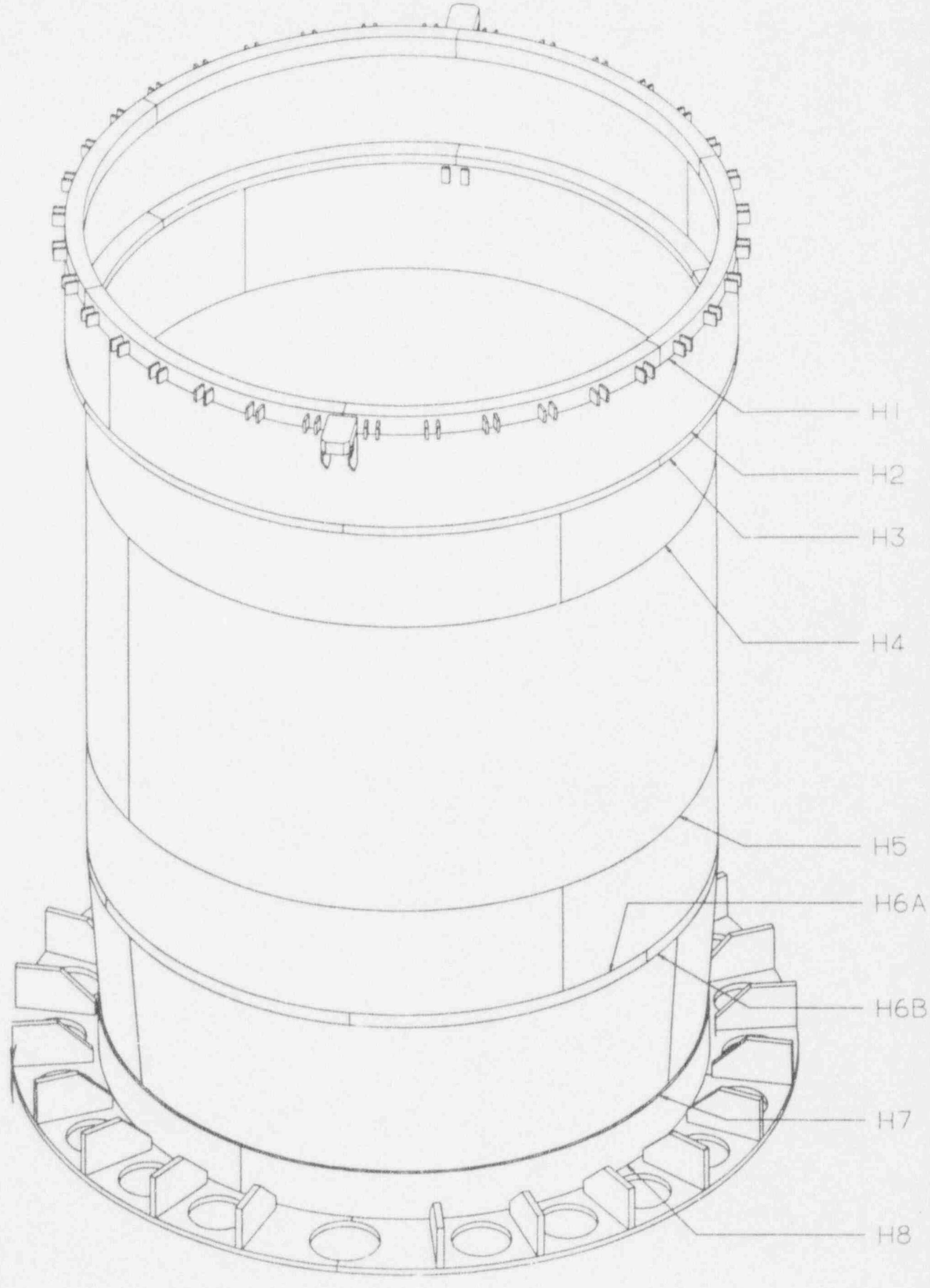
For this PRA evaluation, five limiting initiating events are examined concurrent with a postulated 360° through-wall shroud crack at the H2 to H6A welds. (See Figure 2.1). The potential separation at the H1 location does not affect the capability to insert the control rods or the emergency core cooling systems (ECCS) safety function [Reference 2]. The potential separation at H2 through H6A welds can affect the ECCS safety function (provided shroud displacement of greater than two inches occurs). In addition, shroud separation at the H3 to H6A welds has the potential to affect control rod insertion. However, shroud separation at welds' H6B through H8, will not affect ECCS or control rod insertion [Reference 2]. In performing this PRA, five initiating events are examined. The five initiating events and their respective frequency per year are as follows:

- main steam line break — 1.0×10^{-4} / year
- recirculation line break — 1.0×10^{-4} /year
- design basis earthquake (0.15g) — 9.2×10^{-5} / year
- seismic-induced main steam line break — 7.7×10^{-7} / year
- seismic-induced recirculation line break — 7.7×10^{-7} / year

The values for the recirculation and main steam line breaks are those used in the Fitzpatrick Individual Plant Examination (IPE) [Reference 3]. The seismic events values are based on the revised 1993 Lawrence Livermore National Laboratory (LLNL) seismic hazard curves for Fitzpatrick [Reference 4]. The conditional probability of a seismic-induced large recirculation line break or main steam line break is based on data obtained from NUREG/CR-4550 for Peach Bottom Unit 2 [Reference 5]. The conditional probability is based on seismic-induced failure of the recirculation pump supports. Failure of either recirculation or main steam piping was not included in NUREG/CR-4550, because the seismic loading capacity for these pipe lines are significantly higher than the recirculation pumps supports, therefore, recirculation/main steam line breaks would make a negligible contribution to the initiating event frequency.

The analysis considers the effect of shroud movement during the postulated accident on control rod insertion, reactor core refloodable volume, ECCS availability and standby liquid control system (SLCS) effectiveness.

Figure 2.1
Location of Shroud Welds



2.2 ASSUMPTIONS

- All initiating events occur while the plant is at full power and with a 360° through-wall cracked core shroud.

- Because the Fitzpatrick shroud is fabricated from 304, welded-plate support rings and had a history of early high conductivity; GE concludes, the potential for a 360° shroud crack exists, however, its occurrence is unlikely [Reference 2]. Therefore, different core shroud failure probabilities are selected for each postulated initiating event.

- There is no indication of shroud leakage from control room instrumentation before the postulated initiating event occurring. Therefore, potential operator actions to mitigate the effects of shroud leakage by proceeding to a normal plant shutdown are not considered.

- A shroud displacement of greater than 2 inches is assumed to affect low pressure coolant (LPCI) for welds located below H4, and core spray (CS) ECCS safety function for welds located below H1.

- A postulated main steam line break event could result in significant shroud displacement, and therefore, potentially effect control rod insertion and core spray system function.

- A postulated recirculation line break event could result in significant shroud displacement, and therefore, potentially affect the ability to reflood the reactor vessel to two-thirds core height (ECCS function is precluded). In addition, control rod insertion, SLCS effectiveness, and core spray function are all potentially affected.

- Because the large LOCA break frequency [Reference 3] includes the break frequency for recirculation line break and main steam line break; the same value is used for both main steam and recirculation line break frequency.

- Given successful SLCS performance following a control rod insertion failure, and the unavailability of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC), the failure probability to perform boron mixing with the use of low pressure systems is conservatively estimated to be 0.5 [Reference 6].

- Standby liquid control system operation is precluded (failure probability of 1.0), provided a recirculation line break with 360^0 core shroud through-wall crack and control rod insertion failure occur.

2.3 Fitzpatrick Shroud Through-wall Crack Event Tree

The Fitzpatrick core shroud through-wall crack event tree is shown in Figure 2.2. The top events considered in the event tree are:

IE	Initiating Event.
SHROUD	Probability of core shroud failure.
C	Reactor protection system scram.
C-SHROUD	Shroud movement precludes control rod insertion.
SLCS	Standby liquid control system used for boron injection.
LEVEL-CONTROL	Boron mixing via use of low pressure systems.
CS.	Core spray system vessel makeup.

These events are discussed in more detail as follows:

IE

Initiating events are those disruptions of normal plant operation that cause or require a rapid plant shutdown, with the attendant need to remove heat from the reactor vessel to preclude possible accident sequences leading to core damage. The initiating events shown in the Fitzpatrick core shroud through-wall crack event tree are: main steam line break (MSLB), recirculation line break (RLB), design basis earthquake (SSE), seismic-induced main steam line break (SSE-MSLB), and seismic-induced recirculation line break (SSE-RLB).

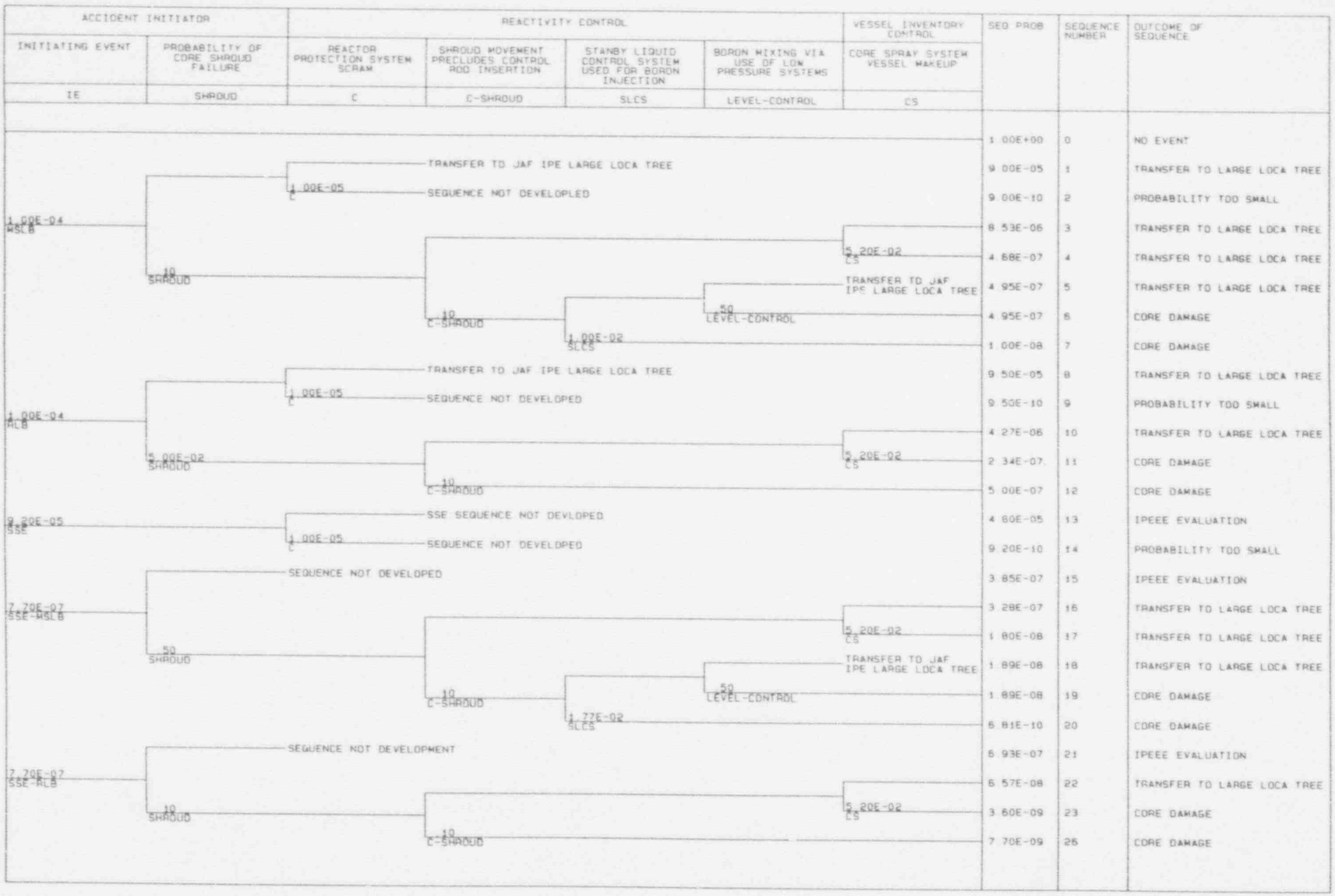
SHROUD

This event defines whether significant shroud displacement occurs, given the occurrence of any one of the above initiating events. The respective core shroud failure probabilities for each of the postulated initiating events are as follows: 0.10 for MSLB, 5.0×10^{-2} for RLB, 0.50 for SSE-MSLB, and 0.10 for SSE-RLB. These values are estimated based on References 2 and 6.

C

This event defines the success of the mechanical insertion of the control rods (successful generation of an electrical scram signal is not considered because alternative means are available). Failure of RPS implies that no control rods can be inserted and, as a result, no

Figure 2.2
Fitzpatrick Core Shroud Through-wall Crack Event Tree



alternate means of mechanically inserting the control rods is available, though reactor subcriticality can be achieved through the use of the standby liquid control system. The probability of a mechanical control rod failure is 1.0×10^{-5} per demand [Reference 3].

<u>C-SHROUD</u>

This event defines whether control rod insertion failure occurs, given significant shroud displacement for a given initiating event. Similar to the C event, failure to insert the control rods is considered a mechanical failure; therefore, rea.tor subcritically can be achieved through the use of the standby liquid control system. The conditional failure probability for this event is 0.1 per demand [Reference 6].

<u>SLCS</u>

This event defines whether the standby liquid control system operation occurs until sufficient boron has been injected into the reactor vessel to achieve and maintain hot shutdown. Failure of the standby liquid control system results in core damage for this analysis (credit for operator action to inject boron using the control rod drive (CRD) system is not considered). The probability of the standby liquid control system failure for non seismic event is 1.0×10^{-2} per demand; it is derived from random mechanical faults [7.7×10^{-3} (Reference 3)] or human error to initiate the standby liquid control system [2.6×10^{-3} (Reference 3)]. The probability during a seismic event is 1.7×10^{-2} per demand; derived from random mechanical faults [7.7×10^{-3} (Reference 3)] or human error to initiate the standby liquid control system [1.0×10^{-2} (Reference 10)].

<u>LEVEL-CONTROL</u>

This event defines whether boron mixing is achieved during an MSLB event with the use of low pressure systems. Failure in this event implies the operator loses control of vessel water level during boron mixing and therefore dilutes the boron concentration. Subsequently reactor shutdown and potential core damage result. The failure probability of boron mixing is 0.50 per demand [Reference 6].

<u>CS</u>.

This event defines whether the use of at least one of two core spray pumps is available to provide sufficient reactor vessel make-up. Failure requires that other reactor vessel make-up systems perform. The probability of core spray system failure is 5.2×10^{-2} per demand; derived from shroud displacement induced core spray faults [5.0×10^{-2} (Reference 6)] or random mechanical faults [2.2×10^{-3} (Reference 3].

Section 3

ANALYSIS

3.1 Main Steam Line Break (MSLB)

A postulated MSLB occurs high (at the main steam nozzle penetration) in the reactor vessel and results in rapid vessel depressurization. Subsequently, a reactor scram and emergency core cooling systems initiate automatically on high drywell pressure (2.7 psig) or low reactor water level [59.5 in. above top of active fuel (TAF)] signals. Although, the HPCI and RCIC systems are both inoperable because of low reactor steam pressure--low pressure systems LPCI and CS rapidly restore reactor vessel water level to above TAF. Operator action ensures the reactor is shutdown and reactor vessel level control within the normal control band is maintained.

Because the postulated MSLB results in the largest lifting loads on the shroud head and lower shroud welds, it has the potential to affect the capability of the control rods to insert and the capability of the core spray system to provide reactor vessel makeup.

From the event tree depicted in Figure 2.2 (sequence 4), the frequency of a postulated MSLB with successful control rod insertion and failure of the core spray system because of shroud displacement is 4.68×10^{-7} /year. However, even though HPCI and RCIC are inoperable because of low reactor steam pressure, low pressure injection systems [e.g., LPCI, residual heat removal service water (RHRSW) cross-tie, and Fire Protection Water cross-tie (FPS)] are available to mitigate the accident, therefore, additional random failures are required for core damage to occur. At this point, the frequency of the sequence is $<10^{-8}$/year; therefore, the sequence would make a negligible contribution toward risk to the public health and safety.

Postulated MSLB with shroud displacement and failure of control rod insertion sequences that result in core damage are sequences 6 and 7, shown in Figure 2.2. Sequence 6, involves successful SLCS performance, however, because HPCI and RCIC are inoperable (low reactor steam supply), boron mixing with the use of low pressure systems fails because of the inability to control level and subsequent born dilution. Without adequate boron mixing, the potential for core damage increases. The core damage frequency (CDF) for sequence 6 is 4.95×10^{-7} /year. This is based on the following: CDF = [1.0×10^{-4} (initiating event frequency)] x [0.1 (probability of core shroud displacement)] x [0.1 (conditional probability of control rod insertion failure given core shroud displacement)] x [0.99 (SLCS success)] x [0.50 (probability of not performing boron mixing)].

Sequence 7, involves SLCS failure to mitigate a failure of control rod insertion given shroud displacement. The probability of SLCS failure is 1.0×10^{-2} /demand. This failure is dominated by random mechanical faults [7.7×10^{-3} (Reference 3)] or human error to initiate

SLCS [2.6 x 10^{-3} (Reference 3)]. The core damage frequency for sequence 7 is 1.0 x 10^{-8} /year. This is based on the following: CDF = [1.0 x 10^{-4} (initiating event frequency)] x [0.1 (probability of core shroud displacement)] x [0.1 (conditional probability of control rod insertion failure given core shroud displacement)] x [1.0 x 10^{-2} (probability of SLCS failure)].

The combined core damage frequency for sequences 6 and 7, is conservatively estimated to be 5.05 x 10^{-7} /year. Therefore, because the predicted core damage frequency is lower than the NRC safety goal of 1.0 x 10^{-4} /year, and in addition, is lower than the Fitzpatrick IPE internal events core damage frequency of 1.92 x 10^{-6} /year, a postulated MSLB with a 360° shroud through-wall crack event will not pose excessive risk to public health and safety.

3.2 Recirculation Suction Line Break (RLB)

A postulated recirculation line break results in rapid depressurization and loss of water inventory in the reactor vessel. Reactor scram and all ECCS will be initiated by high drywell pressure or low reactor water level signals. HPCI and RCIC are both inoperable because of low reactor steam pressure--core reflood will commence once low pressure injection systems start. Similar to the MSLB event, operator action ensures the reactor is shutdown and reactor vessel level restored within TAF (by performing primary containment flooding).

Analyses performed for a postulated RLB with a 360° shroud through-wall crack by GE [Reference 2] and Commonwealth Edison Company [Reference 7] concluded that no shroud displacement will occur (based on calculated blowdown loads), therefore, refloodable core volume, control rod insertion, and ECCS performance are not affected. However, because of the uncertainty in blowdown loads, a conservative assumption of significant shroud displacement is considered. Therefore, the inability to reflood the reactor vessel to two-thirds core height and SLCS effectiveness given control rod insertion failure is assumed. Based on these criteria, two core damage sequences (11 and 12) are postulated. Sequence 11, involves significant shroud displacement and subsequent loss of reflood capability from the use of LPCI, RHRSW and FPS cross-ties. The control rods are inserted into the core, however, core spray system vessel makeup fails, and core damage results. The probability of core spray system failure is 5.2 x 10^{-2} /demand . This failure is dominated by shroud displacement induced core spray faults [5.0 x 10^{-2} (Reference 6)] or random mechanical faults [2.2 x 10^{-3} (Reference 3)]. The core damage frequency for sequence 11 is 2.34 x 10^{-7} /year. This is based on the following: CDF = [1.0 x 10^{-4} (initiating event frequency)] x [5.0 x 10^{-2} (probability of core shroud displacement)] x [0.90 (successful control rod insertion)] x [5.2 x 10^{-2} (probability of CS failure)].

Sequence 12 is similar to sequence 11, except that control rod insertion failure occurs. Although core spray system operation for vessel makeup is possible, core damage occurs, because shroud displacement results in SLCS flow being bypassed, precluding reactor shutdown. The core damage frequency for sequence 12 is 5.0 x 10^{-7} /year. This is based on the following: CDF = [1.0 x 10^{-4} (initiating event frequency)] x [5.0 x 10^{-2} (probability of core shroud displacement)] x [0.1 conditional probability of control rod insertion failure given

shroud displacement)].

The combined core damage frequency for sequences 11 and 12, is conservatively estimated to be 7.34 x 10^{-7} /year. Similar to the MSLB event, the predicted core damage frequency is lower than both the NRC safety goal, and Fitzpatrick IPE internal events core damage frequency. Therefore, a postulated RLB with a 360° through-wall crack event will not pose excessive risk to public health and safety.

3.3 Design Basis Earthquake (DBE)

In case of a DBE of 0.15g magnitude, [frequency of 9.2 x 10^{-5} /year (Reference 4)] with a postulated 360° shroud through-wall crack, shroud displacement could potentially result from the DBE loads. However, without a concurrent MSLB or RLB, the potential shroud displacement is small and is not expected to adversely effect core cooling, control rod insertion, and SLCS performance. (This is based on the NRC evaluation for this event, from Dresden, Unit 3 and Quad Cities, Unit 1 submittal for resolution of core shroud cracking [Reference 7]). Therefore, earthquakes larger than the DBE (hence a lower initiating event frequency) or multiple random systems failure must occur before core damage results. As a result, the shroud-induced CDF for a DBE is a non-dominant event and does not affect public health and safety.

3.4 Seismic-induced Main Steam Line Break (SSE-MSLB)

During a postulated MSLB and DBE with a 360° shroud through-wall crack, the additional loads exerted by the DBE is expected to result in greater shroud displacement as compared to the MSLB event. The greater shroud displacement will increase the likelihood of control rod insertion, SLCS effectiveness and core spray operation failure. However, because main steam line piping is seismically qualified, it is expected to remain intact when exposed to DBE loads, although, the potential of a seismic-induced pipe break cannot be eliminated. From NUREG/CR-4550 [Reference 5], the conditional probability of a seismic-induced recirculation break is 8.4 x 10^{-3}. With a DBE frequency of 9.2 x 10^{-5} /year, the combined DBE and MSLB frequency is 7.7 x 10^{-7} /year. The CDF for seismic-induced MSLB sequences that result in core damage is 1.96 x 10^{-8} /year. Therefore, because the likelihood of a seismic-induced MSLB event is small (as compared to the MSLB or RLB postulated events), the risk associated with this event will not affect public health or safety.

3.5 Seismic-induced Recirculation Line Break (SSE-RLB)

For a postulated RLB and DBE with a 360° shroud through-wall crack, in addition to the impact regarding two-third core height reflood ability (described in RLB event), a failure of control rod insertion and SLCS effectiveness may also occur. However, because a seismic-induced RLB frequency is assumed to be similar to that of a seismic-induced MSLB event (7.7 x 10^{-7}/year), the risks identify with this event is small and well below the MSLB or RLB postulated events (the CDF for seismic-induced RLB sequences that result in core damage is

1.13×10^{-8} /year). Therefore, this event will not affect public health and safety.

3.6 Other Risk Perspective

The above analysis represents a conservative assessment of the core damage frequency of the initiating events of concern combined with a 360° shroud through-wall crack. A more realistic assessment that involves the current perspective on recirculation/main steam line break frequency is also examined.

From NUREG/CR-4792 [Reference 8], the double end guillotine break (DEGB) frequency is estimated as 1.0×10^{-12} /year, excluding the effects of intergranular stress corrosion cracking (IGSCC). The DEGB frequency for IGSCC effects is estimated to be 1.0×10^{-3} /year. However, this value was calculated with no credit for inservice inspection or alternative mitigative actions (e.g., weld overlay, stress improvement, or hydrogen water chemistry control). At Fitzpatrick, an aggressive inspection program as required by NRC Generic Letter 88-01 to preclude the occurrence of recirculation pipe cracking and rupture has been initiated. Cracking detected previously has been repaired or evaluated according to the requirements of NRC Generic Letter 88-01. Furthermore, the hydrogen water chemistry program initiated at Fitzpatrick in 1989 reduces crack initiation and growth. Evidence of the strength of the IGSCC program is documented during the 1992 refueling outage--no new incidence of cracking was found during the NRC mandated IGSCC inspection program. The IGSCC inspection program is documented in Engineering Report JAF-RPT-MULT-01120.
In addition, a recent NRC letter dated 2/14/94 (DSR Number 284200) documenting the NRC's review of the Fitzpatrick IGSCC program on the Recirculation System, concluded that the Fitzpatrick IGSCC program has met the intent of Generic Letter 88-01, Supplement 1.

Because of the constructive action taken to mitigate IGSCC, from references 2 and 9, the large break frequency can be estimated as 7.51×10^{-6} /year. Subsequently, the cumulative core damage frequency of the MSLB and RLB postulated events will be lower. The cumulative core damage frequency of the MSLB and RLB postulated initiating events for this case is presented in Table 3.1.

In addition, because the current Fitzpatrick refueling outage is scheduled to begin November 29, 1994, the recirculation/main steam line break frequency is only 1/4 of the frequency per year. As a result, a 1/4 reduction in break frequency would further lower the predicted CDFs. The cumulative core damage frequency of the MSLB and RLB postulated initiating events for these cases are presented in Table 3.1.

Given the more realistic recirculation/main steam line break frequency and the time remaining to the scheduled refueling outage, the risk contribution to the public health and safety is even lower. Therefore, there is a reasonable basis for continued operation to November 29, 1994.

Table 3.1

Comparison of Core Damage Frequencies

RLB/MSLB Frequency	MSLB (CDF)	RLB (CDF)	DBE (CDF)*	SSE-RLB (CDF)*	SSE-MSLB (CDF)*	Cumulative (CDF)
JAF IPE large LOCA (1.0×10^{-4} /year)	5.05×10^{-7}	7.34×10^{-7}	$<<9.2 \times 10^{-5}$	1.96×10^{-8}	1.13×10^{-8}	1.27×10^{-6}
NUREG/CR-4407 with Updated Operational Experience (7.51×10^{-6} /year)	3.79×10^{-8}	5.52×10^{-8}	$<<9.2 \times 10^{-5}$	1.96×10^{-8}	1.13×10^{-8}	1.25×10^{-7}
JAF IPE large LOCA Frequency Reduced by Time to Refueling Outage (2.5×10^{-5} /qtr)**	1.26×10^{-7}	1.83×10^{-7}	$<<2.3 \times 10^{-5}$	4.90×10^{-9}	2.83×10^{-9}	3.17×10^{-7}
NUREG/CR-4407 Reduced by Time to Refueling Outage (1.9×10^{-6} /qtr)**	9.60×10^{-9}	1.39×10^{-8}	$<<2.3 \times 10^{-5}$	4.90×10^{-9}	2.83×10^{-9}	3.15×10^{-8}

* Core damage frequency for design basis earthquake (DBE), design basis earthquake induced recirculation line (SSE-RLB), or main steam line break (SSE-MSLB) is not affected by the change in recirculation line break frequency.

** Break frequency reduced by 1/4, to reflect time remaining to next refueling outage (August 25, 1994 to November 29, 1994).

Section 4

CONCLUSIONS

From this PRA analysis the conclusions are as follows:

- All five initiating events examined have predicted core damage frequency 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, a 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.

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

Section 5

REFERENCES

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[2] BWR Shroud Cracking Generic Safety Assessment, GENE-523-A-107P-0794, July 1994 Revision 0 and August 1994 Revision 1).

[3] New York Power Authority, "James A. Fitzpatrick, Individual Plant Examination," August 1991.

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[5] J.A. Lambright, et al., "Analysis of Core Damage Frequency: Peach Bottom, Unit 2, External Events," prepared by Sandia National Laboratories for the United States Nuclear Regulatory Commission, NUREG/CR-4550, SAND86-2084, Vol. 4, Rev. 1, Part 3, August 1990.

[6] New York Power Authority, "Indian Point 3 Nuclear Power Plant, Individual Plant Examination," June 1994.

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[8] G.S. Holman and C.K. Chou, "Probability of Failure in BWR Reactor Coolant Piping," prepared by Lawrence Livermore National Laboratory for the United States Nuclear Regulatory Commission, NUREG/CR-4792, UCID-20914, Volume 1: Summary Report, March 1989.

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[10] A. D. Swain, "Accident Sequence Evaluation Program--Human Reliability Analysis Procedure," prepared by Sandia National Laboratories for the United States Nuclear Regulatory Commission, NUREG/CR-4772, SAND86-1996, February 1987.

ATTACHMENT VI to JPN-94-043

Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

REFERENCES

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

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ATTACHMENT VII to JPN-94-043

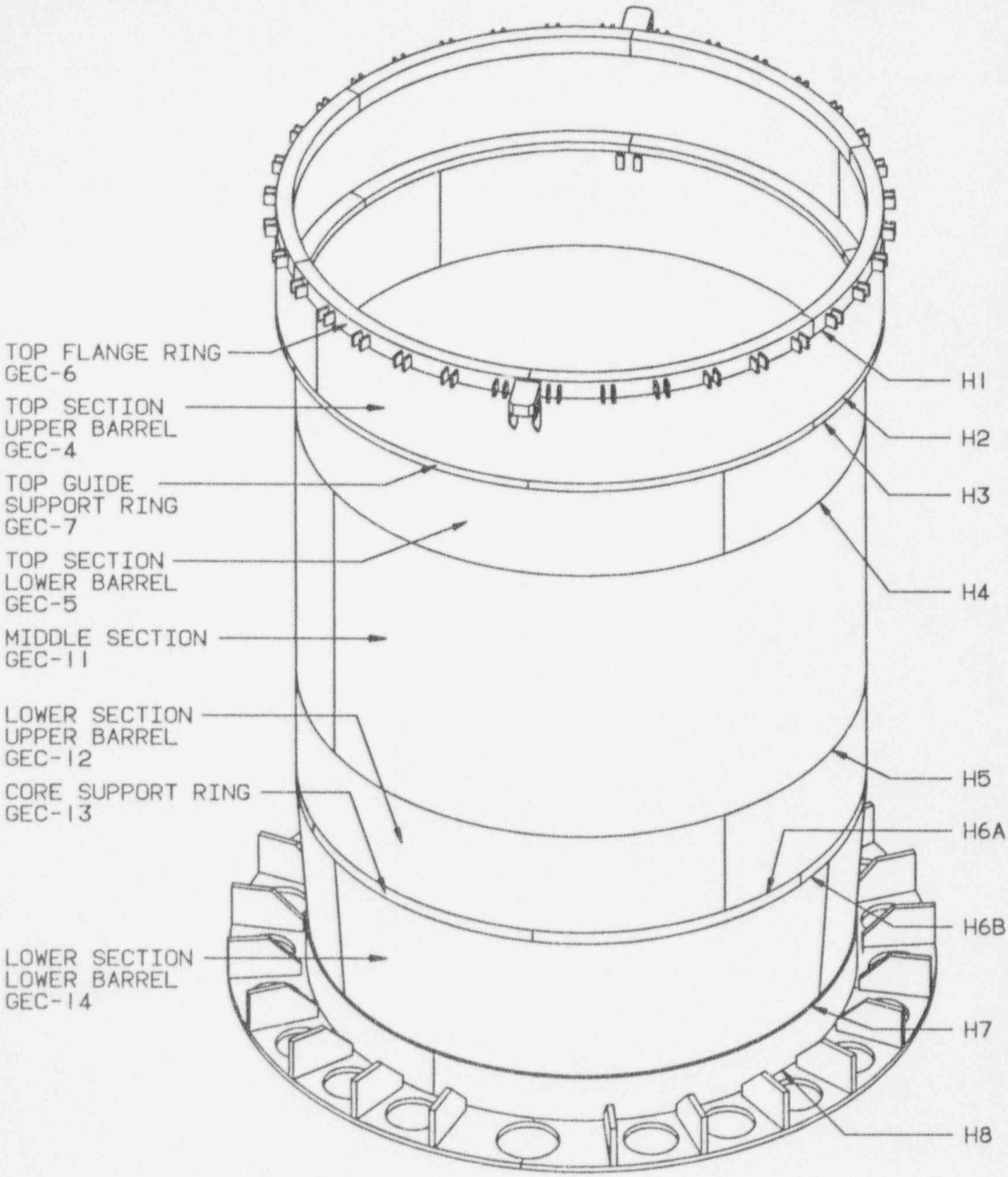
Generic Letter 94-03

Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

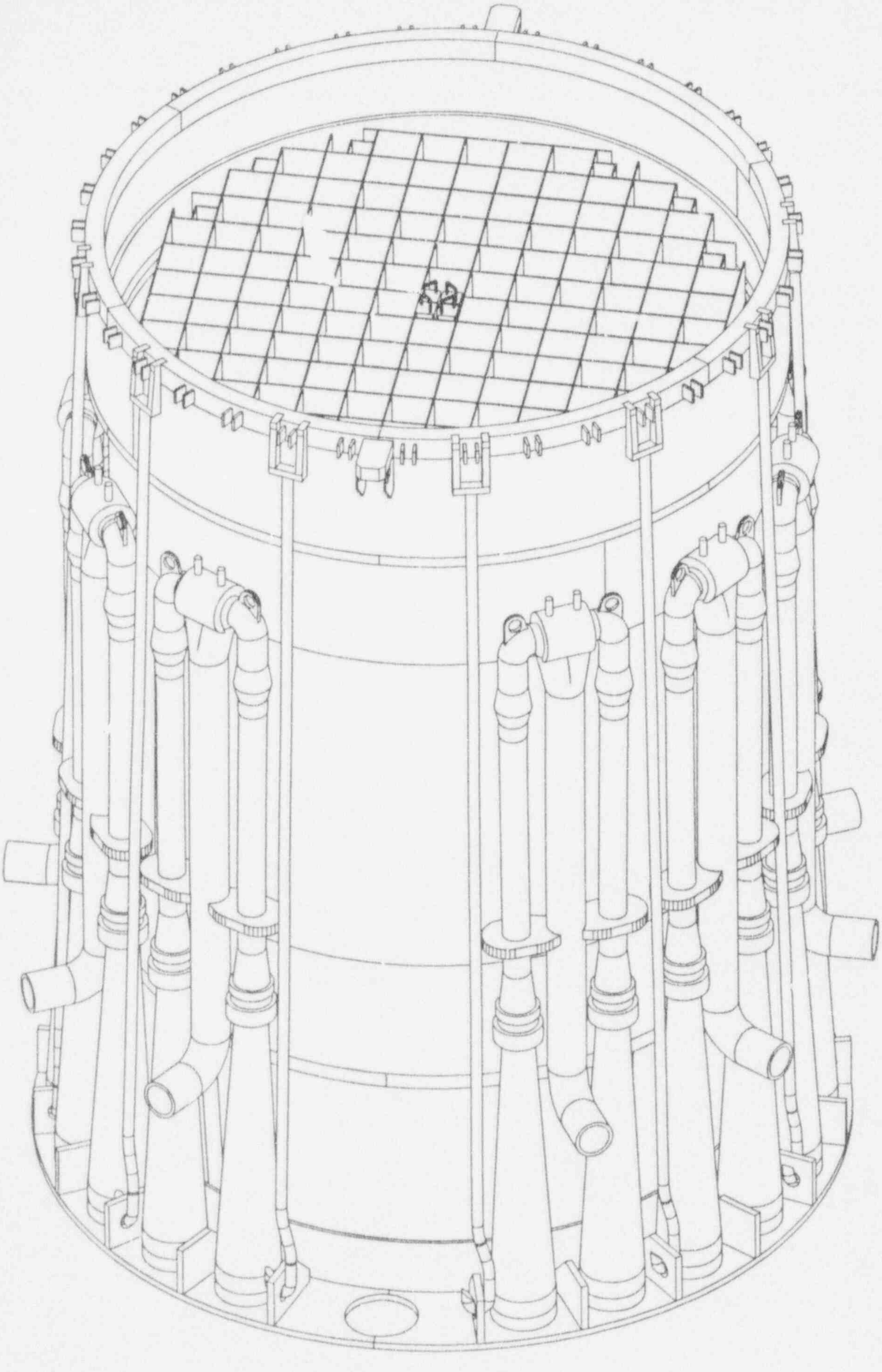
DRAWINGS OF THE CORE SHROUD SUPPORT CONFIGURATION

Requirement 1. (c)

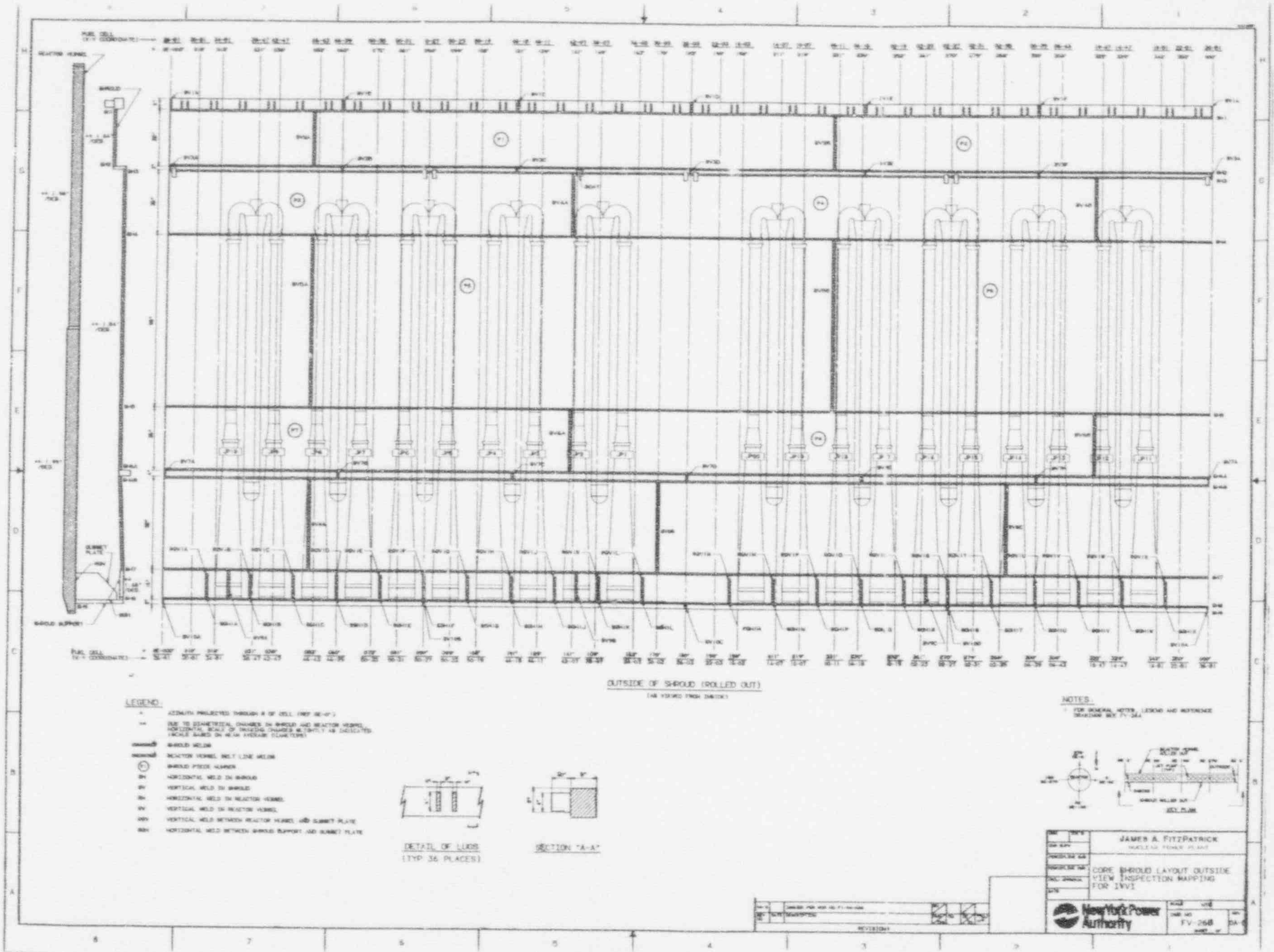
New York Power Authority
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333



JAMES A. FITZPATRICK
NUCLEAR POWER PLANT
SHROUD SHOWING WELDS H1 THROUGH H8



JAMES A. FITZPATRICK
NUCLEAR POWER PLANT
PROPOSED REPAIR TO REACTOR SHROUD



OUTSIDE OF SHROUD (ROLLED OUT)
LEGEND:
NOTES:
DETAIL OF LUGS
(TYP 36 PLACES)
SECTION "A-A"
JAMES A. FITZPATRICK
CORE SHROUD LAYOUT OUTSIDE VIEW INSPECTION MAPPING FOR IVVI
New York Power Authority
FV-268