Fitzpatrick Nuclear Station Shroud Safety Assessment

October 27th, 1994

Prepared By: Renée DuBord
Renée M. DuBord, Engineer
Engineering & Licensing Consulting Services

José L. Casillas
José L. Casillas, Principal Engineer
Engineering & Licensing Consulting Services

Approved By: S Ranganath
Sampath Ranganath, Ph.D., Projects Manager
Engineering & Licensing Consulting Services

GE Nuclear Energy
San Jose, CA

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

The only undertakings of General Electric Company respecting information in this document are contained in the contract between the Power Authority of the State of New York and General Electric Company, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than the Power Authority of the State of New York, or for any purpose other than that for which it is intended, is not authorized: and with respect to any unauthorized use, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

Table of Contents

List of Figures

List of Tables

1. INTRODUCTION

Generic Letter (GL) 94-03 requires BWR utilities to provide a plant specific safety assessment supporting continued operation pending a complete inspection of their core shrouds. The GL also requires that, among other items, this assessment include a safety evaluation considering Main Steam Line Break (MSLB) and Recirculation Line Break (RLB), and an assessment of the plant response during MSLB and RLB, i.e., Control Rod Insertion and ECCS injection.

This report provides the safety evaluation of the shroud assuming one 360 degree through-wall crack at any circumferential weld. Specifically, it provides information on the response to a postulated steam line and recirculation line break and an assessment of the operability of the plant safety features (e.g., control rod insertion). Additionally, it includes sensitivity analyses regarding operation during power coastdown and operation at power uprate conditions. Finally, the report provides overall conclusions concerning plant safety.

The analysis described here is plant unique and is based on Fitzpatrick specific evaluations.

2. SHROUD SAFETY ASSESSMENT

This report contains the results of the plant specific evaluation considering the impact of core shroud cracks. The evaluation conservatively assumes through thickness fully circumferential cracking at any one particular weld location. Figure 2-1 shows the weld locations, along with their designations for the Fitzpatrick plant. In subsection 2.1 are the results of the calculated shroud displacement as a result of plant operations. These operations are classified into normal operation, anticipated operational events, and design basis accidents. In subsection 2.2 are the results of the impact of the shroud condition on the plant safety systems and analyses. The evaluated safety functions are the SCRAM, SLCS, and ECCS capabilities. Subsection 2.3 provides sensitivity analyses at different operating conditions.

2.1 SHROUD RESPONSE TO PLANT OPERATIONS

2.1.1 NORMAL OPERATION

If it is postulated that the shroud may be sufficiently cracked at any of the horizontal weld locations, such that an upward load may cause the upper portion of the shroud to lift, anomalous core characteristics resulting from the flow through the gap will be detected. This anomaly will be the result of reduced moderation in the core due to either increased coolant temperatures or reduced coolant flow. An increased coolant temperature will be the result of flow escaping to the outside shroud region through shroud separation at locations above the fuel top guide (e.g. H1, H2, and possibly H3), where two phase coolant is present. For example, considering a 360 degree one quarter inch gap, the leakage flow is calculated to be approximately 3% of rated core flow. The resulting thermal power loss would also be 3% of rated. This impact on core power is about twice the normal instrumentation uncertainty of 1 to 2% of full power and would therefore be readily detected.

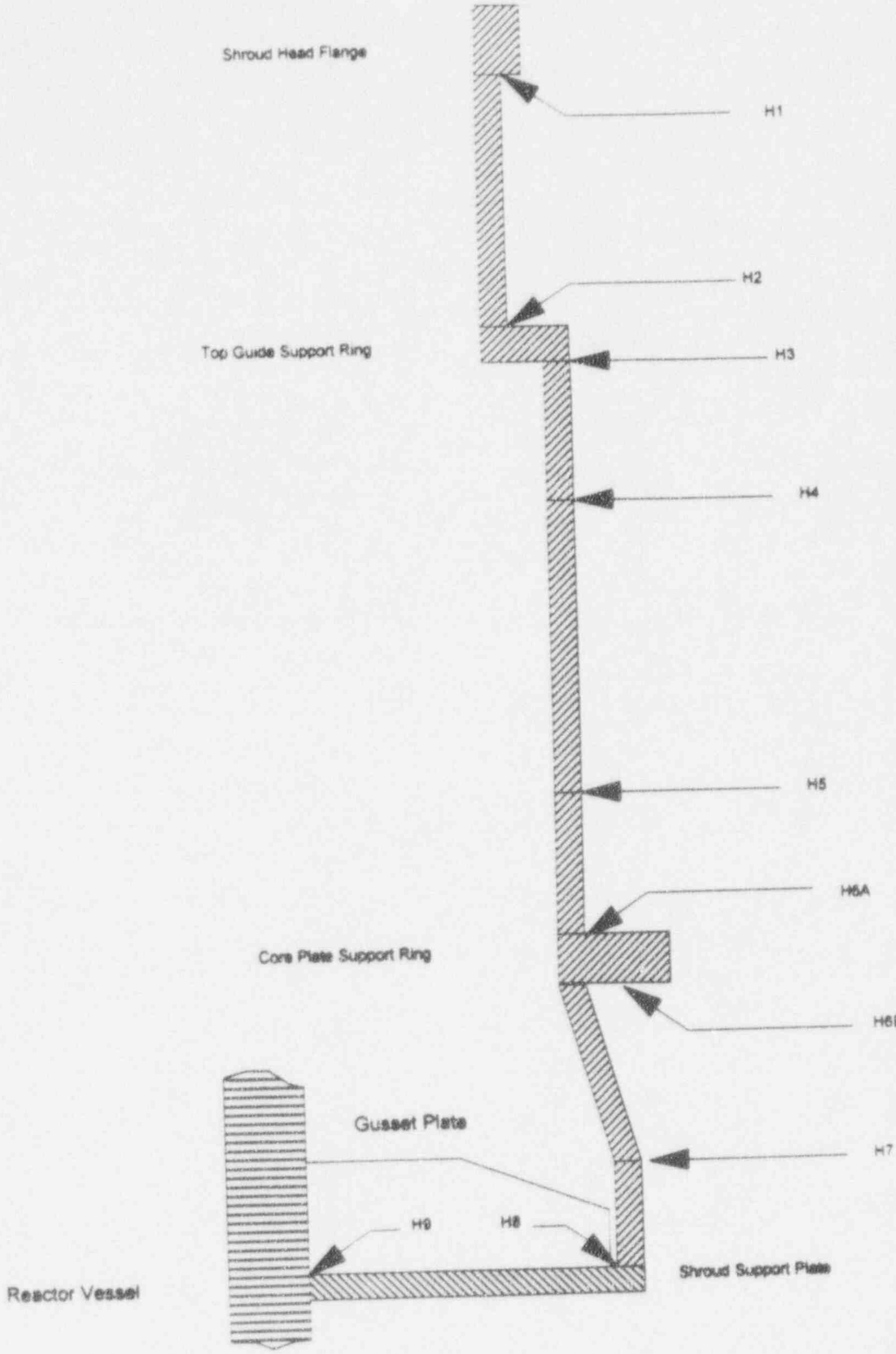

Figure 2-1: Shroud Weld Locations

A decreased coolant flow will be the result of flow escaping to the outside shroud region through shroud separation at locations below the fuel top guide (e.g. H3 through H7), where subcooled coolant is present. For example, considering a 360 degree one quarter inch gap below the core support plate (e.g., H6b, H7, and H8), the leakage flow is calculated to be 9% of rated core flow. The resulting thermal power loss would be 6% of rated. This impact on core power is about four times greater than the normal instrumentation uncertainty of 1 to 2% of full power. These magnitudes of power anomalies will be detected and will result in a plant shutdown. Also present will be other significant abnormal core monitoring indications, such as measured core support plate pressure difference vs. core flow, and measured recirculation flow vs. core flow.

Analogous situations have previously been observed in BWRs. In 1984, a plant began startup with shroud head bolts improperly engaged, resulting in bypass flow paths similar to those that would result from through-wall cracking of the shroud. A similar situation also occurred at a different plant in 1991. In both cases, anomalies such as those described above were detected and the operators shut the plant down.

With respect to a vertical shroud displacement, the two key locations for pressure difference are the shroud head and the shroud support. The shroud head pressure difference applies an upward load at weld locations above the core support plate. The shroud support pressure difference applies an upward load at weld locations below the core support plate. Under normal operating conditions, at 100% power and flow, the pressure difference across the shroud is calculated to be a maximum of 7.2 psi at the shroud head, and 31.4 psi at the shroud support. Listed in Table 2-1 are the approximate pressure differences at which separation is expected for the various weld locations, as well as the maximum pressure difference and separation calculated for each location at normal operating conditions of 100% power and flow. As can be seen from the data, the welds located farther down do not experience as large of a separation. This is the result of the higher shroud weight resting on them. This maximum separation assumes that the upper shroud section suddenly breaks off the bottom section. This separation is greater than the separation that would occur if the break occurred more slowly, or occurred prior to the full power/flow operation. The key weights used in the calculations for the maximum shroud separation are given in Table 2-2.

Table 2-1: Maximum Shroud Separation Under Normal Operating Conditions

Weld Location	Separation DP (psi)	Maximum Operating DP. (psi)	Maximum Separation (inches)
H1	3.9	7.2	5.0[1]
H2	4.3	7.2	3.9[1]
H3	5.3	7.0	1.4[2]
H4	5.7	7.0	0.9[2]
H5	6.7	7.0	0.4[2]
H6a	7.0	7.0	0.0[2]
H6b	7.8	31.4[4]	0.5[3]
H7	9.0	31.4[4]	0.5[3]
H8	9.2	31.4[4]	0.5[3]

[1] Separation at H1 or H2 can not affect core geometry.
[2] Core alignment is assured if the separation of welds H3 to H6a is less than 15 inches.
[3] This separation is limited by the clearance between the core support plate and the top edge of the control rod guide tubes.
[4] This pressure differential is an upper bounding number for all current fuel designs.

Table 2-2: Key Shroud Weights

Component	Dry Weight (kips)	Submerged Weight (kips)
Shroud Head and Separators	101.2	96.8*
Shroud Studs and Guide Rods	14.3	12.9
Core Spray	2.9	2.6
Top Guide	10.7	9.6
Shroud	77.2	69.5
Core Plate	20.7	18.6
Jet Pumps	16.3	14.7

*The separators are conservatively assumed to be submerged.

The vertical separation for the H1 weld will not be obstructed by any other vessel components. However, some interference is expected for the other weld locations by the Core Spray piping penetrating to the inner shroud region. This interference is not strong because the pipe coupling allows some displacement. For purposes of this evaluation, it is conservatively assumed that the obstruction does not affect the amount of separation. For welds below the top guide (H3 to H6a), proper alignment of the core is assured if the separation is less than 15 inches, as the top guide would need to lift 15 inches to lose contact with the top of the fuel channels. For weld locations below the core support plate (H6b, H7 and H8), the separation is limited to one half inch due to the interference of the control rod guide tubes. The guide tubes are assisted by the weight of the fuel support casting and the fuel in limiting the maximum lift to only one half inch.

As shown above, the initial indications will occur as a function of shroud pressure difference and will be apparent at core flows as low as 50% of rated. For example, separation at H6b and H7 welds may become apparent when the core flow exceeds 50% of rated. However, no separation at the H6a weld will occur even at core flows of 100% of rated. Also, once the conditions for separation exist, a gap will develop such that significant flow will escape to the outer shroud region. The calculated displacements shown above demonstrate that in the event of a 360 degree through-wall circumferential shroud crack during normal operation, the plant operators will be able to detect the abnormal conditions and proceed with a normal shut-down. This is true for all near rated flow operation at all welds, except for the H6a location. At the H6a weld location, no separation occurs, and the tight gap present at the crack location has no consequence for the plant operation.

2.1.2 ANTICIPATED OPERATIONAL EVENTS

Assuming there are no indications of shroud leakage during normal operation, this section discusses the possible impact on anticipated operational events on the shroud condition. For this evaluation it is assumed that the shroud is sufficiently cracked at any of the horizontal welds, such that an increased upward load may cause the upper portion of the shroud to lift. Two types of events are evaluated, first those which are considered limiting events for the Fitzpatrick plant, and then those which impose highest loads on the shroud. The limiting events may not be affected significantly by the condition of the shroud, however, since these events determine the maximum fuel overpowers and consequently produce the minimum margin to fuel thermal limits (e.g. Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR)) and vessel pressure limits, they have the greatest potential to impact safety limits. The highest shroud load events may not lead to limiting conditions, however they may determine the maximum shroud displacement and consequently have the greatest potential to affect the shroud functions.

A total of seven unique limiting anticipated operational events were analyzed for the Fitzpatrick plant for the cycle 11 (Reference 2-1). These events are the Feedwater Controller Failure (FWCF), the Loss of Feedwater Heating (LFWH), the Generator Load Rejection without Bypass (LRNBP), the Fuel Loading Error (FLE), the Inadvertent High Pressure Coolant Injection (HPCI), Control Rod Withdrawal Error (RWE), and the Main Steam Isolation Valve Closure with High Flux Scram (MSIVF). The most limiting of these events are characterized by a rapid pressure increase resulting in a core overpower condition. The event does not result in an appreciable increase in core flow or steam flow through the steam separators. Therefore, no increase in shroud loads is predicted and shroud separation is not expected (e.g., since no separation exists prior to the event and load is not increased during the event, the shroud is not expected to separate during the event). Thus the results of the current analyses remain unchanged and no impact on safety limits exists.

Two Safety Analysis Report (SAR) anticipated operational events and the limiting infrequent operational event are identified which result in the highest shroud loads. These are the pressure regulator failure - open, and the recirculation flow control failure - increasing to maximum flow. The inadvertent actuation of the Automatic Depressurization System (ADS) is also analyzed. These events are discussed and evaluated in the following sub-sections

2.1.2.1 Pressure Regulator Failure - Open

This postulated Safety Analysis Report (SAR) event involves a failure in the pressure controls such that the turbine control valves and the turbine bypass valves are opened as far as the maximum combined steam flow limit allows. For the Fitzpatrick plant the bypass capacity is 25% of rated steam flow, and the additional turbine control valve capacity is 7% of rated steam flow. Therefore, the maximum steam flow during the event is about 132% of rated. The increase in shroud loads due to this event is bounded by the Main Steam Line Break, and shroud lift is bounded by the values listed in section 2.1.3. Because the fuel remains properly aligned, core geometry is maintained and successful scram assured.

2.1.2.2 Recirculation Flow Control Failure

This postulated SAR event involves a recirculation control failure that causes all recirculation loops to increase to maximum flow. In this type of case, the upward pressure will change from a part-load condition to the high/maximum system flow capability condition over a time period of about 30 seconds. The increased lifting forces are bounded by the Pressure Regulator Failure discussed in Section 2.1.2.1. However, because this event results in increasing core power (instead of decreasing power as for the Pressure Regulator Failure event), fuel thermal limits are affected. Shroud separation at the upper welds will lead to greater thermal margins because the hotter coolant will limit core power increase. Shroud separation at the middle and lower welds will not affect the event as core power will correspond to the core flow which successfully enters the core and increases reactivity. In this case the shroud separation leads to core flow leakage, making the power increase less severe. Because the fuel remains properly aligned, core geometry is maintained and successful scram assured. The consequences of this event meet the applicable licensing criteria.

2.1.2.3 Inadvertent Actuation of ADS

Inadvertent actuation of the ADS valves is also an event that increases load on the shroud. The maximum steam flow and the depressurization rate are bounded by the main steamline break. This event results in a short-term increase in the steam flow to 150% of rated steam flow (based on seven ADS valves) for the Fitzpatrick plant. The increase in the shroud ΔP resulting from the opening of the ADS valves would occur over a period of about one second, as compared to 0.5 seconds for a MSLB, spreading the effect of the change in load. Table 2-3 shows the shroud separation due to an inadvertent ADS actuation based upon the original Design Basis Calculations. Both other events (e.g. the Pressure Regulator Failure - Open and the Recirculation Flow Control Failure) are bounded by these results. The increase in shroud loads due to this event is bounded by the Main Steam Line Break, and shroud lift is bounded by the values listed in section 2.1.3. Because the fuel remains properly aligned, core geometry is maintained and successful scram assured, and therefore, the conclusions of Section 2.1.2.1 are also applicable for this event.

Table 2-3: Maximum Shroud Separation Due to Inadvertent ADS Actuation

Weld Location	Maximum DP (psi)	Maximum Separation (inches)
H1	13.0	15.1[1]
H2	13.0	13.0[1]
H3	13.0	9.0[2]
H4	13.0	7.9[2]
H5	13.0	5.6[2]
H6a	13.0	4.9[2]
H6b	41.8	0.5[3]
H7	41.8	0.5[3]
H8	41.8	0.5[3]

[1] Separation at H1 or H2 can not affect core geometry.
[2] Core alignment is assured if the separation of welds H3 to H6a is less than 15 inches.
[3] This separation limited by the clearance between the core support plate and the top edge of the control rod guide tubes.

2.1.3 DESIGN BASIS ACCIDENTS

Two Design Basis Accidents (DBAs) are evaluated: a Main Steam Line Break (MSLB) and a Recirculation Line Break (RLB). The MSLB inside primary containment is the worst case because it results in the most severe depressurization. During this event, the reactor is rapidly depressurized as a result of a postulated double-ended break of the main steamline. Thus a maximum pressure difference develops across the shroud as fluid flow is drawn from the core region toward the break. The MSLB imposes the largest lifting loads on the shroud head and shroud support, and has the greater potential to defeat the shroud functions. The RLB does not impose large pressure drops on the shroud, and in fact the shroud pressure drop decreases from its initial value. However, this break results in maximum fuel temperatures, and consequently challenges the Emergency Core Cooling System (ECCS) functions to a greater degree. Additionally, the RLB imposes lateral forces on the shroud.

The potential for shroud displacement is increased if a seismic event is considered coincident with the DBA. While a seismic event coincident with a DBA is beyond the design basis for the Fitzpatrick plant, the following evaluates accidents both with and without the seismic loads from a safe shutdown earthquake.

2.1.3.1 Main Steam Line Break

For this evaluation, the MSLB was calculated for the Fitzpatrick plant using the TRACG model. This model is a best-estimate computer program for the analysis of Boiling Water Reactors (BWRs). TRACG is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model. The two-fluid model used for the thermal hydraulics solves the conservation equations for mass, momentum and energy for both the gas and liquid phases. The thermal- hydraulic model is a multi-dimensional formulation for the vessel component and a one-dimensional formulation for all other components. The conservation equations are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface, as well as at the flow surface boundary. The TRACG structure is based on a modular approach. The thermal-hydraulic model contains a set of basic components, such as pipes, valves, tees, fuel channels, and vessel. Additionally, TRACG contains a control system model capable of simulating the major BWR control systems such as the pressure and water level controllers. Reactor simulations are performed by constructing a model using the basic components as building blocks. Any number of these components may be combined. The number of components, their interaction, as well as the detail in each component, are specified through code input. Therefore, TRACG has the capability of accurately simulating most BWR phenomena. Additional details on the TRACG model are provided in Reference 2-3, and information on the model qualification is documented in Reference 2-4.

The TRACG model prepared for the Fitzpatrick plant consists of a reactor vessel, which is divided into sixteen axial levels and four radial rings. This nodalization was selected based on TRACG qualification results, as well as certain refinements necessary to accurately simulate the performance of the Fitzpatrick plant to a MSLB event.

The initial conditions used for the MSLB calculation correspond to the most limiting core power and flow operating condition. Several event characteristics, which are normally conservatively ignored or simplified, were factored into this calculation. Most of these characteristics are facilitated by the TRACG model capabilities, and others were specifically determined for the Fitzpatrick plant. The Fitzpatrick TRACG MSLB inputs are given in Table 2-4.

Table 2-4 Fitzpatrick TRACG MSLB Inputs

Key Initial Conditions:	**Normal**	**Power Uprate**
Core Power	2436 MWTh	2536 MWTh
Core Flow	77 Mlbs/hr	77 Mlbs/hr
Vessel Steam Flow	10.47 Mlbs/hr	10.98 Mlbs/hr
Dome Pressure	1020 psia	1055 psia
Turbine Pressure	970 psia	1000 psia
Feedwater Temperature	420 Deg F	424 Deg F
Shroud Head DP	7.2	7.2
Shroud Support DP	31*	31*
Normal Water Level	554 above vessel zero	554 above vessel zero
Pump Flow	33.0 Mlbs/hr	33.0 Mlbs/hr
Pump Speed	1437 rpm	1437 rpm
Key MSLB Characteristics:		
Steam Line Diameter	21.564 in	
Recirc Line Diameter	25.867 in	
Vessel Steam Line Safe End	2.536 sq ft	
Stm Line Flow Limiter Area	0.6783 sq ft	
MSIV Closure	5.5 sec	

*This pressure differential is an upper bounding number for all current fuel designs

The results of this calculation for normal and power uprate conditions are shown in Figure 2-2 and indicates a maximum pressure difference of 15.8 psi for the shroud head. The pressure difference for the shroud support is also higher than for normal operation, however, the separation is limited by the control rod guide tubes. This calculation shows the lifting load to last less than three seconds. Listed in Table 2-5 are the maximum pressure differences along with separations calculated for each weld.

The magnitude of these separations for MSLB is somewhat greater than those shown in section 2.1.1 for normal operation. As before, some interference is expected for the weld locations H2 through H6a by the Core Spray piping penetrating to the inner shroud region. This interference is not strong because the pipe coupling allows some displacement. For purposes of this evaluation, it is conservatively assumed that the obstruction does not affect the amount of separation. For weld locations below the top guide (H3 to H6a), proper alignment of the core is assured if the separation is less than 15 inches, as the top guide would need to lift 15 inches to lose contact with the top of the fuel channels. For the Fitzpatrick plant, the maximum lift of the top guide is calculated to be 11.5 inches at the H3 weld location, and thus core alignment is assured. For weld locations below the core support plate (H6b to H8), the separation is limited to one half inch due to the interference of the control rod guide tubes. The guide tubes are assisted by the weight of the fuel support casting and the fuel in limiting the maximum lift to only one half inch. Therefore, the impact on the core internals for a MSLB is the same as for normal operation.

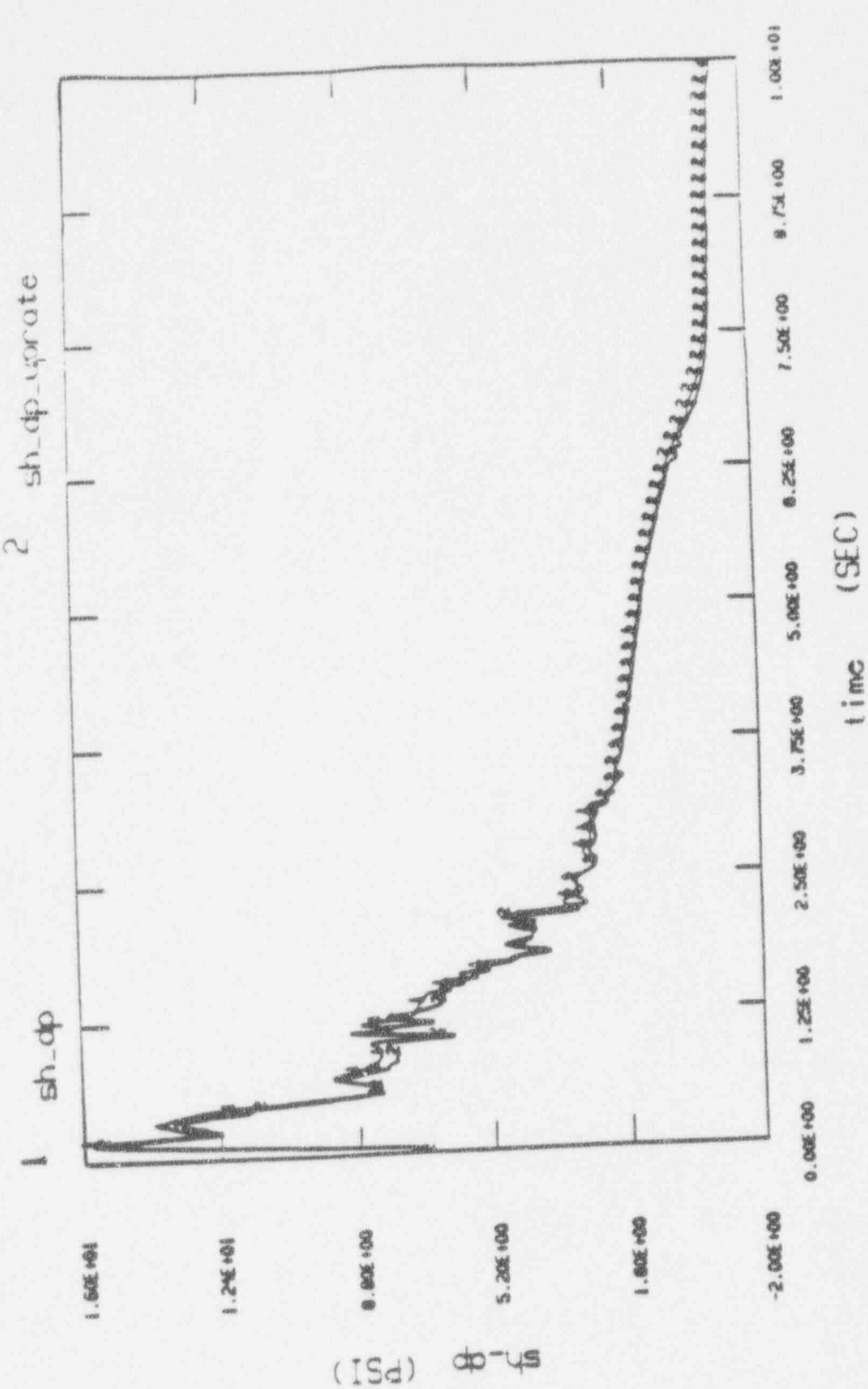

Figure 2-2: Shroud Head DP for MSLB at Normal and Power Uprate Conditions

Table 2-5: Maximum Shroud Separation Under MSLB Conditions

Weld Location	Maximum DP, (psi)	Maximum Separation, (inches)
H1	15.8	18.5[1]
H2	15.8	16.1[1]
H3	15.8	11.5[2]
H4	15.8	10.2[2]
H5	15.8	7.4[2]
H6a	15.8	6.6[2]
H6b	38.9	0.5[3]
H7	38.9	0.5[3]
H8	38.9	0.5[3]

[1] Separation at H1 or H2 can not affect core geometry.
[2] Core alignment is assured if the separation of welds H3 to H6a is less than 15 inches.
[3] This separation limited by the clearance between the core support plate and the top edge of the control rod guide tubes.

A seismic event coincident with a MSLB may result in additional vertical and/or lateral displacement of the upper shroud portion which becomes detached at a weld location, with respect to the lower shroud section. Additionally, after a short duration lift, once the upper shroud portion again rests on the lower shroud portion, the lateral seismic loads apply a shroud tipping moment.

Calculations performed to simulate the possible shroud displacement due to the seismic loads, for the lifting portion of the event, result in less than 1.0 inch additional vertical displacement. This displacement is only temporary, as the upper shroud portion is expected to return to rest on the lower shroud portion after a few seconds. Even if a design basis seismic event is postulated along with the MSLB the maximum vertical displacement is 11.5 + 1.0 = 12.5 inches, which is still less than the 15 inches needed to cause separation of the top guide from the fuel.

Corresponding horizontal calculations also show that the maximum lateral relative displacement of the shroud with respect to the vessel at the core plate is less than 1.0 inch. A portion of this displacement may be permanent as the upper shroud may not be perfectly aligned when it returns to rest on the lower shroud. For this lateral displacement determination, the limiting seismic condition is characterized by the low frequency content of the seismic excitation. Under these assumptions, the maximum relative displacement of the core plate with respect to the vessel during MSLB does not exceed 1.0 inch. Recent dynamic Control Rod Drive insertion tests have shown that the scram function is assured if the displacement at the core plate location is 1.5 inches or less. For the top guide location, the allowed displacement can be up to 4.8 inches. For the seismic higher frequency content, the maximum displacement during the complete seismic event is also less than 1.0 inch. This lateral displacement represents the maximum relative motion of the shroud with respect to the vessel.

For the portion of the event when the upper shroud rests on the lower shroud, the lateral seismic loads apply a tipping moment on the upper shroud. The maximum seismic tipping moment is experienced at the H6a weld location. However, since the restoring moment of the shroud weight is greater, no tipping or rotation will occur.

Therefore, a seismic event in conjunction with a MSLB does not result in a condition which will impact either the core geometry or the scram function. The added displacement calculated for the seismic event does not result in any additional reactor components being affected.

2.1.3.2 Recirculation Line Break

For the RLB, the differential pressure across the shroud decreases from the initial value as the core flow is reduced due to the break and upward forces are reduced. Thus there is no significant threat to core shroud integrity. Any initial shroud separation along a particular weld location will be limited to a tight crack, since any significant separation, prior to the accident, would be detected during normal operation as discussed in section 2.1.1.

Lateral forces for weld locations in the beginning of a RLB are large acoustic forces of short duration, about three milliseconds, followed by smaller blowdown forces for several seconds. Horizontal motion is not expected because of the resistance of the irregular crack surface to horizontal motion without lifting. If sufficient lifting occurs prior to the accident, it will be detected during normal operation as discussed in section 2.1.1. No significant tipping (i.e., rotation) is expected from the acoustic loading because it is of very short duration. The acoustic load calculated for the shroud is conservatively applied to the rotation.

A calculation of the blowdown force for the Fitzpatrick plant results in the limiting tipping occurring at the H6a location, since motion at welds H6b-H8 is limited by the control rod guide tubes. Tipping moments at each weld location are shown in Table 2-6. This Fitzpatrick-specific calculation was performed by scaling a special RLB TRACG calculation for a typical BWR plant to Fitzpatrick dimensions. For this RLB TRACG calculation, the outer region of the reactor vessel was subdivided from the standard 15 sections into approximately 210 sections. This detail was needed to accurately calculate the transient pressure distribution around the shroud during the early portion of the RLB. The resultant shroud forces and tipping moments are shown in Figures 2-3 and 2-4, respectively. The blowdown forces shown in Figure 2-3 are the maximum total shear forces produced during the first five seconds of the RLB blowdown. After the first five seconds the blowdown loads decrease significantly when the subcooled flow is completed. The TRACG calculation gives a range of forces and moments; the resulting figures show the upper and lower bounds for these forces and moments.

Table 2-6: Shroud Weld Tipping Moments

Weld	Weld Elevation (in)	Tipping Moment (E6 in-lbf)	Lateral Displacement (in)
H1	388.5	0.05	0
H2	356.5	0.3	0
H3	353.375	0.3	0
H4	317.375	0.7	0
H5	219.25	3.4	<1.5*
H6a	183.25	5.6	<1.5*
H6b	179.25	5.9	<0.5**
H7	127.1875	11.4	<0.5**
H8	110.875	13.4	<0.5**

* Lateral motion limited by the jet pump riser braces.

** The lateral rotational movement of these weld locations is limited by the vertical displacement of the core support plate against the control rod guide tubes.

A review of the calculated limiting tipping moments shows that the magnitudes may lead to shroud tipping. However, this tipping is of very short duration, about 1 to 2 seconds before the shroud head pressure drop decreases, and by its restoring moment the shroud returns to its original position. Shroud tipping at the H5 and H6a weld locations will also be limited within one to one and a half inches by the jet pump riser braces. The impact of the shroud on the braces has been evaluated. This impact does not cause the braces either to buckle or to exceed their yield strength. Thus these braces are not adversely affected by this load and the restoring moment of the shroud weight will prevent permanent displacement.

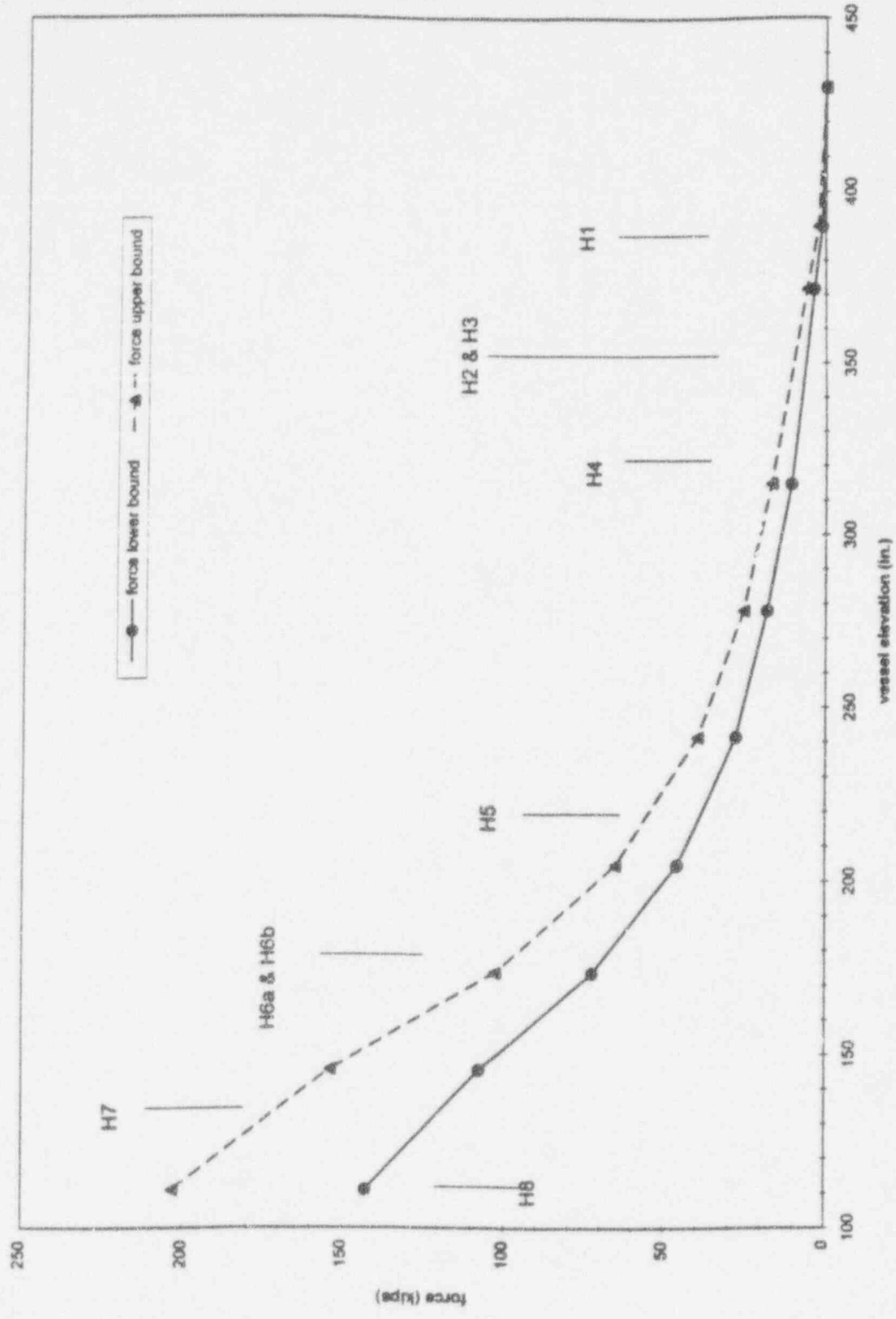

Figure 2-3: Total Force vs. Elevation in Primary Plane for Fitzpatrick Blowdown Analysis

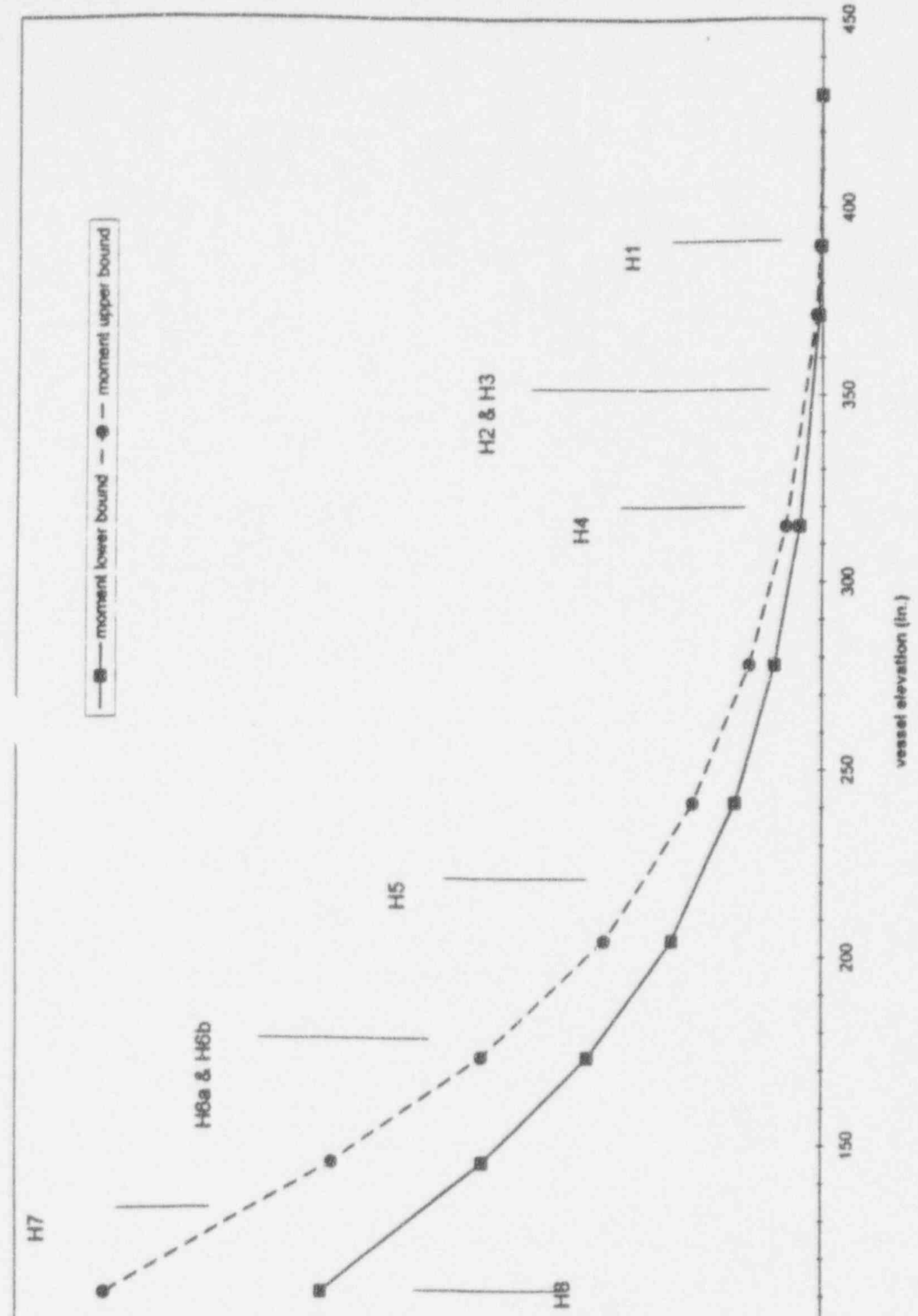

Figure 2-4: Moment vs. Elevation in Primary Plane for Fitzpatrick Blowdown Analysis

For the RLB simultaneous with a seismic event, additional vertical and lateral forces will exist. The lateral seismic loads, combined with the asymmetric blowdown loads result in a bigger tipping moment. As discussed above, the tipping will be limited by the jet pump braces, and the restoring moment returns the shroud to its original position after one to two seconds. The vertical displacement is prevented by the downward pull on the shroud exerted by the RLB. Therefore the results of a RLB remain unchanged. Current Loss of Coolant Accident analyses (Reference 2-2) are unaffected, and limiting calculated results are applicable.

2.2 PLANT SAFETY SYSTEMS

This section addresses the impact of an undetected crack as it affects the key safety functions of control rod insertability, coolable geometry, ECCS performance, and SLCS effectiveness. The primary factor in this determination is the expected movement of the shroud during the postulated event. The basis for the shroud movement is as documented in sections 2.1.2 and 2.1.3.

Control rod insertability will be assured if the fuel remains properly arranged in the core, and the guide tubes and shroud remain aligned. No misalignment occurs during either the MSLB or the RLB, without seismic effects, and thus the control rod motion will not be affected. With seismic effects, the maximum misalignment is only 1.0 inch at the top guide. This horizontal displacement results in only 0.3 degrees misalignment at the core support plate. The control rod motion will not be affected by this small misalignment. It should be noted that since the top guide remains in contact with the fuel channels, for all events, the fuel assemblies will remain properly arranged in the core

A coolable geometry will be maintained if the fuel and vessel internals do not impede the normal flow of coolant to the fuel. For vessel pipe break locations above the Top of Active Fuel (TAF), short and long term cooling is accomplished by ECCS injection anywhere in the reactor vessel in a flow amount equal to the steaming rate of the core. For vessel pipe break locations below the TAF, short term cooling is accomplished by ECCS injection inside the shroud, over the fuel and elsewhere. Long term cooling is accomplished by maintaining the level inside the shroud, to the jet pump suction level, by ECCS injection. Therefore, for a MSLB event, ECCS injection into the vessel is not precluded due to a degraded shroud condition. Also, for a RLB event, ECCS injection inside the shroud is not precluded due to a degraded shroud condition. The affected Core Spray piping during a postulated MSLB shroud lift does not impact the consequences of a MSLB, as the fuel remains covered, and the Core Spray is still able to provide coolant into the vessel. For the RLB, the Core Spray is essential to core cooling, however, for this event the Core Spray water delivery function is not affected by one to one and a half inch tipping of the shroud because of the mechanical flexibility in the core spray lines.

Proper ECCS performance is achieved if the ECCS coolant is available when and where needed. The ECCS for Fitzpatrick consists of Core Sprays (CS), Low Pressure Coolant Injection (LPCI), and High Pressure Core Injection (HPCI) systems. The CS and LPCI inject through upper shroud penetrations and jet pumps. The HPCI injects into the outer shroud region through the feedwater line. The impact of a postulated shroud failure on the consequences of MSLB is limited to some ECCS injection degradation. This is the result of CS line interference by a lifting shroud. However, the amount of ECCS flow required under a MSLB event is minimal, and thus no impact in overall ECCS performance exists. The impact of a postulated shroud failure on a RLB is limited to some additional ECCS flow needed to maintain a two thirds water level in the core. This is the result of coolant leakage through lower weld cracks. This leakage is calculated to be approximately 70 gpm. However, this leakage is minimal compared to the single-pump ECCS capacity of 8910 gpm for the RHR pump, 4265 gpm for the CS pumps, and the normal overflow through the jet pumps, and therefore no impact in overall ECCS performance exists.

Effectiveness of the SLCS is maintained if injection and mixing of boron is possible. The SLCS is used to shut down the reactor if control rods are not adequately inserted (for a RLB, the nature of the accident may limit the effectiveness of SLCS by draining the boron from the core). The SLCS injects through lower plenum lines for the Fitzpatrick plant. For a shroud separation above TAF, shroud cracks and displacement will not prevent SLCS from performing its intended function for either the RLB or MSLB. For a shroud separation below TAF, shroud cracks and displacement do not affect SLCS performance for the MSLB, and SLCS effectiveness during the RLB is also not affected as some boron may be drained through the vessel break with or without shroud cracks.

2.3 SENSITIVITY ANALYSES

A sensitivity analysis was performed for two different operating conditions: operation during power coastdown and operation at power uprate conditions.

Plant operation during power coastdown results in reduced steam flow. This reduced steam flow does result in a slightly higher shroud DP during a postulated MSLB. Previous MSLB design calculations for Fitzpatrick show an increase of only 4% in shroud DP for a core power of 23% of rated, compared to the value at 104% power. Therefore, the effect of reduced steam flow due to a power coastdown to 60% power is less than 0.25 psi in shroud DP. An increase of 0.25 psi DP results in a maximum additional vertical displacement of 0.32 inch (at the H3 location, smaller at lower locations). This effect is not significant.

The TRACG MSLB analysis was also calculated for the proposed Fitzpatrick power uprate conditions given in the Fitzpatrick TRACG MSLB Inputs table. The results shown in Figure 2-2 indicate no increase in the maximum reactor shroud head DP or in the resulting maximum shroud lift. Therefore the effect of operating at power uprate conditions does not affect possible shroud displacement.

2.4 REFERENCES

2-1 "Supplemental Reload Licensing Report for Fitzpatrick Nuclear Station Reload 10 Cycle 11", 23A7114, July 1992.

2-2 "James A. Fitzpatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis", NEDC-31317P, April 1993.

2-3 "TRACG Model Description - Licensing Topical Report", NEDE-32176P, February 1993. (GE PROPRIETARY).

2-4 "TRACG Qualification - Licensing Topical Report", NEDE-32177P, Revision 1, June 1993. (GE PROPRIETARY)

3. SUMMARY AND CONCLUSIONS

The key conclusions from the evaluation performed for the Fitzpatrick shroud are summarized here.

The applied loads to the shroud are low such that the shroud design margins are maintained even with significant cracks present. The susceptibility of the shroud to cracking and the associated crack growth rates were recently studied in Reference 3-1 on a generic basis. This generic evaluation on materials and chemistry indicates that IGSCC initiation in the plate welded ring could occur. However, the minimum required ligament for maintaining the ASME Code margins is 5 to 10% of the wall thickness. While the initial crack depth cannot be definitely established, the generic evaluation performed for BWR's indicates that the current crack depth could be up to 80% of the wall thickness. Even if this limiting value is assumed, sufficient structural margin is maintained for the current operating cycle since the predicted crack growth rate for the current cycle is small. Fitzpatrick is currently operating under hydrogen water chemistry (HWC) and growth at most locations would be less than that expected under normal water chemistry (NWC) conditions.

The safety consequences of shroud failure for the Fitzpatrick plant are not significant. The postulated shroud separation during normal operation will be readily detectable through normal measurements of plant performance parameters. The detected anomaly will lead to a normal plant shut-down.

The postulated shroud failure during abnormal and accident event conditions will not lead to greater challenges to either transient safety limits or accident consequences. Specifically:

- The safety assessment for the main steam line break combined with a seismic event shows that the lift of the shroud will not exceed the depth of the top guide for the postulated steam line break. Thus, the ability to scram is maintained. The lateral loads for the recirculation line break combined with a seismic event were also determined and found to cause at worst, limited tipping and crack opening followed by subsequent return to the original geometry in one to two seconds. While a seismic event coincident with a DBA is beyond the design basis for the Fitzpatrick plant, both accidents were evaluated with and without the loads from a safe shutdown earthquake.

- A sensitivity analysis shows that the shroud lift event is not sensitive to either operation during power coastdown or operation at power uprate conditions.

In summary, this analysis confirms the ability to operate safely with limiting crack depth assumptions.

3.1 REFERENCES

3-1 BWR Shroud Cracking Generic Safety Assessment, GENE-523-A107P-0794, August 1994.

3-2 Evaluation and Screening Criteria for the Fitzpatrick Shroud, GENE-523-154-1093, October 1993.