50-249 DRESDEN 3 CEC

SAFETY ASSESSMENT OF H2 AND H3 CORE SHROUD WELDS FOR CYCLE 14

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Attachment A Safety Assessment of H2 and H3 Core Shroud Welds for Cycle 14 Operation of Dresden Unit 2

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1.0 <u>Introduction</u>

This report contains the results of a plant specific safety analysis supporting the continued operation of Dresden 2 until the end of Cycle 14. This plant specific safety assessment has been prepared as a response to the September 27, 1994, NRC request for additional information with regard to item 1b of Generic Letter 94-03. This safety analysis for the H2 and H3 welds supplements the previous assessment (Reference 1) that was based on an assessment of the critical H5 weld location. This analysis provides a summary of the existing structural margin based on the information obtained from the detailed inspections and evaluations of the two sister plants (Dresden 3 and Quad Cities 1) as well as an assessment of a postulated separation at the H2 and H3 weld locations. Included is an evaluation of the response of the shroud to the structural loadings resulting from design basis events as well as postulated accidents beyond the design basis (e.g. steam line break, recirculation line break, seismic loads and combined accident and seismic loads), and an assessment of the ability of the plants' safety features to perform their intended functions considering the shroud response to structural loadings (e.g. control rod insertion, ECCS injection). A vertical separation at the H1 weld will not be obstructed by any other vessel components nor will it affect core geometry or coolability, therefore it need not be evaluated any further.

2.0 Summary and Conclusions

Potential core shroud cracking associated with the H2 and H3 welds at Dresden Unit 2, was evaluated for both design basis and beyond design basis accident conditions and it has been determined that the potential cracking does not prevent the safe operation of Unit 2 for the remainder of the present cycle 14. Provided below is a summary of the key areas reviewed and the conclusions reached.

1. The combination of high ductility, high toughness and low stresses makes the shroud extremely flaw tolerant. Even for the assumed case of 360 degree circumferential cracking, crack depths greater than 90% (2.70 inches considering the 1 " fillet weld and 1.80 inches neglecting the fillet weld) of the available material can be tolerated while maintaining the structural integrity for normal operation, postulated design basis accident conditions and postulated accidents beyond the design basis, including ASME Code safety factors. The analysis considers the extra one inch of ligament provided by the fillet weld in addition to the two inch shroud wall thickness as well as an assessment without the fillet weld.

2. The maximum bounding crack depth of 0.64 inches (based on operation up to the present) was calculated for Dresden 2 using an analytical modeling approach, and is within the computed allowable crack depth. The calculation of the bounding crack depth is based on a conservative estimate that the crack initiated at the end of 3 effective full power years (EFPY). The plant water chemistry history at Dresden 2 is significantly better than Dresden 3 and the effects of hydrogen water chemistry (since 1983) have been included in the calculation of the plant specific crack growth rates.

3. The realistic crack growth rate associated with the H2 and H3 weld area is slightly higher than that at the H5 weld, but is still bounded by the prediction made by the PLEDGE model for the Dresden 2 plant specific hydrogen water chemistry, and is approximately 1/30 of the values calculated for Dresden Unit 3. The Dresden 2 plant specific water chemistry was used in conjunction with the system availability (90%) to determine a crack growth rate (CGR) of 1.68 E -6 inches/hour for the upper core region (e.g. H2 and H3).
4. If the shroud assembly is postulated to have a 360° through-wall crack at H2 or H3, it would lift a maximum of 3.8 inches at H2 and 1.6 inches at H3 during normal operating conditions and the operator would be able to detect the crack, and safely shut down the plant. If a 360° crack is postulated with a Main Steam Line Break (MSLB), the maximum lift is 14.6 inches at H2 and 10.1 inches at H3. The MSLB induced lift associated with a crack at H2 results in no lateral movement of the top guide and thus lateral support of the core is not affected. The MSLB induced lift associated with a crack at H3 results in a lift of the shroud and top guide, that is less than the 14.5 inch height of the top guide and thus alignment of the core is assured and insertion of the control rods can be achieved.

5. In the unlikely occurrence of a design basis accident, safe reactor shutdown will be achieved, and the short term and long term emergency core cooling requirements will be satisfied. Table 2-1 provides a summary of the function of the plants safety features under all of the postulated events.

6. In the very unlikely occurrence of a design basis accident concurrent with a safe shutdown earthquake (SSE), which is beyond the plant licensing basis, safe reactor shutdown can be achieved and the short term and long term emergency core cooling requirements will be satisfied. The results of the ongoing ComEd and industry efforts to better define the seismic response of a shroud with a 360° through-wall flaw have been included in this analysis and are described in Sections 6.0 and 7.0.

Table 2-1 Dresden Unit 2 Safety Assessment With Loss of H2 or H3 Weld

Design Basis Accidents	Anticpated Movement			Rod Insertion	Core Reflood	Core Spray	SBLC
	Lateral	Vertical	Moment(Tip)				
Normal Operation	None	3.8" at H2 and 1.6" at H3	None	Insertion Completed After Shroud Comes Down, Timing Not Significantly Affected	Floodable Volume Maintained	Potential Damage Of CS Riser Or Sparger, CS delivery function not effected	No Boron Density Change
Design Basis Earthquake (SSE) Combined with Normal Operation Uplift Pressures	See Section 7.1	3.8" at H2 and 1.6" at H3	None	Rod Insertion Complete After and While Shroud Comes Down, Oscillitory Velocity Profile Timing Affected	Floodable Volume Maintained	Potential Failure Of CS Riser Or Sparger, Injection Into RPV Allows Long Term Cooling	No Boron Density Change
Main Steam Line Break	None	14.6" at H2 and 10.1" at H3	None	Insertion Completed After Shroud Comes Down, Timing Not Significantly Affected	Floodable Volume Maintained	Potential Failure Of CS Riser Or Sparger, Injection Into RPV Allows Long Term Cooling	No Boron Density Change
Recirculation Line Break	None	None Additional due to RLB	None	Rods Insert, Timing Not Significantly Affected	Floodable Volume Maintained	Potential Damage Of CS Riser Or Sparger, CS delivery function not effected	Injection Ability Not Affected (see note)

Additional Scenarios Considered	Anticpated Movement			Rod Insertion	Core Reflood	Core Spray	SBLC
	Lateral	Vertical	Moment(Tip)				
Main Steam Line Break Plus DBE	See Section 7.1	15.6" at H2 and 11.1" at H3	None	Rod Insertion Complete After and While Shroud Comes Down, Oscillitory Velocity Profile, Timing Affected	Floodable Volume Maintained	Potential Failure Of CS Riser Or Sparger, Injection Into RPV Allows Long Term Cooling	No Boron Density Change
Recirc. Line Break Plus DBE (Low PRA Without Adding Single Failure Criteria)	None	None Additional due to RLB	None	Rods Insert, Timing Affected	Floodable Volume Maintained	Potential Damage Of CS Riser Or Sparger, CS delivery function not effected	Injection Ability Not Affected (see note)

Note: SBLC is not designed to function during a recirculation line break.

3.0 Existing Structural Margin

3.1 Background

The extent of cracking that is expected at Dresden 2 is much less than that at Quad Cities Unit 1 and Dresden Unit 3. The bounding crack depth was estimated using the same analytical technique in Reference 2. Though the postulated cracking at Dresden Unit 2 is not insignificant, it has been demonstrated that there remains sufficient structural margin in the shroud to meet all of its design functions. The postulated cracking at the H2 and H3 welds is considered to be a conservative estimate of the conditions for Dresden 2 based on the justifications as provided in the following sections.

3.2 Crack Initiation and Effect of Water Chemistry on Crack Growth Rate (CGR)

There are several factors that differentiate the likelihood of cracking between the Dresden 2 and Dresden 3 Units. While it is known that Unit 2 has two more years of hot operation than Unit 3 (17 years versus 15 years), the first five cycle conductivity level for Unit 2 was significantly lower than that for Unit 3 (0.299 υS/cm versus 0.399 υS/cm). Shroud inspection data indicates a strong correlation between the first five cycle conductivity level and the likelihood of significant cracking. Consequently, the early conductivity level at Dresden Unit 2 more than compensates for the two years of additional operation over that of Unit 3. In fact, the Unit 2 first five cycle conductivity level is under the current EPRI guidelines of 0.300 υS/cm.

Another factor that differentiates the susceptibility of the units to IGSCC is that Unit 2 has been operating with hydrogen water chemistry since 1983. This provides a substantially improved environment for the shroud, which would significantly reduce the growth rates of any cracks that may have initiated in this region as compared to the growth rates experienced in Unit 3. Also, although the exact amount of protection that would be afforded is not clearly defined, the RPV Internals IGSCC Event Comparison suggests that Unit 2 is much less susceptible to IGSCC in this region than Unit 3 (see Table 3-1). Finally, the history of IGSCC in the primary coolant piping at Dresden Unit 2 is significantly less than that experienced at Unit 3.

Table 3-1 Dresden RPV Internals IGSCC Event Comparison

Area Inspected	Unit 2 Events	Unit 3 Events
Jet Pump Beams	2	12
Jet Pump Riser Braces	0 [1]	3
Shroud Head Bolts	1	30
Shroud Access Hole Covers	0	0
Core Spray Sparger/Piping	0 [1]	2
In Core SRM/IRM Tubes	1	1
Top Guide Bolts	0 [1]	3

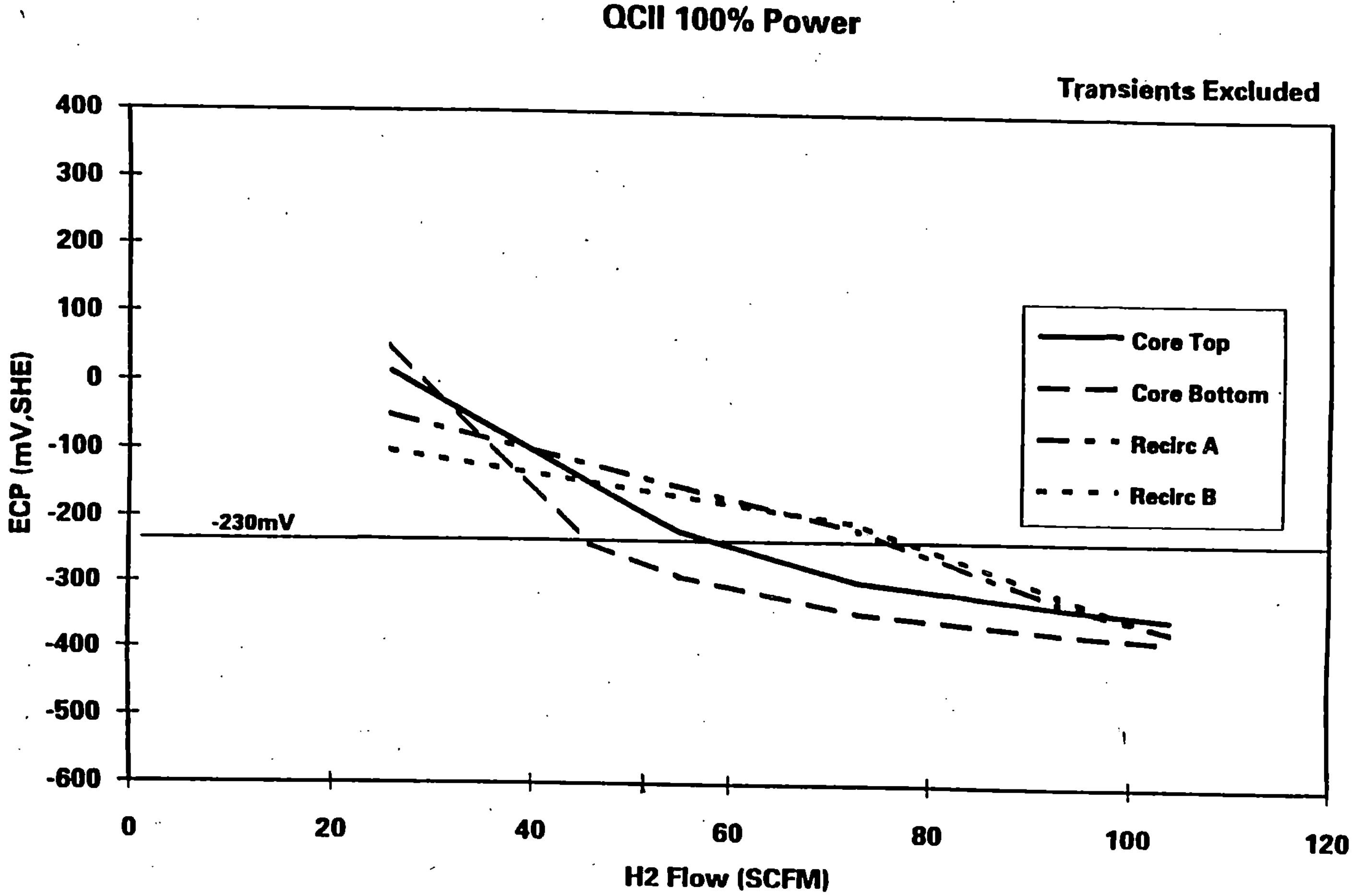
Notes:

1. Previous inspections for Unit 2 may not have been performed at the sensitivity used to detect the cracking at Unit 3.

3.3 Crack Growth Rate

The realistic crack growth rate at the H2 and H3 welds has been determined based on the electrochemical potential (ECP) measurements at Quad Cities 2 and the PLEDGE model predictions used in Reference 1 (see Figures 3-1 and 3-2).

Preliminary Data



<u>Figure 3-1 Quad Cities Unit 2 ECP as a Function of Hydrogen Injection</u>

PLEDGE Model Prediction for Dresden 2/3 Sensitized Type 304 Crack Growth Rate

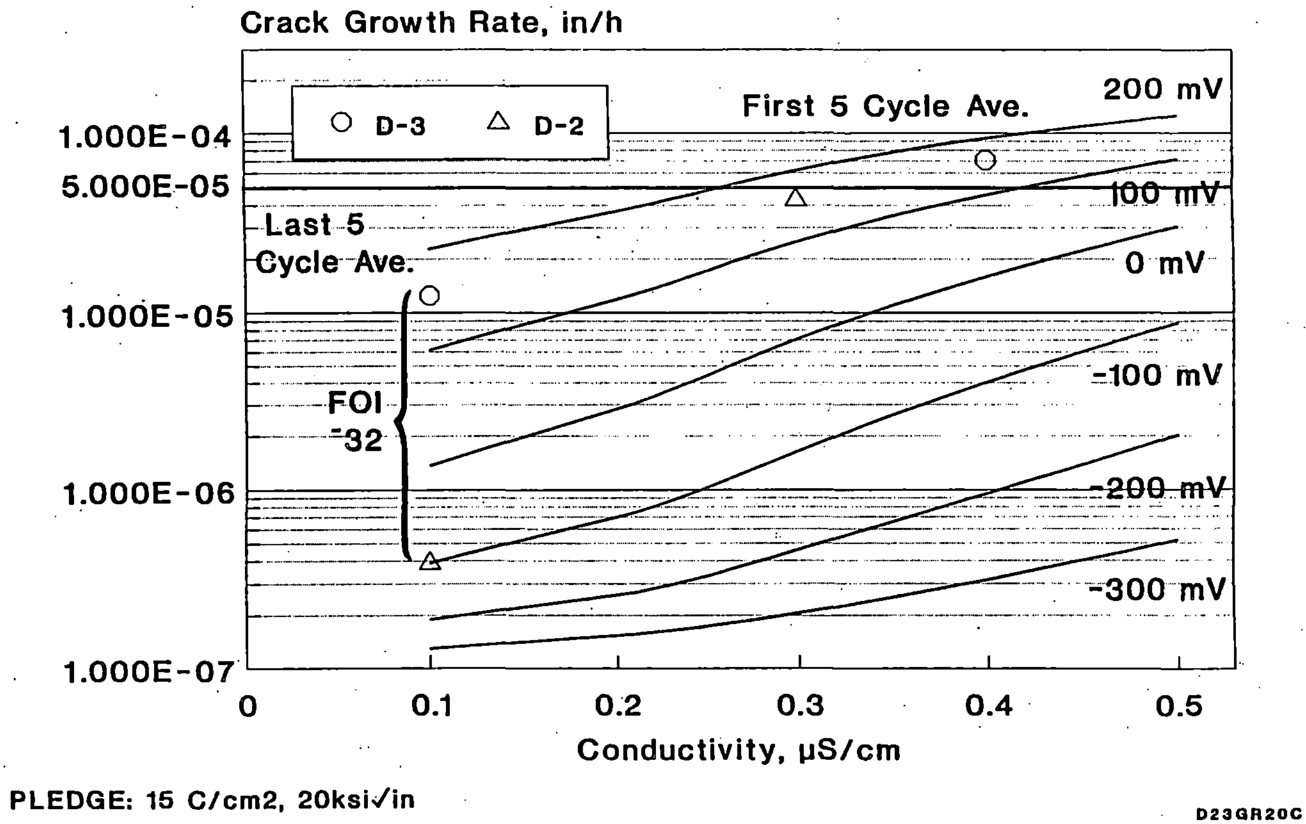


Figure 3-2 Crack Growth Rate as a Function of Conductivity and ECP

Figure 3-1 shows the Quad Cities Unit 2 ECP measurements at several locations at full power. Dresden Unit 2 has a hydrogen injection rate of 1.0 to 1.5 ppm, and as can be seen from this figure the ECP level is slightly lower near the core bottom (H5) than at the top of the core (H2 and H3) for this range of hydrogen injection. The average ECP (at a 1.25 ppm or 40.6 SCFM hydrogen injection rate) is approximately -110 mV(SHE) for H2 and H3 and -160 mV(SHE) for H5. Figure 3-2 shows the crack growth rate as a function of conductivity and ECP. The crack growth difference between -110 mV(H2 and H3) and -160 mV (H5) is approximately 30 % at 0.1 μS/cm of conductivity and with 1.25 ppm hydrogen injection. The difference between crack growth rates becomes very small as the rate of hydrogen injection decreases to 1.0 ppm and the ECP increases to approximately -25 mV(SHE). Summarized below in Table 3-2 are the key input parameters and results of the crack growth rate analysis. Note that the bounding CGR of 1.68 E-6 as shown in Table 3-3 is based on a weighted average of the CGR with hydrogen injection (90% operation) and without hydrogen injection (10% operation).

Table 3-2 Summary of Crack Growth Rate Analysis With HWC At H2, H3 and H5

Hydrogen Injection (ppm)	Conductivity (μS/cm)	ECP (mV SHE) at H2 and H3	ECP (mV SHE) at H5	Crack Growth Rates (Inches/Hour)
1.00 (32.5 SCFM)	0.10	-25	-25	1.00 E-6 (H2, H3, and H5)
1.50 (48.7 SCFM)	0.10	-180	-260	2.40 E-7 (H2 & H3) 1.60 E-7 (H5)
1.25 Average (40.6 SCFM)	0.10	-110	-160	4.00 E-7 (H2 & H3) 2.80 E-7 (H5)

3.4 Estimated Structural Margin

The new plant specific crack growth rates and the estimated crack initiation data as defined in the previous sections were used to calculate the bounding crack depth and the corresponding remaining operating margin. Note that the results of the TRACG analysis for the Recirculation line break blowdown loads (Reference 3) as well as the ComEd and BWR-VIP industry efforts to more accurately quantify the shroud loads (Reference 4) have been used to determine the primary membrane and bending stresses at H2 and H3. The methodology as defined on page 13 of the SER (Reference 2) was used in conjunction with the crack growth rates as previously defined in Table 3-2 to determine bounding crack depth. Provided below in Table 3-3 is a summary of the effective full power years and plant specific H2/H3 crack growth rates that were used to calculate the bounding crack depth and remaining operating margin.

Table 3-3 Dresden Unit 2 Summary of EFPY and CGR Based on HWC Availability

Date	Effective Full Power Years (EFPY)	HWC System Availability	Crack Growth Rate at H2 and H3 (Inches/Hour)
January 1983	7.48	None	1.32 E-5 (From SER)
June 1983	7.73	90%	1.68 E-6 [1]
December 1983	7.98	90%	1.68 E-6 [1]
March 1994	14.05	90%	1.68 E-6 [1]

Notes:

1. CGR of 1.68 E-6 is equal to 90% of 4.0 E-7 plus 10% of 1.32 E-5 which represents a weighted average that is based on HWC system availability.

A. Determination of the Bounding Crack Depth Through EFPY 14.05

- Conservatively assuming that the crack initiates at the end of EFPY 3.0 the bounding crack depth (BCD) becomes:

BCD_3 = [(7.73 - 3.0) 1.32 E-5 + (14.05 - 7.73) 1.68 E-6] 365 x 24 = 0.64 Inches

B. Minimum Ligament Required

Summarized below in Table 3-4 are the minimum required ligaments based on the maximum primary membrane and bending stresses calculated at the H2 and H3 welds for the critical loading cases (beyond design basis). The minimum required ligaments were calculated using the same limit load analysis methods as previously submitted for the H5 flaw evaluations and accepted in the SER (Reference 2). The loads used in performing this ligament determination as well as the lift calculations are based on the latest results of the analysis for a recirculation line break asymmetric loads.

Table 3-4 Dresden 2 Summary of Required Ligament and Remaining Margin

Weld Location	Critical Loading Case	Maximum d/t Ratio	Required Ligament t=2"	Required Ligament t=3" (fillet)	Operating[1] Margin Ligament t=2"/t=3"	Years of[2] Remaining Margin t=2"/t=3"
H2	Normal	0.9969	0.0062"	0.0093"	1.354" 2.351"	218 252
H2	SSE	0.9951	0.0098"	0.0147"	1.350" 2.345"	138 159
H2	MSLOCA	0.9936	0.0128"	0.0192"	1.347" 2.341"	105 122
H2	SSE+MSLOCA	0.9907	0.0186"	0.0279"	1.341" 2.332"	72 83
H2	SSE+RR LOCA	0.9950	0.0100"	0.0150"	1.350" 2.345"	135 156
H3	Normal	0.9972	0.0056"	0.0084"	1.354" 2.352"	241 280
H3	SSE	0.9947	0.0106"	0.0159"	1.349" 2.344"	127 147
H3	MSLOCA	0.9935	0.0130"	0.0195"	1.347" 2.341"	104 120
H3	SSE+MSLOCA	0.9900	0.0200"	0.0300"	1.340" 2.330"	67 78
H3	SSE+RR LOCA	0.9946	0.0108"	0.0162"	1.349" 2.344"	125 145

Notes:

1. Values shown are based on a crack initiation at the end of 3 EFPY and a bounding crack depth of 0.64".
2. Margin on remaining ligament is the ratio of Operating Margin Ligament divided by the Required Ligament.

C. Conclusion

Based on the plant specific crack growth rates at H2 and H3 as well as the bounding crack depth at the EFPY 14.04 (current status) and maximum anticipated stresses, sufficient margin exists to continue operation until the end of cycle 14. Even with a postulated crack depth of more than 90% through-wall instead of the above calculated crack depth, and without taking credit for the fillet weld thickness, an operating margin of more than 65 times the required ligament exists.

4.0 Normal Operation

As discussed in the preceding sections, the postulation of significant through-wall cracking, leakage, or separation of the core shroud assembly at the H2 and H3 weld area is extremely improbable. Assuming that the shroud is sufficiently cracked at either the H2 or H3 welds such that an upward load would result in a lift of the upper portion of the shroud, an anomalous core characteristic resulting from the flow through the gap can be detected. This anomaly would be the result of reduced moderation in the core due to either increased coolant temperatures or reduced coolant flow. An increased coolant temperature would result from flow escaping to the outside region through shroud separation at locations above the fuel top guide (e.g. H2 and possibly H3) where two phase coolant is present. For example, considering a one quarter inch gap , the leakage flow is calculated to be 4% of the rated core flow. The resulting thermal power loss would also be 4% of rated. These power anomalies are detectable. Also present will be other significant abnormal core monitoring indications, such as measured recirculation flow versus core flow.

Analogous situations have previously been observed in BWRs. In 1991, Dresden 2 began startup without the shroud head bolts properly engaged, resulting in bypass flow paths similar to those that would result from through-wall cracking of the shroud. The lift of the shroud head was identified by the operators as a core flow anomaly at approximately 80% of the rated core flow. A similar situation occurred at a different plant in 1984. In both cases, anomalies such as those described were detected and the operators safely shut down the plant.

The lifting of the shroud is induced by the differential pressure exerted on the shroud head which exceeds the weight of the shroud above the failed weld section. Under the most limiting normal operating conditions (100% power and 100% flow) the pressure difference across the shroud head is calculated to be a maximum of 7 psi. The maximum differential pressure required to initiate a separation of the shroud at the H2 and H3 welds along with the maximum lift is listed in Table 4-1. Some interference with the core spray piping is expected for the H2 and H3 welds. The core spray interference is not very rigid as the pipe coupling is designed and constructed to allow for some displacement. For purposes of this evaluation, it is conservatively assumed that the obstruction does not limit the amount of separation. For weld locations above the top guide (H1 and H2) the top guide does not move and the lateral support of the core is not affected. For weld H3, proper alignment of the core is assured since the separation is less than 2 inches and the top guide would need to lift 14.5 inches to lose contact with the top of the fuel channels. Since no significant rotational or lateral loads exist, the shroud would return to its original location when the differential pressure is reduced.

As shown in Table 4.1, a shroud with a 360° through-wall crack at H2 or H3 would be readily detectible prior to achieving full power operation and thus will be identified and appropriate corrective action can be initiated. This is further substantiated by the previous experience from the Dresden event in 1991.

Table 4.1 Shroud Differential Pressures and Lifts - Normal Operation

Weld Location	Shroud ΔP To Initiate Separation (psi)	Maximum Normal Shroud ΔP (psi)	Maximum * Vertical Separation (Inches)
H2	4.5	7.0	3.8
H3	5.7	7.0	1.6

* Separation at H1 and H2 does not affect core geometry.

5.0 Anticipated Operational Events

The previous sections demonstrate that cracks which grow through the shroud wall or cause complete separation of the shroud assembly at H2 and H3 are improbable. If such a shroud crack and separation occurs, the operators can detect it and safely shutdown the plant. This section discusses the possible impact of anticipated operational events on the shroud assuming that through wall cracks exist and have not been previously detected through normal operation. Two types of events are reviewed, first those which are considered limiting events for the Dresden 2 plant, and then those which impose the highest loads on the shroud. The FSAR limiting events are not necessarily affected significantly by the condition of the shroud, since these events are typically evaluated to determine the minimum margin to fuel thermal limits (e.g. MCPR and LHGR) and vessel pressure limits. The highest shroud load events do not necessarily lead to limiting fuel and vessel conditions, however they may determine the maximum shroud displacement and consequently have the greatest potential to affect the shroud functions.

The limiting anticipated operational events were evaluated for the Dresden plant for cycle 14 operation (Reference 5). These events are the Feedwater controller Failure (FWCF) and the Main Steam Isolation Valve closure with High Flux Scram (MSIVF). These events are characterized by a core over power condition or a rapid pressure increase respectively. These events do not result in an appreciable increase in core flow or flow through the steam separators. Therefore, no increase in shroud loads is predicted and shroud separation will not exceed that expected during normal operation. The results of the current analysis remain unchanged and no impact on safety functions exist.

5.1 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 Flow Limiter (MCFL) allows. For the Dresden units, with a bypass capacity of 40% of rated steam flow, the worst case involves inadvertently increasing the steam flow to about 150% of rated. This would not happen because the steam flow limit is set at 105%. A depressurization and cooldown occurs when isolated by Main Steamline Isolation Valve (MSIV) closure. This steam flow increase is small enough that the increased force on the shroud head (approximately 50% above the normal pressure drop) is less than the pressure differential of 14 psid due to the main steam line break (see Section 6.1). The weight of the core shroud and separators will partially resist the uplift force on the shroud head. The duration of the upper shroud portion lift is only a few seconds, and the shroud is then expected to rest again on the lower shroud portion. This event is bounded by the loads addressed in Section 6.1.

5.2 Recirculation Flow Control Failure

This postulated event involves a recirculation control failure that causes both recirculation loops to increase to maximum flow. In this type of case, the pressure drop could change from a part-load condition to the high/maximum flow condition over a time period of several seconds, but it should not significantly exceed the pressure drop expected for normal full power, high core flow operating conditions (7 psid). Normal operating procedures are considered sufficient to minimize the consequences of this potential transient, and the force on the shroud head is bounded by the value predicted for the main steam line break LOCA (14 psid in Section 6.1).

5.3 Inadvertent Actuation of ADS

Inadvertent actuation of the Automatic Depressurization System (ADS) valves is another postulated event that could put an increased load on the upper shroud. The maximum steam flow and the depressurization rate are significantly smaller than for the postulated main steamline break, causing a short-term increase in steam flow of approximately 30% of rated steam flow. The increase in the shroud ΔP resulting from the opening of the ADS valves would occur over a period of about one second, spreading the effect of the change in load. Inadvertent ADS is also a very low probability event; it is considered to be in the ASME Emergency category in the vessel thermal duty design. It has been used as the design basis Emergency event for the Dresden shroud. The effect of this event is bounded by loads addressed in Section 6.1.

6.0 Design Basis Accidents

This section discusses the possible impact of a design basis accident on a shroud with through-wall cracks (assuming that the cracking has not been previously detected through normal operation). The Main Steamline Break (MSLB) Accident imposes the largest potential lifting loads on the shroud head. Liquid breaks such as a recirculation line break (RLB) do not impose large pressure drops on the shroud head, and, in fact, the shroud pressure drop decreases from its initial value. However, this break results in maximum fuel temperature and consequently challenges the Emergency Core Cooling System (ECCS) functions. Additionally, the RLB imposes asymmetric lateral loads on the shroud.

6.1 Main Steamline Break

The MSLB inside primary containment is the postulated worst case because it results in the largest depressurization rate. During this SAR event, the reactor is rapidly depressurized as a result of a postulated instantaneous, double-ended break of the largest steamline. Thus a larger than normal pressure difference could develop across the shroud as fluid flow is drawn from the core region toward the break. For Dresden 2, the design basis pressure difference defined in the UFSAR is 12 psid for the guillotine break of a main steam line (Reference 6). As discussed previously with the NRC staff during the June 27, 1994 meeting the realistic shroud head pressure from a MSLB is expected to be slightly higher. Provided below is a description of the procedure used to estimate a more realistic shroud head pressure during a MSLB (approximately 14 psid).

The increase of the shroud head pressure difference is caused by the depressurization of the steam dome, which results from the steam blowdown through the main steam line (MSL). Therefore, one of the most important parameters that affects the shroud head ΔP is the ratio of the main steam line diameter to the reactor pressure vessel (RPV) diameter. Figure 6-1 shows the shroud head ΔP calculated for several plants by the TRACG computer code as a function of the MSL to RPV diameter ratio. This data shows a strong correlation between the MSL to RPV diameter ratio and the ΔP for six plants. Dresden Unit 2 has a relatively small diameter MSL in comparison to the RPV diameter and therefore is expected to have a relatively low shroud head ΔP. Considering the uncertainty associated with this type of analysis 14 psid was selected as the upper bound value of the shroud head pressure during the MSLB for use in performing an analysis of a postulated 360° through-wall flaw.

The amount of lifting and the potential effect of these postulated occurrences on emergency operation is described in the following paragraphs. One of the key considerations of this postulated accident case is the ability of the control rods to insert before or during the postulated accident. Specifically, sufficient lifting of the top guide prior to control rod

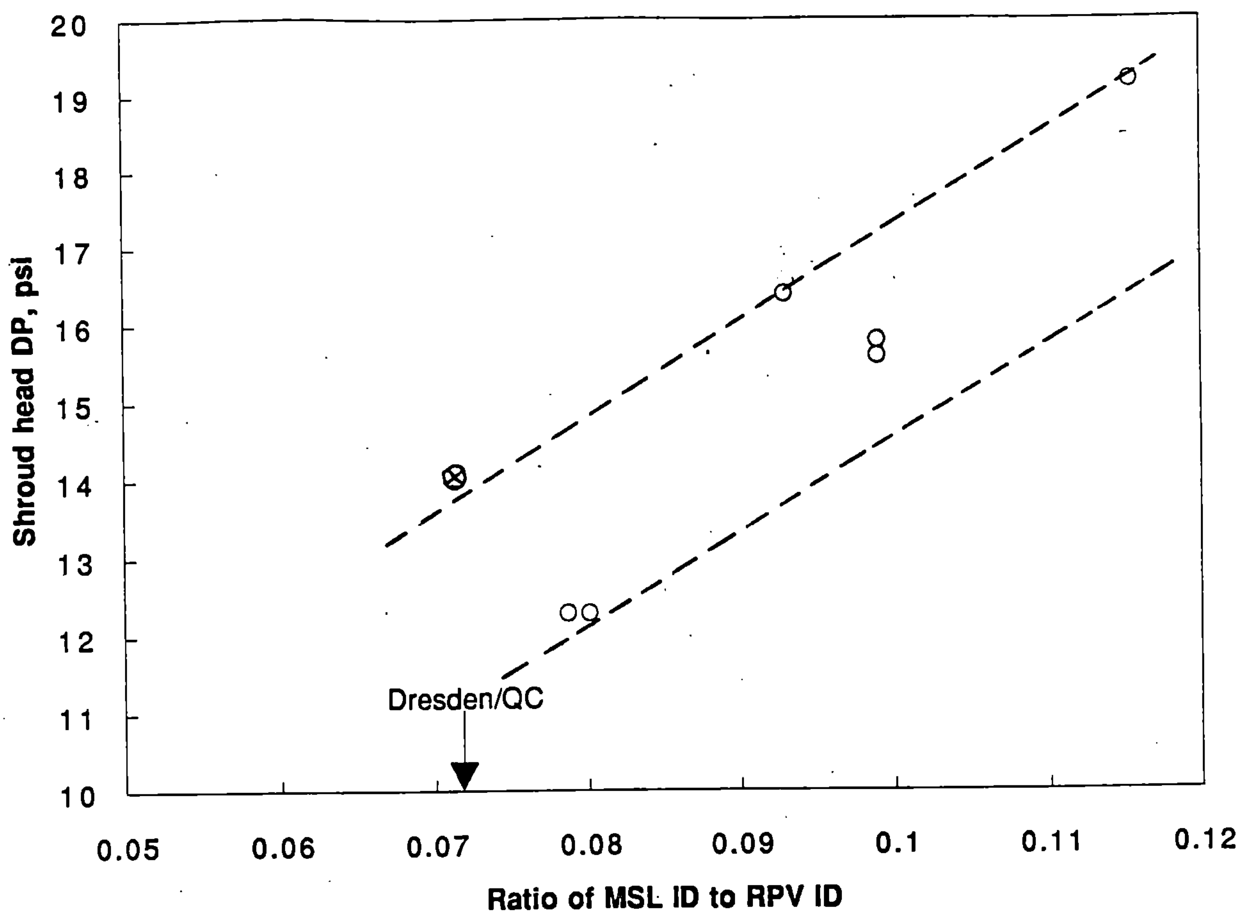


Figure 6-1 Shroud Head ΔP as a Function of the Ratio of MSL Diameter to RPV Inside Diameter

insertion could cause reorientation of the fuel bundles and thus impede the insertion of control rods.

The shroud head pressure drop characteristics calculated for the instantaneous, double-ended steamline break accident were evaluated for a typical BWR (Reference 7). 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 dome in the short term. The 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 small (see Section 3.0), the structural integrity of the shroud will remain intact for this postulated limiting event. Note that if a 360° through wall crack is assumed, the operator will detect the shroud lift during normal operation and will shut the plant down, thus the likelihood of a lift occurring during a MSLB is even further reduced. If it is postulated that the initial load pulse causes the shroud to separate, the last part of the pressure loading could cause the shroud assembly to lift. The flow path created by any separation reduces the upward lifting forces. For this postulated scenario the core shroud assembly will lift and the result is shown in Table 6-1.

Table 6-1 Maximum Shroud Lift Under MSLB Conditions

Weld Location	Shroud ΔP To Initiate Separation (psi)	Maximum Shroud ΔP (psi)	Maximum * Vertical Separation (Inches)
H2	4.5	14.0	14.6
H3	5.7	14.0	10.1

* Separation at H1 and H2 does not affect core geometry.

The magnitude of the lifts shown in Table 6-1 are greater than those shown in Section 4.0 for the normal operating condition. As noted before, some interference is expected by the core spray piping penetrating the inner shroud region. This interference is not strong because the pipe coupling allows some displacement. For the purpose of this evaluation, it is conservatively assumed that the interference does not affect the magnitude of the lift . For the H3 weld, proper alignment of the core is assured if the lift is less than 14.5 inches, as the top guide would need to lift over 14.5 inches to lose contact with the top of the fuel channels. The lift calculated is slightly more than 10 inches, and thus alignment of the core is assured. The 14.6 inch lift at the H2 weld does not affect the movement of the top guide and thus alignment of the core is not affected and the lifting at H2 is acceptable.

Scram is initiated during the main steamline break (inside containment) accident by the high drywell pressure trip signal. Drywell pressure exceeds the setpoint almost instantaneously, so the only delays in the rod insertion come from the sensors, the Reactor Protection System, and rod motion. For the main steamline break outside containment, shroud loads are reduced, MSIV closure is initiated by high steam flow, and scram is initiated from the MSIV closure. For either postulated steamline break scenario, the insertion of all control rods will occur because core alignment is assured. With the main steam line break alone, the core shroud assembly would not move laterally, and no significant degradation of scram performance is expected.

Movement of the upper shroud assembly will affect the core spray system and as a result coolant flow to the two core spray spargers is affected. For the MSLB core cooling can be assured as long as coolant reaches the RPV. Therefore the core spray line function of water delivery to the spray header inside the shroud is not required as long as the coolant is injected into the RPV.

The main steamline break has also been evaluated for radiological release consequences in the SAR. For a main steamline break inside of containment, the radiological consequences are bounded by the Loss of Coolant Accident. For the main steamline break outside of containment, the magnitude of the pressure loads that potentially could lead to separation of the upper shroud are less than that for breaks inside the containment, due to attenuation of the depressurization wave along the steamline. MSIV closure is initiated before any potentially increased radiological release outside containment from such a scenario could occur. The radiological consequences of this main steamline break scenario are thus still bounded by the plant SAR results.

6.2 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, upward forces are reduced, and thus there is no significant threat to core shroud integrity. No additional shroud lift over than experienced during normal operation will occur, because any significant lift prior to the RLB, would be detected during normal operation as discussed in Section 4.0. Lateral forces on the welds at the beginning of a RLB are large acoustic forces of very short duration, approximately five 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 a corresponding lifting force. If sufficient lifting occurs prior to the accident, it would be detected during normal operation as discussed in Section 4.0.

Both the acoustic and blowdown loads are small at the H2 and H3 weld locations because it is a substantial distance from the recirculation suction line nozzle. The acoustic load at the H2 and H3 welds is significantly less than that at H5 (less than one quarter) and the blowdown loads are much less (less than one tenth). Therefore, the consequences of a RLB for the H5 weld will bound the H2 and H3 weld locations. Even if the shroud were to lift at the H2 or H3 weld, core coolant supply below H3 would be intact and therefore long term core cooling is assured.

6.3 Safe Shutdown Earthquake

Reviews of the effects of partial through wall flaws on the seismic response of the RPV and internals indicate that the effect of the change in stiffness is not significant provided that sufficient ligament remains to transfer shear across the plane of the flaw. As previously noted a 360° through wall flaw at H2 or H3 would be detected during normal operation and thus the evaluation of the SSE with uplift due to normal operating pressure is not necessary and is bounded by the analysis provided in section 7.1 for a combined SSE and MSLB.

The design basis earthquake for Dresden Unit 2 is defined in section 3.7 of the UFSAR and the smoothed design response spectra are shown in Figures 3.7-1 through 3.7-3 of the UFSAR for four values of oscillator damping. The design response spectrum at 2% oscillator damping for Dresden Units 2 and 3 bounds the corresponding response spectrum from the time history for the 1940 North/South component of the El Centro earthquake below 1.21 hertz (0.826 second period). Analysis of the response of a shroud with a postulated 360° through wall flaw and a lift (i.e. floating shroud) indicates that the maximum relative displacement of the core shroud with respect to the vessel is approximately equal to the maximum ground displacement. The results of this analysis correlates well with the theoretical prediction that for very low frequency systems (periods greater than 10 seconds), the separated floating portion of the shroud will not respond initially with the main structure and thus a differential displacement less than or equal to the maximum ground displacement is expected. Recent efforts to develop a synthetic time history that bounds the Housner response spectra have shown that the ground displacement characteristics for natural earthquake records have significant variability in magnitude. Table 6-2 is a summary of the displacement characteristics for the eight earthquake records that were used to develop the Housner Spectra. The corresponding maximum ground displacement for Dresden would be 2.465 inches (El Centro 1940 S00E normalized to 0.2g).

Table 6-2 Summary Of Information For The 8 Earthquakes That Were Used To Develop The Housner Spectra

Earthquake Record No.	Location & Date	Direction	Acceleration (cm/s^2)	Acceleration Perecent Of Gravity	Maximum Ground Displacement (cm)	Ground Displacement Normalized To 0.24g (cm)	Ground Displacement Normalized To 0.24g (Inches)
IIA001	El Centro, Imperial Valley Earthquake, 5/18/40	S00E	341.7	0.348	10.9	7.514	2.958
IIA001	El Centro, Imperial Valley Earthquake, 5/18/40	S90W	210.1	0.214	19.8	22.198	8.740
IIA004	Kern County, Taft Lincoln School Tunnel, 7/21/52	S69E	175.9	0.179	9.2	12.320	4.850
IIA004	Kern County, Taft Lincoln School Tunnel, 7/21/52	N21E	152.7	0.156	6.7	10.335	4.069
[illegible]024	El Centro, Imperial Valley, Lower California Earthquake, 12/30/34	N90E	179.1	0.182	3.7	4.866	1.916
IIB024	El Centro, Imperial Valley, Lower California Earthquake, 12/30/34	N00E	156.8	0.160	4.2	6.309	2.484
IIB029	Western Washington Earthquake, Olympia Washington Hwy. Test Lab, 4/13/49	S86W	274.6	0.280	10.4	8.921	3.512
IIB029	Western Washington Earthquake, Olympia Washington Hwy. Test Lab, 4/13/49	S04E	161.6	0.165	8.5	12.390	4.878
					Averages	10.607	4.176

Reference: EERL 71-50, Stong Motion Earthquake Accelerogram, September 1971, California Institute of Technology

7.0 Postulated Accidents Beyond Design Basis

7.1 Main Steam Line Break Plus SSE

If the main steam line break occurs simultaneously with the design basis earthquake, this added load is postulated to cause separation of the upper shroud assembly at H2 or H3 weld locations, leading to an upward displacement of this structure and the associated top guide. The vertical lift is only minimally affected by the seismic excitation because the upper shroud assembly is not subjected to the seismic excitation while it is separated from the lower portion of the shroud. Calculations performed to simulate the possible shroud vertical displacement due to a seismic excitation concurrent with a MSLB results in less than 1.0 inch of additional lift. This lift is temporary as the shroud assembly above the crack returns to the lower shroud portion after approximately one second.

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. However for the portion of the shroud above the H2 and H3 welds, no tipping or rotation will occur because the resisting moment due to the shroud weight is greater than the seismic overturning moment.

The primary consideration of this postulated accident case is the capability to insert the control rods before or during the postulated accident. Specifically, the combination of lifting and lateral movement of the top guide prior to or during control rod insertion could impede the insertion of control rods. With random displacement anticipated during a seismic event (see Reference 8), the control rod alignment in the core region would undergo intermittent periods of misalignment. Hence, the CRD scram speed would vary with time as the control rods are being inserted.

The relative horizontal displacement time histories between the shroud and the vessel at the top guide and core support plate elevations, corresponding to a 360° through-wall crack at weld H3, are given in Figures 7-1 through 7-4. The input motion to the analysis was a synthetic time history generated from the Dresden smoothed design spectra given in Figures 3.7-1 and 3.7-3 of the Dresden UFSAR. The synthetic time history was normalized to a Zero Period Acceleration (ZPA) of 0.24g and has a duration of 40 seconds. The 40 second duration was required to adequately capture the low frequency characteristics. The Dresden SSE ZPA is 0.20g and the Quad Cities SSE ZPA is 0.24g. The value of 0.24g was used so that the resulting relative displacement time histories would be bounding for both Dresden and Quad Cities.

Based on actual static deflection control rod insertion tests, the dynamic factor from dynamic deflection control rod insertion tests, a 10% reduction to account for computational affects, and a 1.125 safety factor for faulted load cases, the allowable relative displacement between the shroud and the vessel is 1.50 inches at the core plate elevation and 4.80 inches at the top guide elevation.

Time history analysis were performed for two shroud connectivity conditions at the H3 weld location. The roller connectivity condition is the bounding case. However, it is a very conservative assumption since it assumes separation of the upper shroud assembly (above the H3 weld) occurs for the full 40 second duration of the SSE event. The separation actually occurs for approximately 1 second. Consequently, the relative displacement time histories corresponding to the shroud pinned connectivity condition are more representative after separation has occurred.

From Figures 7-3 and 7-4, it is observed that the maximum relative displacements at the top guide and the core plate are well below the allowables for the pinned condition for the full 40 second duration of the SSE event. Furthermore, from Figure 7-2, the maximum relative displacement at the core plate is also well below the allowable for the roller condition for the full 40 second duration of the SSE event.

Only at the top guide location for the roller condition does the relative displacement exceed the allowable value. However, assuming one second to the initiation of the scram and a one second duration for the separation of the shroud, the roller connectivity condition is over at approximately two seconds. From Figure 7-1, the peak relative displacement at the top guide before 2 seconds is less than 2.5 inches. After that time the shroud separation has closed and the actual relative displacement at the top guide will not follow the Figure 7-1 time history. However, even if it did, with a control rod insertion time of between 2 and 3 seconds, the control rods would be almost fully inserted by the time the relative displacement reached the 4.8 inch allowable, which occurs a little after 3 seconds. In any event the control rods would be fully inserted during the next oscillatory cycle.

Figure 7-1 Displacement Time History at the Top Guide with a Through-Wall Crack at H3 (Roller Connected Joint)

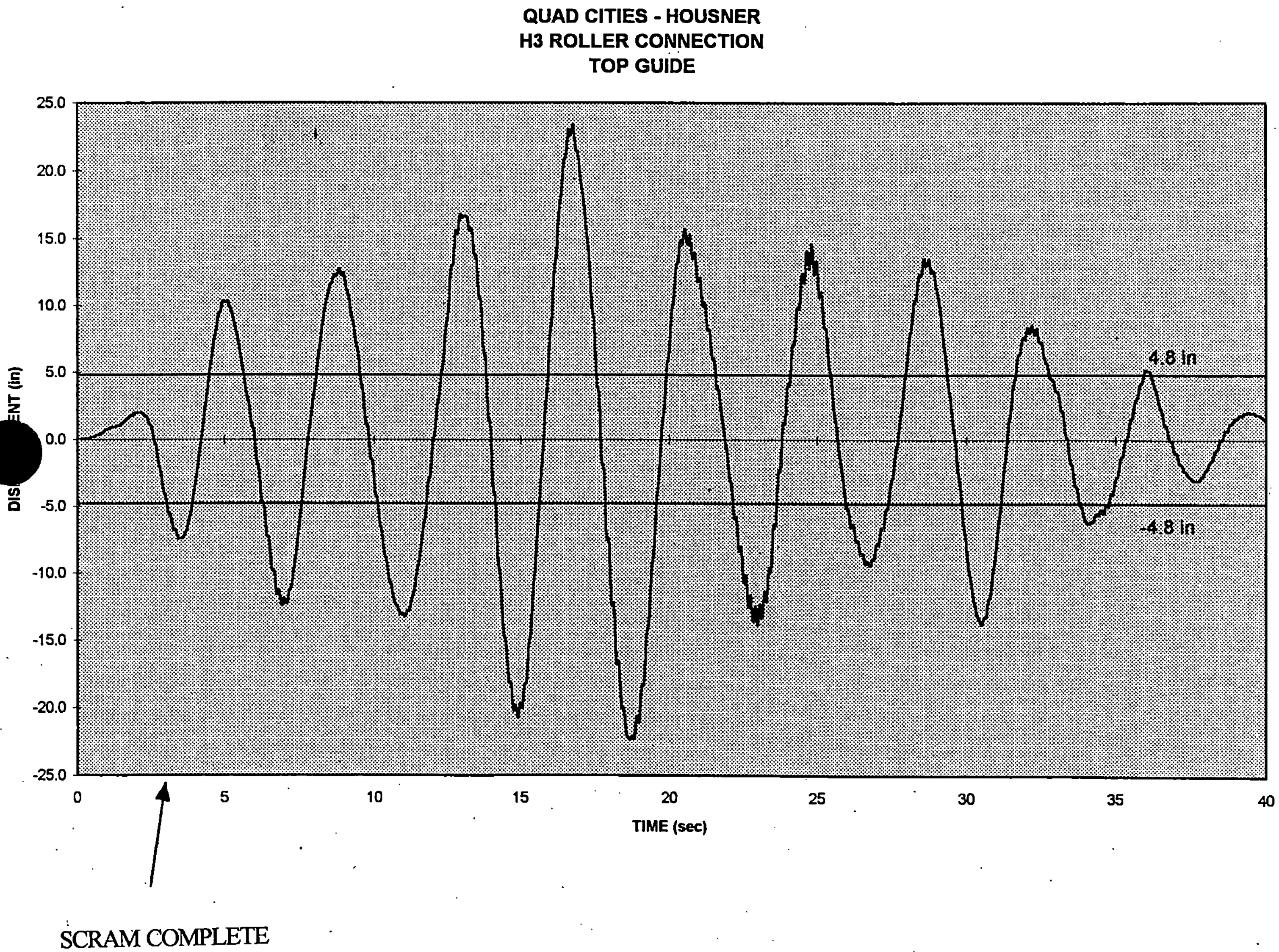


Figure 7-2 Displacement Time History at the Core Plate with a Through-Wall Crack at H3 (Roller Connected Joint)

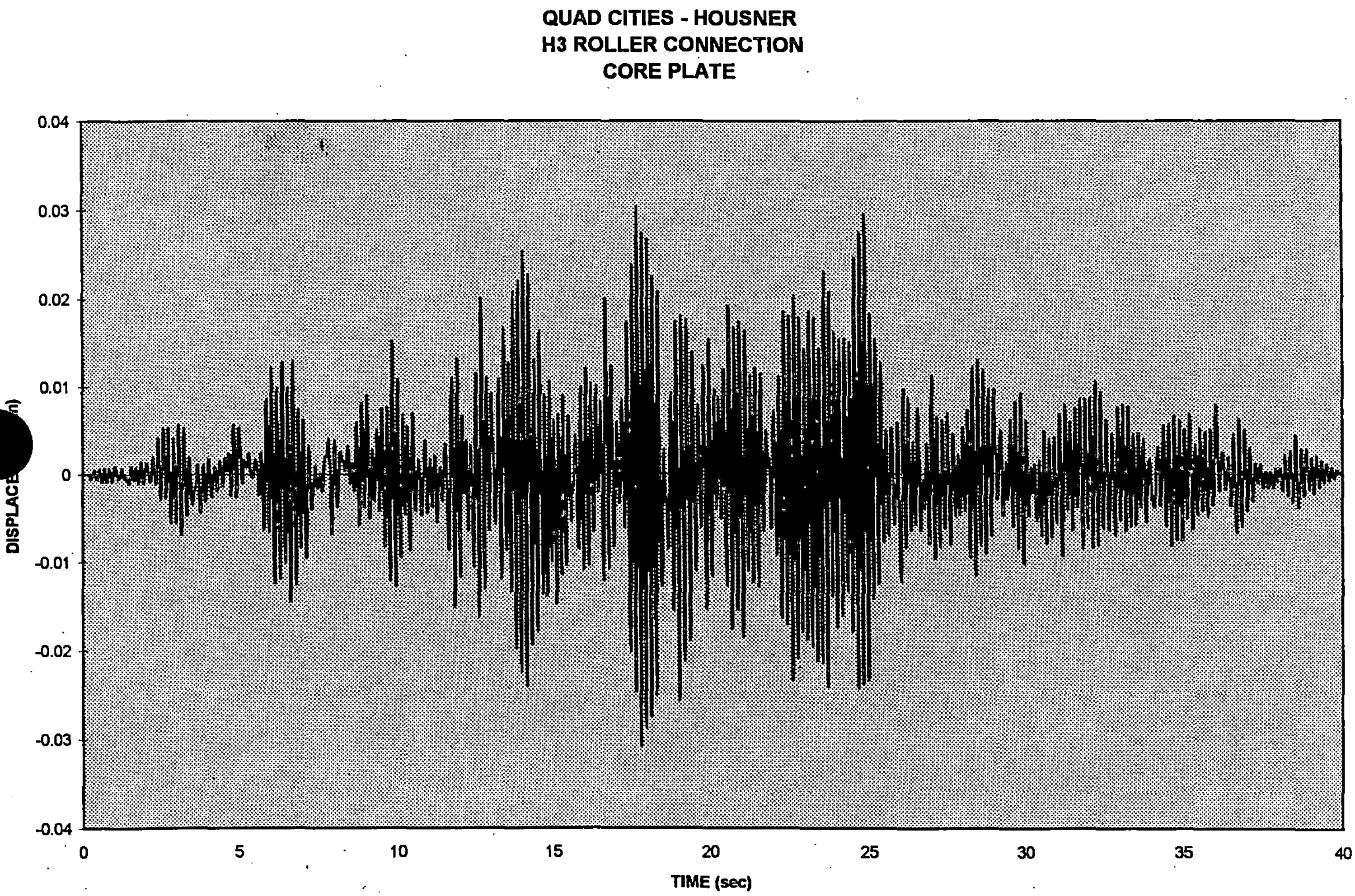


Figure 7-3 Displacement Time History at the Top Guide with a Through-Wall Crack at H3 (Pin Connected Joint)

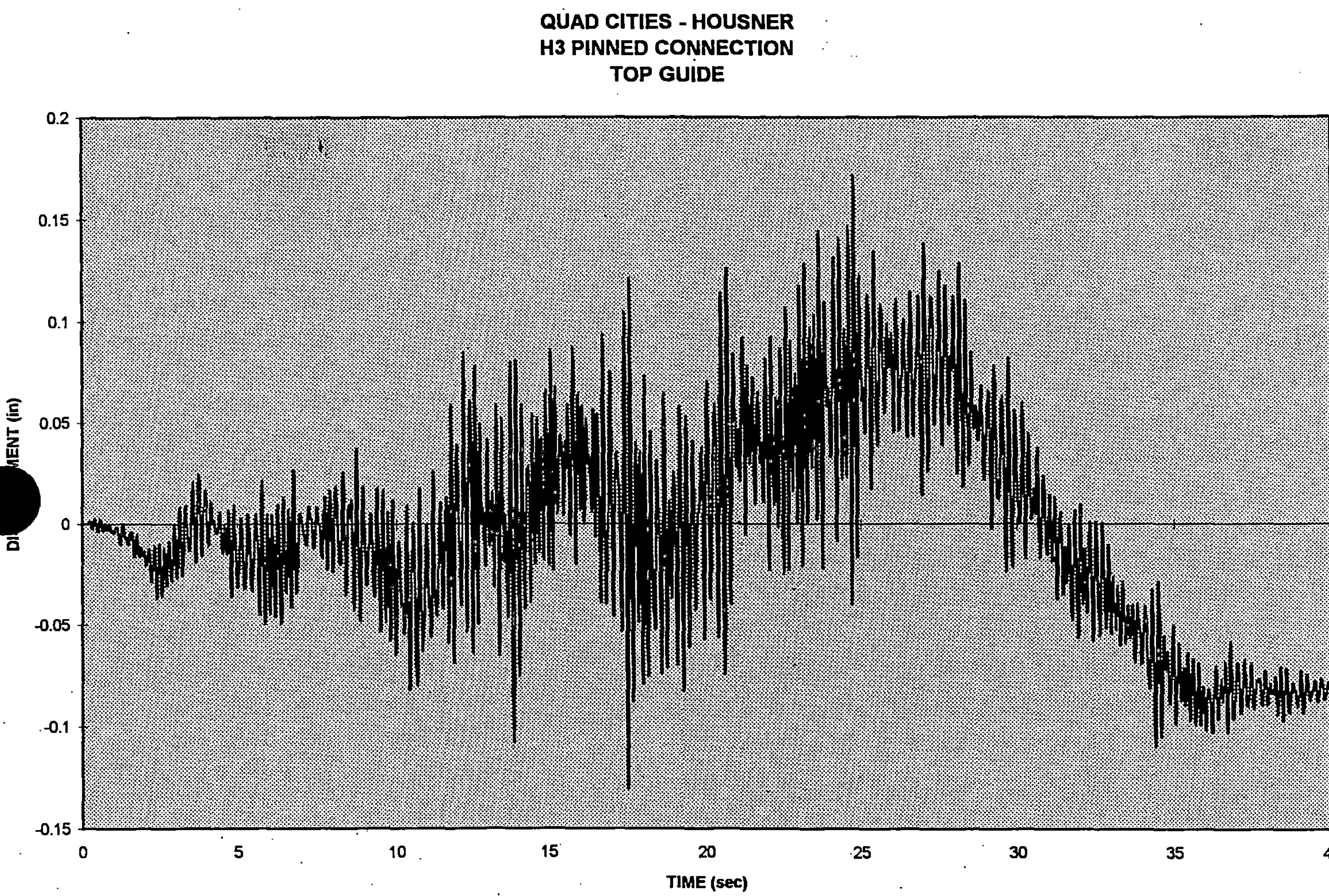
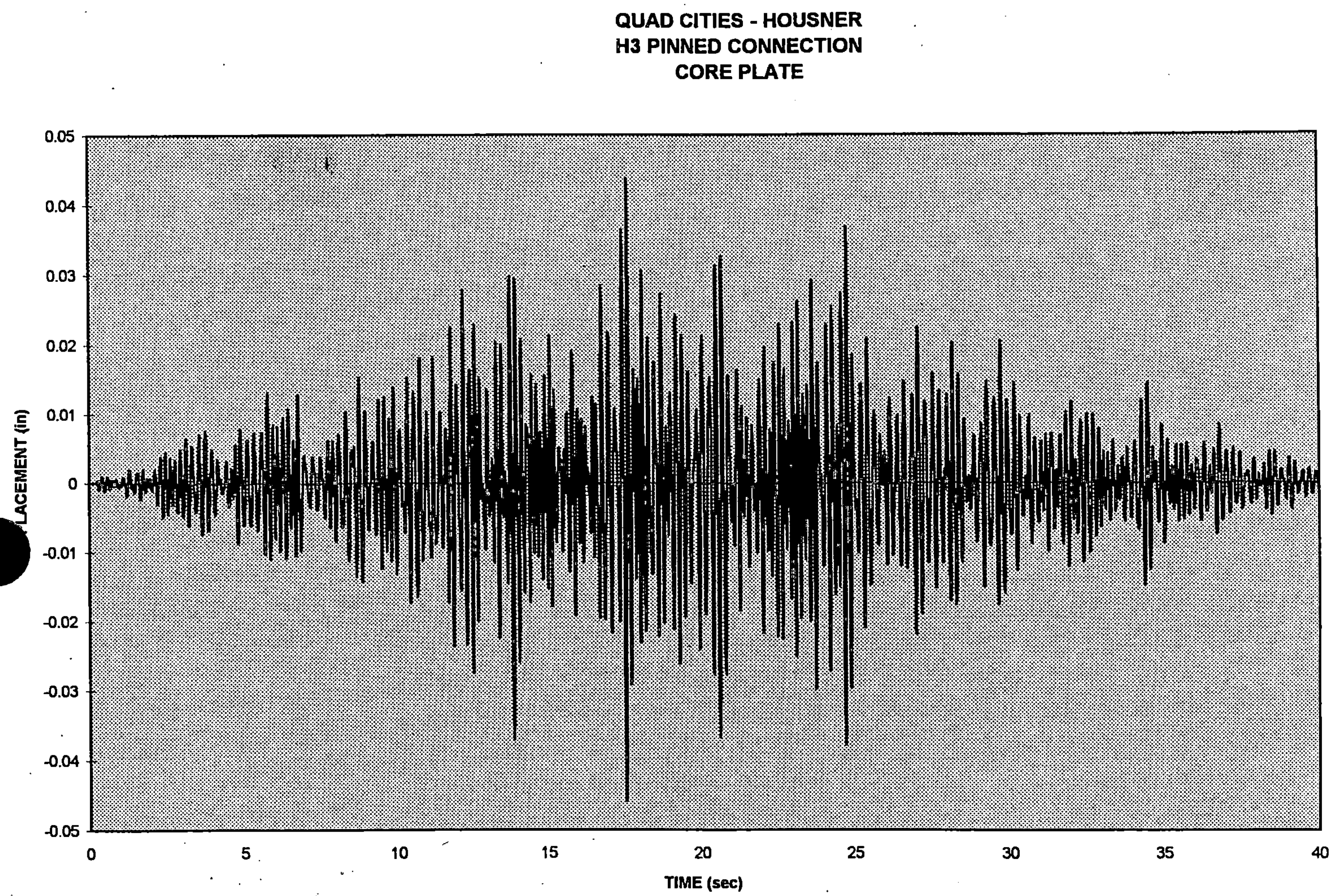


Figure 7-4 Displacement Time History at the Core Plate with a Through-Wall Crack at H3 (Pin Connected Joint)



7.2 Recirculation Line Break Plus SSE

For the RLB accident simultaneous with a seismic event, additional vertical and lateral forces will exist. The lateral seismic loads when combined with the asymmetric blowdown loads result in a larger tipping moment. As discussed in Section 6.2, the lateral load is small at the H2 and H3 weld locations during a RLB and thus the primary load will be due to the seismic excitation. The portion of the shroud above the H2 and H3 welds will not tip or rotate because the resisting moment due to the shroud weight is greater than the combined recirculation line break and seismic overturning moment. Vertical displacement of the shroud will be resisted by the downward force on the shroud exerted by the RLB. The vertical seismic excitation of 0.13 g is much less than gravity and thus will be offset by the combination of the pulldown force and the dead weight. Therefore, the combination of the RLB with the SSE does not result in a loading case or a motion that is more critical than what has been evaluated for the other events.

7.3 Probabilities of Events

The probabilities of the design basis and beyond design basis events were provided to the NRC in Reference 9 (question PR-1) and for your convenience are summarized below.

Event	Dresden Frequency(1)
SSE	5.0 E-5 /year
Main Steam Line Break (MSLB)	4.1 E-8/year
Recirculation Line Break (RLB)	3.0 E-4/year
SSE coincident with MSLB	5.6 E-15/year
SSE coincident with RLB	4.1 E-11/year

Note 1 - For purposes of these responses 'coincident' is defined as occurring in the same 24 hour period.

These event probabilities for the beyond design basis accidents are extremely small and thus provide substantiation to the unlikeliness of the occurrence of these combined events. The ability to detect a 360° through wall flaw at H2 and H3 during normal operation rules out the possibility of having an undetected flaw prior to these events and thus concludes that the crack would have to be initiated by the accident. The probability associated with an event that would include a (1) initiation of a through wall flaw, plus (2) a MSLB of a RLB, and (3) a full SSE, is less at H2 or H3 than at H5 because of the ability to detect a 360 degree through-wall flaw at H2 and H3.

8.0 Emergency Operator Actions

The Emergency Procedure Guidelines (EPGs) are the basis for plant specific Emergency Operating Procedures (EOPs). The EPGs 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 discussed above could result in separation of the upper shroud assembly from the section of the shroud below the H3 weld, which has minimal impact to scram performance. Therefore, no further consideration is necessary for the impact of this postulated event on the EPGs.

The EPGs provide instructions for reactor pressure, water level, and power control, as well as control of key primary containment parameters. Actions specified in the EPGs 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 value (typically 110 F). EPG instructions are for water level to be controlled below the high water level setpoint; thus there would not be dilution of the liquid boron by flooding to the steamline elevation or loss of vessel inventory out the break in case SLCS injection were to occur. Water level would be controlled after the postulated event because the break is high in the vessel and a large compliment of water injection systems would be available.

9.0 References

1. GE report GE-NE-A00-05652-03 Rev. 1, DRF A00-05652 (15), "Preliminary Safety Assessment of Core Shroud Indications for Cycle 14 Operation of Dresden Unit 2", June 1994.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation, Related to Core Shroud Cracking at Dresden Unit 3 and Quad Cities Unit 1, July 21, 1994.

3. GE Report GE-NE-L12-00819-05 Rev. 0, DRF L12-00819 (26), "Core Shroud Blowdown Load Calculation During Recirculation Suction Line Break by TRACG Analysis, for Dresden Nuclear Power Station Units 2 and 3, and Quad Cities Nuclear Power Stations Units 1 and 2", August 1994, (Proprietary Information is Included).

4. BWR-VIP Draft Report SL-4942 Rev. C, "BWR Core Shroud Evaluation Load Definition Guideline", Calculation 9511-00, September 23, 1994.

5. Dresden Unit 2 Cycle 14, Plant Transient Analysis, EMF-92-126, Siemens Power Corporation, October 1992.

6. Updated Final Safety Analysis Report, Dresden 2 and 3, Tables 3.9-19 and 3.9-20, Figure 3.9-5.

7. Letter dated November 9, 1994, from L. A. England to USNRC, "BWROG Safety Assessment-BWR Shroud Crack Indications".

8. Strong Motion Earthquake Accelograms, Digitized and Plotted Data, Corrected Accelograms and Integrated Ground Velocity and Displacement Curves, EERL 71-50, California Institute of Technology, Earthquake Engineering Research Laboratory, September 1971.

9. Letter form ComEd to W. Russell (NRC), dated July 8, 1994, transmitting the response to the June 23, 1994 NRC request for additional information.

Attachment B Safety Assessment of H2 and H3 Core Shroud Welds for Cycle 14 Operation of Quad Cities Unit 2

Table of Contents

1.0 Introduction

This report contains the results of a plant specific safety analysis supporting the continued operation of Quad Cities Unit 2 until the end of Cycle 13. This plant specific safety assessment has been prepared as a response to the September 27, 1994 NRC request for additional information with regard to item 1b of Generic Letter 94-03. This safety analysis for the H2 and H3 welds supplements the previous assessment (Reference 1) that was based on an assessment of the critical H5 weld location. This analysis provides a comparison of the existing structural margin based on the information obtained from the detailed inspections and evaluations of the two sister plants (Dresden 3 and Quad Cities 1)as well as an assessment of a postulated separation at the H2 and H3 weld locations. Included is an evaluation of the response of the shroud to the structural loadings resulting from design basis events as well as postulated accidents beyond the design basis (e.g., steam line break, recirculation line break, seismic loads, and combined accident and seismic loads), and an assessment of the ability of the plant's safety features to perform their intended functions considering the shroud response to structural loadings (e.g., control rod insertion, ECCS injection). A vertical separation at the H1 weld will not be obstructed by any other vessel components nor will it affect core geometry or coolability, therefore it need not be evaluated any further.

2.0 Summary and Conclusions

Potential core shroud cracking associated with the H2 and H3 welds at Quad Cities Unit 2, was evaluated for both design basis and beyond design basis accident conditions. It has been determined that the potential cracking does not prevent the safe operation of Unit 2 for the remainder of the present cycle 13. Provided below is a summary of the key areas reviewed and the conclusions.

1. The combination of high ductility, high toughness and low stresses makes the shroud extremely flaw tolerant. Even for the assumed case of 360 degree circumferential cracking, crack depths greater than 90% (2.70 inches considering the 1 " fillet weld and 1.80 inches neglecting the fillet weld) of the available material can be tolerated while maintaining the structural integrity for normal operation, postulated design basis accident conditions and postulated accidents beyond the design basis, including ASME Code safety factors. The analysis considers the extra one inch of ligament provided by the weldment in addition to the two inch shroud wall thickness as well as an assessment without the fillet weld.

2. The crack growth rates associated with the H2 and H3 weld areas for Quad Cities Unit 2 will be similar to those for Unit 1. This is based on similar water chemistry histories and average conductivities for both of the two units. Even discounting the Hydrogen Water Chemistry (HWC) benifits that both units have had since 1990, and using the bounding crack growth rate of 5.0 E-5 inches/hour and the cracking identified for Unit 1, the crack depth is not expected to exceed the allowable crack depth prior to Q2R13.

3. If the shroud assembly is postulated to have a 360° through-wall crack at H2 or H3, it would lift a maximum of 5.0 inches at H2 and 2.6 inches at H3 during normal operating conditions and the operator would be able to detect the crack, and safely shut down the plant. If a 360° crack is postulated with a Main Steam Line Break (MSLB), the maximum lift is 14.6 inches at H2 and 10.1 inches at H3. The lift during a MSLB with a crack at H2 results in no movement of the top guide and thus lateral support of the core is not affected. The lift during a MSLB with a crack at H3 results in a lift of the shroud and top guide, that is less than the 14.5 inch height of the top guide and thus alignment of the core is assured and insertion of the control rods can be achieved.

4. In the unlikely occurrence of a design basis accident, safe reactor shutdown will be achieved, and the short term and long term emergency core cooling requirements will be satisfied. Table 2-1 provides a summary of the function of the plant's safety features under all of the postulated events.

5. In the unlikely occurrence of a design basis accident concurrent with a safe shutdown earthquake (SSE), which is beyond the plant licensing basis, safe reactor shutdown can be achieved, and the short term and long term emergency core cooling requirements will be satisfied. The results of the ongoing ComEd and industry efforts to better define the seismic response of a shroud with a 360° through-wall flaw have been included in this analysis and are described in Sections 6.0 and 7.0.

Table 2-1 Quad Cities Unit 2 Safety Assessment With Loss of H2 or H3 Weld

Design Basis Accidents	Anticipated Movement Lateral	Vertical	Moment(Tip)	Rod Insertion	Core Reflood	Core Spray	SBLC
Normal Operation	None	5.0" at H2 and 2.6" at H3	None	Insertion Completed After Shroud Comes Down, Timing Not Significantly Affected	Floodable Volume Maintained	Potential Damage to CS Riser Or Sparger, CS Delivery Function not Affected.	No Boron Density Change
Design Basis Earthquake (SSE) Combined with Normal Operation Uplift Pressures	See Section 7.1	5.0" at H2 and 2.6" at H3	None	Rod Insertion Complete After and While Shroud Comes Down, Oscillitory Velocity Profile Timing Affected	Floodable Volume Maintained	Potential Failure Of CS Riser Or Sparger, Injection Into RPV Allows Long Term Cooling	No Boron Density Change
Main Steam Line Break	None	14.6" at H2 and 10.1" at H3	None	Insertion Completed After Shroud Comes Down, Timing Not Significantly Affected	Floodable Volume Maintained	Potential Failure Of CS Riser Or Sparger, Injection Into RPV Allows Long Term Cooling	No Boron Density Change
Recirculation Line Break	None	None Additional Due to RLB	None	Rods Insert, Timing Not Affected	Floodable Volume Maintained	Potential Damage to CS Riser Or Sparger, CS Delivery Function not Affected.	Injection Ability Not Affected (See Note (1))

Additional Scenarios Considered	Anticpated Movement Lateral	Vertical	Moment(Tip)	Rod Insertion	Core Reflood	Core Spray	SBLC
Main Steam Line Break Plus DBE	See Section 7.1	15.6" at H2 and 11.1" at H3 (See Note (2))	None	Rod Insertion Complete After and While Shroud Comes Down, Oscillitory Velocity Profile, Timing Affected	Floodable Volume Maintained	Potential Failure Of CS Riser Or Sparger, Injection Into RPV Allows Long Term Cooling	No Boron Density Change
Recirc. Line Break Plus DBE (Low PRA Without Adding Single Failure Criteria)	None	None Additional Due to RLB	None	Rods Will Insert, Timing Affected	Floodable Volume Maintained	Potential Damage to CS Riser Or Sparger, CS Delivery Function not Affected.	Injection Ability Not Affected (See Note (1))

Note: (1) SBLC is not designed to function during a recirculation line break.
(2) H2 does not lift top guide

3.0 Existing Structural Margin

3.1 Water Chemistry Considerations

The BWR oxidizing environment can provide the electrochemical driving force for intergranular stress corrosion cracking (IGSCC) of BWR structural materials. Also, the conductivity of the BWR coolant is sufficiently high to allow the corrosion reaction to occur (References 6, 7 and 8).

The crack growth rate depends on the water chemistry and the conductivity of the reactor water. Both Quad Cities Units 1 and 2 have operated on HWC continuously since the third quarter of 1990. The levels of hydrogen injected range from 1.4 to 1.5 ppm in the feedwater line with a corresponding concentration in the reactor vessel of approximately 180ppb (Reference 7). The HWC system was available for approximately 57% of the time that the reactor was above 20% power for Unit 1 and 44% of the time for Unit 2. While this duration and availability is not high enough to protect reactor internals from IGSCC or IASCC, it could retard crack propagation. A comparison of conductivity measurements for Units 1 and 2 shows that the average cycle conductivities through cycle 12 are 0.257 μS/cm for Unit 1 and 0.258 μS/cm for Unit 2 (Reference 10). Unit 1 has operated for one more cycle than Unit 2 with the number of critical reactor hours (through April 30,1994) being 151,487 hours for Unit 1 and 146,195 hours for Unit 2. Given that the total hours of hot operation are greater for Unit 1, coupled with the fact that the water chemistry history is similar for both units, the cracking in Unit 2 would be expected to be similar to what was identified in Unit 1 which was determined to be acceptable.

3.2 Estimated Structural Margin

The extent of cracking that is expected at Quad Cities Unit 2 should be very similar to what was identified at Quad Cities Unit 1 (Reference 2), and while this cracking is not insignificant, sufficient structural strength remains to meet all of its design functions. The shroud is 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, the transient differential pressure increase due to design basis accident loading, design basis seismic loads, and asymmetric loads due to a recirculation suction line break. These loads are generally small and are well within the remaining structural integrity of the shroud.

The applied loads during normal operation, anticipated operational events, and the main steam line break are in the upward direction and result in a net uplift force on the shroud at H2 and H3. The applied loads during a recirculation line break will result in a net downward force on the upper shroud, but for the purpose of flaw evaluation at H2 and H3 the uplift forces corresponding to the normal operating differential pressure have been considered.

The combination of high ductility and low applied stresses make the shroud extremely flaw tolerant. It has been calculated that 360° circumferential cracking of greater than 90 % of the 2.0 inches of material can be tolerated while maintaining the industry accepted ASME Code allowable safety factors based on limit load evaluations of the H2 and H3 welds. The analysis that determines the 90 % criterion conservatively ignores the available material of the extra one inch of ligament provide by the fillet weld in addition to the two inch shroud wall thickness. Table 3-1 provides a summary of the required ligament for the H2 and H3 welds under the critical design basis and beyond design basis loading conditions. Note that the results of the TRACG analysis for the Recirculation Line Break blowdown loads (Reference 3) as well as the ComEd and BWR-VIP efforts to more accurately quantify the shroud loads (Reference 4) have been used to determine the primary membrane and bending stresses at H2 and H3.

Table 3-1 Quad Cities 2 Summary of Required Ligament

Weld Location	Critical [1] Loading Case	Maximum d/t Ratio	Required Ligament t=2"	Required Ligament t=3" (fillet)
H2	Normal	0.9959	0.0082"	0.0123"
H2	SSE	0.9926	0.0148"	0.0222"
H2	MSLOCA	0.9898	0.0204"	0.0306"
H2	SSE+MSL OCA	0.9850	0.0300"	0.0450"
H2	SSE+RR LOCA	0.9925	0.0150"	0.0225"
H3	Normal	0.9961	0.0078"	0.0117"
H3	SSE	0.9918	0.0164"	0.0246"
H3	MSLOCA	0.9895	0.0210"	0.0315"
H3	SSE+MSL OCA	0.9837	0.0326"	0.0489"
H3	SSE+RR LOCA	0.9917	0.0166"	0.0249"

Notes:

1. These required ligament sizes were computed based on the conservative differential pressures for normal, upset and faulted conditions as defined in the UFSAR.

4.0 Normal Operation

As discussed in the preceding sections, the postulation of significant through-wall cracking, leakage, or separation of the core shroud assembly at the H2 and H3 weld area is extremely improbable. Assuming that the shroud is sufficiently cracked at either the H2 or H3 welds such that an upward load would result in a lift of the upper portion of the shroud, an anomalous core characteristic resulting from the flow through the gap can be detected. This anomaly would be the result of reduced moderation in the core due to either increased coolant temperatures or reduced coolant flow. An increased coolant temperature would result from flow escaping to the outside region through shroud separation at locations above the fuel top guide (e.g. H2 and possibly H3) where two phase coolant is present. For example, considering a one quarter inch gap , the leakage flow is calculated to be 4% of the rated core flow. The resulting thermal power loss would also be 4% of rated. These power anomalies are detectable. Also present will be other abnormal core monitoring indications, such as measured recirculation flow versus core flow.

Analogous situations have previously been observed in BWRs. In 1991, Dresden 2 began startup without the shroud head bolts properly engaged, resulting in bypass flow paths similar to those that would result from through-wall cracking of the shroud. The lift of the shroud head was initiated at 70% of flow and was identified by the operators. A similar situation occurred at a different plant in 1984. In both cases, anomalies such as those described were detected and the operators safely shutdown the plant.

The lifting of the shroud is induced by the differential pressure exerted on the shroud head which exceeds the weight of the shroud above the failed weld section. Under the most limiting normal operating conditions (100% power and 100% flow) the pressure difference across the shroud head is calculated to be a maximum of 8 psi. The maximum differential pressure required to initiate a separation of the shroud at the H2 and H3 welds along with the maximum lift is listed in Table 4-1.

Some interference with the core spray piping is expected for the H2 and H3 welds. The core spray interference is not very rigid as the pipe coupling is designed and constructed to allow for some displacement. For purposes of this evaluation, it is conservatively assumed that the obstruction does not limit the amount of separation. For weld locations above the top guide (H1 and H2) the top guide does not move and the lateral support of the core is not affected. For weld H3, proper alignment of the core is assured since the separation is less than 3 inches and the top guide would need to lift 14.5 inches to lose contact with the top of the fuel channels. Since no significant rotational or lateral loads exist, the shroud would return to its original location when the differential pressure is reduced.

As shown in Table 4-1, a shroud with a 360° through-wall crack at H2 or H3 would be readily detectible prior to achieving full power operation and thus will be identified and appropriate corrective action can be initiated. This is further substantiated by the previous experience from the Dresden event in 1991.

Table 4.1 Shroud Differential Pressures and Lifts - Normal Operation

Weld Location	Shroud ΔP To Initiate Separation (psi)	Maximum Normal Shroud ΔP (psi)	Maximum * Vertical Separation (Inches)
H2	4.5	8.0	5.0
H3	5.7	8.0	2.6

* Separation at H1 and H2 does not affect core geometry.

5.0 **Anticipated Operational Events**

The previous sections demonstrate that cracks which grow through the shroud wall or cause complete separation of the shroud assembly at H2 and H3 are improbable. If such a shroud crack and separation occurs, the operators can detect it and safely shutdown the plant. This section discusses the possible impact of anticipated operational events on the shroud assuming that through wall cracks exist and have not been previously detected through normal operation. Two types of events are reviewed, first those which are considered limiting events for Quad Cities Unit 2, and then those which impose the highest loads on the shroud. The FSAR limiting events are not necessarily affected significantly by the condition of the shroud, since these events are typically evaluated to determine the minimum margin to fuel thermal limits (e.g. MCPR and LHGR) and vessel pressure limits. The highest shroud load events do not necessarily lead to limiting fuel and vessel conditions, however they may determine the maximum shroud displacement and consequently have the greatest potential to affect the shroud functions.

The limiting anticipated operational events were evaluated for the Quad Cities Unit 2 for cycle 13 operation (Reference 11). These events are the Feedwater controller Failure (FWCF) and the Main Steam Isolation Valve closure with High Flux Scram (MSIVF). These events are characterized by a core overpower condition or a rapid pressure increase, respectively. These events do not result in an appreciable increase in core flow or flow through the steam separators. Therefore, no increase in shroud loads is predicted and shroud separation will not exceed that expected during normal operation. The results of the current analysis remain unchanged and no impact on safety functions exist.

5.1 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 Flow Limiter (MCFL) allows. For the Quad Cities units, with a bypass capacity of 40% of rated steam flow, the worst case involves inadvertently increasing the steam flow. This would not happen because the steam flow limit is set at 105%. A depressurization and cooldown occurs when isolated by Main Steamline Isolation Valves (MSIV). This steam flow increase is small enough that the increased force on the shroud head (approximately 50% above the normal pressure drop) is less than the pressure differential of 14 psid due to the main steam line break (see Section 6.1). The weight of the core shroud and separators will partially resist the uplift force on the shroud head. The duration of the upper shroud portion lift is only a few seconds, and the shroud is then expected to rest again on the lower shroud portion. This event is bounded by the loads addressed in Section 6.1.

5.2 Recirculation Flow Control Failure

This postulated event involves a recirculation control failure that causes both recirculation loops to increase to maximum flow. For this event, the pressure drop could change from a part-load condition to the high/maximum flow condition over a time period of several seconds, but it should not significantly exceed the pressure drop expected for normal full power, high core flow operating conditions (8 psid). Normal operating procedures are considered sufficient to minimize the consequences of this potential transient, and the force on the shroud head is bounded by the value predicted for the main steam line break LOCA (14 psid in Section 6.1).

5.3 Inadvertent Actuation of ADS

Inadvertent actuation of the Automatic Depressurization System (ADS) valves is another postulated event that could put an increased load on the upper shroud. The maximum steam flow and the depressurization rate are significantly smaller than for the postulated main steamline break, causing a short-term increase in steam flow to approximately 130% of rated steam flow. The increase in the shroud ΔP resulting from the opening of the ADS valves would occur over a period of about one second, spreading the effect of the change in load. Inadvertent ADS is also a very low probability event; it is considered to be in the ASME Emergency category in the vessel thermal duty design. The effect of this event is bounded by loads addressed in Section 6.1.

6.0 Design Basis Accidents

This section discusses the possible impact of a design basis accident on a shroud with through-wall cracks (assuming that the cracking has not been previously detected through normal operation). The Main Steamline Break (MSLB) Accident imposes the largest potential lifting loads on the shroud head. Liquid breaks such as a recirculation line break (RLB) do not impose large pressure drops on the shroud head, and, in fact, the shroud pressure drop decreases from its initial value. However, this break results in maximum fuel temperature and consequently challenges the Emergency Core Cooling System (ECCS) functions. Additionally, the RLB imposes asymmetric lateral loads on the shroud.

6.1 Main Steamline Break

The MSLB inside primary containment is the postulated worst case because it results in the largest depressurization rate. During this SAR event, the reactor is rapidly depressurized as a result of a postulated instantaneous, double-ended break of the largest steamline. Thus a larger than normal pressure difference could develop across the shroud as fluid flow is drawn from the core region toward the break. For Quad Cities Unit 2, the design basis pressure difference defined in the UFSAR is 20 psid for the guillotine break of a main steam line (Reference 12). As discussed previously with the NRC staff during the June 27, 1994 meeting, the realistic shroud head pressure from a MSLB is expected to be slightly lower. Provided below is a description of the procedure used to estimate a more realistic shroud head pressure during a MSLB (approximately 14 psid).

The increase of the shroud head pressure difference is caused by the depressurization of the steam dome, which results from the steam blowdown through the main steam line (MSL). Therefore, one of the most important parameters that affects the shroud head ΔP is the ratio of the main steam line diameter to the reactor pressure vessel (RPV) diameter. Figure 6-1 shows the shroud head ΔP calculated for several plants by the TRACG computer code as a function of the MSL to RPV diameter ratio. This data shows a correlation between the MSL to RPV diameter ratio and the ΔP for six plants. Quad Cities Unit 2 has a relatively small diameter MSL in comparison to the RPV diameter and therefore is expected to have a relatively low shroud head ΔP. Considering the uncertainty associated with this type of analysis 14 psid was selected as the upper bound value for the shroud head pressure during the MSLB.

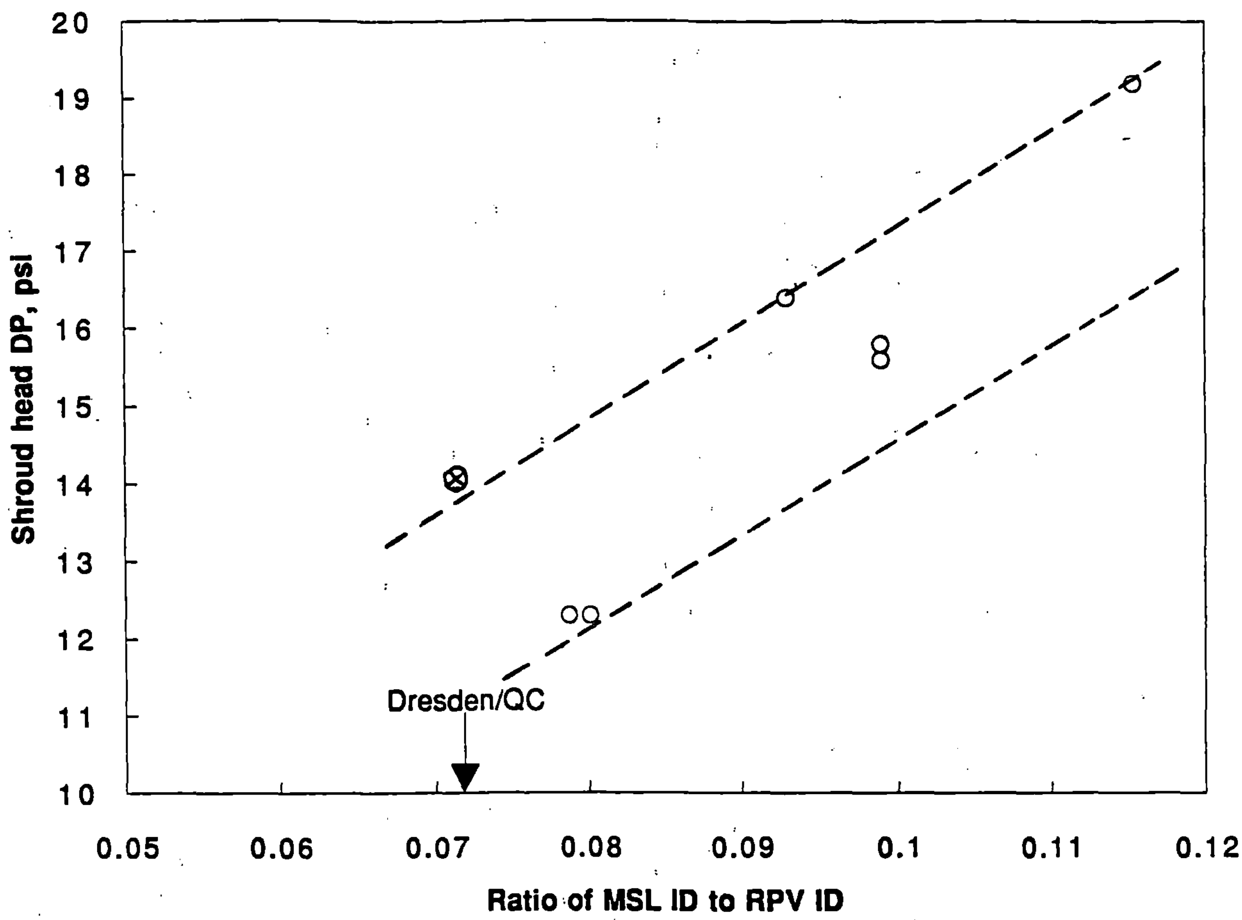


Figure 6-1 Shroud Head ΔP as a Function of the Ratio of MSL Diameter to RPV Inside Diameter

The amount of lifting and the potential effect of these postulated occurrences on emergency operation is described in the following paragraphs. One of the key considerations of this postulated accident case is the ability of the control rods to insert 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.

The shroud head pressure drop characteristics calculated for the instantaneous, double-ended steamline break accident were evaluated for a typical BWR (Reference 13). 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 dome in the short term. The 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 small (see Section 3.0), the structural integrity of the shroud will remain intact for this postulated limiting event. Note that if a 360° through wall crack is assumed, detection is possible during normal unit operation and the plant will be shut down, thus the likelihood of a lift occurring during a MSLB is even further reduced. If it is postulated that the initial load pulse causes the shroud to separate, the last part of the pressure loading could cause the shroud assembly to lift. The flow path created by any separation reduces the upward lifting forces. For this postulated scenario the core shroud assembly will lift and the result is shown in Table 6-2.

<u>Table 6-2 Maximum Shroud Lift Under MSLB Conditions</u>

Weld Location	Shroud ΔP To Initiate Separation (psi)	Maximum Shroud ΔP (psi)	Maximum * Vertical Separation (Inches)
H2	4.5	14.0	14.6
H3	5.7	14.0	10.1

* Separation at H1 and H2 does not affect core geometry.

The magnitude of the lifts shown in Table 6-2 are greater than those shown in Section 4.0 for the normal operating condition. As noted before, some interference is expected by the core spray piping penetrating the inner shroud region. This interference is not strong because the pipe coupling allows some displacement. For the purpose of this evaluation, it is conservatively assumed that the interference does not affect the magnitude of the lift . For the H3 weld, proper alignment of the core is assured if the lift is less than 14.5 inches, as the top guide would need to lift over 14.5 inches to lose contact with the top of the fuel channels. The lift calculated is slightly more than 10 inches, and thus alignment of the core is assured.

The 14.6 inch lift at the H2 weld does not affect the movement of the top guide and thus the alignment of the core is not affected and the lifting at H2 is acceptable.

Scram is initiated during the main steamline break (inside containment) accident by the high drywell pressure trip signal. Drywell pressure exceeds the setpoint almost instantaneously, so the only delays in the rod insertion come from the sensors, the Reactor Protection System, and rod motion. For the main steamline break outside containment, shroud loads are reduced, MSIV closure is initiated by high steam flow, and scram is initiated from the MSIV closure. For either postulated steamline break scenario, the insertion of all control rods will occur because core alignment is assured. With the main steam line break alone, the core shroud assembly would not move laterally, and no significant degradation of scram performance is expected.

Movement of the upper shroud assembly will affect the core spray system and as a result coolant flow to the two core spray spargers is affected. For the MSLB core cooling can be assured as long as coolant reaches the RPV. Therefore the core spray line function of water delivery to ring spray header inside the shroud is not required as long as the coolant is injected into the RPV.

The main steamline break has also been evaluated for radiological release consequences in the SAR. For a main steamline break inside of containment, the radiological consequences are bounded by the Loss of Coolant Accident. For the main steamline break outside of containment, the magnitude of the pressure loads that potentially could lead to separation of the upper shroud are less than that for breaks inside the containment, due to attenuation of the depressurization wave along the steamline. MSIV closure is initiated before any potentially increased radiological release outside containment from such a scenario could occur. The radiological consequences of this main steamline break scenario are thus still bounded by the plant SAR results.

6.2 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, upward forces are reduced, and thus there is no significant threat to core shroud integrity. No additional shroud lift over that experienced during normal operation will occur, because any significant lift prior to the RLB would be detected during normal operation as discussed in section 4.0. Lateral forces on the welds at the beginning of a RLB are large acoustic forces of very short duration, approximately five 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 a corresponding lifting force. It is possible to detect a 360° through wall flaw at H2 and H3 which rules out the possiblilty of an undetected flaw.

Both the acoustic and blowdown loads are small at the H2 and H3 weld locations because it is a substantial distance from the recirculation suction line nozzle. The acoustic load at the H2 and H3 welds is significantly less than that at H5 (less than one quarter) and the blowdown loads are much less (less than one tenth). Therefore, the consequences of a RLB for the H5 weld will bound the H2 and H3 weld locations. Even if the shroud were to lift at the H2 or H3 weld, core coolant supply below H3 would be intact and therefore long term core cooling is assured.

6.3 Safe Shutdown Earthquake

Reviews of the effects of partial through wall flaws on the seismic response of the RPV and internals indicate that the effect of the change in stiffness is not significant provided that sufficient ligament remains to transfer shear across the plane of the flaw. As previously noted, detection of a 360° through wall flaw at the H2 and H3 weld location is possible during normal unit operation and the plant will be shut down. Therefore, the evaluation of the SSE with uplift due to normal operating pressure is not necessary and is bounded by the analysis provided in section 7.1 for a combined SSE and MSLB.

The design basis earthquake for Quad Cities Unit 2 is defined in section 3.7 of the UFSAR and the design response spectra is shown in Figures 3.7-1 and 3.7-2 of the UFSAR. The design response spectrum at 5% damping for Quad Cities Units 1 and 2 is bounded by the spectrum for the 1957 San Francisco earthquake (Golden Gate Park S80E component) for periods less than 0.265 seconds and by the Housner spectra for periods greater than 0.265 seconds. Analysis of the response of a shroud with a postulated 360° through wall flaw and a lift (i.e. floating shroud) indicates that the maximum relative displacement of the core shroud with respect to the vessel is approximately equal to the maximum ground displacement. The results of this analysis correlate well with the theoretical prediction that for very low frequency systems (periods greater than 10 seconds), the separated floating portion of the shroud will not respond initially with the main structure and thus a differential displacement less than or equal to the maximum ground displacement is expected. Recent efforts to develop a synthetic time history that bounds the Housner response spectra have shown that the ground displacement characteristics for natural earthquake records have significant variability in magnitude. Table 6-3 is a summary of the displacement characteristics for the eight earthquake records that were used to develop the Housner Spectra. The corresponding maximum ground displacement for Quad Cities would be 0.71 inches for the Golden Gate time history, approximately 8 inches for the synthetic time history representing the Housner spectra and 4.18 inches for the average of the eight earthquakes comprising the Housner spectra (all values normalized to 0.24g).

Table 6-3 Summary Of Information For The 8 Earthquakes That Were Used To Develop The Housner Spectra

Earthquake Record No.	Location & Date	Direction	Acceleration (cm/s^2)	Acceleration Percent Of Gravity	Maximum Ground Displacement (cm)	Ground Displacement Normalized To 0.24g (cm)	Ground Displacement Normalized To 0.24g (Inches)
IIA001	El Centro, Imperial Valley Earthquake, 5/18/40	S00E	341.7	0.348	10.9	7.514	2.958
IIA001	El Centro, Imperial Valley Earthquake, 5/18/40	S90W	210.1	0.214	19.8	22.198	8.740
IIA004	Kern County, Taft Lincoln School Tunnel, 7/21/52	S69E	175.9	0.179	9.2	12.320	4.850
IIA004	Kern County, Taft Lincoln School Tunnel, 7/21/52	N21E	152.7	0.156	6.7	10.335	4.069
IIB024	El Centro, Imperial Valley, Lower California Earthquake, 12/30/34	N90E	179.1	0.182	3.7	4.866	1.916
IIB024	El Centro, Imperial Valley, Lower California Earthquake, 12/30/34	N00E	156.8	0.160	4.2	6.309	2.484
IIB029	Western Washington Earthquake, Olympia Washington Hwy. Test Lab, 4/13/49	S86W	274.6	0.280	10.4	8.921	3.512
IIB029	Western Washington Earthquake, Olympia Washington Hwy. Test Lab, 4/13/49	S04E	161.6	0.165	8.5	12.390	4.878
					Averages	10.607	4.176

Reference: EERL 71-50, Stong Motion Earthquake Accelerogram, September 1971, California Institute of Technology

7.0 Postulated Accidents Beyond Design Basis

7.1 Main Steam Line Break Plus SSE

If the main steam line break occurs simultaneously with the design basis earthquake, this added load is postulated to cause separation of the upper shroud assembly near the H2 and H3 weld locations, leading to an upward displacement of this structure and the associated top guide. The vertical lift is only minimally affected by the seismic excitation because the upper shroud assembly is not subjected to the seismic excitation while it is separated from the lower portion of the shroud. Calculations performed to simulate the possible shroud vertical displacement due to a seismic excitation concurrent with a MSLB results in less than 1.0 inch of additional lift. This lift is temporary as the shroud assembly above the crack returns to the lower shroud portion after approximately one second.

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. However for the portion of the shroud above the H2 and H3 welds, no tipping or rotation will occur because the resisting moment due to the shroud weight is greater than the seismic overturning moment.

The primary consideration of this postulated accident case is the capability to insert the control rods before or during the postulated accident. Specifically, the combination of lifting and lateral movement of the top guide prior to or during control rod insertion could impede the insertion of control rods. With random displacement anticipated during a seismic event (see Reference 14), the control rod alignment in the core region would undergo intermittent periods of misalignment. Hence, the CRD scram speed would vary with time as the control rods are being inserted.

The relative horizontal displacement time histories between the shroud and the vessel at the top guide and core support plate elevations, corresponding to a 360° through-wall crack at weld H3, are given in Figures 7-1 through 7-4. The input motion to the analysis was a synthetic time history generated from the Dresden smoothed design spectra given in Figures 3.7-1 and 3.7-3 of the Dresden UFSAR. The synthetic time history was normalized to a Zero Period Acceleration (ZPA) of 0.24g and has a duration of 40 seconds. The 40 second duration was required to adequately capture the low frequency characteristics. The Dresden SSE ZPA is 0.20g and the Quad Cities SSE ZPA is 0.24g. The value of 0.24g was used so that the resulting relative displacement time histories would be bounding for both Dresden and Quad Cities.

Based on actual static deflection control rod insertion tests, the dynamic factor from dynamic deflection control rod insertion tests, a 10% reduction to account for computational affects, and a 1.125 safety factor for faulted load cases, the allowable relative displacement between the shroud and the vessel is 1.50 inches at the core plate elevation and 4.80 inches at the top guide elevation.

Time history analysis were performed for two shroud connectivity conditions at the H3 weld location. The roller connectivity condition is the bounding case. However, it is a very conservative assumption since it assumes separation of the upper shroud assembly (above the H3 weld) occurs for the full 40 second duration of the SSE event. The separation actually occurs for approximately 1 second. Consequently, the relative displacement time histories corresponding to the shroud pinned connectivity condition are more representative after separation has occurred.

From Figures 7-3 and 7-4, it is observed that the maximum relative displacements at the top guide and the core plate are well below the allowables for the pinned condition for the full 40 second duration of the SSE event. Furthermore, from Figure 7-2, the maximum relative displacement at the core plate is also well below the allowable for the roller condition for the full 40 second duration of the SSE event.

Only at the top guide location for the roller condition does the relative displacement exceed the allowable value. However, assuming one second to the initiation of the scram and a one second duration for the separation of the shroud, the roller connectivity condition is over at approximately two seconds. From Figure 7-1, the peak relative displacement at the top guide before 2 seconds is less than 2.5 inches. After that time the shroud separation has closed and the actual relative displacement at the top guide will not follow the Figure 7-1 time history. However, even if it did, with a control rod insertion time of between 2 and 3 seconds, the control rods would be almost fully inserted by the time the relative displacement reached the 4.8 inch allowable, which occurs a little after 3 seconds. In any event the control rods would be fully inserted during the next oscillatory cycle.

Figure 7-1 Displacement Time History at the Top Guide with a Through-Wall Crack at H3 (Roller Connected Joint)

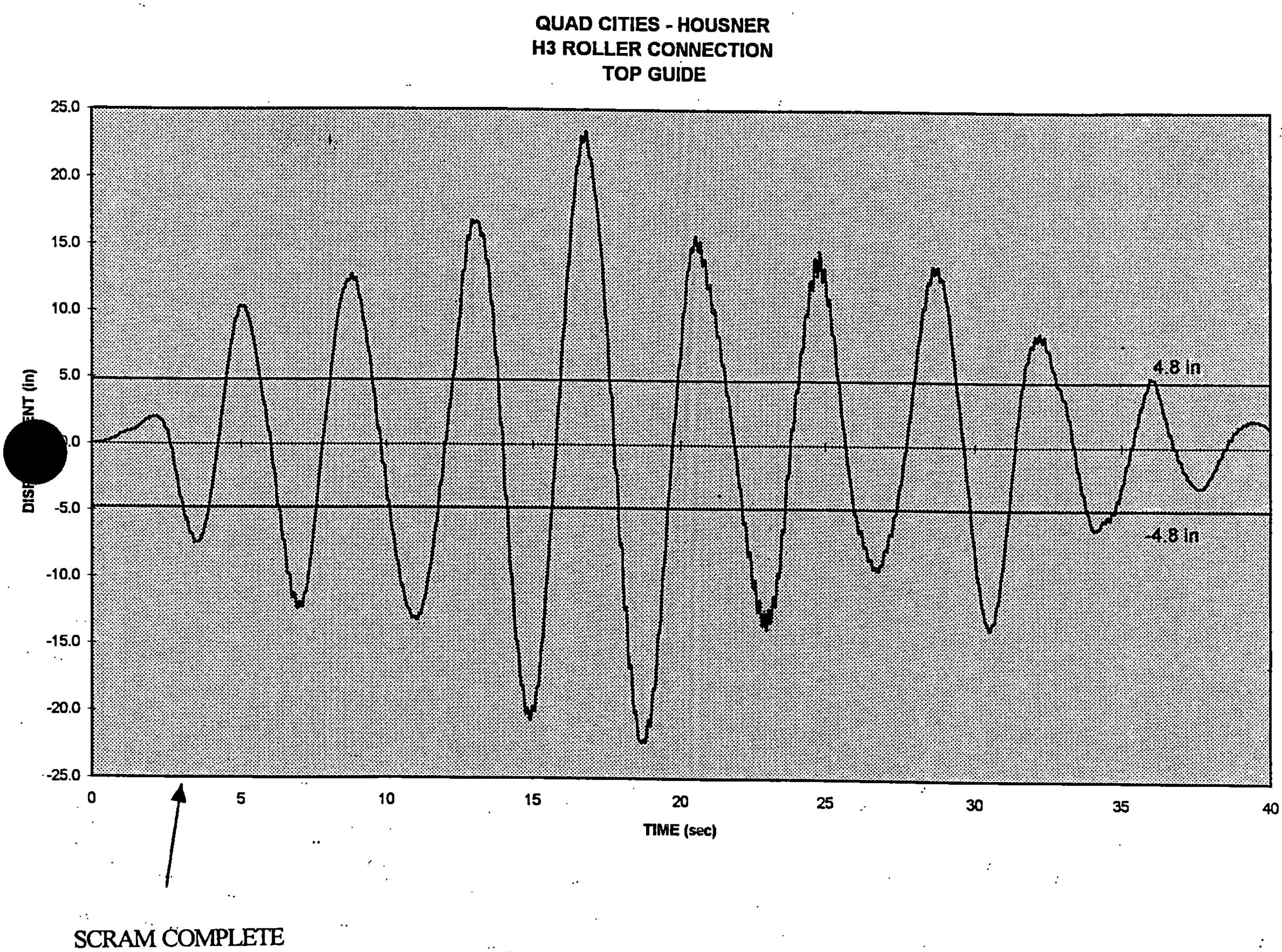


Figure 7-2 Displacement Time History at the Core Plate with a Through-Wall Crack at H3 (Roller Connected Joint)

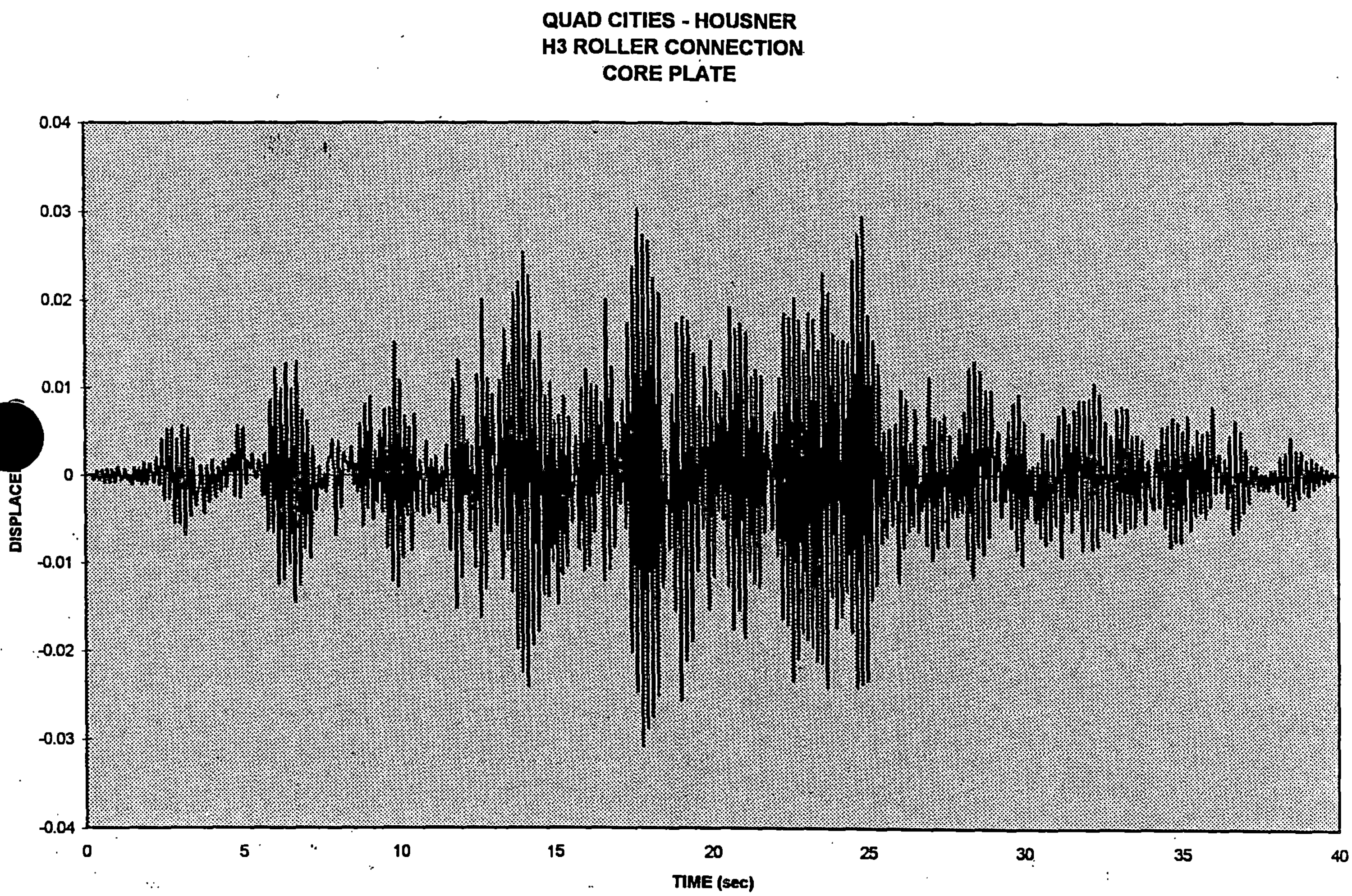


Figure 7-3 Displacement Time History at the Top Guide with a Through-Wall Crack at H3 (Pin Connected Joint)

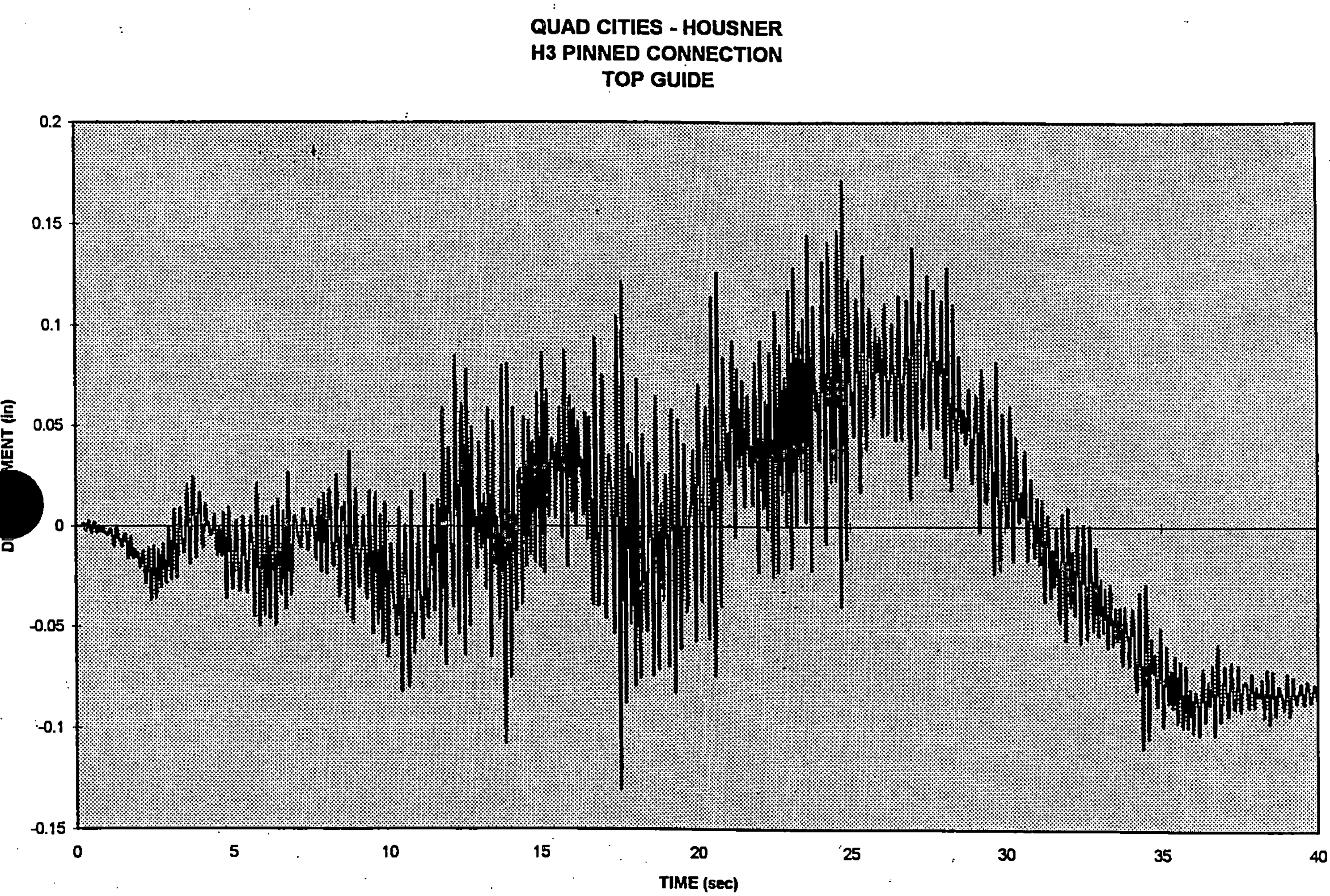
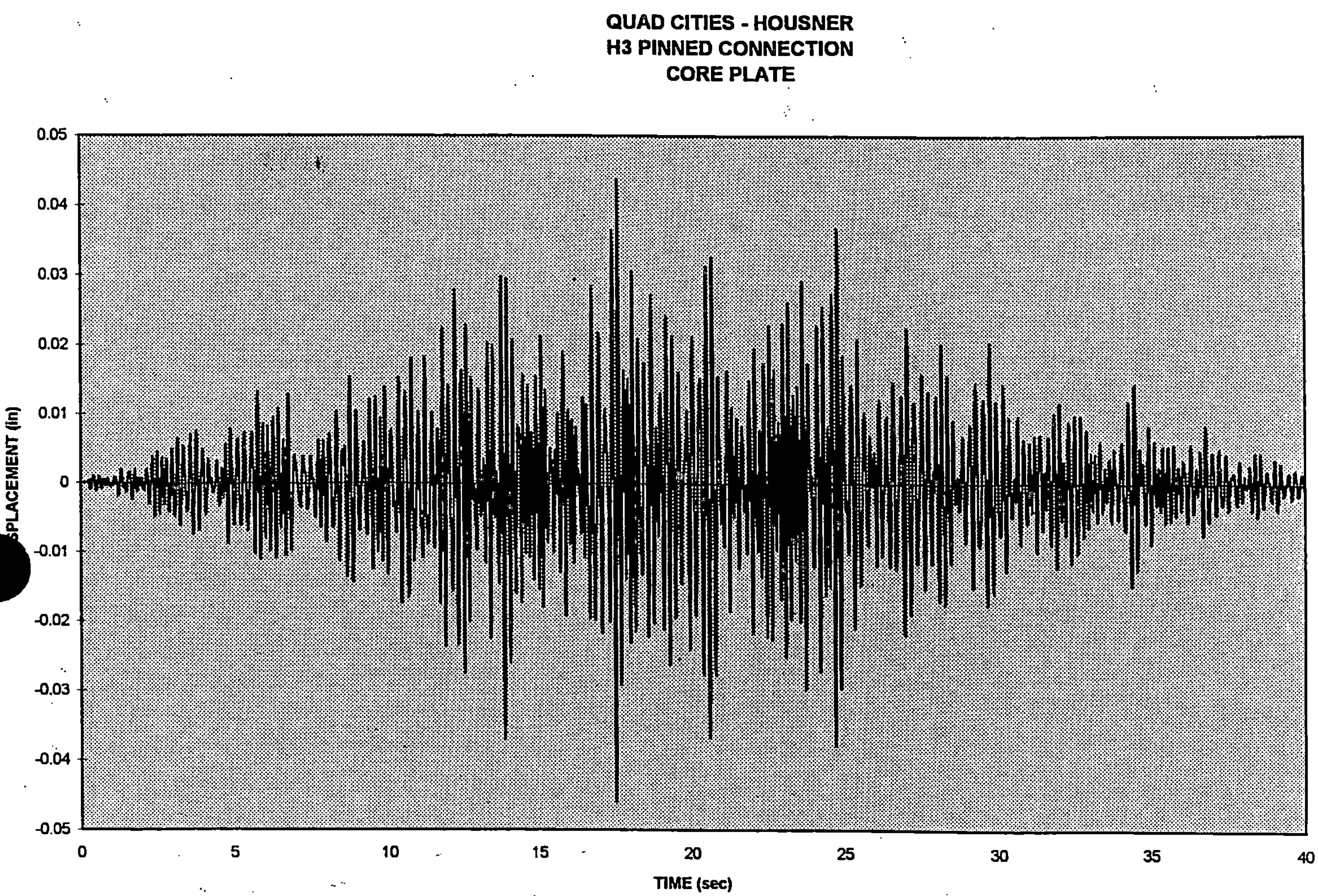


Figure 7-4 Displacement Time History at the Core Plate with a Through-Wall Crack at H3 (Pin Connected Joint)



7.2 Recirculation Line Break Plus SSE

For the RLB accident simultaneous with a seismic event, additional vertical and lateral forces will exist. The lateral seismic loads when combined with the asymmetric blowdown loads result in a larger tipping moment. As discussed in Section 6.2, the lateral load is small at the H2 and H3 weld locations during a RLB and thus the primary load will be due to the seismic excitation. The portion of the shroud above the H2 and H3 welds will not tip or rotate because the resisting moment due to the shroud weight is greater than the combined recirculation line break and seismic overturning moment. Vertical displacement of the shroud will be resisted by the downward force on the shroud exerted by the RLB. The vertical seismic excitation of 0.16 g is much less than gravity and thus will be offset by the combination of the pulldown force and the dead weight. Therefore, the combination of the RLB with the SSE does not result in a loading case or a motion that is more critical than what has been evaluated for the other events.

7.3 Probabilities of Events

The probabilities of the design basis and beyond design basis events were provided to the NRC in Reference 15 (question PR-1) and for your convenience are summarized below.

Event	Quad Cities Frequency
SSE	2.2 E-5 /year
Main Steam Line Break (MSLB)	4.1 E-8/year
Recirculation Line Break (RLB)	3.0 E-4/year
SSE coincident with MSLB*	2.5 E-15/year
SSE coincident with RLB*	1.8 E-11/year

*-For purposes of these responses 'coincident' is defined as occurring in the same 24 hour period.

These event probabilities for the beyond design basis accidents are extremely small and thus provide substantiation to the unlikeliness of the occurrence of these combined events. The ability to detect a 360° through wall flaw at H2 and H3 during normal operation rules out the possibility of having an undetected flaw prior to these events and thus concludes that the crack would have to be initiated by the accident. The probability of a through wall flaw coupled with a MSLB or RLB and a full SSE is less at H2 or H3 than at H5 because the flaw at H2 and H3 is detectable during normal operation.

8.0 Emergency Operator Actions

The Emergency Procedure Guidelines (EPGs) are the basis for plant specific Emergency Operating Procedures (EOPs). The EPGs 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 discussed above could result in separation of the upper shroud assembly from the section of the shroud below the H3 weld, which has minimal impact to scram performance. Therefore, no further consideration is necessary for the impact of this postulated event on the EPGs.

The EPGs provide instructions for reactor pressure, water level, and power control, as well as control of key primary containment parameters. Actions specified in the EPGs 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 value (typically 110 F). EPG instructions are for water level to be controlled below the high water level setpoint; thus there would not be dilution of the liquid boron by flooding to the steamline elevation or loss of vessel inventory out the break in case SLCS injection were to occur.

Water level would be controlled after the postulated event because the most challenging break, MSLB, is high in the vessel and a large compliment of water injection systems would be available.

9.0 References

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3. GE Report GE-NE-L12-00819-05 Rev. 0, DRF L12-00819 (26), "Core Shroud Blowdown Load Calculation During Recirculation Suction Line Break by TRACG Analysis, for Dresden Nuclear Power Station Units 2 and 3, and Quad Cities Nuclear Power Stations Units 1 and 2", August 1994, (Proprietary Information is Included).

4. BWR-VIP Draft report SL-4942 Rev. C, "BWR Core Shroud Evaluation Load Definition Guideline", Calculation 9511-00, September 23, 1994.

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6. GE Rapid Information Communication Services Information Letter (RICSIL) 068, Revision 2, Update on Core Shroud Cracking, May 6, 1994.

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9. Telecon, Dilip Rao (GE) and Tracy Finnegan (ComEd), May 5, 1994.

10. Internal Communication, Barry Gordon/Dillip Rao (GE), May 5, 1994.

11. GE/NFS Letter 23A7197 Rev. 0, November 1992, "Supplemental Reload License Report For Quad Cities Nuclear Power Station Unit 2".

12. Updated Final Safety Analysis Report, Quad Cities Units 1 and 2, Tables 3.9-19 and 3.9-20, and Figure 3.9-5.

13. Letter dated November 9, 1993, from L. A. England to US NRC, "BWROG Safety Assessment - BWR Shroud Crack Indications".

14. Strong Motion Earthquake Accelograms, Digitized and Plotted Data, Corrected Accelograms and Integrated Ground Velocity and Displacement Curves, EERL 71-50, California Institute of Technology, Earthquake Engineering Research Laboratory, September 1971.

15. Letter from ComEd to W. Russell (NRC), dated July 8, 1994, transmitting the response to the June 23, 1994 NRC request for additional information.