UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 22, 1994

MEMORANDUM TO: Jack R. Strosnider, Chief
Materials and Chemical Engineering Branch
Division of Engineering

FROM: Robert Jones, Chief
Reactor Systems Branch
Division of Systems Safety & Analysis

Timothy Collins for

SUBJECT: DSSA SER INPUT TO EMCB REGARDING MOST SUSCEPTIBLE PLANTS TO CORE SHROUD CRACKING GENERIC LETTER 94-03 SUBMITTALS AND THE BWR SHROUD CRACKING GENERIC SAFETY ASSESSMENT

SRXB has reviewed the submittals for the plants EMCB considered to be the most susceptible to core shroud cracking. The submittals reviewed were Pilgrim, Nine Mile Point 1, Hatch 1, Oyster Creek, Dresden 2, Quad Cities 2, and FitzPatrick. Dresden 2 and Quad Cities 2 response referenced previously submitted documents and analyses of the H5 weld as justification for continued operation. A request for additional information was sent to Dresden 2 and Quad Cities 2 which is required to be answered by October 10, 1994. The staff has determined that two plants', Nine Mile Point 1 and Hatch 1, submittals were unsatisfactory with regards to plant specific assessment of core shroud response to structural loadings resulting from design basis events. Hatch 1 has been in a refueling outage since September 20, 1994, and therefore, the staff concluded that no safety concerns existed due to the low probability of a design basis event occurring during that limited time frame. Nine Mile Point 1 was sent a request for additional information which is required to be answered by September 26, 1994.

In addition, SRXB has reviewed the BWROG BWR Shroud Cracking Generic Safety Assessment specifically Appendix A. The staff has concluded that plant specific information is required to judge whether control rod insertion can be assured during a main steamline break (MSLB) and a MSLB plus seismic event. SRXB also had concerns regarding the conclusions for the recirculation line break (RLB) and the RLB plus seismic event since GE had not completed a detailed calculation of the magnitude of the blowdown loads resulting from the events. Although Appendix A to the generic safety assessment provided valuable insights regarding the safety consequences associated with operation with 360° through-wall shroud cracks, the consequence evaluation alone does not provide adequate justification for the continued operation of any one individual plant.

cc: G. Holahan
M. Virgilio

Attachments:
As noted

Contact: K. Kavanagh, SRXB/DSSA, 504-3743

DISTRIBUTION
Docket File
SRXB R/F
RJones
MRubin
[redacted]
Nine Mile Point 1 P/F
KKavanagh R/F
ACubbage
Pilgrim P/F
Oyster Creek P/F
FitzPatrick P/F
Hatch 1 P/F
Dresden 2 P/F
Quad Cities 2 P/F

H-6

SRXB:DSSA	SRXB:DSSA	SRXB:DSSA
KKAVANAGH	MRUBIN	RJONES
9/ -7/94	9/22/94	9/ /94

DOC NAME: G:\EMCB.SER

9409280326 XA

<u>Pilgrim Licensing Basis Events</u>

The licensee did not believe that 360° through-wall crack could exist without being detected but postulated complete weld failures at various locations to provide a bounding consequence assessment.

The licensee provided a plant specific evaluation of a large recirculation line break (RLB) assuming applied loads are sufficient to cause shroud separation along any one particular weld location. The licensee stated that the RLB does not impose large pressure drops on the shroud, but does impose lateral forces. The licensee believed that there is no significant threat to core shroud integrity since upward forces are immediately reduced following the RLB. The licensee performed a detailed analysis of the lateral forces, acoustic and blowdown, during the RLB. The licensee stated that lateral displacement and tipping of the large shroud mass will not occur due to the short duration of the acoustic loading. Lateral motion during blowdown was not expected due to the resistance of the irregular crack surface to lateral motion without lifting. The licensee used a new calculation for the blowdown force that resulted in a limiting tipping moment at the H7 location. However, the shroud tipping, which would be 1 to 2 seconds in duration, has an associated restoring moment that would return the shroud to its original position.

The licensee performed a main steamline break (MSLB) evaluation using the TRACG model. The postulated MSLB event would result in a large upward load on the shroud which could impact the ability of the control rods to insert and the ability of the core spray system to perform its safety function. For Pilgrim, a maximum displacement of 11.5 inches is required to clear the top of the active fuel (TAF). The licensee's TRACG model analysis established that the maximum vertical displacement of the core shroud upon a MSLB was 10.3 inches. The calculation demonstrated that the lifting load lasted less than three seconds and then the shroud rested on the lower shroud portion again. Based on these findings, the licensee believed that proper alignment of the core was assured and no additional reactor components were affected.

The licensee provided an evaluation of a seismic event assuming a postulated 360° through-wall crack in combination with a MSLB and a RLB, respectively. The possible shroud displacement due to the seismic loads during the lifting portion of a MSLB resulted in an additional 1 inch vertical displacement, i.e. 10.3 inches for a MSLB and 11.3 inches for a MSLB plus seismic event. This vertical displacement of 11.3 inches was 0.2 inches below the maximum displacement of 11.5 inches required to clear the TAF. An additional 1 inch lateral displacement also resulted from the MSLB plus seismic event. The licensee postulated that the lateral displacement may be permanent as the upper shroud may not be perfectly aligned when it returns to rest on the lower shroud portion. The lateral seismic loads will also apply a tipping moment at the H7 location. The tipping moment will not result in shroud tipping or rotation due to the restoring moment of the shroud. The licensee concluded that the vertical and lateral displacements resulting from the MSLB plus seismic event does not impact core geometry or affect any additional reactor components. For the RLB plus a seismic event, additional vertical and lateral forces exert an almost instantaneous downward pull on the shroud that would prevent vertical and lateral displacement along a weld location. However, lateral seismic loads combined with asymmetric blowdown loads result in momentary tipping at H7. The licensee stated that the largest tipping at H7 was expected to be 1.5 inches and was limited by the jet pump riser braces. Furthermore, the jet pump riser braces are not adversely affected by the loads and the restoring moment would prevent permanent displacement. The licensee concluded that the added displacement calculated for the seismic event momentarily impacts the jet pump riser braces, but the results of a RLB remain unchanged.

Additionally, the licensee stated that the operators or the reactor engineers would detect a failed weld based on a power reduction of 4-7% through routine log taking or review of core performance data. The licensee stated that every 24 hours the reactor engineers review the process computer data, where they would observe the 4-7% power reduction. Training has not been provided to the reactor engineers or the reactor operators such that they would recognize the cause of the power reduction. The licensee stated that special instructions on recognizing a core shroud crack would be provided to the reactor operators prior to returning to full power operations. The reactor operators have been trained on the shroud head lifting scenario. The licensee intends to review and incorporate, if appropriate, the BWROG guidance for operator actions to detect potential shroud cracks during normal operation when the guidance is issued.

The staff performed a qualitative assessment of the licensee's submittal. The staff found the submittal to be a relatively complete assessment of the consequences of a RLB with acoustic and blowdown loads, a MSLB, a MSLB plus seismic event, and a RLB plus seismic event with acoustic and blowdown loads with regards to the Pilgrim design. The licensee did not provide an assessment of a seismic event alone assuming a 360° through-wall crack. The staff believes that this scenario should be bounded by the MSLB plus seismic and the RLB plus seismic evaluations.

Due to the relatively small clearances remaining at the top guide structure for the postulated failure of the H3 weld during a MSLB plus seismic event (0.2 inches), the staff does not believe that lateral fuel movement can be ruled out. However, the staff recognizes that the probability of occurrence of the combined events is low and that the standby liquid control system should provide reactivity control if rod insertion is impacted for this scenario.

At this time, Pilgrim is in cold shutdown for a main generator maintenance outage and is not schedule to return on line until December 2, 1994. The licensee's next planned refueling outage is scheduled for April 1995. The licensee provided the results of a plant specific probabilistic risk assessment (PRA) assuming the combination of the worst case pipe break, i.e. a RLB, and shroud weld failure causes the loss of the ability to insert control rods, the loss of the SLCS, and the failure of core spray. For Pilgrim, the probability of core damage for this worst case scenario for seven months of operation was 4.4E-6. Currently, Pilgrim will operate for five months and not seven as assumed in the PRA before the next refueling outage in April, 1995. The staff concludes that continued operation for the five month period from December, 1994, to April, 1995, is justified based on the results of the plant specific licensing basis events assessment and the results of the plant specific PRA.

Nine Mile Point 1 Licensing Basis Events

The licensee stated that Nine Mile Point 1 was bounded by the BWROG generic safety assessment, dated August 5, 1994, and that finding a 360° crack with an average depth in excess of 90% during the Fall 1994 inspections at plants in refueling outages was unlikely. Therefore, the licensee will use the Fall 1994 outage inspection results of Oyster Creek to determine the uncertainty associated with the potential for finding a 360° crack with a depth in excess of 90%. The licensee stated that the consequences of 360° through-wall crack was reviewed as part of the BWROG generic safety assessment.

The licensee did not provide a plant specific evaluation of a large recirculation line break (RLB) assuming a postulated 360° through-wall crack at the H8 weld. The licensee stated that a vertical displacement of the core shroud upon H8 failure during a RLB was not a credible failure based on reduced weld tensile residual stresses in H8. To support this claim, the licensee estimated that 1/4 inch of ligament was required in H8 to support the RLB download of two million pounds, which is the limiting load for H8. The licensee concluded that cracking in H8 would have to be about 90% through-wall for 360° in order for the weld to fail under limiting accident conditions. The licensee did not evaluate the plant specific acoustic and blowdown loads associated with a RLB.

The licensee did not perform a main steamline break (MSLB) evaluation for Nine Mile Point 1 but referenced the RELAP 5 Oyster Creek model. The staff has reviewed Oyster Creek's MSLB evaluation which stated that an upward displacement of 13.31 inches was required to clear the top of the active fuel (TAF). The RELAP 5 Oyster Creek model analysis established that the maximum displacement of the core shroud upon an instantaneous guillotine failure of the main steamline is 13.285 inches. Oyster Creek concluded that since the shroud displacement was below the TAF, the postulated MSLB with a 360°through-wall crack was not significant. The staff does not agree that 0.025 inches provided sufficient margin of safety to consider the postulated MSLB with a 360° through-wall crack as not significant.

The licensee did not provide an evaluation of a design basis earthquake (DBE), a RLB plus a seismic event with acoustic and blowdown loads, and a MSLB plus a seismic event assuming a postulated 360° through-wall crack at any of the core shroud welds.

The licensee assessed the core damage frequency (CDF) of a shroud weld failure combined with the design basis seismic, or RLB, or MSLB. The CDF was estimated to be 4.8E-8 /six months. The staff could not evaluate this CDF assessment since all the assumptions and initiating frequencies were not presented. Furthermore, the licensee provided the probability of failure of the H8 weld to be 1.0E-3 per RLB event. The licensee stated that this failure probability was conservatively assigned due to the high reliability of the H8 weld. The staff does not agree with this assumption and concludes that no basis exists for the 1.0E-3 per RLB event failure frequency of the H8 weld.

Generic guidance regarding through-wall crack indication during normal operations has been provided to the reactor operators and has been incorporated into the normal operating procedure for the Nuclear Steam Supply System (N1-OP-1). The licensee stated that the procedure has been revised to alert operators of the expected plant response should a through-wall shroud crack develop. This training was provided to all operations crews prior to their resumption of shift duties.

The staff performed a qualitative assessment of the licensee's submittal. The staff found the submittal was incomplete for assessment of the unique vulnerabilities of Nine Mile Point 1. The licensee's submittal was dependant on the results of the Oyster Creek inspections. The staff acknowledges the similarities between the Oyster Creek and Nine Mile Point 1 design. However, Oyster Creek had installed 36 brackets symmetrically around the base of the shroud to structurally replace furnace sensitized components during construction. These brackets are not part of the Nine Mile Point 1 design and therefore a plant specific assessment of the shroud response to the structural loading resulting from design basis events is required for continued operation.

The staff concluded that a plant specific assessment of seismic loads, the RLB with acoustic and blowdown loads with regards to the H8 weld, and the MSLB with regards to the H3 weld must be done as a minimum to justify continued operation until February 1995.

Hatch 1 Licensing Basis Events

The licensee has proposed to structurally replace all circumferential shroud welds, H1 through H8, with four low tension tie rod assemblies in the annular space between the core shroud and the reactor pressure vessel wall during the refueling outage scheduled to begin September 21, 1994. The licensee assumed that all welds have 360° through-wall cracks and therefore, no inspections of the shroud welds are planned.

The licensee stated that a plant specific analysis was not performed due to the short response time required by Generic Letter 94-03. The licensee has reviewed the BWROG's generic safety assessment, dated August 5, 1994, and believes that Plant Hatch is bounded by the susceptibility grouping and analysis presented in the report. The licensee performed a probabilistic evaluation of the potential core damage contribution of operating with an undetected cracked shroud for a one month period. Using a 0.75 plant availability, the probability of core damage was determined to be 1E-5. The licensee also compared the Hatch individual plant evaluation (IPE) results to other documents, such as the Dresden NRC SER, NUREG-1150, and NUREG/CR-4792, which contained PRAs of core damage frequency due to a main steamline break (MSLB), a recirculation line break (RLB), seismic events, and MSLB or RLB plus seismic events. The staff could not evaluate these comparisons since the all the assumptions and initiating frequencies were not known.

The licensee did not provide plant specific assessment of the consequences of a RLB with acoustic and blowdown loads, a MSLB, seismic event, a MSLB plus seismic event, and a RLB plus seismic event with acoustic and blowdown loads with regards to the Hatch 1 design. Therefore, the staff could not perform a qualitative assessment of the licensee's submittal.

The staff concluded that the licensee's submittal did not fully address all information requests of Generic Letter 94-03. However, the staff acknowledges that the licensee has put forth considerable effort to prepare for the repair of the Hatch 1 core shroud during the refueling outage scheduled to begin September 21, 1994. Furthermore, the staff believes that the operation of Hatch 1 prior to the shutdown was not a safety concern due to the low probability of a design basis event occurring during that limited time frame.

Oyster Creek Licensing Basis Events

The licensee did not believe that 360° through-wall crack could exist without being detected but postulated complete weld failures at various locations to provide a bounding consequence assessment.

The licensee provided an evaluation of a large recirculation line break (RLB) assuming a postulated 360° through-wall crack at the H8 weld. The licensee had installed 36 brackets symmetrically around the base of the shroud to structurally replace furnace sensitized components during construction. The licensee concluded that the existence of the 36 brackets prevents any vertical displacement of the shroud and prevents damaging the core spray system piping upon H8 and/or H7 weld failures. The licensee also evaluated the acoustic loads resulting from a recirculation suction line failure. The licensee stated that the very short acoustic loading interval (milliseconds) will not be sufficient to tip the shroud with a postulated 360° through-wall crack. The licensee did not evaluate the blowdown loads which continue past the duration of the acoustic loads resulting from a RLB.

The licensee performed a main steamline break (MSLB) evaluation using the RELAP 5 computer code. The postulated MSLB event would result in a large upward load on the shroud which could impact the ability of the control rods to insert and the ability of the core spray system to perform its safety function. For Oyster Creek, an upward displacement of 13.31 inches is required to clear the top of the active fuel (TAF). The licensee's RELAP 5 analysis established that the maximum vertical displacement of the core shroud upon an instantaneous guillotine failure of the main steamline was 13.285 inches. The licensee concluded that since the shroud displacement was below the TAF, the postulated MSLB with a 360° through-wall crack is not significant. The staff does not agree with the licensee that 0.025 inches provides sufficient margin of safety to consider the postulated MSLB with a 360° through-wall crack as not significant.

The licensee provided an evaluation of a design basis earthquake (OBE) assuming a postulated 360° through-wall crack at the H3 weld as the bounding seismic event. The maximum seismic ground displacement is 3.96 inches for the OBE. The licensee concluded that the actual differential displacement of the core shroud at H3 would not exceed 3.96 inches and would not prevent control rod insertion during an OBE. The licensee did not provide an assessment of a DBA LOCA plus a seismic event since it was not design or licensing basis. The staff believes that the probability of a DBA LOCA plus a seismic event is low for the short duration of operation, and therefore has not requested an assessment of these events.

Additionally, the licensee stated that the operators have the ability to recognize a failed weld by observing power reductions without a corresponding reduction in recirculation flow. Oyster Creek staff participated in the development of the BWROG guidance for operator actions to detect potential shroud cracks during normal operation. The licensee stated that both the shift technical advisors (STAs) and the reactor operators have attended an hour session on core shroud cracking, failure modes, and indications. The licensee stated that the STAs have a program by which they monitor core performance for core shroud cracking indications.

The staff performed a qualitative assessment of the licensee's submittal. The staff found the submittal to be a complete assessment of the unique vulnerabilities in the BWR-2 design as described in the BWROG generic safety assessment submittal dated August 5, 1994. The presence of the H8/H7 brackets should preclude the most significant shroud vulnerability to RLB, even though the licensee's evaluation was incomplete since they neglected blowdown loads.

Due to the small clearances remaining at the top guide structure for a postulated failure of the H3 weld for MSLB events (0.025 inches), the staff does not believe that lateral fuel movement can be ruled out. However, the standby liquid control system (SLCS) operation should provide reactivity control if rod insertion is impacted for this scenario.

The staff believes that the low likelihood of pipe breaks that could challenge a failed shroud weld results in low risk for the short operation period planned prior to inspection. Therefore, the staff concluded that continued operation until the refueling outage of September 10, 1994, was not a safety concern.

Dresden 2 and Quad Cities 2 Licensing Basis Events

The licensee's response referenced previously submitted documents and analyses of the H5 weld as justification for continued operation for Dresden 2 and Quad Cities 2 until March, 1995 and January, 1995, respectively. Reference (e) in the licensee's response, ComEd letter (P. Piet) to the NRC (W. Russell), "Dresden Nuclear Power Station Unit 3, Quad Cities Nuclear Power Station Unit 1, Shroud Cracking Issue Response to Request for Additional Information (RAI)," dated July 8, 1994, attachments ME-1-5 and ME-1-6, were evaluated with regard to Generic Letter 94-03. In this document, the licensee assessed the consequences of a main steamline break (MSLB), a MSLB plus seismic event, and a recirculation line break (RLB) for a postulated 360° through-wall crack at H5. The licensee did not assess the acoustic and blowdown loads which occur during a RLB or assess a RLB plus seismic event.

The staff performed a qualitative assessment of the licensee's submittal. The staff found the submittal to be an incomplete assessment of the unique vulnerabilities of Dresden 2 and Quad Cities 2 designs. The staff acknowledges the fact that the licensee does not believe that the postulated shroud cracking could be any worse than that seen at Dresden 3 and Quad Cities 1. However, the staff concluded that a plant specific assessment of the RLB with acoustic and blowdown loads and the MSLB must be performed for the H2 and H3 welds, as a minimum requirement, to justify continued operation until March, 1995 for Dresden 2, and January, 1995 for Quad Cities 2.

FitzPatrick Licensing Basis Events

The licensee did not believe that 360° through-wall cracks could exist without being detected by operators trained to identify the characteristic flow and power anomalies during full power operation. The licensee has contracted General Electric (GE) to perform plant specific analysis of the main steamline break (MSLB) and recirculation line break (RLB) loads assuming 360° through-wall cracking of welds. The licensee has committed to provide the results of the GE analysis to the NRC by October 17, 1994. Until that time, the licensee provided a plant specific probabilistic risk assessment of the core damage frequency of the five initiating events.

The licensee modeled a recirculation line break (RLB), a main steamline break (MSLB), a design basis earthquake (DBE), a seismic-induced MSLB, and a seismic-induced RLB concurrent with a postulated 360° through-wall crack at the H2 to H6A welds. The licensee did not model H1, H6B, H7, and H8 based on the BWROG generic safety assessment, dated August 5, 1994, that weld separation at those locations do not affect the capability to insert control rods or the emergency core cooling systems (ECCS) safety function. The licensee provided the initiating event frequencies per year and assumptions used in the models. Different core shroud failure probabilities for each postulated initiating event were selected since core shroud failure is considered to be unlikely.

For the RLB, the licensee assumed significant shroud displacement existed because of the uncertainty of blowdown loads and also assumed the inability to reflood the reactor vessel to two-thirds core height and standby liquid control system (SLCS) effectiveness given control rod insertion failure. Two core damage sequences, 11 and 12, were postulated based on those criteria. Sequence 11 consisted of significant shroud displacement and the subsequent loss of reflood capability, control rod insertion, the failure of core spray, and core damage. Sequence 12 was similar to sequence 11 except that core spray was available and control rod insertion fails. The core damage frequency (CDF) was estimated to be 2.34E-7 /year and 5.0E-7 /year for sequences 11 and 12, respectively. The combined CDF for sequences 11 and 12 was estimated to be 7.34E-7 /year.

For the MSLB, the licensee postulated shroud displacement and failure of control rod insertion and assumed high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) are inoperable. Two core damage sequences, 6 and 7, were postulated based on those criteria. Sequence 6 involved successful SLCS performance but inadequate boron mixing due to the use of low pressure systems. Sequence 7 considered SLCS failure to mitigate a failure of control rod insertion given shroud displacement. The CDF was estimated to be 4.95E-7 /year and 1.0E-8 /year for sequences 6 and 7, respectively. The combined CDF for sequences 6 and 7 was estimated to be 5.05E-7 /year.

The licensee provided the CDF for a DBE of 0.15 g magnitude with a postulated 360° through-wall crack as 9.2E-5 /year which was calculated in a previous FitzPatrick document. The licensee stated that potential shroud displacement was small without a concurrent MSLB or RLB and was not expected to adversely effect core cooling, control rod insertion, and SLCS effectiveness.

The licensee postulated a MSLB and DBE with a 360° through-wall crack. The licensee stated that the additional loads exerted by the DBE would result in greater shroud displacement than a MSLB, and increase the likelihood of control rod insertion, SLCS effectiveness, and core spray operation failure. However, since the main steamline is seismically qualified, it is expected to remain intact during a DBE. The CDF for seismic-induced MSLB sequences that result in core damage was estimated to be 1.96E-8 /year. The staff agrees that the main steamline is seismically qualified up to the main steam isolation valves (MSIV). However, the staff believes that a MSLB downstream of the MSIVs will produce the same amount of lift of the top guide due to the closure times of the MSIVs. Therefore, the staff has concluded that a MSLB downstream of the MSIVs coincident with a DBE should be considered even though the probability of occurrence is low.

The licensee also postulated a RLB and DBE with a 360° through-wall crack in addition to the inability to reflood to two-third core height and failure of control rod insertion and SLCS effectiveness. The CDF for seismic-induced RLB sequences that result in core damage was estimated to be 1.13E-8 /year.

The licensee stated that even if circumferential through-wall cracking occurred, plant operators have received specific training to detect and respond to cracking during normal operations using existing instrumentation. Appropriate procedures and lesson plans have been revised to address the potential of the occurrence of through-wall cracking. Furthermore, a computer based video animation of the onset and progression of core shroud cracking was developed for use in conjunction with the training materials. The licensee stated that they would review and incorporate, if applicable, the BWROG guidance for operator actions to detect potential shroud cracks during normal operation when the guidance is issued.

The staff performed a qualitative assessment of the licensee submittal. The staff found the submittal to be a complete assessment the risk aspects of a RLB, a MSLB, a DBE, a seismic-induced MSLB, and a seismic-induced RLB with regards the FitzPatrick design. There are some assumptions in FitzPatrick's risk assessment that still need to be verified. However, the staff concluded that the risk assessment prepared by the staff for Dresden 3 and Quad Cities 1 appeared to be the upper boundary for FitzPatrick and would justify acceptably small risk until the next refueling outage scheduled for November 29, 1994. Nonetheless, Generic Letter 94-03 required a plant specific safety assessment of the shroud response to the structural loading resulting from design basis events. Since the licensee has committed to providing a plant specific assessment by October 17, 1994, and since the probability of a design basis event coupled with a 360° through-wall crack is low, the staff believes that continued operation is justified until the plant specific safety assessment is confirmed to agree with the above conclusions.

SRXB EVALUATION OF THE BWR SHROUD CRACKING GENERIC SAFETY ASSESSMENT

General Electric concluded in the BWR Shroud Cracking Generic Safety Assessment that there is a very low likelihood of a 360°, greater than 90% deep crack existing in conjunction with a design basis LOCA event. For information, the report provided an assessment of the safety consequences associated with 360° through-wall cracking. This assessment is provided in Appendix A, "Shroud Cracking Safety Assessment," of the BWR Shroud Cracking Generic Safety Assessment Report. The staff's evaluation of this report is discussed below.

The safety assessment provides a discussion of the detectability of 360° through-wall cracking at the different weld locations. They have concluded that through wall cracking would be identifiable during normal operation for most weld locations at most plants. The report states, however, that additional training or special procedures may be needed to facilitate identification of characteristics that would lead to detection. The BWROG Emergency Procedure Committee (EPC) is currently working on this activity, and will be providing guidance on this matter to the BWR utilities in the near term.

The safety assessment also provided a discussion of the safety consequences associated with normal, transient and accident conditions with 360° through-wall cracking in the shroud at various weld locations. The report concluded that for normal operation and transient conditions there were no safety consequences associated with 360° through-wall cracks. For accident conditions, four scenarios were evaluated: main steam line break, main steam line break plus seismic event, recirculation line break, and recirculation line break plus seismic event.

For the main steam line break scenario, the results reported by GE indicate that the maximum attainable lift of the detached upper shroud is less than that required to clear the fuel channels for BWR/3, 4, and 5 plant designs. These results were obtained using TRACG. Acceptable results were not shown, however, for the BWR/2 and BWR/6 design plants. GE expects that further evaluations with less bounding assumptions would yield acceptable results. Unacceptable lifts could impact control rod insertion, but the SLCS system would still be available for reactor shutdown. Core spray may also be impacted, but because of the nature of the main steam line break accident, core cooling would not be impacted as the ability to reflood the core would not be compromised. The staff concludes that plant specific information is required to judge whether the control rod insertion can be assured during this accident scenario, particularly for the BWR/2 and BWR/6 plant designs.

For the main steam line break plus seismic scenario, additional shroud motion to that assumed for main steam line break alone would be expected. GE has concluded that the effect of seismic loads would be less than 1 inch in both the vertical and lateral directions, and also for rotation (tipping). This was based on specific plant evaluation and engineering judgement. Again, the staff concludes that plant specific information must be provided to judge whether the control rod insertion can be assured during this accident scenario

For the recirculation line break scenario, GE has concluded that shroud motion is not expected. There are two components of the loads during a recirculation line break, acoustic and blowdown. Since GE had not completed detailed calculation of the magnitude of the blowdown loads at the time the report was submitted, the staff cannot judge the conclusion made by GE that no shroud motion will result during a recirculation line break.

Lastly, GE evaluated the combined recirculation line break and seismic event. While GE concluded that only small momentary tipping of the shroud (3/4 inch) could be expected, the staff has the same concerns regarding this conclusion as with the recirculation line break scenario because detailed blowdown load calculation loads have not been completed.

While Appendix A to the generic safety assessment provides valuable insights regarding the safety consequences associated with operation with 360° through-wall shroud cracks, for the reasons discussed above, the consequence evaluation alone does not provide adequate justification for the continued operation of any one individual plant. It is for this reason, in part, that after rev. 0 of this document was submitted on July 13, 1994, the staff issued Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds In Boiling Water Reactors," on July 25, 1994. In this document, the staff requested all BWR licensees with the exception of Big Rock Point in reporting requirement 1 (b) to provide "a safety analysis, including a plant-specific safety assessment, as appropriate, supporting continued operation of the facility until inspections are conducted." Individual licensees have responded to this request in their Generic Letter responses which were due by August 25, 1994. The staff is in the process of reviewing the safety assessment provided in these responses on a plant specific basis.