SAFETY ASSESSMENT

BWR SHROUD CRACK INDICATIONS

Prepared for the BWR Owners' Group
by GE - Nuclear Energy

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Executive Summary

This document provides information to address for all Boiling Water Reactors the safety significance of the circumferential crack indications in the heat-affected zone of the top guide support ring weld of the core shroud assembly. The report concludes that the observed phenomenon does not represent a threat to the safe operation of the plant.

The shroud provides a partition between the core region and the downcomer annulus to separate the upward flow of core coolant from the downward recirculation flow. The shroud is not a primary pressure boundary component.

The shroud is made of ductile material with high toughness properties even after accounting for any effects due to neutron fluence, while the applied loads on the shroud are generally small. The combination of ductility and low stresses makes the shroud extremely flaw tolerant.

Assuming 360 degree circumferential cracking, and utilizing ASME Code safety factors, crack depths of up to an average of 90% of the shroud thickness can be tolerated while maintaining the structural integrity for normal operation and postulated accident conditions (the worst observed crack indications are an average of about 60% of the shroud thickness). Even with only 10% thickness remaining, the ASME Code safety margins are maintained.

Should significant through-wall cracking occur, it would be detected during normal operation using existing instrumentation and normal plant shutdown could be initiated. Even under very conservative assumptions, safe reactor shutdown is achieved automatically and adequate core cooling is provided, with manual backup available using the existing Emergency Operating Procedures.

Acknowledgment

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1.0 Introduction

Circumferential and axial crack indications have been reported at various locations in the core shroud assembly of a BWR/4 located in the US. The circumferential crack indications located in the inside surface in the heat-affected zone of the top guide support ring horizontal weld (referred to as the H3 weld) are of the most interest because they appear to extend 360 degrees around the circumference of the shroud. GE Services Information Letter 572, Revision 1 has been issued to assist utilities in their individual evaluation of this situation. The USNRC has also issued Information Notice 93-79.

This document provides information for all Boiling Water Reactors (BWRs) to address the safety significance of the concerns related to crack indications of the H3 weld. A generic shroud cracking evaluation procedure, which will describe the process for performing detailed plant-specific evaluations of the shroud, is being prepared under the sponsorship of the BWR Owners' Group.

2.0 Summary and Conclusions

Crack indications near the H3 weld do not represent a threat to the safe operation of a plant:

1. The combination of ductile material and low stresses makes the shroud extremely flaw tolerant. Assuming 360 degree circumferential cracking, and utilizing ASME Code safety factors, crack depths of up to an average of 90% of the shroud thickness can be tolerated while maintaining the structural integrity for normal operation and postulated accident conditions (the worst observed crack indications are an average of about 60% of the shroud thickness). Even with only 10% thickness remaining, the ASME Code safety margins are maintained.

2. The probability of postulated separation of the top guide assembly from the shroud is negligible. A more likely but still improbable scenario would be that the crack grows through the shroud and allows some flow to be bypassed from the core to the downcomer. If it is postulated that the average crack depth is greater than 90% and significant leakage flow occurs, it would be detected during normal operation using available instrumentation. The operator would initiate a normal shutdown.

3. In the unlikely occurrence of a design basis accident or seismic condition with undetected 360 degree circumferential cracking up to an average of greater than 90% of the shroud thickness, but with the top guide assembly still attached to the shroud, safe reactor shutdown is achieved and adequate core cooling is available. In the unlikely scenario that the shroud mechanical integrity is severely distorted such that complete control rod insertion does not occur, current Emergency Operating Procedures adequately direct the operator to use the Standby Liquid Control System to shut down the reactor, and to maintain other aspects of safe shutdown.

3.0 Shroud and Top Guide Functions

The shroud support and shroud make up a stainless steel cylindrical assembly that provides a partition between the core region and the downcomer annulus to separate the upward flow of coolant through the core from the downward recirculation flow. The shroud also provides (in conjunction with other components) a floodable region following a postulated recirculation line break. The shroud is not a primary pressure boundary component.

The top guide consists of a circular grid plate with square openings secured to the bottom of the top guide cylinder. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies.

4.0 Shroud Structural Evaluation

Crack indications have been observed in various locations of the shroud. The circumferential crack indications located in the inside surface of the heat-affected zone for the top guide support ring horizontal weld are of the most potential significance because they appear to extend 360 degrees around the circumference of the shroud. The vertical welds in the top guide support ring are relatively short (on the order of a couple of inches) compared to the length of the horizontal weld and therefore are not of concern.

4.1 Characteristics of the Crack Indications

While the extent of the crack indications that have been reported at the BWR/4 plant is significant, there remains sufficient structural strength in the components to meet their intended function. Testing demonstrates that the shroud and support ring are made of ductile material with high toughness properties even after accounting for any effects due to neutron fluence. The applied loading on the shroud is mainly from the differential pressure during normal operation and the transient differential pressure increase due to design basis accident loading and design basis seismic loads. The applied load during normal, high power (>~ 80%) operation is in the upward direction. The accident and seismic loads are generally small relative to the Code allowable loads and well within the remaining structural integrity of the shroud.

The combination of high ductility and low applied stresses makes the shroud extremely flaw tolerant. In fact, it can be shown that through-wall cracking of over 50% of the shroud circumference can be tolerated while maintaining normal ASME Code allowable design safety factors. Typical allowable flaw sizes range from 75 - 110 inches for each 90 degree sector of the shroud (the length of a 90 degree sector is plant-specific but typically about 150 inches). If 360 degree circumferential cracking is postulated, an allowable flaw size of up to an average of 90% of the thickness can be tolerated with sufficient remaining industry-accepted Code margins. Even if the crack depth is greater than 90% of the shroud thickness (up to the "critical flaw size", with no additional safety factor), the full design basis and seismic loads can be accommodated.

4.2 Potential for Further Structural Degradation

Even if relatively deep cracks occur, it is important to consider the nonuniformity of the crack growth around the circumference. Because of differences in sensitization, fluence, cold work and weld residual stresses around the circumference, uniform crack growth at different crack locations is not expected. This means that any further crack growth will not be uniform and the growth rate will be higher at some locations than others. Even if the growth continues until it is through-wall, this would only occur at selected locations (similar to the leak before break scenario in piping). Under the core internal pressure load, this would lead to a crack opening and leakage from the core. Leakage due to significant cracking will relieve the differential pressure loading and retard the subsequent crack growth rate. While the exact amount of leakage is difficult to predict, the fact remains that if leakage occurs (especially when the remaining ligament is small) it will eventually lead to detection as described in Section 5.0.

In summary, the low stresses and high material ductility make postulation of a 360 degree crack leading to separation of the top guide assembly from the shroud extremely unrealistic.

5.0 Normal Operation

As discussed in the preceding section, the postulated separation of the top guide assembly from the shroud is an extremely low probability event. A more likely but still improbable scenario would be for the crack to create some flow from the core region to the downcomer.

If separation of the top guide and shroud assembly did occur during normal operation, the upward displacement of the top guide and shroud assembly would be less than a few inches, and the core assembly and fuel bundle orientation would be held intact. Moreover, flow through the resulting gap would be detected during normal operation by the reactor operator using available instrumentation for monitoring reactor performance, as described below.

If the crack and leakage occurred on one side of the shroud only, the indications would be asymmetrical which would facilitate detection. The process computer calculations of power/flow operating conditions would not match expected conditions. Additionally, for example, differences will develop in the relationships between recirculation drive flow to core flow and in power level relative to the core flow. If the leakage flow is large enough, those plants with recirculation loop cavitation monitoring instruments will indicate low subcooling of recirculation loop fluid, while all plants should indicate higher than normal recirculation loop temperature(s).

After detecting such an anomaly, a normal shutdown would be initiated until the cause of the anomaly is found and corrected.

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

6.0 Anticipated Operational Events Related to Increased Shroud Head Pressure Loads

The previous sections demonstrate that postulated cracks that grow through the shroud wall or cause complete separation of the top guide from the shroud are improbable, but should either occur it would be detectable during normal operation. Assuming there are no indications of shroud leakage, this section discusses anticipated operational occurrences that could increase shroud loads above those experienced during normal operation: pressure regulator failure - open, recirculation flow control failure - increasing to maximum flow, and inadvertent actuation of the Automatic Depressurization System (ADS).

6.1 Pressure Regulator Failure - Open

This postulated Safety Analysis Report (SAR) event involves a failure in the pressure controls such that the turbine control valves and the turbine bypass valves are opened as far as the maximum combined steam flow limit allows. For units with standard bypass capacity (about 25% of rated steam flow), the worst case involves inadvertently increasing the steam flow to about 130% of rated. This is also true for units with larger bypass capacity if the steam flow limit is set at 130% or less. A depressurization and cooldown occurs which is isolated by Main Steamline Isolation Valve (MSIV) closure. This steam flow increase is small enough that the increased force on the shroud head (about 50% above the normal pressure drop) is within the load capability of the shroud as discussed in Section 4.0.

6.2 Recirculation Flow Control Failure

This postulated event involves a recirculation control failure that causes all 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 about 30 seconds, but it should not significantly exceed the pressure drop expected for normal full power, high core flow operating conditions. Normal operating procedures are considered sufficient to minimize the consequences of this potential transient, and the force on the shroud head is within the shroud capability as discussed in Section 4.0.

6.3 Inadvertent Actuation of ADS

Inadvertent actuation of the 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 about 50% of rated steam flow (plant dependent). 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. This 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 also within the shroud capability as discussed in Section 4.0.

7.0 Design Basis Accidents

Sections 4.0 and 5.0 demonstrate that cracks that might grow through the shroud wall or cause complete separation of the top guide from the shroud are highly improbable, but should either occur it would be detectable during normal operation. Although the combined probability of an accident occurring when a severe (360 degree circumferential crack of uniform depth greater than 90% of the shroud thickness) undetected crack exists is thus very low, such a postulated event is addressed in this section.

The Main Steamline Break Accident imposes the largest potential lifting loads on the shroud head. Liquid breaks (e.g., recirculation line breaks) do not impose large pressure drops on the shroud head, and in fact the shroud pressure drop decreases from its initial value.

7.1 Main Steamline Break

The main steamline break 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. If a sufficient ΔP is developed across the top guide support ring weld (H3) area, and sufficient cracking exists, it is postulated that this added differential pressure might cause separation of the shroud leading to an upward displacement of this structure and the associated top guide. The amount of lifting and the potential effect of these postulated occurrences on emergency operation are described below.

One of the key considerations of this postulated accident case is the ability 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 the potential to 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. The initial shroud head pressure drop loading is a result of the depressurization of the steam dome region which reduces system pressure overall, but which increases differential pressure across the shroud in the short term. This pressure loading increase is short-lived (less than two seconds) and decreases to below normal steady state loads. Even if the remaining shroud ligament is enough so that significant cracking is undetected, but the ligament is less than an average of 10% (see Section 4.2), the structural integrity of the shroud will remain intact for this postulated limiting event plus seismic loads. If it is even further postulated that the initial load pulse causes the shroud to separate, the last part of the pressure loading could cause the top guide assembly to lift. The flow path created by any separation reduces the upward lifting forces. For this postulated scenario the top guide assembly would remain engaged with the fuel channels.

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 accident 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. Even if the first loading pulse causes the upper shroud assembly to break free, control rod motion will be started before the upper shroud assembly and top guide lift significantly. It is likely that the top guide will remain engaged with the tops of the fuel bundles. Any control rods that are partially inserted as part of normal operation are already in position to initiate shutdown. Insertion of fully withdrawn control rods to 5% of full stroke will occur by 0.9 second, early enough for the control rods to be moving up between the bundles before any significant lifting of the top guide could take place. The remainder of the insertion [illegible] because the fuel will remain properly oriented. Reactor shutdown would thus be complete with all drives inserted.

In the very unlikely case that scram may not be complete, the Standby Liquid Control System is available to provide shutdown capability, as discussed in Section 8.0.

Movement of the upper shroud assembly (in the very unlikely case that it occurs) could affect the core spray system if it impacts the core spray line connections. If this were to occur, core spray flow sufficient to provide long term cooling would still be expected. Any one Emergency Core Cooling System (ECCS) pump is sufficient to provide adequate makeup and maintain reactor water level.

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 recirculation line break 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. Therefore, separation and disengagement of the fuel from the top guide is even more unlikely. Nevertheless, if it is further postulated that the top guide assembly is lifted and then is repositioned on the fuel assemblies, there is a potential to mechanically damage some of the fuel cladding leading to some fission product release within the core. However, assuming closure of the MSIVs within the time permitted by Technical Specifications (typically three to five seconds), this scenario results in MSIV closure before a potential release outside containment from such an improbable scenario could occur. The radiological consequences of this very conservative main steamline break scenario are thus still bounded by the plant SAR results.

7.2 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. With the shroud integrity maintained, a floodable core region is also preserved. Therefore, the recirculation line break analysis results are unchanged.

8.0 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 and potential disengagement of the top guide from the fuel channels, which is further postulated to prevent a full scram. This event (a large steamline break with failure to completely insert the control rods) is beyond the design basis of the plant. Nonetheless, it is adequately addressed by EOPs.

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 pool temperature increases to the allowable value (typically 110 °F). The postulated event would clearly lead to SLCS injection within a very few minutes, resulting in safe shutdown. 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.

Water level would be controlled after the postulated event because the break is high in the vessel and a large complement of water injection systems would be available. Separation of the shroud above the top of the fuel channels would not prevent maintaining the core in a flooded condition.

Even if the core spray delivery system were damaged by the shroud or top guide displacement, some core spray flow would be expected. Any one ECCS pump would be sufficient to provide adequate makeup. For some plants, SLCS injection occurs through the High Pressure Core Spray system. For these plants, boron injection would still occur through the spray flow even if the system flow path was changed by the shroud or top guide displacement.