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Synergistic Failure of BWR Internals

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Boiling Water Reactor (BWR) core shrouds and other reactor internals important to safety are experiencing intergranular stress corrosion cracking (IGSCC). The United States Nuclear Regulatory Commission has followed the problem, and as part of its investigations, contracted with the Idaho National Engineering and Environmental Laboratory to conduct a risk assessment. The overall project objective is to assess the potential consequences and risks associated with the failure of IGSCC-susceptible BWR vessel internals, with specific consideration given to potential cascading and common mode effects. An initial phase has been completed in which background material was gathered and evaluated, and potential accident sequences were identified. A second phase is underway to perform a simplified, quantitative probabilistic risk assessment on a representative high-power BWR/4. Results of the initial study conducted on the jet pumps show that any cascading failures would not result in a significant increase in the core damage frequency. The methodology is currently being extended to other major reactor internals components.

INTRODUCTION

General Design Criteria 2 and 4 require that commercial nuclear reactor structures, systems, and components important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, and the effects of postulated accidents, including loss-of-coolant accidents (LOCAs). Boiling water reactor (BWR) internals components were originally believed to have been designed to accommodate these requirements. However, intergranular stress corrosion cracking (IGSCC) degradation has been observed in both core shrouds as well as a number of other BWR reactor internals components, many of which are important to plant safety.

Although IGSCC of reactor internals had been recognized for over 20 years, this phenomenon received increased attention, beginning when crack indications were reported at core shroud welds located in the beltline region of an overseas BWR in 1990. The core shroud is a stainless steel cylinder that is located inside the reactor vessel. It serves to both provide lateral support to the reactor core and to direct the flow of water inside the reactor vessel, and is generally regarded as a component whose integrity is critical to maintaining core safety. Later, a visual inspection of a U.S. BWR core shroud revealed crack indications at several weld regions. Subsequently, General Electric and the NRC[1,2,3] issued correspondence regarding core shroud cracking.

In addition to the BWR core shroud degradation, other BWR reactor internals components, including shroud support access hole cover welds, jet pump hold-down beams, core spray systems, and top guides

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have also been experiencing IGSCC degradation over the years.[4] These instances have for the most part been sporadic, were not believed to be of major safety importance, and were addressed by General Electric through notices such as Safety Information Letters (SILs) and by the NRC through Information Notices and a Bulletin that have been issued from time-to-time since about 1980.[5,6,7,8,9,10,11] However, the instances of core shroud cracking served to escalate attention as to the seriousness of the IGSCC problem in BWR reactor internals.

The NRC Office of Nuclear Reactor Regulation has followed the problem and issued a Bulletin, Information Notices, and Safety Evaluation Reports (SERs), as well as a Generic Letter.[2] The primary emphasis has been placed on core shroud degradation, but common mode or cascading failure of other components could also have safety significance. Consequently, this NRC-sponsored program described in this paper has been initiated to conduct a risk assessment investigating the concern of cascading failures of BWR vessel internals.

The objective of the study is to assess the potential consequences associated with the failure of IGSCC-susceptible BWR reactor internals components, both singly and in combination with the failures of others. Specific consideration is given to potential cascading and common mode effects on system performance stemming from cracking of core shrouds and other BWR reactor internals components when subjected to design-basis and beyond-design-basis accident loading conditions such as seismic events.

The focus is on mechanical design, failure locations, consequences, potential accident scenarios, and characterization of risk associated with IGSCC degradation of BWR vessel internals. The scope is limited to the basic risk evaluation, including the following:

- The only degradation mechanism considered in this study is IGSCC, including contributing SCC mechanisms such as irradiation-assisted SCC (IASCC). It is recognized that other degradation mechanisms such as fatigue can act synergistically with IGSCC in that a crack which is initiated by IGSCC can propagate to failure from fatigue.

- The NRC is investigating the causes and contributing aspects of the IGSCC problem in separate programs. Industry groups are investigating inspection, mitigation, repair, or replacement, as well as the causes and contributing aspects of IGSCC.

- There are five currently operating types of BWRs in the U.S., designated BWR/2 through BWR/6. Since there is only a single BWR/1 in operation, Big Rock Point, and it is expected to be decommissioned in the near future, the scope of work in this study does not include the unique BWR/1.

BACKGROUND

The three basic elements that must *all* be present for IGSCC to occur are:

1. a susceptible material
2. a chemically aggressive environment
3. a high tensile stress

Under normal circumstances, the stress must be above the yield stress, which can occur at locations such as residual stresses around welds. However, if certain other factors are present, the conditions for the three basic elements listed above (such as the need for the tensile stress to be above the yield stress), may be somewhat altered.

INITIAL PROGRAM PHASE

The first phase of the study involved acquiring and evaluating relevant background information on IGSCC of BWR reactor internals, to qualitatively assess potential accident scenarios, and to identify scenarios for detailed analyses, that is, those expected to have large effects on Core Damage Frequency (CDF). This phase has been completed.

Accident Sequences

Differences in reactor internals designs and accident mitigation systems for the various BWR types were categorized, the degradation of BWR internals to date was catalogued, and management practices to deal with aging were reviewed. From this background study, various types of *systems* failure modes that could result from simultaneous common mode failures of various combinations of reactor internals were catalogued. This included the consideration of functional losses or significant degradations of certain inside-reactor vessel systems. A similar assessment for various types of *mechanistic* failure modes (i.e., the potential results of physical impacts/interactions, common mode and cascading failures, etc., between various reactor internals components due to failures and/or degradations of the components that are subject to IGSCC degradation), was made. This included the various ways that these components might fail and how those types of failures might affect other components inside the reactor vessel. Approximately 250 different and unique scenarios were identified.

The safety significance was also evaluated. This is generally component specific; however, one common safety significance is a loose part which can result from IGSCC. There are three basic safety consequences:

1. a loose part can inhibit control rod motion
2. a loose part can block or partially block a coolant flow channel
3. a large loose part can impact adjacent components and impair their function

There are other safety consequences not associated with loose parts which could be caused by damage to any of several reactor internals components, such as:

1. increased coolant leakage between plenums
2. damage to emergency coolant or shutdown systems
3. damage to control rods or prevent their motion
4. cause of a reactor coolant system leak

In order to reduce the number of scenarios to be considered, a screening logic based on the safety consequences identified above was prepared. Five criteria were developed that are believed to cover the most important issues necessary to adequately address public safety with regards to reactor vessel internals failures. The screening logic was applied to all 250 initially identified scenarios. Of these, 148 remained after the screening, which reduced the work scope somewhat, but still left a large number of sequences to evaluate. A quantitative risk assessment was subsequently conducted on these remaining 148 sequences, as described in the following section.

Preliminary Qualitative Risk Assessment

A qualitative ranking (based on potential contributions to CDF) was made of potential accident scenarios which can be exacerbated by IGSCC degradation of reactor internals. Various possibilities of single, common mode, and cascading failure sequences were postulated for the high, medium, or low rankings. Although the rankings were qualitative, a NUREG-1150 PRA was used to assist in providing for an estimate for the rankings.

A preliminary risk assessment including a list of potential safety concerns (i.e., possible accident scenarios), deterministically developed, was made. Specific areas were described for each potential accident sequence where additional analyses are needed to provide a more definitive understanding of accident scenarios that involve either simultaneous (i.e., common mode) or cascading (i.e., sequentially caused by other failures). The scope included both deterministic failure considerations and qualitative risk assessments.

Most (about 100) of the scenarios were ranked high. Although there appears to be a large number of scenarios, there is a great deal of redundancy in that the high-ranking scenarios fall into variations of two basic categories:

1. loss of the reactor protection system (RPS)
2. loss of coolant to the core

The high-ranking categories were broken into subcategories to further differentiate between the various causes of loss of the RPS and the various ways that coolant to the core could be lost. The following seven subcategories were chosen:

1. loss of scram capability
2. standby liquid control (SLC) system nonfunctional
3. both RPS and SLC nonfunctional
4. medium LOCA with loss of SLC
5. both high-pressure coolant injection (HPCI) and low-pressure coolant injection LPCI ineffective [no redundant emergency core cooling system (ECCS)]
6. core reflood to two-thirds level cannot be maintained (treated as loss of ECCS)
7. high-pressure coolant system (HPCS) and SLC (through sparger) eliminated (several BWR/5 and BWR/6 plants)

Each of the 100 high-ranked scenarios can be placed into one or more of these subcategories.

CONTINUED PROGRAM PHASE

The initial phase of the project, which was primarily to scope the overall effort, has been completed. The second phase is now underway, in which quantitative calculations will be performed. However, there are many difficulties in carrying out this program, such as:

(a) the large number of components and failure sequences
(b) the different types of BWRs
(c) the difficulty in estimating crack sizes and growth rates
(d) there are a large number of disciplines involved
(e) there are limited "good" PRA and thermal-hydraulic models available

It was decided to narrow the research scope to provide a simplified, cost-effective approach. The following simplifications were proposed as an initial approach:

(a) select a single plant for study
(b) select a single component and probable failure locations for initial calculations
(c) perform minimal calculations and research
(d) develop a methodology to introduce IGSCC-induced failures into an existing PRA which can then be applied to the failure of any BWR vessel internals component
(e) convert an existing TRAC-B model to a representative plant to determine flow characteristics

(f) estimate the failure probabilities and insert events associated with the failure of the selected component into an existing PRA, considering a single failure at the most likely locations, common mode failures, and cascading failure sequences. If successful, then apply to other components
(g) use expert panels to critique methods, offer suggestions for approach, and help in estimating probabilities and uncertainties

Plant Selection

A number of criteria were used to select a plant for study, including:
(a) is there an existing PRA model for the plant (internal and external events)
(b) is there an existing TRAC-B model for the plant (or one for a similar plant)
(c) is the plant typical
(d) older plants were preferred

The Peach Bottom BWR/4 was chosen for study. As a high-power BWR/4, it is the most representative type of BWR. There is a fairly good (but not ideal) PRA, and there exists a BWR/4 TRAC-B thermal-hydraulics model which can be modified to represent the selected plant. Figure 1 shows the general arrangement of a typical BWR/3-BWR/4 plant.

Initial Component Selection

A number of criteria were used to select a specific reactor internals component for study, including:
(a) degradation to date
(b) cascading possibilities
(c) safety significance
(d) typicality

The jet pump was the reactor internals component selected for study, as there has been recently discovered cracking in jet pump riser inlet welds[11] and jet pump failure could lead to a variety of cascading failure sequences. IGSCC failures have also initiated at the jet pump holddown beams, the first instance occurring in a BWR/3 in 1980. Cracking also was found in the beams of two BWR/6 plants. Subsequently, the beams have been redesigned. Figure 2 shows the general arrangement of a jet pump.

PRA Modification

Three sets of PRA models are available for the Peach Bottom nuclear power plant: the NUREG-1150 models, the Individual Plant Examination (IPE) and Individual Plant Examination for External Events (IPEEE) models, and the Accident Sequence Precursor (ASP) model. The NUREG-1150 PRA models for internal and external events at full power were generated in the late 1980s as part of an NRC-sponsored program to consistently analyze five different nuclear power plants. Generally, the NUREG-1150 studies included limited plant-specific data collection, and the external events analyses were performed with a streamlined and somewhat simplified methodology. In contrast, the IPE (for full-power internal events and internal flooding) was performed in the early 1990s and included more recent plant-specific data and design information. However, the IPEEE, performed in the mid 1990s, utilized screening and seismic margins approaches that did not result in models capable of predicting CDFs from external events. Finally, the ASP model for Peach Bottom is a simplified model compared with the NUREG-1150 and IPE models. The ASP model covers only full-power internal events and is considered to be too simplified to be of much use in this study.

Unfortunately, none of the PRA studies are ideal choices to support the evaluation of IGSCC of BWR internals. The NUREG-1150 studies address the plant design as it existed in the late 1980s and contain very little plant-specific data, and the external events analyses suffer from various deficiencies. In contrast, the IPE/IPEEE studies reflect the plant design as it existed in the early 1990s, but again contain limited plant-specific data. Also, the IPEEE studies did not use methodologies that result in CDF predictions. In general, the ASP model is too simplified for the purposes of this study and does not include external events. Finally, all three types of studies did not address the CDF from low power and shutdown operations.

From past studies of IGSCC of BWR internals, it was expected that initiating events such as recirculation line breaks (RLBs) and seismic events will be important. Therefore, it was important that the Peach Bottom PRA chosen for use in this project be accurate with respect to RLBs and seismic events. Because the IPEEE did not use a methodology that could predict a seismic CDF, the IPE/IPEEE set of models is not appropriate for this project. Therefore, the NUREG-1150 set of PRA models were used. However, in order to update these models, several changes were made.

The existing PRA for the selected plant was modified for the IGSCC-induced failures. The following are examples of the modifications that were made:

1. introducing information from the latest Individual Plant Evaluation (IPE)
2. modifying seismic hazard curves to reflect more up-to-date seismic hazard information
3. including internal and external event PRA branches that were not used in previous PRA studies because of low probabilities
4. separating the main steam line, main feedwater line, and RLBs into three individual events, each with its own probability of occurrence
5. adding an event tree for IGSCC-induced initiating events

These modifications are applicable to all of the potential IGSCC-induced failures of reactor internals. However, the model was first run only with the probabilities of IGSCC-induced failures of jet pump components included.

RESULTS OF JET PUMP STUDY

Calculations that have been performed considered sequences initiated by jet-pump failures. To help establish which sequences would have negligible effects on core-damage frequency (CDF), and to identify the sequences that were more probable in increasing the CDF, parameter studies were conducted with the PRA model to determine which sequences could be screened as negligible contributors to CDF, and with structural analysis to determine probabilities of cascading failures.

The generic BWR CDF for internal events and internal flooding, internal fires, seismic events, and low-power and shutdown operation was estimated to be on the order of 5×10^{-5} events/rx/yr. Based on the assumption that a 10% increase in CDF is significant to risk (based on an interpretation of the acceptance guidelines in draft DG-1061),[12] and that there were approximately 50-100 scenarios to evaluate, a screening level for each sequence of 1×10^{-7} events/rx/yr was chosen for the preliminary calculations. Sequences involving failure of the reactor protection system and emergency core cooling system were evaluated.

The study included an assessment of IGSCC damage to date that has been detected in the various jet pumps components (Table 1) and the potential targets from a large jet pump loose part that might result in cascading failure sequences (Table 2).

Table 1. IGSCC damage in jet pumps

Jet pump component	IGSCC	Failure	Mitigation	Notes
Inlet riser	Y	N		Detected visually
Riser brace	Possibly	N		Primarily fatigue cracking
Holddown beam	Y	Y	Redesign	Material & size changes
Transition piece	N	N		
Nozzle	N	N		
Inlet mixer, throat	N	N		
Diffuser, adapter	N	N		

Table 2. Potential targets from failed jet pump

Component	IGSCC Damage	Damage Potential from Jet Pump	Safety Concern	Comment on Safety Concern
Baffle plate	N	N	_	
Baffle plate cover	Y	Y	Lower reflood level	Only for recirculation line break
Tie rod	N	Y	RPS, LPCS/HPCS diversion	Could fail core shroud
Core shroud	Y	N	_	Only if tie rods fail
Shroud head bolts	Y	Y	None	Only if upward migration possible; most bolts would have to fail
Core spray line	Y	Y	LPCS failure	Only if upward migration possible
Adjacent jet pump	Y	Y	Lower reflood level	Only for recirculation line break

Structural calculations were performed to assess damage to adjacent components that could result from jet pump failures. Thermal-hydraulic studies were conducted by Scientech to calculate the flow rate through the jet pumps, and the flow velocities in the annular region surrounding the jet pumps. IGSCC history, energy required for failed jet pump parts to migrate to nearby components, and energy required to damage adjacent components were considered.

The results showed that there was insufficient energy for loose jet pump parts to migrate upward to damage components such as the core spray system. Loose jet pump parts could contact adjacent jet pumps, the reactor vessel wall, the baffle plate and covers, the core shroud, and core shroud tie rods. Damage to these components would not be expected except for the following:

1. If the adjacent jet pump, baffle plate cover, and core shroud were already very severely damaged by IGSCC, these components could fail, possibly resulting in further cascading failures
2. The plant chosen for evaluation did not contain core shroud tie rods, but calculations showed that impacts from failed jet pumps should not fail properly installed tie rods

Preliminary results show that the cascading failure of an adjacent jet pump or baffle plate covers would not result in a significant increase in CDF. If a core shroud were very severely damaged, it is expected that inservice inspection would have detected the degradation, and repair methods such as tie rods would have been installed. If a tie rod were impacted by loose jet pump part, the rod would not fail if properly installed.

Loose parts from a failed jet pump might migrate through the lower core plenum, and back up into the core, which could block control rod insertion or block coolant from sufficiently reaching a fuel channel. Although probability estimate of loose parts is very difficult to quantify, the studies show that there is a very low probability of this sequence affecting the CDF.

APPLICATION TO OTHER COMPONENTS

Once the methodology was developed, it was applied to the remaining components. The 148 sequences identified in the initial phase of the project included any plausible damage from any of the reactor internals for all BWR types, and without regard to safety analysis or PRA results. In order to reduce the number of sequences from 148 to a more manageable number, the following results and assumptions were used to reduce the list to about 36:

1. A number of the sequences did not affect BWR/4s. This reduced the list by 9.
2. A number could immediately be screened from the plant-specific PRA. This reduced the list by 18.
3. A number of the sequences involved loose parts from various postulated failed reactor internals breaking up, migrating to the core, and entering the small holes in the CRD housings to inhibit control rod insertion or block fuel channels. Studies showed that while probabilities for these sequences were very difficult to quantify, the overall assessment was that damage to a sufficient number of control rod drive mechanisms or fuel elements was below the screening value. This reduced the list by about 45.
4. Safety studies have shown that for core damage to occur, approximately 1/3 of the control rods would have to fail to insert on a random basis, or 5 to 10 control rods grouped together.[13,14] Since a number of the sequences involved only a single rod failure, this reduced the list by about 20.
5. For some events that involved a SBLOCA, the plant makeup system could easily supply makeup during the subsequent plant shutdown, and the event could be screened. This reduced by list by 6.
6. A number of the remaining sequences had been evaluated during the jet pump study. This reduced the list by an additional 17.

Of the remaining sequences, only a few are considered major, and involve either a reactivity accident caused by failure of rods to insert because of misalignment of core internals, or diversion or loss of coolant to the core. These involve the most important internals, such as the core shroud, core plate, top guide, and core spray system. The accident initiators are line breaks and seismic events.

The analyses are dependent on the frequency and magnitude of seismic events and the frequency of line breaks. The mean frequencies for various ground level accelerations were taken from the Peach Bottom seismic hazard curve. The accelerations at the levels of the reactor internals supports were calculated by multiplying the ground level accelerations factors which account for the amplification between ground level and the elevation at the supports, based on a simplified approach recommended by Dr. R.P. Kennedy, a consultant to the project. Various estimates of pipe break probabilities have been made over the years. The values in NUREG-1150[15,16] have received widespread review and have been used in many studies. A recent study by the INEEL[17] recommended lower probabilities (events/rx-yr) for line breaks in U.S. BWR plants. BWRVIP studies[18] recommend an even lower probability for a BWR recirculation system LBLOCA. Sensitivity studies are being included to assess the results from using the various estimates.

Table 3. Estimates of BWR recirculation line break frequency (events/yr)

Source	Mean frequency
NUREG-1150	1E-4
Poloski et al.	2E-5
BWRVIP	7.5E-6

The probability of component failure can be simply stated by the following:

Probability of Failure = Pci x Pmbi x Pgtf

Where Pci = probability of crack initiation
Pmbi = probability crack missed by inspection
Pgtf = probability of crack growth through wall until fragility depth is reached for loading condition under study

A conditional failure probability of 1 could be used, but this would result in a significant increase in CDF for the failure of several of the major components and is too conservative. However, this serves to emphasize that if the degradation if left unmanaged, there could be a significant increase in the probability of CDF.

Inspection crack detection probabilities, crack growth rates, and structural calculations are currently being used to provide estimates of these probabilities as input to the PRA. Figure 3 shows the structural finite element model being used in the calculations. The cutaway view shows the reactor vessel and support, the core shroud and head, the top guide, and the core plate. For clarity, the detailed finite element mesh is not shown.

SUMMARY

A program is underway to perform an independent risk assessment of accident sequences initiated by IGSCC-induced failures of BWR reactor internals components. An initial phase has been completed in which background material was gathered and evaluated, potential accident sequences were identified, and a qualitative PRA was performed to rank the sequences as having a high, medium, or low potential to significantly change the core damage frequency. A second phase is underway to perform a simplified, quantitative PRA on a representative high-power BWR/4. The existing PRA for the plant has been upgraded and modified for the project, including introducing an event tree associated with reactor internals failures. Failures associated with jet pumps were first to establish an analysis methodology. Preliminary results show low probabilities for sequences initiating from jet pump failures significantly affecting the CDF. The methodology is being extended to other major reactor internals components.

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Figure 1. BWR/3-BWR/4 general arrangement

Figure 2. Typical jet pump arrangement

Figure 3. General cutaway view of finite element model, showing reactor vessel and support, core shroud and head, top guide, and core plate