



Preliminary Risk Assessment Associated with IGSCC of BWR Vessel Internals

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ABSTRACT

BWR core shrouds and other reactor internals important to safety are experiencing intergranular stress corrosion cracking (IGSCC). The United States Nuclear Regulatory Commission (NRC) has followed the problem, and as part of its investigations, contracted with the Idaho National Engineering and Environmental Laboratory (INEEL) 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. The paper presents an overview of the program, discusses the results of a preliminary qualitative assessment, and summarizes a simplified risk assessment that was conducted on sequences resulting from failures of jet pump components of a BWR/4 plant.

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

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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 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 aging management practices 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

A 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.

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 welds¹¹ and jet pump failure could lead to a variety of cascading failure sequences. IGSCC failures have also initiated at the jet pump hold-down 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.

PRA Modification

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 component seismic fragilities to reflect more recent seismic hazard fragility 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 recirculation line breaks 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.

PRELIMINARY RESULTS

The initial 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, 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.

In parallel, structural calculations were performed to assess damage to adjacent components that could result from jet pump failures. Thermal-hydraulic studies were conducted 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
2. The plant chosen for evaluation did not contain core shroud tie rods, and therefore the strength of the tie rod end connections was not evaluated

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 itself would not fail, but since end connections may vary in design, these locations require further evaluation.

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. However, the preliminary studies show that there is a very low probability of this sequence affecting CDF.

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 have been addressed first. 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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