December 8, 2000

MEMORANDUM TO: Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

FROM: Ashok C. Thadani, Director Original signed by M.V. Federline for
Office of Nuclear Regulatory Research

SUBJECT: TRANSMITTAL OF RESEARCH INFORMATION LETTER RIL-0003 AND NUREG/CR-6677 "EVALUATION OF RISKS ASSOCIATED WITH IGSCC INDUCED FAILURE OF BWR REACTOR INTERNAL COMPONENTS"

Attached is a copy of the subject Research Information Letter and the associated NUREG/CR report. The two documents present the results of recent research in the evaluation of risks associated with intergranular stress corrosion cracking (IGSCC) induced failure of BWR reactor internal components. The draft version of NUREG/CR-6677 was reviewed by R. Hermann and G. Carpenter of your staff in DE/EMCB, and their comments were resolved and incorporated in the final report.

This research was started in April 1996 to address one part of a user need request from NRR ("NRR User Need Request for Support of Resolving Problem of Stress Corrosion Cracking of Reactor Vessel Internal Components," Memorandum from W. Russell to E. Beckjord, December 2, 1994). This user need request consists of two parts, the first is concerned with issues related to the materials aspect, and the second is concerned with the assessment of potential consequences and risks of the failure of BWR internal components caused by IGSCC, including the synergistic effects of multiple cracks in one or more internal components. The work reported in NUREG/CR-6677 closed out the part of the user need request related to the consequences and risks of the failures of BWR internal components caused by IGSCC. The status of the first part of the user need request is discussed at the end of this memorandum.

This program was conducted in two phases. Using a qualitative approach, Phase I identified about 150 potentially important accident scenarios associated with IGSCC-induced failures (single, common mode, and cascading failure modes) that have the potential to increase plant core damage frequency (CDF) or offsite risk to the public for all five types of BWR designs (BWR/2 to BWR/6). The initiating events could either be failures of IGSCC-degraded reactor internal components, internal events [such as large LOCA (recirculation line break, main steam line break, or feedwater line break)], or external events (e.g., earthquakes).

At the beginning of Phase II in 1997, a multi-discipline NRC Review Panel with members from both RES and NRR [with expertise in reactor systems, Probabilistic Risk Assessment (PRA), materials, mechanical, and structural engineering] was formed to provide guidance and review of the program on a regular basis until the completion of the program in May 2000. Notably, G. Carpenter, R. Hermann, R. Caruso, K. Kavanagh, A. Cubbage and U. Shoop of your staff made significant contributions to the program as members of the NRC Review Panel. In addition, two outside expert panels, one on Materials and one on Structural/PRA were also formed in 1997 to provide estimates on crack growth rates and to provide advice on structural/PRA methodologies and assumptions.

For Phase II of the program, a representative BWR/4 plant was chosen for further quantitative evaluation. A PRA screening criterion was used to reduce the number of scenarios to a more manageable number and to narrow the focus of attention to the potentially most important accident scenarios. The existing plant PRA model was modified to add branches to account for IGSCC-induced failures of BWR reactor internal components, and limited thermal-hydraulic analyses were performed for the large LOCA events. The effectiveness of crack detection in inspection, operating experience data on crack growth rates, as well as structural calculations were used to estimate the various probabilities associated with IGSCC-induced failures in this PRA study. In order to truly assess the consequences and risks associated with IGSCC-induced failures of BWR reactor internal components, the initiating events in Phase II considered not only IGSCC-induced failure of reactor internal components but also internal events (large LOCAs) and external events including design-basis and beyond-design-basis earthquakes.

The results of this program indicated that a number of BWR reactor internal components, such as the top guide, core plate, core spray system, jet pump, core shroud, and core shroud support, could potentially fail either in a common mode or cascading manner under an internal or external event (large LOCAs or earthquakes) if it were allowed to be severely degraded by IGSCC. These failures could lead to an increase in plant CDF on the magnitude of 5E-6/RY or higher.

Nonetheless, the increase in risk can be kept to an acceptable level through a proper aging management program [such as the BWR Vessel and Internals Project (BWRVIP) for inspecting, monitoring, evaluating, and repairing IGSCC-related degradation] and the associated regulatory oversight of this program. With the continued implementation of a robust aging management program (such as the BWRVIP) and the associated regulatory oversight, IGSCC in BWR internals can be identified, evaluated, and corrected in time to preclude undesirable increase in CDF (i.e., the increase in CDF will be less than 5E-6/RY).

Work is continuing on the part related to the materials aspect of the user need request, which includes the following issues to be addressed for both BWR and PWR internal structural components: (1) causes and mechanisms of intergranular cracking; (2) potential mitigation measures to limit the extent of cracking; (3) crack initiation and growth analysis including the potential development of through-wall cracks; (4) assessment of industry crack growth models; (5) flaw acceptance criteria; (6) post-weld heat treatment effects on crack initiation and growth; and (7) effectiveness of nondestructive examination methods in detecting and sizing cracks. Principally, the Environmentally Assisted Cracking (EAC) research program at the Argonne National Laboratory addresses the above issues.

Based on insights gained in the NUREG/CR-6677 risk study and emerging issues, our current and planned activities in the EAC program are as follows: Because of risk insights, lesser emphasis is placed on potential mitigation measures and flaw acceptance criteria in the BWR context. Mechanistic research on the effects of impurities and grain boundary structures, and data collection (from Japanese and EPRI programs) on weldments, will continue to address BWR problems with a focus to use what is known or can easily be learned in the BWR environment to help address the much more difficult problems in PWRs. Research on BWR cracking continues in a highly leveraged Cooperative IASCC Research Program with much of the financial support from the reactor operators. The research on crack growth models and on in-reactor crack detection and measurement continues. The interim results as well as results of completed tasks have been provided to EMEB/NRR as available.

Please contact Sher Bahadur (415-6010) for any questions on the enclosed RIL and NUREG/CR-6677. For the materials aspect of the user need request, please call Nilesh Chokshi (415-6013).

Attachments: As stated

cc: W. Travers
C. Paperiello
F. Miraglia

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RESEARCH INFORMATION LETTER RIL-0003

RISK ASSOCIATED WITH IGSCC-INDUCED FAILURE OF BWR REACTOR INTERNAL COMPONENTS

Background

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, such as loss-of-coolant accidents (LOCAs). Boiling water reactor (BWR) internal components were originally believed to have been designed to accommodate these requirements. However, intergranular stress corrosion cracking (IGSCC) degradation has been observed in a number of BWR reactor internal components, many of which are important to plant safety.

Although IGSCC of reactor internal components had been recognized for over 20 years, this phenomenon received increased attention, beginning when crack indications were reported at core shroud horizontal welds located in the beltline region of an overseas BWR in 1990. In response, General Electric (GE) issued RICSIL 054 (Reference 1) to all owners of GE BWRs. A visual inspection of a domestic BWR core shroud revealed crack indications at several horizontal welds in 1993, and the NRC promptly issued Information Notices (INs) 93-79, 94-42 (References 2, 3), and Generic Letter (GL) 94-03 (Reference 4). In GL 94-03, the NRC requested the group of licensees with BWRs susceptible to IGSCC degradation to inspect the core shrouds during their next outages, and to perform safety assessments to justify their interim operation. Subsequently in 1997, cracks at core shroud vertical welds were also detected in some of the domestic plants, and the NRC issued IN 97-17 (Reference 5).

In addition to the BWR core shroud degradation, other BWR reactor internal 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. Some of the more recent NRC INs addressing components other than core shrouds are IN 92-57 (Reference 6), IN 93-101 (Reference 7), and IN 95-17 (Reference 8). In response to NRC GL 94-03 and the increasing IGSCC cracking of BWR internal components discovered in the domestic plants, the BWR Vessel and Internals Project (BWRVIP) was formed by an association of utilities in June 1994 to address the BWR vessel and internals issues.

The number of different types of components that have experienced cracking has increased over the years. At the end of 1999, IGSCC degradation has been detected in at least 15 different types of BWR reactor internal components. To date, most of the safety concerns have been focused on core shroud degradation. However, it is recognized by the NRC that there is a need to assess the potential consequences and risks of the failure of BWR internal components caused by IGSCC, including the synergistic effects of multiple cracks in one or more internal components. This RES program was therefore established in response to a user need request from the NRR (Reference 9) to evaluate the potential safety consequences and risks associated with IGSCC-induced failures of BWR reactor internal components, both singly and in combination with the failure of other reactor internal components. Idaho National Engineering and Environmental Laboratory was selected as the contractor.

Research Summary

The focus of this program is on the mechanical design of reactor internal components, potential failure locations, failure consequences (e.g., cascading effects), potential accident scenarios and the characterization of risk. The degradation mechanism considered is IGSCC.

This program was conducted in two phases. Phase I catalogued differences in reactor design and accident mitigation systems, reviewed aging management practices for BWR reactor internal components, and documented available information on degradation of BWR reactor internal components. Using a qualitative approach, Phase I also identified 148 potential important accident scenarios associated with IGSCC-induced failures (single, common mode, and cascading failure modes) that have the potential to increase plant core damage frequency (CDF) or offsite dose risk to the public for all five types of BWR designs (BWR/2 to BWR/6). The initiating events could either be failures of IGSCC-degraded internal components, or events such as large LOCAs (recirculation line break, main steam line break, or feedwater line break) or earthquakes.

Due to the large number of potential failures of reactor internal components and accident sequences, design diversity among the BWR reactor types, difficulty in estimating crack growth rates, and lack of failure data to assist in estimating probabilities, a multi-discipline NRC Review Panel with members from both RES and NRR [with expertise in reactor systems, Probabilistic Risk Assessment (PRA), materials, mechanical, and structural engineering] was formed to provide guidance and review for Phase II of the program on a regular basis. The scope of the Phase II program was first narrowed to investigate a representative single plant (a BWR/4 plant) with potential failure modes and risk consequences of a single component type (jet pumps). Based on Regulatory Guide 1.174 (Reference 10), a quantitative PRA screening criterion was developed and used to reduce the 148 potentially important accident scenarios to a more manageable number, and to narrow the focus to the most potentially important accident scenarios for further, more detailed, quantitative analysis. Two outside expert panels, one on Materials (Professors Gary S. Was and Roger W. Staehle) and one on Structural/PRA (Drs. Robert P. Kennedy and Robert J. Budnitz) were also formed to provide estimates on crack growth rates and to advise on structural/PRA methodology and assumptions. After completing the initial study for failure of jet pumps, the methodology was applied subsequently to include failure of other reactor internal components and the cascading effects. Phase II of the program was completed at the end of 1999.

A BWR/4 plant was chosen for the Phase II study because it represents the largest number (19) and the most typical of the total 36 BWR plants in service in the U.S. today. The existing plant PRA model was modified to add branches to account for IGSCC-induced failures of BWR reactor internal components, and limited thermal-hydraulic analyses were performed for the large LOCA events to determine the loads on the components. Inspection crack detection probabilities, crack growth rates, and structural calculations were used to provide estimates of probabilities as input to the PRA. In addition to single, common mode and cascading failure modes, uncertainty was also considered.

The last NRC Review Panel meeting of this RES program took place on February 8, 2000. After reviewing the findings of this program, the Panel agreed that this program has been successfully completed and the original objective of the program has been met. The findings of this RES program was presented to the annual NRC/BWROG/BWRVIP management meeting on March 16, 2000. Subsequently, after the process of review, comments and the resolution of comments by the staff of RES and NRR, the final NUREG/CR-6677 report of this RES program (Reference 11) was published and issued in July 2000.

<u>Research Results</u>

As mentioned above, this program was conducted in two phases. In Phase I, the qualitative evaluation of 148 potential important accident scenarios provided some insights related to the design of different BWR types that will have bearings on the consequences of the IGSCC-induced failures of BWR internal components. One insight was the locations of emergency coolant water injection as BWR designs changed over the years during which the BWR/2 through BWR/6 lines were developed. Whereas each design has adequate provision for emergency core cooling, should a beyond-design-basis accident occur that includes an IGSCC-induced failure of a reactor internal component, some designs would provide greater safety margins (lower probability of core damage) than others. These can be summarized as follows:

1) Of the internal and external initiating events considered (LOCAs and earthquakes), the recirculation line break potentially creates more safety concerns than any other event, due to the necessity of maintaining the two-thirds core reflood capability and the loss of low pressure coolant injection (LPCI) in BWR/3s and 4s (since LPCI injects water from the suppression pool into the reactor core through the recirculation line, and one of the LPCI trains would not function properly under the recirculation line break event).

2) The BWR/2s, 3s, and 4s typically do not have multiple-redundant emergency core cooling (ECC) injection inside the core shroud beyond that provided by core spray (two independent lines). This reduces the number of ECC options available to the plant operator when various accident situations arise. BWR/5s and 6s have a single high pressure core spray (HPCS), a single LPCS, and three independent LPCI lines, one of which injects inside the core shroud. Thus, the later BWR designs (5s and 6s) have been improved in certain safety design aspects.

3) The BWR/2s do not have a LPCI system (no low-pressure ECC flood capability) which reduces the ECC options for accident mitigation. The BWR/2s rely on two trains of core spray for emergency core cooling.

In Phase II, the scope of the study was narrowed down to a representative single BWR/4 plant. This Phase was divided into two parts: a jet pump study to fully develop methodologies for the estimation of structural fragilities and modeling of PRA with the consideration of degradation of jet pump subcomponents, and a second study to apply this methodology to include failure of other reactor internal components and the cascading effects. Since IGSCC has been detected at many locations in BWR reactor internal components, and IGSCC cracking is expected to continue as the U.S. BWR plants age, this program assigned the probability of 1 for developing IGSCC cracks in IGSCC-susceptible reactor internal components. In order to truly assess the potential consequences and risks associated with IGSCC-induced failures of BWR reactor internal components, the initiating events considered not only IGSCC-induced failure of internal components and large LOCAs, but also both design-basis and beyond-design-basis earthquakes (using the plant-specific seismic hazard and in-structure response spectra). The quantitative analyses of Phase II provided the following insights:

1) Two possible types of loose parts from IGSCC-induced failure of BWR reactor internal components were identified, namely large parts that could result in cascading failure, or smaller parts that could migrate to the lower plenum and result in failure of parts of the reactor protection system (RPS) or isolated direct fuel damage. Assigning probability to smaller migrating loose parts has very high uncertainty. Based on engineering judgment and RPS functioning criterion from two safety studies (References 12 and 13), it was concluded that the smaller loose parts causing flow blockage of coolant to the fuel (thereby damaging fuel) or inhibiting control rod motion are below the PRA screening level.

2) The study identified three major accident scenario categories that should be focused on: both RPS and standby liquid control (SLC) nonfunctional; diversion and resulting inadequate coolant injection to the core; and core reflood to two-third level cannot be maintained. The 148 potential important accident scenarios were reduced to a more manageable number by focusing on the three major scenario categories and by using the quantitative PRA screening criterion. These major accident scenarios involve the most important internal components such as top guide, core plate, core spray system, jet pump, core shroud and core shroud support. However, calculations for seismic and large LOCA events indicated that components generally would not fail unless they are already severely degraded by IGSCC. Similarly, the study showed that cascading effects of large components interacting (e.g., forming a missile and causing impact) with other components in general would not damage the functionality of other components unless they are already severely degraded by IGSCC.

3) For multiple failures to occur, inspections would have to miss very severe cracking at several locations as well as having ineffective monitoring. The probability of this is estimated to be low if both inspections and monitoring are being performed in accordance with the BWRVIP aging management program.

4) Numerous BWRVIP submittals provided by NRR, including the quantitative safety assessment report (Reference 14), were reviewed in this program. In general, the calculations and estimates made in the documents submitted by the BWRVIP appear to be reasonable.

5) If no aging management program (which involves inspection, monitoring, evaluations, and repair) is in place, then a number of BWR vessel internals, if severely degraded by IGSCC, could fail either in a common mode or cascading manner, leading to an inability to insert control rods or cool the core in the event of a severe internal or external event.

6) With no credit for an aging management program which involves inspection, monitoring, evaluations, and repair (e.g., BWRVIP program), and a high probability of significant cracks developing in major reactor internal components based on operating experience to date, coupled with the initiating event frequencies and system failure frequencies in the PRA studied, an undesirable increase in the plant CDF (>5E-6/RY) is predicted.

7) With the current BWRVIP aging management program of inspection, monitoring, evaluations, and repair, there is expected to be no undesirable increase in CDF (<5E-6/RY) caused by

failures of BWR reactor internal components. That is, IGSCC problems can be identified and evaluated or corrected in time, to preclude an undesirable increase in the CDF.

8) While this risk study was performed for a BWR/4 plant, the results should be applicable to all BWRs since the inspection and repair methods are generally the same for all types.

Regulatory Implications

A number of BWR reactor internal components, such as top guide, core plate, core spray system, jet pump, core shroud and core shroud support, if allowed to be severely degraded by IGSCC, could potentially fail either in a common mode or cascading manner under an internal or external event (large LOCAs or earthquakes), and lead to an unacceptable increase in the plant CDF of the magnitude of 5E-6/RY or higher.

However, results of this program indicated that with the current BWRVIP aging management program of inspections, monitoring, evaluations, and repairs implemented by the licensees, and NRC/NRR's regulation and review of the BWRVIP/licensees activities, IGSCC problems could be identified, evaluated and corrected in time to preclude significant increase in CDF, i.e., the increase in CDF will be less than 5E-6/RY.

Closure

The conclusion of this study is that, the risk associated with the IGSCC of BWR reactor internal components is low given that a credible aging management program, such as the BWRVIP program of inspections, monitoring, evaluations, and repairs, is currently in place. However, absent a credible aging management program, the CDF has the potential to increase on the order of 5E-6/RY or higher.

REFERENCES

1. RICSIL 054, "Core Support Shroud Crack Indications," Rev.1, General Electric Company, July 21, 1993.

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3. Information Notice 94-42, "Cracking in the Lower Region of the Core Shroud in Boiling Water Reactors," U.S. Nuclear Regulatory Commission, Washington, D.C., June 7, 1994; Supplement 1, July 19, 1994.

4. Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shroud in Boiling Water Reactors," U.S. Nuclear Regulatory Commission, Washington, D.C., July 25, 1994.

5. Information Notice 97-17, "Cracking of Vertical Welds in the Core Shroud and Degraded Repair," U.S. Nuclear Regulatory Commission, Washington, D.C., April 4, 1997.

6. Information Notice 92-57, “Radial Cracking of Shroud Support Access Hole Cover Welds,” U.S. Nuclear Regulatory Commission, Washington, D.C., August 11, 1992.

7. Information Notice 93-101, “Jet Pump Hold-Down Beam Failure,” U.S. Nuclear Regulatory Commission, Washington, D.C., December 17, 1993.

8. Information Notice 95-17, “Reactor Vessel Top Guide and Core Plate Cracking,” U.S. Nuclear Regulatory Commission, Washington, D.C., March 10, 1995.

9. “NRR User Need Request for Support of Resolving Problem of Stress Corrosion Cracking of Reactor Vessel Internal Components,” Memorandum from W. T. Russell to E. S. Beckjord, December 2, 1994.

10. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions of Plant-Specific Changes to the Licensing Basis,” U.S. Nuclear Regulatory Commission, Washington, D.C., July 1998.

11. NUREG/CR-6677, “Evaluation of Risk Associated With Intergranular Stress Corrosion Cracking in Boiling Water Reactor Internals,” U.S. Nuclear Regulatory Commission, Washington, D.C., July 2000.

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