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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO GENERIC LETTER 94-03

DRESDEN, UNIT 2, AND QUAD CITIES, UNIT 2

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-237 AND 50-265

1.0 INTRODUCTION

The core shroud in a Boiling Water Reactor (BWR) is a stainless steel cylindrical component within the reactor pressure vessel (RPV) that surrounds the reactor core. The core shroud serves as a partition between feedwater in the reactor vessel's downcomer annulus region and the cooling water flowing up through the reactor core. In addition, the core shroud provides a refloodable volume for safe shutdown cooling and laterally supports the fuel assemblies to maintain control rod insertion geometry during operational transients and accidents.

In 1990, crack indications were observed at core shroud welds located in the beltline region of an overseas BWR. This reactor had completed approximately 190 months of power operation before discovery of the cracks. As a result of this discovery, General Electric Company (GE), the reactor vendor, issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications," on October 3, 1990, to all owners of GE BWRs. The RICSIL summarized the cracking found in the overseas reactor and recommended that at the next refueling outage plants with high-carbon-type 304 stainless steel shrouds perform a visual examination of the accessible areas of the seam welds and associated heat-affected zone (HAZ) on the inside and outside surfaces of the shroud.

Subsequently, a number of domestic BWR licensees performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic inspection experience. Of the inspections performed to date, significant cracking was reported at several plants. The combined industry experience from these plants indicates that both axial and circumferential cracking can occur in the core shrouds of GE designed BWRs.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03 "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," to all BWR licensees (with the exception of Big Rock Point) to address the potential

ENCLOSURE

for cracking in their core shrouds. Generic Letter 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

- inspect their core shrouds no later than the next scheduled refueling outage;
- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;
- develop an inspection plan that addresses inspections of all shroud welds, and delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject;
- develop a plan for evaluation and/or repair of the core shroud; and
- work closely with the BWROG on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking.

Commonwealth Edison Company (ComEd), the licensee for Dresden, Unit 2, and Quad Cities, Unit 2, responded to GL 94-03 on August 23, 1994. The staff reviewed the licensee's response to the GL and found open items. As a result, the staff issued a request for additional information (RAI) on September 27, 1994. By letters dated October 7, 1994, and October 13, 1994, ComEd responded to the RAI. At the request of the NRC staff, ComEd met with the staff to discuss ComEd's responses to GL 94-03. As a result of the meeting, additional open items were identified. By letter dated November 14, 1994, the staff issued an RAI. By letter dated December 14, 1994, ComEd responded to that RAI. The licensee's response included a schedule for inspection of the core shroud for each unit and a safety assessment supporting continued operation of each facility through the remainder of their current operating cycles. On October 7, 1994, the licensee provided the staff with additional information pertinent to justifying continued operation of both units until the remainder of their fuel cycles.

2.0 JUSTIFICATION FOR CONTINUED OPERATION AND SCHEDULE FOR INSPECTION/REPAIR

The licensee plans to conduct an inspection or repair, as appropriate, of the core shroud at Quad Cities, Unit 2, in March 1995 and at Dresden, Unit 2, in July 1995. The following is the staff's assessment of the licensee's basis for justifying continued operation of Dresden, Unit 2, and Quad Cities, Unit 2.

2.1 Susceptibility of Dresden, Unit 2, and Quad Cities, Unit 2, Core Shrouds to Intergranular Stress Corrosion Cracking (IGSCC)

The core shroud cracks which are the subject of GL 94-03, result from IGSCC which is most often associated with sensitized material near the component welds. Intergraular stress corrosion cracking is a time-dependent phenomenon requiring a susceptible material, a corrosive environment, and a tensile stress within the material.

Industry experience has shown that austenitic stainless steels with low carbon content are less susceptible to IGSCC than stainless steels with higher carbon content. Boiling water reactor core shrouds are constructed from either type 304 or 304L stainless steel. Type 304L stainless steel has a lower carbon content than type 304 stainless steel. During the shroud fabrication process when the sections of the core shroud are welded together, the heating of the material adjacent to the weld metal sensitizes the material. Sensitization involves carbon diffusion out of solution forming carbides at grain boundaries upon moderate heating. The formation of carbides at the grain boundaries depletes the chromium in the adjacent material. Since the corrosion resistance of stainless steel is provided by the presence of chromium in the material, the area adjacent to the grain boundary depleted of chromium is, thereby, susceptible to corrosion. Increased material resistance to IGSCC will result if the carbon content is kept below 0.035 percent, as specified for type 304L grade material.

Currently available inspection data indicate that shrouds fabricated with forged ring segments are more resistant to IGSCC than rings constructed from welded plate sections. The current understanding for this difference is related to the surface condition resulting from the two shroud fabrication processes. Welded shroud rings are constructed by welding together arcs machined from rolled plate. This process exposes the short transverse direction in the material to the reactor coolant. Elongated grains and stringers in the material exposed to the reactor coolant environment are believed to accelerate the initiation of IGSCC.

Water chemistry also plays an important role in regard to IGSCC susceptibility. Industry experience has shown that plants which have operated with a history of high reactor coolant conductivity have been more susceptible to IGSCC than plants which have operated with lower conductivities[1]. Furthermore, industry experience has shown that reactor coolant systems (RCS) that have been operated at highly positive, electrochemical potentials (ECP) have been more susceptible to IGSCC than RCSs that have been operated at more negative ECPs[2]. The industry has made a considerable effort to improve water

---

[1]Conductivity is a measure of the anionic and cationic content of liquids. As a reference, the conductivity of pure water is ~0.05 μs/cm. Reactor coolants with conductivities below 0.20 μs/cm are considered to be relatively ion free; reactor coolants with conductivities above 0.30 μs/cm are considered to have a relatively high ion content.

[2]The ECP is a measure of a material's susceptibility to corrosion. In the absence of an externally applied current, and, therefore, for reactor internals in the RCS, the electrochemical potential is equal to the open circuit potential of the material. Industry experience has shown that crack growth rates in reactor internals are low when the ECP $\leq$ ~-0.230 volts.

chemistry at nuclear facilities over the past ten years. Industry initiatives have included the introduction of hydrogen water chemistry as a means of lowering ECPs (i.e., making the ECPs more negative) in the RCS. The effectiveness of hydrogen water chemistry in reducing the susceptibility of core shrouds to IGSCC initiation has not been fully evaluated; however, its effectiveness in reducing IGSCC in recirculation system piping has been demonstrated.

Welding processes can introduce high residual stresses in the material at the weld joint. The high stresses result from thermal contraction of the weld metal during cooling. A higher residual tensile weld stress will increase the material's susceptibility to IGSCC. Although weld stresses are not easily quantified, previous investigation into weld stresses indicate that tensile stresses on the weld surface may be as high as the yield stress of the material. The stress decreases to compressive levels in the center of the welded section.

Commonwealth Edison Company has reviewed the materials, fabrication and operational histories (on-line years and water chemistry) of the Dresden, Unit 2, and Quad Cities, Unit 2, core shrouds and has submitted this information to the staff in its response to GL 94-03.

The core shrouds of Dresden, Unit 2, and Quad Cities, Unit 2, are susceptible to IGSCC and their susceptibility ranking is considered relatively high among all domestic BWRs. The plant-specific susceptibility factors are summarized below:

(i) The shrouds were fabricated with American Society for Testing and Materials (ASTM) A-240 Type 304 stainless steel plate. Carbon content varies from 0.044 to 0.063. The shroud welds were fabricated using ASTM Type E308 and ER-308 filler metal.

(ii) The core shroud support rings at Quad Cities, Unit 2, and at Dresden, Unit 2, were fabricated by welding rolled plate segments, followed by machining to size.

(iii) The two Quad Cities units have operated for almost the same number of years (16 years for Unit 1 and 16.1 years for Unit 2). Dresden, Unit 2, has operated slightly longer than Dresden, Unit 3 (17 years for Unit 2 and 15 years for Unit 3).

(iv) Both Quad Cities, Unit 2, and Dresden, Unit 2, operated at moderate reactor coolant ionic content levels during the initial years of operation. During the first five cycles of operation, the reactor coolant water conductivity at Quad Cities, Unit 2, averaged 0.377 μs/cm. Dresden, Unit 2, reactor water coolant water conductivity during the initial operating cycles averaged 0.299 μs/cm. The average for the entire population of U.S. BWRs is 0.340 μs/cm with a range from ~0.123 μs/cm to 0.717 μs/cm. The initial average conductivity for Quad Cities, Unit 2, (0.377 μs/cm), is comparable to that of Unit 1 (0.379 μs/cm). The Dresden, Unit 2, conductivity (0.299 μs/cm), was lower than that for Dresden, Unit 3 (0.399 μs/cm).

Considering the above plant-specific susceptibility factors as well as the industry-wide inspection experiences and the uncertainties in the residual stress profile resulting from fabrication, the staff concludes that significant cracking of the Dresden, Unit 2, and Quad Cities, Unit 2, core shrouds can not be ruled out. The above comparison of operating history for the core shrouds at Dresden, Unit 2, and Quad Cities, Unit 2, with those of Dresden, Unit 3, and Quad Cities, Unit 1, that is of similar construction and was recently inspected for cracking, provides some insight into the expected condition of the uninspected core shrouds at Dresden, Unit 2, and Quad Cities, Unit 2. However, Dresden, Unit 2, operated with lower conductivity coolant during the first five cycles of operation than that of Dresden, Unit 3. Furthermore, Dresden, Unit 2, has operated with hydrogen water chemistry (HWC) since 1983 and Quad Cities, Unit 2, has operated with HWC since the third quarter of 1990.

2.2 Basis For Continued Operation

2.2.1 Licensee's Assessment of Structural Integrity

The licensee assumed that cracks initiated in the shrouds at Dresden, Unit 2, and Quad Cities, Unit 2, after 3 effective full-power years (EFPY) of operation. Postulated crack depths were determined analytically for Dresden, Unit 2, considering plant specific water chemistry and hydrogen addition to the primary system. The licensee calculated a bounding crack depth of 0.64 inches for the unit. The licensee states that a remaining ligament of less than 10 percent is required to maintain structural integrity for the shroud under all design conditions. The licensee, using what they consider realistic crack growth rates determined by the PLEDGE model, concluded that the projected remaining ligaments for the Dresden, Unit 2, shroud welds would provide considerably greater margin than that required by the ASME Code. The licensee states that the crack depths measured for Quad Cities, Unit 1, during its recent inspection would be bounding for Unit 2 since the water chemistries and years of operation for the units have been similar. Their basis for their structural integrity assessment for Unit 2 is the evaluation of the cracking found at Unit 1.

2.2.2 Staff's Evaluation of Justification for Continued Operation

The staff has reviewed the inspection results for other BWRs with core shrouds more susceptible to IGSCC and notes that there has been no instance where a 360° through-wall crack existed in any plant that was inspected. Further, no BWR has exhibited any symptoms (power to flow mismatch) caused by leakage through a 360° through-wall crack. All analyses performed by licensees for higher susceptibility plants show that even if cracking did exist, ligaments would exist to assure structural integrity. In addition, there is a low probability for an initiating event which could potentially challenge the integrity of the core shroud, and there is only a short duration of operation until the licensee implements necessary inspections or repairs.

Dresden, Unit 2, has operated two more years than Dresden, Unit 3. However, based on susceptibility criteria discussed above in Section 2.1, Unit 2 appears less susceptible to IGSCC than Unit 3. Dresden, Unit 2, has been operating with hydrogen water chemistry for the past several operating cycles. In addition, the staff compared the history of reactor internals cracking between the Dresden units and found that the licensee has identified more indications of cracking in Unit 3 than in Unit 2 for a number of different components. The licensee's evaluation of the Dresden, Unit 2, core shroud utilized realistic estimates of time to initiation and crack growth rates. The methodologies used by the licensee to obtain these values has not been fully evaluated at this time. Benchmark testing is necessary to quantify any error involved in the licensee's calculations. However, considering the small remaining ligament necessary for adequate core shroud structural integrity and industry experience with shroud cracking, the staff feels that the Dresden, Unit 2, core shroud will have sufficient ligament at the end of the proposed operating period to preclude failure under all conditions. With regard to Quad Cities, both units have operated for approximately the same number of years with similar water chemistries. The staff believes that its safety evaluation dated July 21, 1994, for the cracking found at Quad Cities, Unit 1, would bound any cracking that could occur in the Unit 2 core shroud until the next refueling outage.

2.2.3 Consequence Assessment

Based on the evaluation provided in Section 2.0, the staff finds that the schedule for the inspection or preemptive repair of the core shrouds at Dresden, Unit 2, and Quad Cities, Unit 2, is acceptable. The staff concludes the units can continue to be safely operated until their next refueling outage. The bases are: (1) there has been no 360°, through-wall core shroud cracking observed to date in any U.S. BWR that has performed a shroud inspection; (2) all analyses performed by the licensee for Dresden, Unit 2, and Quad Cities, Unit 2, show that even if cracking does exist in the shrouds, ligaments would remain such that structural integrity would be assured; (3) Dresden, Unit 2, and Quad Cities, Unit 2, have not exhibited any of the symptoms (power to flow ratio mismatch) caused by leakage through a 360°, through-wall shroud crack; (4) there is a low probability of occurrence for either steam line or recirculation line breaks; and (5) there is only a short duration of operation until a repair or inspection is implemented.

Generic Letter 94-03 requested that licensees perform a consequence assessment of the shroud response to design basis loads and their effect on the ability of plant safety features to perform their function assuming 360° through-wall cracking. The licensee's intent of this consequence assessment was to demonstrate that fuel geometry and core cooling would be maintained given the unlikely occurrence of a through-wall failure of any horizontal weld and to identify whether horizontal weld failures would be detectable. Fuel geometry must be maintained to ensure control rod insertion while core cooling is ensured by proper emergency core cooling system (ECCS) performance. The licensee, considering differential pressure across the shroud head which was determined by the TRACG model, concluded that weld separation greater than one quarter of an inch during normal operations, was detectable at the H2 and H3 weld locations. The licensee predicted the weld failure during normal operations at the H5 location would not be detectable. The licensee also stated that the ability to maintain reactivity control, fuel geometry, core cooling, and a refloodable volume was assured with substantial margin for the H2, H3, and H5 weld locations, although some degraded performance was assumed for design basis events. Based on this assessment, the licensee concluded that core shroud separation and/or displacement occurring during normal operations or during anticipated events would have no effect on the primary safety functions of reactivity control and core cooling which are required to mitigate those events.

The staff performed a qualitative assessment of the licensee's consequence assessment. The staff found the submittal to be an adequate assessment of the consequences of a main steamline break (MSLB) and a MSLB plus seismic event for the H2, H3, and H5 weld locations. The staff also reviewed the consequence assessment of a recirculation line break (RLB) and a RLB plus seismic event, with acoustic and blowdown loads, for the H2 and H3 weld locations at Dresden, Unit 2, and Quad Cities, Unit 2. The staff could not entirely verify all the details of the evaluations by the licensee for Dresden, Unit 2, and Quad Cities, Unit 2, such as the inherent uncertainties in the TRACG model, the large uncertainties about the irregular crack surface of the postulated failed welds, and the identified uncertainties in the RLB loads used in the assessment of H5. For a main steamline event, the licensee's calculations demonstrated that the top guide would not lift above the fuel, therefore, assuring no lateral fuel movement. While this conclusion is reasonable, there may be some small likelihood of top guide lift above the fuel for upper weld locations, due to the inherent uncertainties in the analysis methods, as mentioned above. However, even if this were to occur, the staff concluded that safe shutdown of the reactor should be achieved by the activation of the standby liquid control system (SLCS). Assuming the presence of through-wall failures of shroud welds, the other initiating event of concern would be the recirculation line break. The licensee's calculations indicated momentary tipping of the shroud at certain postulated failed weld locations due to the blowdown forces, but no permanent lateral movement. For such shroud response, the staff agrees that adequate core flooding will be maintained since little core/annulus bypass will occur. Modeling the behavior of a cracked shroud during a RLB is quite complex, involving assumptions on crack surface friction and other interacting forces in the vertical and lateral directions. Therefore, the staff is unable to conclude with high confidence that such lateral motion can not occur following a RLB. Lateral motion of less than the thickness of the shroud would only result in small bypass leakages. However, large lateral movement could open up a significant leakage path through the shroud which could possibly prevent adequate 2/3 height core flooding following the RLB. Although the staff could not agree with the licensee's assessment of the RLB with an assumed through-wall crack at the lower shroud weld, the staff has concluded that only the most extreme assumptions, with respect to initiating events, crack locations, and crack depths, would result in unacceptable consequences.

3.0 CONCLUSION

Based on the above evaluation, due to the low frequency of the initiating event, the availability of the SLCS, and the presence of the remaining ligament to assure structural integrity, the staff concludes that there is no undue risk to the public health and safety. Therefore, power operation is acceptable until the next scheduled refueling outages in March 1995 for Quad Cities, Unit 2, and July 1995 for Dresden, Unit 2.

4.0 OUTSTANDING ISSUES/FUTURE ACTIONS

In accordance with the reporting requirements of GL 94-03, the licensee shall submit to the NRC, no later than 3 months prior to performing the core shroud inspections, both the inspection plan and the plan for evaluation and/or repair of the shroud based on the inspection results. In addition, results should be provided to the NRC within 30 days from the completion of the inspection. If the licensee identifies any core shroud cracking requiring an analysis per the ASME Code, details of such an evaluation must also be submitted to the NRC for review.

It should be noted that the industry is currently encountering difficulty in performing comprehensive inspections of lower shroud welds due to NDE equipment accessibility problems. The staff urges licensees to work with various vendors and the EPRI NDE Center in order to develop improved reliable tooling for inspections of shroud welds which are highly obstructed. Should improved inspection techniques become available, the staff's recommendation is for licensees to reinspect the lower shroud welds at the earliest opportunity.

The licensee indicated in their response that they may adjust their core shroud inspection schedule and scope per guidance from the BWRVIP. At present, the NRC has not approved the inspection guidelines proposed by the BWRVIP. Considerable differences remain with regard to the recommended scope of core shroud inspections. The staff cautions the licensee against modifying their plans according to BWRVIP recommendations which have not undergone review and approval by the NRC. The staff's current position with regard to the scope of inspections is a recommendation for the inspection of 100% of the accessible core shroud welds. Should the licensee opt to install a preemptive repair in lieu of performing a comprehensive core shroud inspection, the only required inspection is that mandated in the staff approval of the repair option.

Principal Contributor: P. Rush

Date: January 31, 1995