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

October 5, 1993

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Albert F. Gibson, Director
Division of Reactor Safety, RII

William L. Forney, Acting Director
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Samuel J. Collins, Director
Division of Reactor Safety, RIV

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Division of Reactor Safety and Projects, RV

FROM: James T. Wiggins, Acting Director
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: POTENTIAL CORE SHROUD CRACKING AT BOILING WATER REACTORS

As a result of in-vessel visual inspections performed during the current refueling outage for the Brunswick Unit 1 reactor, the Carolina Power and Light (CP&L, the licensee) staff informed the NRC of numerous cracks contained in the Brunswick 1 core shroud. The inspections were performed by the licensee in accordance with the recommendations contained in General Electric (GE) Corporation Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications," which was issued as a result of cracking previously discovered in the core shroud of an overseas boiling water reactor (BWR).

The report of a circumferential crack found in horizontal weld H-3, which fuses the top guide support ring to the lower shroud, was particularly significant. This crack is located in the heat affected zone of the weld and extends nearly 360° circumferentially around the shroud. Subsequent evaluations of the H-3 crack by ultrasonic testing (UT) methods and by boat sample analyses indicated that the crack has significant depth. Numerous other axial and circumferential cracks were also discovered by the licensee at welds H-2, H-4 and H-5 of the core shroud; however, these cracks were determined by the licensee to be of lesser safety significance than the crack discovered at weld H-3.

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The licensee and GE have kept us well informed of developments in regard to the Brunswick Unit 1 core shroud cracking. The Office of Nuclear Reactor Regulation has issued Information Notice 93-79 to inform the holders of BWR operating licenses of the cracking. Furthermore, GE has updated RICSIL 054 (Revision 1) to incorporate the core shroud cracking information compiled by the CP&L and GE.

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The core shroud cracking issue at the Brunswick Unit 1 reactor has attracted the attention of the press, the U.S. Congress, and of several intervenors. We are enclosing a packet for your information, to provide a preliminary view as to its safety significance, and to elicit your help in our follow-up of the issue. The packet includes Information Notice 93-79, "Core Shroud Cracking At Beltline Region Welds in Boiling-Water Reactors," GE RICSIL 054, "Core support shroud crack indications," and a report issued by the Division of Systems Safety and Analysis, NRR, titled "Core Shroud Cracking Preliminary Safety Assessment." As indicated, the safety assessment is both preliminary in nature and for internal use only. Please do not distribute it outside of the NRC.

I would appreciate if you would discuss shroud inspection plans with BWR licensees which are in or will be in a refueling outage, and provide insights, via E mail, to Jack R. Strosnider, Chief, Materials and Chemical Engineering Branch, NRR. Please feel free to contact any of the Technical Contacts listed in the Information Notice, Don Brinkman, the lead PM on the issue (at 301-504-1409), or Mr. Strosnider (at 301-504-2795) should you have any questions in regard to the core shroud cracking.

ORIGINAL SIGNED BY:
James T. Wiggins

James T. Wiggins, Acting Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosures: As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

September 30, 1993

NRC INFORMATION NOTICE 93-79: CORE SHROUD CRACKING AT BELTLINE REGION WELDS IN BOILING-WATER REACTORS

Addressees

All holders of operating licenses or construction permits for boiling-water reactors (BWRs).

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees that cracks have been observed in the weld regions of the core support shroud in boiling water reactors. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

During the current refueling outage at Brunswick Unit 1 (a BWR-4 reactor), in-vessel visual inspection revealed cracks at weld regions of the core support shroud. The shroud is a stainless steel cylinder that serves to direct the flow of water inside the reactor vessel. The shroud is completely contained inside the 15.2 centimeter [6 inch] thick reactor vessel. The structural integrity of the reactor vessel is not impacted by the cracks in the shroud.

Carolina Power and Light Company (CP&L), the licensee for Brunswick, found both circumferential and axial cracks in the shroud. The circumferential cracks were located in the inside shroud surface in the heat-affected zone (HAZ) of weld H-3 and extended 360° around the circumference of the shroud (see Figures 1 and 2). Weld H-3 is a horizontal weld which fuses the top guide support ring to the lower shroud. The first axial crack discovered was located on the outside shroud surface of weld H-4 in the lower shroud. CP&L performed additional visual testing (VT) and ultrasonic testing (UT) of the shroud and removed boat samples to evaluate the length and size of the cracks.

Discussion

In 1990, crack indications were reported at core shroud welds located in the beltline region of an overseas reactor (BWR4). This reactor had completed approximately 190 months of power operation before the cracks were discovered. As a result of this discovery, General Electric (GE) issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud

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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 HAZ on the inside and outside surfaces of the shroud.

Since early July, CP&L has performed VT inspections of the Unit 1 inside and outside shroud surface in the vicinity of welds. These inspections were performed in accordance with GE RICSIL 054 and discovered cracks in the weld regions. CP&L determined that in order to perform an adequate VT it was necessary to remove the outer blade guides, pre-clean inspection areas, and obtain an improved resolution of "1-millimeter wire" (in lieu of the Code-prescribed resolution). Camera and lighting positions were also found to be crucial in performing adequate VTs. Also, CP&L has worked with GE to develop more sophisticated UT equipment to identify how deeply into the shroud metal the crack extends.

Additional VT inspections revealed more axial cracks at the inside surface of weld H-4 as well as cracks at welds H-1, H-2, and H-5 of the shroud. One of the additional cracks, a circumferential crack at weld H-5, appears to be approximately 76.2 centimeters [30 inches] in length. The majority of the cracks are located in the HAZ of the welds, although one crack was discovered in the central region of shroud plate P-6. The crack in P-6, however, may be associated with a possible weld repair of a surface defect in the plate after its fabrication at the mill.

The results from the boat samples indicated intergranular stress-corrosion cracking (IGSCC) as the mechanism. Preliminary results suggest that the crack in the HAZ of weld H-3 may be 3.8 centimeters [1.5 inches] or more in depth. The location of this crack is shown in Figure 2.

As a result of the shroud cracks being discovered on Unit 1, CP&L re-examined the results of the inspection performed during the 1991 refueling outage of Unit 2. The re-examination revealed three minor crack indications in the HAZ of weld H-2. The licensee concluded that the cracks do not pose a concern to normal operation of the reactor.

CP&L plans to repair the Brunswick Unit 1 core shroud before the plant is brought back into service. CP&L intends to restore the strength of the shroud by adding stiffening braces around the top portion of the shroud. However, the licensee will continue to examine and evaluate the cracks in the core shroud.

General Electric issued Revision 1 to RICSIL 054 on July 21, 1993, to update the information on the core support shroud cracks and to provide revised interim recommendations to perform visual examination of accessible areas of the shroud at all GE BWRs during the next scheduled outage. The NRC has been informed by GE that they are evaluating the Brunswick results and will provide updated information to owners of GE BWRs. The NRC staff is evaluating the implications of the shroud cracks for reactor core configuration and emergency core cooling system performance under accident conditions at operating plants and will consider the need for additional generic communications.

This information notice requires no specific action or w .. ten response. If you have any questions about the information in this not.ce, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contacts: R. A. Hermann, NRR (301) 504-2768

J. Medoff, NRR (301) 504-2715

P. Byron, Region II (919) 457-9531

T. Greene, NRR (301) 504-1175

Attachments:
1. Figure 1: Weld and Plate Locations in the Beltline Region of the Brunswick Unit 1 Core Shroud
2. Figure 2: Details of Weld Locations H-2 and H-3 in the Brunswick Unit 1 Core Shroud
3. List of Recently Issued NRC Information Notices

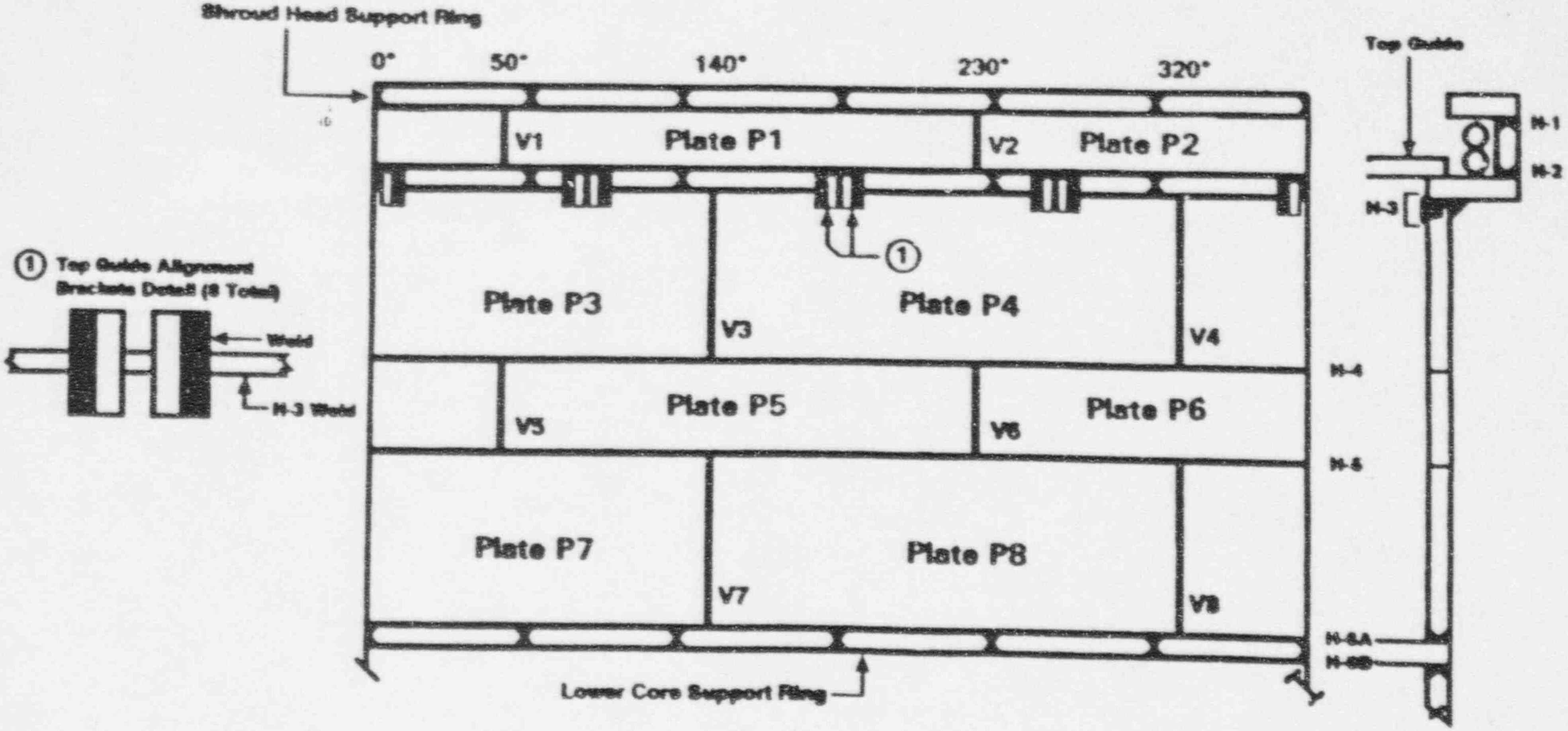

Figure 1. Weld and Plate Locations in the Beltline Region of the Brunswick Unit 1 Core Shroud (From CP&L)

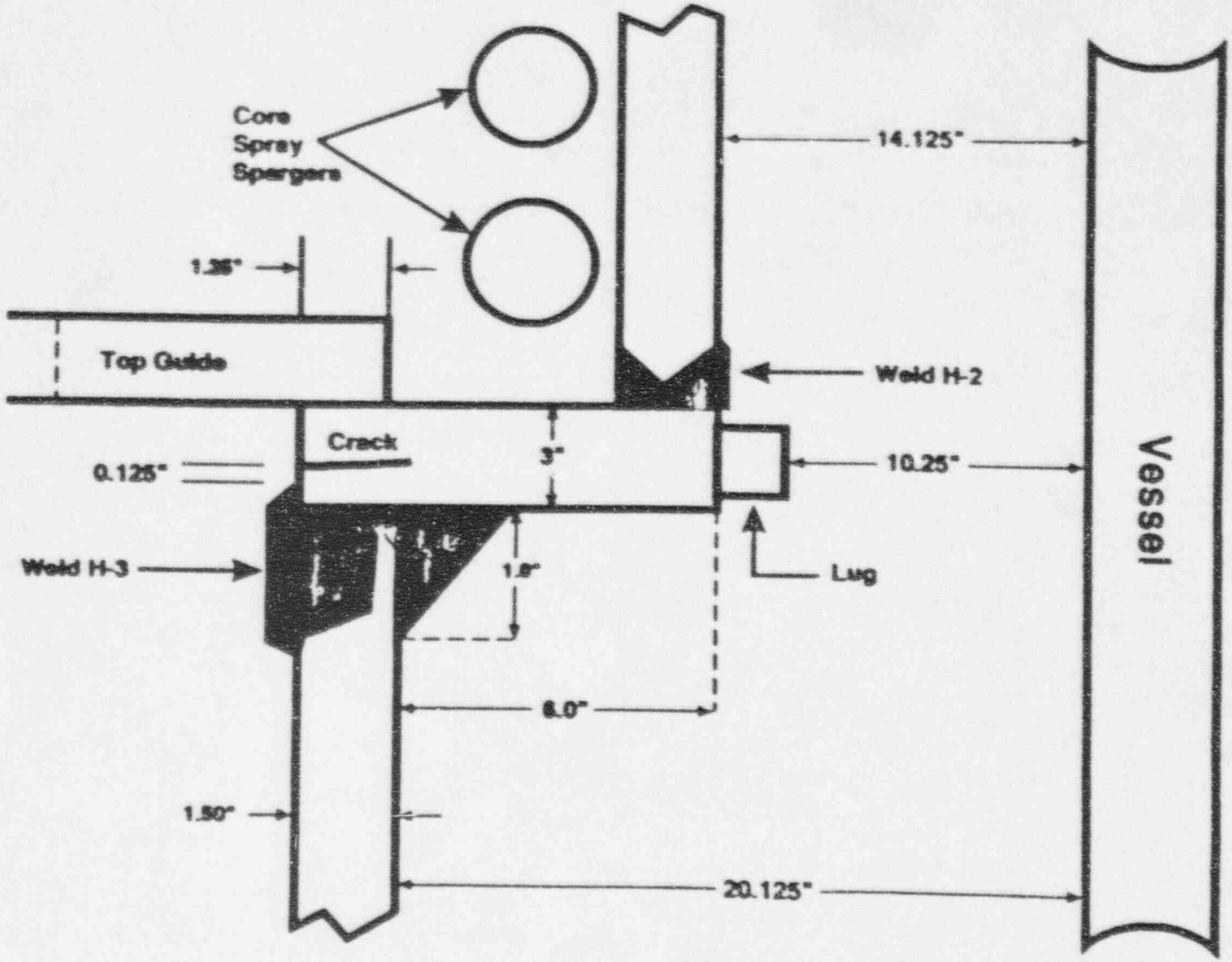

Figure 2. Details of Weld Locations H-2 and H-3 in the Brunswick Unit 1 Core Shroud (From CP&L)

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-78	Inoperable Safety Systems At A Non-Power Reactor	10/04/93	All holders of OLs or CPs for test and research reactors.
93-77	Human Errors that Result in Inadvertent Transfers of Special Nuclear Material at Fuel Cycle Facilities	10/04/93	All nuclear fuel cycle licensees.
93-76	Inadequate Control of Paint and Cleaners for Safety-Related Equipment	09/21/93	All holders of OLs or CPs for nuclear power reactors.
93-75	Spurious Tripping of Low-Voltage Power Circuit Breakers with GE RMS-9 Digital Trip Units	09/17/93	All holders of OLs or CPs for nuclear power reactors.
93-74	High Temperatures Reduce Limitorque AC Motor Operator Torque	09/16/93	All holders of OLs or CPs for nuclear power reactors.
93-73	Criminal Prosecution of Nuclear Suppliers for Wrongdoing	09/15/93	All NRC licensees.
93-72	Observations from Recent Shutdown Risk and Outage Management Pilot Team Inspections	09/14/93	All holders of OLs or CPs for nuclear power reactors.
93-71	Fire at Chernobyl Unit 2	09/13/93	All holders of OLs or CPs for nuclear power reactors.
93-70	Degradation of Boraflex Neutron Absorber Coupons	09/10/93	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

GE Nuclear Energy

RICSIL
Rapid Information Communication Services Information Letter

Core support shroud crack indications

RICSIL No 054
Revision 1
July 21, 1993

In October 1990, GE Nuclear Energy reported in RICSIL No. 054 that cracking had been observed near the circumferential seam weld at the core midplane of the type 304 stainless steel core support shroud in a GE BWR/4 located outside the United States. The crack indications, which were observed initially at three locations on the inside surface of the shroud, were confined to the heat affected zone (HAZ) of the circumferential seam weld. The cracking was judged initially to be IGSCC. Ultrasonic examination (UT) performed in 1990 in the area of the longest indication—at the 100 degrees vessel azimuth location—confirmed continuous circumferential cracking with depths estimated to vary from about 0.08 inch to about 0.18 inch.

RICSIL No. 054 furnished GE's interim recommendations that owners of all GE BWRs review fabrication records for shroud material type and location of shroud seam welds. For plants with high carbon type 304 stainless steel shrouds, GE recommended that owners perform a visual examination of accessible areas of the seam welds and associated HAZs on the inside and outside surfaces of the shroud during the next scheduled outage.

The seam weld of the affected BWR was re-inspected during scheduled refueling outages both in 1991 and 1992. UTs performed with improved equipment and techniques have confirmed that the cracking in the region of the longest visible crack indication was as deep as 0.70 inch.

During the 1992 outage a two-inch diameter through-wall "plug" sample was removed from the cracked region of the shroud for metallurgical analysis to establish the root cause of cracking.

Results of these recently completed metallurgical evaluations and a shroud cracking occurrence at a second GE BWR have led GE to revise its original interim recommendations. This RICSIL No. 054 Revision 1 discusses current knowledge of the nature of the first occurrence of shroud cracking and presents current knowledge of a more recent occurrence of shroud cracking. This RICSIL No. 054 Revision 1 voids and closes the original RICSIL No. 054.

Discussion

Results of metallurgical evaluation

The shroud at the plant in which the first shroud cracking occurred was fabricated from four sections—two in each shell course—of 1.25 inches thick, roll-formed type 304 stainless steel plate. (Fabrication records show that two of the plates contained 0.060 percent carbon and that the other two contained 0.045 percent carbon.) The plates are joined by one circumferential weld at core midplane and four vertical seam welds. All indications of cracking were associated with the circumferential weld that join the upper and lower shroud segments. Most cracking was oriented circumferentially in higher fluence regions—$8x10^{20}$ nvt (E > 1 MeV).

Metallurgical evaluations were completed on a two-inch diameter "plug" sample removed from the shroud crack at the 100 degrees vessel azimuth location. Key results of the metallurg' ·l evaluations performed on the plug samp.e were as follows:

- The material affected with cracking is the lower carbon heat—0.045 percent carbon.
- The material is not thermally sensitized; an HAZ is not present. The cracking is multibranched, intergranular, and affects the base metal as well as the weld structure.

• There is no surface cold work, but there may be some initial fabrication-related bulk cold work.

• Based on the results of the metallurgical investigative work, GE's understanding of the root cause of cracking is Irradiation Assisted Stress Corrosion Cracking (IASCC) with propagation promoted by weld residual stress and possible corrosion oxide wedging stresses.

Recent occurrence of shroud cracking

In July 1993, a second occurrence of shroud cracking was reported. Crack indications were found in a U. S. GE BWR/4 in the top guide support ring near the weld between the shroud and the ring—designated the H3 weld at this plant. The cracking, which was found by in-vessel visual inspection (IVVI), was circumferential on the inside diameter and relatively long—assumed to be 360 degrees of cracking. Because of the rolling direction of the original fabrication of the ring, the orientation of laminar inclusions within the ring would coincide with the orientation of the observed cracking. Subsequent UT from the inner and outer surfaces using a pole-mounted transducer showed that crack depth ranged from 0.18 inch (4.5 mm) to about 0.40 inch (10 mm.) The material is type 304 stainless steel with carbon content around 0.06 percent.

Cracking also was found near the horizontal weld that joins two sections of the shroud—designated the H4 weld at this plant—36 inches below the top guide ring weld. The crack was axial, one inch long and located on the outside surface of the shroud. Subsequent UT from the outside showed that the depth was about 0.25 inch. Visual examination of the corresponding location on the shroud's inner surface did not show cracking. Preliminary estimates are that fluence on the inner surface is approximately 1.8×10^{20} nvt (E > 1 MeV) for the H3 weld and 5.1×10^{20} for the H4 weld.

At the time this RICSIL No. 054 Revision 1 was issued, evaluations were continuing with particular focus on how these two occurrences of shroud cracking may be related. The first occurrence was the result of an IASCC mechanism. However, the second occurrence may have been caused by the combined effects of IGSCC and IASCC with possible aggravation from planer material inclusions in the region of the H3 weld HAZ. GE Nuclear Energy expects to furnish related recommendations to owners of GE BWRs when the nature of the recently observed cracking is understood adequately. Until that time, GE's revised interim recommendation is that owners of all GE BWRs perform a visual examination of accessible areas of the seam welds and associated HAZs on the inner and outer surfaces of the shroud during the next scheduled outage.

To receive additional information on this subject, please contact your local GE Nuclear Energy service representative.

This RICSIL pertains only to GE BWRs. The conditions under which GE Nuclear Energy issues RICSILs are stated in RICSIL No. 001 Revision 1, the provisions of which are incorporated into this SIL by reference.

Product reference

B11, B12, B13—Reactor Pressure Vessel

Technical source

D. E. Delwiche 4110

Issued by

JGMoore

J. G. Moore, Manager
Customer Service Communications
GE Nuclear Energy
175 Curtner Avenue, San Jose, CA 95125

CORE SHROUD CRACKING

PRELIMINARY SAFETY ASSESSMENT

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1. INTRODUCTION

During the current Brunswick Unit 1 refueling outage, the licensee performed an inspection of the core shroud and has found numerous crack indications. Inspection of the shroud is on going. The most significant of these indications was a 360° circumferential crack at the edge of the 2.25 inch thick H3 weld. This crack was discovered visually. Ultrasonic testing (UT) revealed this crack ranges in depth from 0.8 to 1.7 inches. The staff has had several meetings with the licensee and GE to discuss the findings of the core shroud inspection and the licensee's repair plans. The applicability of these findings to other BWRs was also discussed.

As a result of the degree of cracking observed in the core shroud, the staff has performed an initial scoping safety assessment of the plant specific and generic implications of the inspection findings. This preliminary safety assessment reports the initial staff conclusions concerning the safety significance of the core shroud cracking on operating BWRs. The staff will continue to evaluate this issue and monitor future core shroud inspections.

This report first discusses the cracking observed at Brunswick Unit 1 and the safety impact of the as-found cracks. The evaluation is then extended to evaluate the safety significance of the core shroud cracks assuming that they had continued to propagate through the shroud wall. The resulting behavior of the core shroud under normal operation, accidents and earthquakes is then evaluated and the resulting plant response is discussed. A preliminary probabilistic perspective is provided for these scenarios to estimate the probability of a core damage resulting during these conditions. Finally, a discussion of the planned industry and staff actions is provided.

2. BRUNSWICK UNIT 1

During the current Brunswick Unit 1 refueling outage, an inspection of the core shroud was performed. This inspection was performed in response to GE Rapid Information Communication Service Information Letter (RICSIL) No. 054, "Core Support Shroud Crack Indications," as a result of core shroud cracking observed at mid-core elevations at a foreign plant. Cracking indications at a similar location were observed in Brunswick Unit 1. The licensee also performed a visual inspection of the H3 weld (not an area specified in the RICSIL), located near the top of the core, and found a 360° circumferential crack. Subsequent UT measurements revealed the H3 crack to have a depth ranging from 0.8 to 1.7 inches. The staff

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has issued Information Notice 93-79 on the findings from the Brunswick Unit 1 core shroud inspections. A copy of the Information Notice is attached.

The core shroud is illustrated in Figure 1. The core shroud separates the core region from the downcomer annulus, which contains the jet pumps, and assures that feedwater flow is directed down the downcomer annulus, through the jet pumps, through the lower plenum, and up through the core region. From the core region, the exiting two-phase flow is directed through the steam separators and dryers and the steam then exits the vessel through the steam lines while the liquid is recirculated to the downcomer annulus. The core shroud is welded at the bottom to the ledge at the bottom of the downcomer annulus upon which the jet pumps sit. The shroud is further supported by 14 pedestals welded to the bottom head of the reactor vessel. The steam separators and the dryers are mechanically attached to the core shroud at the ledge region near the top of the core above the core spray spargers. The core top guide structure, which provides lateral support for the fuel assemblies and assures that the core geometry is maintained to allow for control rod insertion, is supported by a second ledge below the core spray spargers. The spray header for the high pressure and low pressure core spray systems is contained within and supported by the core shroud and the connecting piping enters through the vertical portion of the core shroud above the top guide support ring (ledge).

The specific cracking observed at Brunswick Unit 1 is illustrated on Figures 2 and 3. Numerous cracks were observed in the lower elevations of the vertical portion of the core shroud surrounding the core. These cracks predominantly ranged from one to three inches in length, with several longer cracks of 9 and 30 inches in length. None of these cracks penetrated through the shroud wall. A 360° circumferential crack was observed at weld location H3 which connects the upper ledge of the core shroud to the vertical portion of the core shroud which surrounds the core. At this location, the weld thickness is 2.25 inches. Using UT, the crack depth was found to range from 0.8 to 1.7 inches.

The licensee was questioned on whether this cracking had been previously observed and noted that no inspections of the core shroud had been performed until this outage, nor are such inspections required by the ASME code. The licensee believes that it is likely that these cracks initiated early in the operating period (the plant has been operating since 1976) when water chemistry was not well controlled. Further, it was the licensee's position that growth observed for IGSCC cracks is slow, and it is likely that many years of operation would be required to obtain the amount of cracking observed.

With the available information, the staff cannot assess the growth rates for the cracking observed at Brunswick. However, based on the staff's knowledge of IGSCC crack behavior, the staff finds that the licensee's conclusion that many years of operation were

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necessary to achieve the amount of cracking observed is reasonable.

The licensee and GE also noted that IGSCC is also more likely to occur for the high carbon 304 stainless steel material used at Brunswick -- and other early BWR units including the BWR3 and some early BWR4 designs -- and the specific construction method used to fabricate the top guide support ring. The ring at Brunswick was constructed from a series of segments flame cut from a plate, butt welded together and then machined to the final dimensions. The ring was then submerged-arc welded to the core shroud. This welding process and the weldment design created a significant amount of residual stress in the support ring. The heavy surface machining and the plate rolling orientation also contributed to stress risers and crack initiation. GE estimated that 25% of the operating BWRs may have similar fabrication specifications or conditions and could be susceptible to the type of cracking observed at Brunswick Unit 1.

The licensee has assessed the significance of the observed core shroud cracking. Since the cracks in the lower core elevation do not appear to be through wall, no safety significance was attributed to these cracks. The licensee indicated that even if these cracks had penetrated through wall, it expected that the cracks would be "tight", typical of IGSCC cracks, and only small leakages would result. Further, given the low pressure differential across the shroud wall at these locations, the licensee estimated that the leakage would be limited to a maximum of approximately 50 gpm. This small amount of leakage would have no impact on normal plant operation. The licensee also concluded that even under accident conditions, gross leakage would not be expected. Therefore, core coolability under accident conditions would be maintained.

The staff has not performed a detailed review of the licensee's assessment, but based on experience with other IGSCC cracks at BWRs, the staff believes that the licensee's conclusions are reasonable. Therefore, the staff initially believe that these cracks are unlikely to be of safety significance.

With respect to the cracking observed at the H3 weld location, the licensee has evaluated the structural integrity of this weld under normal operation, accident, and earthquake conditions. The licensee has estimated that if a ½ inch ligament remained around the circumference, the structural margins , including an allowance for crack growth, required by the ASME code would not be exceeded. Therefore, the licensee concluded that the as-found cracks would not impair structural integrity and were not safety significant.

The staff does not believe that sufficiently conservative margins of strength will remain for an additional operating cycle. Additionally, the staff felt that the UT examination was not sufficiently accurate in its indication of crack depth.

The licensee has elected to perform a repair that will provide the

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structural requirements for the attachment of the top guide support ring to the lower core shroud. Specifically, the licensee is evaluating structural stiffeners to compensate for the loss of margin in the structural integrity of the H3 weld. In addition, the licensee will start up with H_2 water chemistry (20 scfm) which will reduce the growth rate for the IGSCC cracks and reduce the potential for future crack development. The staff will continue to evaluate the licensee's repair actions.

The staff has performed an assessment of the consequences of the cracking observed at the H3 weld if it had proceeded to propagate through wall. This assessment is discussed in the following sections.

3. GENERIC SAFETY ASSESSMENT

The preceding discussion centered on the safety assessment of the specific cracking observed at Brunswick Unit 1. As noted above, the licensee concluded, and the staff concurs, that the cracking observed in the portion of the core shroud surrounding the core is not safety significant. Thus, this section will not discuss these cracks any further. The crack at the H3 weld is of potential safety significance. In the evaluation which follows, the staff has assumed that the H3 weld cracking would continue to propagate and lose its structural integrity either during normal operation, an accident condition or a seismic event. The resultant effect of the failure of the H3 weld is then discussed, including an assessment on any plant unique features, and the conditions needed to progress to a severe core damage event is provided. A initial probabilistic perspective is then provided. Finally, our views drawn from this preliminary safety assessment are provided.

3.1 PLANT RESPONSE WITH H3 WELD FAILURE

3.1.1 Normal Operation

Under normal operating conditions, sufficient hydraulic forces exist to lift the upper internals if the H3 weld was completely failed. GE stated that if the upper internals lifted by approximately one-eight of an inch, a large bypass flow would occur which would be observable to the operators as a power-flow mismatch and would result in the operator initiating a plant shutdown. GE noted that such lifting has occurred in operating plants when the upper internals have not been properly bolted in place, and was readily observable. If a complete failure of the H3 weld was postulated, a larger lift than previously observed would occur, and an automatic reactor trip would occur on high water level. The staff believes this assessment is reasonable.

Of particular interest if a complete failure of the weld occurred was the response of the top guide core structure. Lifting of the top guide above the top of the fuel assemblies would result in the loss of lateral support for the fuel assemblies and a loss of the

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spacing between the fuel bundles such that full control rod insertion may not be possible. GE calculated that given the complete failure of the H3 weld, the top guide would be lifted approximately 5 incnes. This is significantly less than the 14 inch lift needed to lift the top guide above the fuel assemblies. Thus, GE concluded that the alignment needed for rod insertion during normal operation would be maintained. It should be noted that the shroud portion that would contain the top guide would retain the shroud lateral (seismic) supports intact. While the staff has not reviewed the GE calculations, given the large margins which exist, the staff believes this conclusion is reasonable.

It is also possible that the H3 weld may not completely fail, and that an asymmetric, hinge type, lift of the core shroud may occur. GE has not fully considered this possibility. Such failures may result in oscillatory flow through the core and may lead to core power instabilities. While this condition is not fully analyzed, the staff believes that any significant lifting of the core shroud will be observable and the potential core consequences are unlikely to lead to severe core damage. Further assessment of this issue is continuing.

3.1.2 Accident Conditions

GE has evaluated the potential results of a failure of the H3 weld under accident conditions. Specifically examined were the responses to recirculation line and main steam line breaks.

For a recirculation line break, GE concluded that the forces on the H3 weld would be smaller than that observed during normal operation. Therefore, GE concluded that if no failure had previously been observed during normal operation, structural integrity would be maintained during the recirculation line break. The staff questioned GE about the asymmetric loadings that the core shroud would be exposed to during this accident. GE stated that the acoustic wave is significantly attenuated by the time it reaches the upper portions of the core shroud and that the subsequent loadings would be low. Further, the support provided by the spray system piping would restrict motion of the upper portions of the core shroud. While the staff has not reviewed GE's assessment, the staff believes these conclusions are reasonable given the core flow pattern that would result from such a break.

For steam line break conditions, the hydrodynamic loads across the shroud are sufficient to result in a significant lift of the upper internals if the H3 weld completely fails. This lift is expected to result in the top guide core structure to lift above the fuel assemblies and lateral support to the assemblies will no longer be provided. In addition, the spray headers are likely to be damaged and Emergency Core Cooling System (ECCS) injection inside of the shroud through these headers could be lost. However, the weakest point in the portion of the core spray system in the vessel is the section of pipe immediately inside the vessel. Thus, it is anticipated that as a result of this failure, the injected flow

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would still be delivered to the vessel and would flow into the downcomer annulus and through the lower plenum to the core. Further, for Brunswick Unit 1 and plants with similar ECCS system, the four low pressure coolant injection pumps would remain available to provide core cooling.

Because of the large driving force provided by the scram accumulators, GE believes that the control rods will insert. GE stated that the control rods have sufficient force to possibly even lift the assemblies off the lower core support, but the control rods would be maintained within the core volume and a coolable geometry would be maintained.

The staff generally agrees with GE's assessment for plants with ECCS equipment similar to Brunswick Unit 1, but is not assured that control rod insertion will occur. Therefore, while core cooling would initially be adequate, it is the staff's view that ultimately boron injection via the standby liquid control system (SLCS) may be necessary to maintain the core subcritical and assure core cooling by ECCS. GE stated that while this specific scenario has not been analyzed, operators have been trained to respond to a depressurization accident with an Anticipated Transient Without Scram (ATWS). Using the existing emergency procedure guidelines (EPG), the operator would be expected to limit the Low Pressure Coolant Injection (LPCI) flow to minimize reactor power, while maintaining core cooling. While this specific scenario has not been analyzed, the staff believes that the actions specified in the EPGs should provide adequate operator response for this scenario.

The staff has assessed whether differences in ECCS designs may result in conditions different than that discussed above for the Brunswick plant. Other BWR plants have more reliance on ECCS spray systems which would be damaged as a result of lifting of the core shroud by a steam line break. However, as noted above, it is expected that injection flow would be maintained except it would be provided via the downcomer instead of through the spray system. Since injection would be maintained, the staff believes that adequate core cooling would likely be maintained for all BWRs for this scenario.

3.1.3 Earthquakes

Given an earthquake, a concern arises that the core shroud may move and result in a displacement of the upper guide structure such that the control rods may be incapable of fully inserting following the event. GE noted that because of the design of the internals, the maximum movement that may occur would be about 4 inches at which time the fuel rods would be contacting the vertical portions of the core shroud surrounding the core. GE also opined that it would take an earthquake approximately two to three times the design basis earthquake to result in such a large movement.

GE believes that the control rods are likely to insert even given a 4 inch lateral movement of the upper guide structure. Because

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the control rods are connected to the drive by a ball-like support, sufficient flexibility would exist to drive the control rods even given this displacement. GE supported this assertion by noting it had performed rod insertions using fuel assemblies bowed by approximately 2.5 inches at the core center. For these experiments, control rod insertion occurred to within a few inches of complete insertion even with this bowing. It was GE's position that these tests represent a much more extreme condition than that which would occur for a lateral displacement of the upper guide structure.

While the staff believes GE's position on control rod insertion is reasonable, it has inadequate data to fully assess whether control rod insertion would occur given the large lateral displacements postulated above. Therefore, for this assessment, the staff has assumed at this time that the SLCS would be needed to shut down the reactor given a seismic event.

1.2 PROBABILISTIC PERSPECTIVE

The issue of concern is that displacement of the upper guide structure could cause fuel misalignments sufficient to prevent insertion of the control rods. Large main steam line breaks and a large seismic event have been postulated to produce enough force to create sufficient misalignment which may prevent insertion. However, there is no analysis that can be used to provide a conditional probability of failure to scram, given either of these events. Such analysis would have to address: 1) the length of time that the H3 weld would be in an unbroken condition, but sufficiently weakened that it would fail during the event; and 2) the probability that any resulting displacement would prevent control rod insertion. Without this analysis, only the sensitivity to this unknown parameter can be provided.

Main steam line breaks outside the MSIVs are considered capable of creating the guide structure displacements before MSIV closure. The frequency of occurrence for large steam line breaks (inside and outside containment) is estimated to be on the order of 10^{-4} to 10^{-3}/year, respectively. Seismic events with intensities in the range of 2 to 3 times beyond dusign basis-levels are estimated to have a frequency of occurrence on the order of 10^{-5}/year. Therefore, steam line breaks are expected to dominate the risk.

The NUREG-1150 model for Peach Bottom and the draft of a revised ASP model for Peach Bottom were reviewed to provide insights on the probability of mitigating a failure of control rod insertion. This probability is estimated to be on the order of 10^{-1}. Failure to achieve timely shutdown with the Standby Liquid Control System (SLCS) dominates this probability. Human errors, including failure to initiate SLCS in time and failure to restore SLCS after maintenance or test, are important contributors to this failure probability.

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Therefore, the core damage frequency, given the assumption that the H3 is sufficiently degraded to fail during the event and no rod insertion would occur, is on the order of 10^{-4} to 10^{-5}/year. This estimate of 10^{-4}-10^{-5}/year is a conditional probability (and believed to be conservative) in that it assumes the existence of a completely broken shroud that would be free to move in an accident occurring during a year of plant operation at power. These conservative estimates indicate that a precipitous shutdown action is not warranted. Also in recognition of the factors discussed earlier in this paper (e.g. detectability through power/flow mismatch) a more realistic frequency of core damage would likely be much lower. We plan to collect additional information to better assess the significance of this issue.

3.3 SUMMARY

For a 360° through weld IGSCC crack at the H3 weld location, the staff has performed an initial assessment which indicates that under certain low probability accident conditions; 1) the core configuration may not be maintained, 2) control rods may not be inserted, 3) some ECCS equipment may be damaged and unavailable for core cooling, and 4) severe core damage may occur. For these scenarios, the staff has preliminarily estimated that the probability of core damage is on the order of 10^{-4} to 10^{-5}/reactor year. Although not quantified, the staff also believes that the significant cracking required to result in a loss of structural integrity of the H3 weld would likely be observable during normal plant operation and would result in a plant shutdown.

4. FUTURE ACTIONS

It should be noted that all the above information was obtained from conference calls with GE and the licensee. The staff plans to follow up by reviewing GE's and the licensee's final assessment. Therefore, the views presented above are preliminary in nature. It also should be noted the staff does not traditionally analyze these types of accidents.

General Electric (GE) has issued RICSIL No. 054 advising BWRs of the observed core shroud cracking. This RICSIL is being revised to include the Brunswick findings. Further, GE is preparing a more detailed SIL on the event which will recommend inspection of the H3 weld at the next refueling for all plants. Also, INPO has issued a Significant Evant Notification (SEN) describing the Brunswick shroud cracks.

As a result of the evaluations performed to date, the staff is considering the need for further regulatory action. The staff is considering the need for a generic letter or bulletin which will request all BWR licensees to perform this inspection at the next refueling outage. The staff is also reviewing the current regulatory requirements relating to inservice vessel inspection to determine whether additional action is warranted to require

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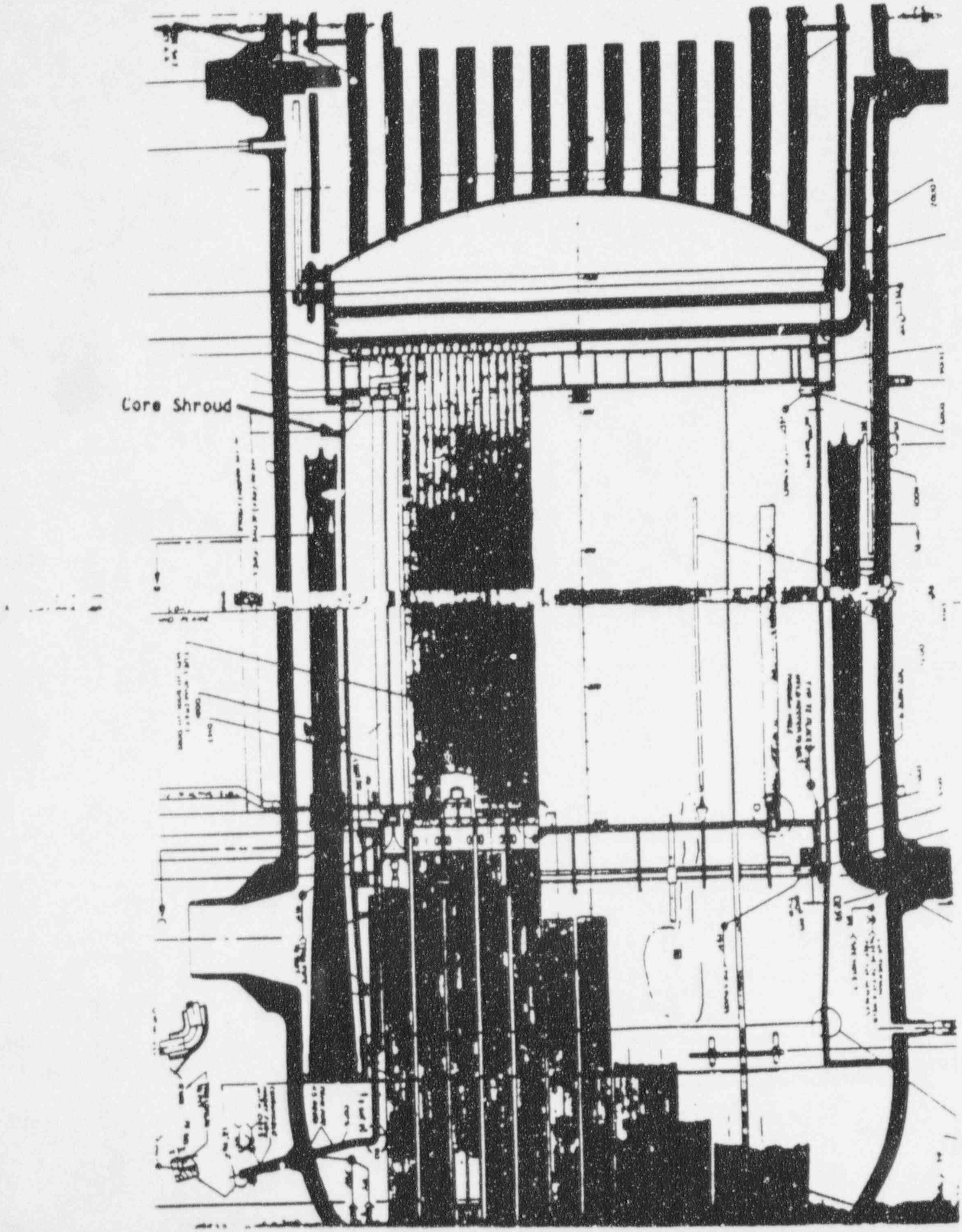
periodic inspection of the reactor vessel internals. It should be noted that any additional action beyond the issuance of the IN will be based upon a more thorough and systematic evaluation of the core shroud cracking concerns.

FIGURE 1
BWR REACTOR VESSEL



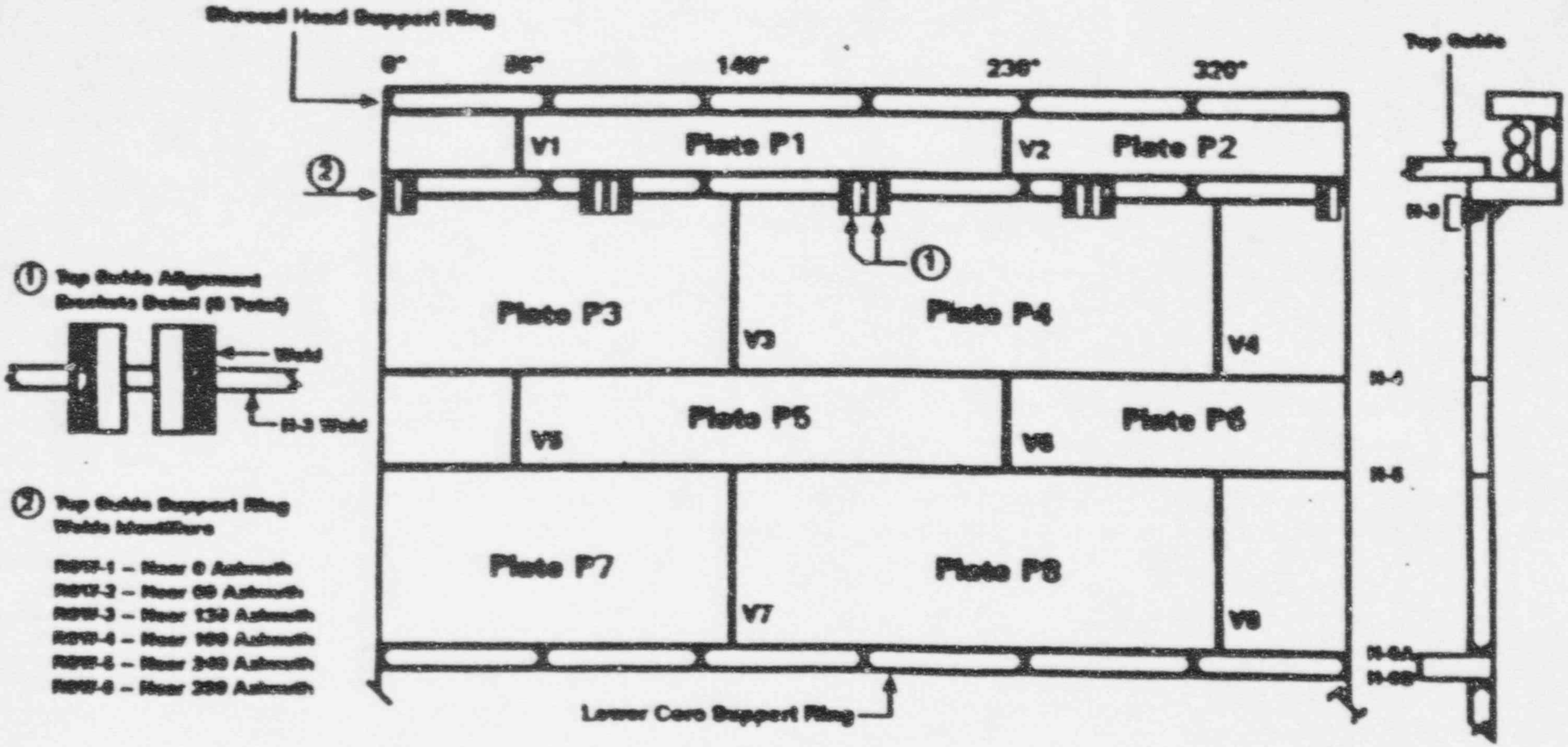


Figure 2. Weld and Plate Locations in the Beltline Region of the Brunswick Unit Core Shroud (From CP&L)

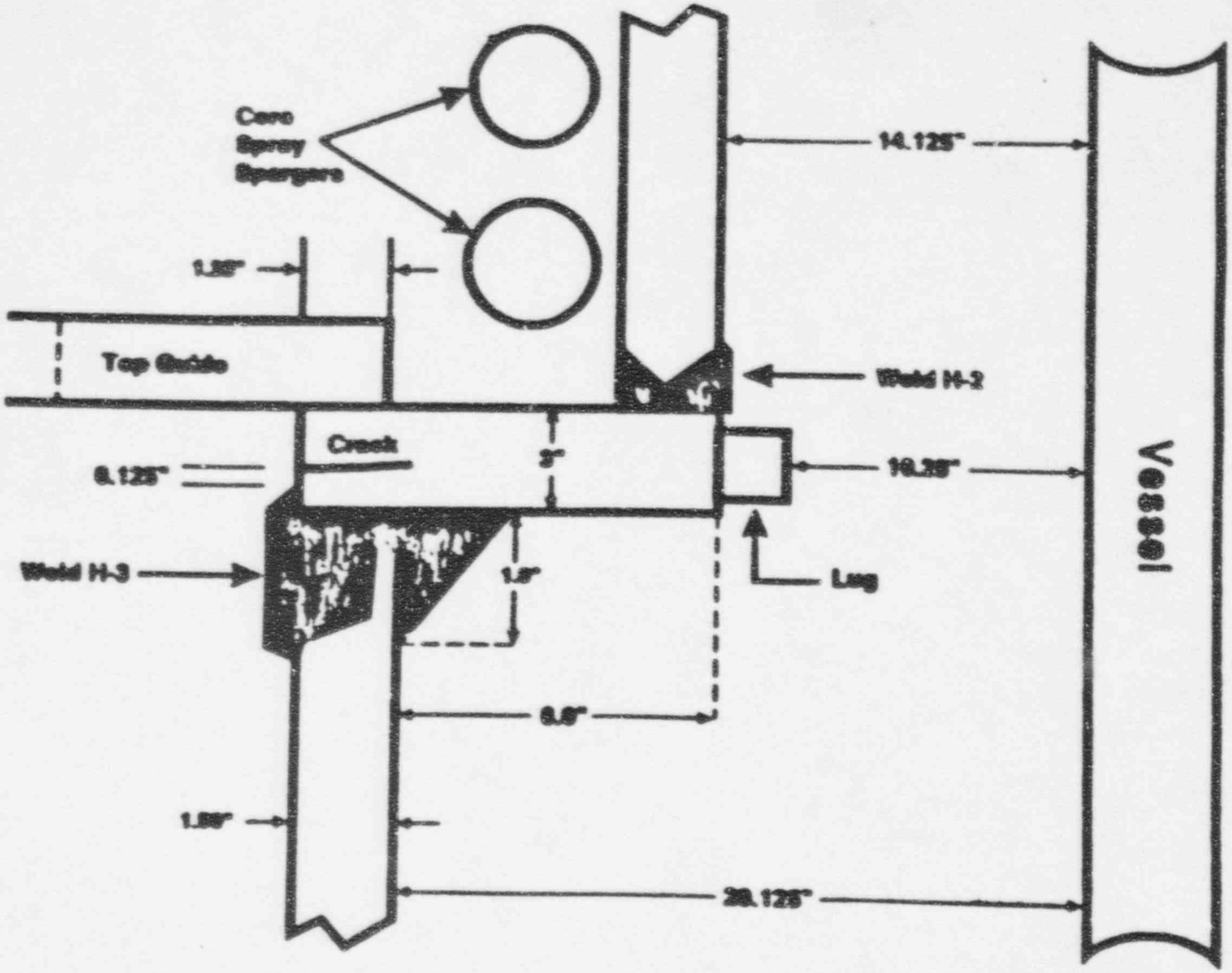


Figure 3. Details of Welds and Cracks at Weld Locations H-2 and H-3 in the Brunswick Unit 1 Core Shroud (From CP&L)