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ComEd

June 19, 1995

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Quad Cities Station Units 1 and 2
Core Shroud Repair Hardware Inadvertent Loading
NRC Docket Nos. 50-254 and 50-265

References: (1) R.M. Pulsifer to D.L Farrar letter dated June 8, 1995

(2) Teleconferences between USNRC (P. Hiland, R. Capra, et al) and ComEd (L.W. Pearce, et al) on June 14, 1995 and June 16, 1995

In the Reference (1) letter, the NRC Staff issued a Safety Evaluation regarding the Core Shroud Repair at Quad Cities Nuclear Station. During reassembly of the reactor vessel internals for Quad Cities Unit 2, Commonwealth Edison (ComEd) discovered that the Shroud Head/Moisture Separator support legs directly impinged upon the Core Shroud Repair at two separate bracket locations. ComEd immediately halted the reactor reassembly and initiated an evaluation of the inadvertent loading upon the Core Shroud repair hardware, and the options for resolution. This event, and the preliminary results of the evaluation, were discussed with the NRC staff during the referenced teleconferences. This letter transmits ComEd's revised 10 CFR 50.59 Safety Evaluation (and supporting calculations) for the Quad Cities Unit 2 Core Shroud Repair (Attachment). This revision addressed the affects of the inadvertent loading of the Core Shroud repair hardware. The revised portions of the 10 CFR 50.59 Safety Evaluation are marked with a vertical bar in the right hand margin

ComEd's evaluation of the inadvertent loading included a remote visual inspection of the core shroud repair hardware with underwater cameras, and an evaluation of the loads that were placed on the repair hardware and the shroud head support ring. The results of the remote visual inspection indicated that the Core Shroud repair hardware was intact and did not sustain any visible deflection or damage.

The revised 10 CFR 50.59 Safety Evaluation (including two separate supporting calculations) validated the results of these visual inspections. The first supporting calculation analyzed the "at-rest" condition, with the entire weight of the separator on two repair brackets. The second

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calculation assumed that the entire separator weight, plus impact force, impinged on one repair bracket. This second analysis resulted in a 7% increase over the stresses in previous ComEd submittals. However, with the extra conservatisms of this second analysis [i.e. use of a high impact factor (see attached ComEd Letter, SLE 95-005) and neglect of buoyancy effects of the water], the result of the calculation is acceptable, as it still provides adequate margin to allowables. As such, the design functions of the shroud repair hardware and shroud head support ring have not been altered from the previous assessments (i.e structural, systems, materials, and fabrication considerations) which were submitted by ComEd, and approved by the NRC staff in Reference (1).

To the best of my knowledge and belief, the analyses and evaluations contained in these documents are true and correct. In some respects these documents are not based on my personal knowledge, but on information furnished by other Commonwealth Edison employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

If there are any questions, please contact John L. Schrage at 708-663-7283.

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC. STATE OF ILLINOIS
MY COMMISSION EXPIRES: 11/29/97

Sincerely,

John L. Schrage
Nuclear Licensing Administrator

Mary Jo Yack 6-19-95

Attachment

cc: J .B. Martin, Regional Administrator - Region III
R. M. Pulsifer, Project Manger - NRR
C. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

ATTACHMENT

Revised 10 CFR 50.59 Safety Evaluation - Core Shroud Repair
Revision 4, June 16, 1995

GENE 771-111-0695, Revision 0
"Shroud Head Contact on Upper Support - Backup Calculations for FDDR #EE2-0505."

GENE 771-113-0695, Revision 0
"Shroud Head Contact on Upper Support - Backup Calculations with Impact Factor for FDDR #EE2-0505."

ComEd Letter, SLE 95-005, June 18, 1995
"Criteria Used to Determine Impact Factor"

ATTACHMENT B

10CFR50.59 SAFETY EVALUATION, REVISION 4

1. Procedure/test/change M04-2-94-007

Station / Unit Quad Cities / 2 Applicable Modes All

Other Relevant Plant Conditions NONE

System(s) affected 0201 Equipment #(s) ______

Equipment Name(s) Core Shroud Horizontal Welds H1 Through H7

2. a. Describe the proposed change.

BACKGROUND INFORMATION (see figure 1):

In 1990, crack indication were reported at core shroud welds located in the beltline region of an overseas reactor (BWR-4). This reactor had completed approximately 190 months of power operation before the cracks were discovered. As a result of this discovery, GE Nuclear Energy (GENE) issued Rapid Information Communication Services Information letter (RICSIL) 054, "Core Support Shroud Crack Indications," on October 3, 1990, to all owners of GE BWRS. This RICSIL summarized 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, on the inside and outside surfaces of the shroud.

During the 1993 refueling outage at Brunswick Unit 1 (BWR-4), in-vessel visual inspection revealed cracks at weld regions of the core shroud. Brunswick found both circumferential and axial cracks in the shroud, although cracking was predominantly circumferential. Circumferential cracks were located on the shroud inside surface in the heat-affected zone (HAZ) of weld H-3 and extended 360 degrees around the circumference of the shroud. Weld H3 is a horizontal weld that attaches the bottom of the Top Guide Support Ring (TGSR) to the top of the shroud cylinder below the ring. The H2 weld that joins the upper shroud cylinder to the top of the other side of the TGSR was also cracked extensively, although the cracking was more shallow. The first axial crack discovered was located on the outer shroud surface at weld H-4 (lower shroud cylinder). Brunswick performed additional visual testing (VT) and ultrasonic testing (UT) of the shroud and removed boat samples at welds H-2, H-3, and H-4 to evaluate the length and size of the cracks, and to validate ultrasonic sizing test procedures. GE 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.

SHROUD PROBLEM DESCRIPTION:

In-vessel inspections found linear indications in the horizontal core shroud welds at Dresden Unit 3 and Quad Cities Unit 1 during the spring 1994 outages. Visual examination and ultrasonic testing at weld H5 indicated the crack extended 360 degrees around the circumference of the shroud. Two boat samples were taken (Quad Cities; azimuths 154 and 342-size 3"x 2"x 1.5": Dresden; azimuths 153 and 324-size 3"x 2"x 1.35") to examine/analyze the root cause of the linear indications and compare measured crack depths in the samples to the depths determined by ultrasonic testing. Metallurgical evaluation determined intergranular stress corrosion cracking to be the root cause of the linear indications due to the application of the welded Type 304 stainless steel components in a strongly oxidizing aqueous environment.

The depth and length of the cracking has made repairs unavoidable at these plants. A conservative evaluation concluded that the cracked shrouds will satisfy ASME Code margins against weld failure for fifteen months of operation above cold shutdown. The NRC approved Quad Cities unit 1 and Dresden unit 3 for fifteen months of operation above cold shutdown on July 15, 1994.

It is anticipated that the two online units, Dresden Unit 2 and Quad Cities Unit 2 will have similar linear indications and will also need repair. The core shroud horizontal welds have a potential of failing through wall.

SHROUD PROBLEM SOLUTION (see figure 2):

The technical design requirement is that the repair design structurally replaces the core shroud horizontal welds HI through H7 if these welds fail completely through wall. In addition, for design purposes the circumferential jet pump support plate H8 weld is to be considered cracked completely through and 360 degrees. Also, the design should not result in a driving mechanism for Intergranular Stress corrosion Cracking (IGSCC) in these welds or any other component in the reactor vessel such that it reduces the operating margin available from the remaining ligaments of the welds.

The core shroud repair is designed to structurally replace the core shroud's horizontal welds H1 through H7 and provide vertical clamping forces on the shroud in the event that any or all the seven shroud horizontal weld joints are cracked through wall. In general the core shroud repair design installs low tension tie rods with spring stabilizers connected between the separator head support ring and the jet pump support plate. Four tie rods will be evenly distributed in the annulus region of the reactor pressure vessel. Spring stabilizers will be mounted at the top guide support ring (welds H2/H3) and the core plate support ring (welds H5/H6) in the annulus area between the core shroud and the reactor pressure vessel wall. A middle spring stabilizer is mounted on the tie rod at the same elevation as the jet pump riser braces.

The upper and lower springs transmit seismic loads from the nuclear core directly to the RPV via the core plate support ring and the top guide support ring. The function of the spring stabilizers is to provide lateral stability for the core shroud to ensure core geometry and refloodable volume are maintained. The spring stiffness in the stabilizers was optimized to provide the minimum possible adverse effect of the seismic loads to the reactor internals (i.e. maximum horizontal support for the fuel assemblies) while meeting the stress and displacement limits. The middle spring provides an intermediate lateral support to the tie rod and keeps the shroud from moving closer than 0.5-inches to the jet pump riser braces. The tie rod function is to provide rotational stability for the core shroud to ensure core geometry and refloodable volume are maintained. (Additional technical functions and design features of the shroud repair are discussed in item #5)

b. **Describe the reason for the change.**

Linear indications were found in the horizontal core shroud welds at Dresden unit 3 and Quad Cities unit 1 during the spring 1994 outages. At weld H5 the crack extended 360° around the circumference of the shroud. The depth and length of the cracking has made repairs unavoidable at these plants. It is anticipated that the two on line units, Dresden unit 2 and Quad Cities unit 2, will have similar linear indications and will also need repair. The core shroud horizontal welds have a potential of failing through wall. A decision was made that the best design approach was a comprehensive repair that included all the core shroud horizontal welds HI through H7. In addition, for design purposes the circumferential jet pump support plate H8 weld is to be considered cracked completely through its thickness and 360 degrees.

3. **Document Review**

List the SAR sections which describe the affected systems, structures, or components (SSCs) operations or activities. List any other controlling documents such as SERS, 10CFRs, Regulatory Guides, Fire Protection Report (FPR), Offsite Dose Calculation (ODCM), Core Operating Limits Report (COLR), previous modifications or Safety Evaluations, etc.

UFSAR
- 3.2 Classification of Structures, Components and Systems
- 3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping
  - 3.6.2 Postulated Piping Failures in Fluid Systems Inside Primary Containment
- 3.7 Seismic Design
- 3.9 Mechanical System and Components
- 4.0 Reactor
- 5.0 Reactor coolant and Connected Systems
- 6.0 Engineered Safety Features
  - 6.3 Emergency Core Cooling Systems
- 7.6 Core and Vessel Instrumentation
- 15.6 Decrease in Reactor Coolant Inventory

4. **Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs. The description should provide all relevant information necessary for a reviewer unfamiliar with the change, to understand plant operational impact without reference to other sources.**

Leakage flow to bypass the steam separators due to machining eight circular holes through the jet pump support plate, cracks in the seven horizontal circumferential welds H1 through H7, cracks in the circumferential weld in the jet pump support plate H8, leakage past the jet pump support plate access hole covers, leakage paths through the shroud head flange pockets/notches, and one of the pockects/notches with a hole cut through the back of the shroud head support ring (Unit 2 only) have been evaluated. The performance impact of the total bypass leakage flow for 100% rated power and 87 to 108% rated core flow is discussed below.

BYPASS LEAKAGE FLOW EVALUATION:

As discussed above, the installation of the shroud hardware will result in the potential for increased leakage through the jet pump support plate at the bolted connections. To assure a bounding estimate, the evaluation of core bypass flow leakage is based on the shroud repair hole leakage, the jet pump support plate access hole covers, leakage paths through the shroud head flange pockets/notches, one of the pockects/notches has a hole cut through the back of the shroud head support ring (Unit 2 only) and the flow calculated to occur simultaneously through one mil gaps in all the circumferential shroud welds including the jet pump support plate weld H8. The leakage flows are predicted based on loss coefficients and reactor internal pressure differences across the applicable shroud components. Leakage flows from the jet pump support plate repair holes, the weld cracks, leakage paths through the shroud head flange pockets/notches, one of the pockects/notches has a hole cut through the back of the shroud head support ring and the jet pump support plate access hole covers, for 100% rated power and 87 to 108% rated core flow [corresponding up to maximum increased core flow (ICF)] result in a total / combined leakage value of about 0.44% of total core flow. The steam portion of the leakage flows will contribute to increasing the total carry under from the steam separators. The impacts of the total leakage on the steam separation system performance, jet pump performance, core monitoring, fuel thermal margin, Emergency Core Cooling System (ECCS) performance and fuel cycle length are evaluated below;

- STEAM SEPARATION SYSTEM:
  The leakage flow above the top guide support ring includes steam flow, which effectively increases the total carryunder in the downcomer by a maximum of about 0.03% at 100% rated power and 87 to 108% rated core flow. The carryunder from the separators is based on the applicable separator test data at the lower limit of the operating water level range. The combined effective carry under from the separators and the shroud head leakage is bounded by the design value.

- JET PUMPS:
  The total carryunder meets the design condition carryunder value. Therefore, there is no impact on jet pump performance compared with the design condition.

- CORE MONITORING:
Measured "total core flow" (actually cumulative flow through the pumps) is an input to the core monitoring computer code's power distribution calculation. These are performed at least daily during steady-state operation above 25% power to demonstrate compliance with the core operating limits as required by Technical Specifications. The code adjusts (reduces) this measured total jet pump flow to account for flow that does not pass active fuel rods (i.e. Ex-channel and water rod flow). The ex-channel bypass flow does not account for the new potential leakage paths associated with the shroud. A conservative estimate on the impact from the various shroud leakage paths on these calculations is an indicated active core flow that is about 0.22% higher than actual. This is small compared to the core flow measurement uncertainty of 2.5% for jet pump plants used in the uncertainty analysis associated with the Minimum Critical Power Ratio (MCPR) Safety Limit. Additionally, the affect of having 0.22% lower core flow than indicated by the core monitoring code is only a 0.1% decrease in MCPR relative to that calculated during these surveillances. Because this small difference only affects operating margin (margin at steady-state compared to the MCPR operating limit), the margin of safety is not affected. The effect on other core surveillance parameters (LHGR and MAPLHGR) would be even smaller and also insignificant.

- FUEL THERMAL MARGIN EFFECT - ANTICIPATED ABNORMAL TRANSIENTS:
The code used to evaluate performance under anticipated abnormal transients and determine fuel thermal margin includes carryunder as one of the inputs. The effect of the increased carryunder due to leakage results in greater compressibility of the downcomer region and, hence, a reduced maximum vessel pressure. Since this is a favorable effect, the thermal limits are not impacted.

- EMERGENCY CORE COOLING SYSTEM (ECCS):
The leakage flow above the top guide support ring results in slightly increased carryunder that causes the initial core enthalpy to increase slightly, with a corresponding decrease in the core inlet subcooling. However, because the total downcomer carryunder still meets the design value, there is no impact on the ECCS performance from this condition. Another effect of the leakage flows from the repair holes and the weld cracks is to decrease the time to core uncovery slightly and, also to increase the time that the core is uncovered. The combined effect has been assessed to increase the Peak Clad Temperature (PCT) for the limiting LOCA event by less than 15 degrees F. The current analysis basis yields LOCA PCTs of approximately 1680 degrees F for the diesel generator failure case. Therefore substantial margin exists to the 10CFR50.46 acceptance criterion of 2200 degrees F. Because the maximum potential effect on the design basis LOCA PCT is very small, there is no adverse effect on the margin of safety. This impact is sufficiently small to be judged insignificant, and, hence, the licensing basis PCT for the normal condition with no shroud leakage is applicable. The sequence of events remains essentially unchanged for the LOCA events with the shroud leakage.

- FUEL CYCLE LENGTH:
  The increased carryunder due to shroud bracket-hole leakage results in a slight increase in the core inlet enthalpy, compared with the no leakage condition. The combined impact of the reduced core inlet subcooling and the reduced core flow due to leakage results in a minor effect (0.8 days) on fuel cycle length and is considered insignificant.

- REACTOR RECIRCULATION PUMPS:
  The total carryunder meets the design condition carryunder value. The increased carryunder due to shroud leakage results in a slight increase in enthalpy in reactor recirculation pumps inlet, compared with the no leakage condition. There is enough margin before cavitation occurs in the reactor recirculation pumps inlet to accommodate the increase in the enthalpy due the maximum possible leakage through the shroud. Hence, this slight increase in enthalpy on the reactor recirculation pumps inlet is considered insignificant and is bounded by the design conditions.

In addition, an evaluation is made below to evaluate the downcomer flow characteristics with the four stabilizers installed inside the annulus, to determine the impact of the additional flow blockage on the recirculation system loop hydraulic resistance, loop pressure drop, reactor coolant level, and the coolant flow rate, as well as any impact of the recirculation line break blowdown calculations, including ECCs performance.

DOWNCOMER FLOW EVALUATION:

The closest distance between the jet pump suction nozzle inlet (at elevation 317.6 inches, where jet pump suction flow enters the jet pump) and the 3.5-inch diameter stabilizer tie rod is over 6 inches. At this distance the predominately downward flow distribution near the jet pump nozzle will not be significantly affected.

The smallest vessel-to-shroud annulus plan flow area between the H1 and H2 weld is at the H1 weld. Although other locations have more shroud repair hardware, they have less flow restrictions from other items already connected to the shroud, such as shroud head bolts and lug sets, core spray piping and guide rod brackets. The end result is that these other locations have larger flow areas.

The four added upper stabilizer springs and their supports block less than 2% of the pre-repair minimum downcomer area. This blockage applies only to the vertical distance corresponding to the length of the upper stabilizer springs and their supports, located between welds H1 and H2. Locations with horizontal flow blockage from shroud stabilizer hardware at other elevations in the shroud-to-vessel annulus will have larger flow areas. The impact of the additional flow blockage on the recirculation system loop hydraulic resistance, loop pressure drop, reactor coolant level, and the coolant flow rate is determined to be negligible.

During a recirculation suction line break there may be a significant horizontal component of flow in the lower vessel annulus. The four lower stabilizer springs are each located between jet pumps 45 degree away from the recirculation outlet nozzle. The net vertical flow area at the lower stabilizer springs will have an insignificant effect on recirculation line break blowdown calculations. Hence, ECCS performance is not impacted as a result of the flow blockage associated with the stabilizer mechanisms.

5. **Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.**

This change will not adversely affect equipment failures nor will it create any new failure modes. The core shroud repair system's only function is to reinforce the shroud in the event that any or all of the shroud horizontal weld joints are cracked through wall. The upper and lower springs transmit seismic loads from the nuclear core directly to the RPV via the core plate support ring and the top guide support ring. The spring stiffness in the stabilizers was optimized to provide the minimum possible adverse effect on the seismic loads to the reactor internals (i.e. maximum horizontal support for the fuel assemblies) while meeting stress and displacements limits. The tie rod function is to provide rotational stability for the core shroud to ensure that core geometry and refloodable volume are maintained. In addition, the tie rods will structurally replace the core shroud horizontal welds H1 through H7 and provide vertical clamping forces on the shroud.

The natural vibration frequency of the tie rod with the intermediate lateral support is well removed from the flow-induced forcing frequency. The shroud stress analysis demonstrates that the core shroud and the shroud repair assembly structural integrity are maintained if any or all of the seven horizontal (H1-H7) welded joints and / or circumferential jet pump support plate (H8) weld joints are cracked completely through their thickness and completely around their entire 360 degree circumference. The structural integrity of the shroud and the shroud repair assembly is also demonstrated in the event that the shroud is uncracked but the repair assembly is installed.

An Evaluation on the seismic loads on the RPV has been performed with the shroud repair hardware in place. All stress intensities due to the new design mechanical loads satisfy the allowable stress intensities of the original code of construction.

The effect of the design repair hardware weight added in the annulus region of the RPV was considered in the evaluations and found to be acceptable. The tie rods assembly dead loads (weight) are transmitted to the jet pump support plate which is connected to the RPV. These loads are transmitted to the rigid foundation via the RPV to the RPV skirt ring to the anchor bolts and high strength bolts down to the RPV pedestal. The repair hardware dead loads are considered to be insignificant.

The seismic analyses were based on the time history method of analysis. The input motions included the 1957 Golden Gate earthquake record and a synthetic time history matching the Housner spectrum curve which is the licensing commitment in the UFSAR, section 3.7.1. The major forces include dead load, buoyant forces, horizontal and vertical seismic, mainsteam LOCA, reactor recirculation LOCA (including blowdown and acoustic) and fluid mass. The forces were combined using the appropriate load combinations from the UFSAR, section 3.9. Also considered was the combination of seismic load concurrent with each LOCA. Analyses were done for the complete range of postulated shroud welded joint cracks as well as for the fully uncracked configuration with the shroud restraint hardware installed. Bounding Design Basis Earth quake (DBE) loads were obtained for use in load combinations for the Emergency and Faulted conditions, and bounding Operating Basis Earthquake (OBE) loads for the Upset condition. The resulting seismic loads were used as input to the design of the shroud repair hardware and to validate the continued structural integrity of the core support structure and the RPV internals.

The seismic analysis on the RPV externals with the shroud repair hardware installed indicate load increases on the RPV lateral support system such as the RPV stabilizer rods, shield wall top ring plate, shield wall to containment wall star truss, RPV skirt ring girder, anchor bolts, high strength bolts and the RPV pedestal. These components with the load increases have been reanalyzed. The results show these components are capable of withstanding the increased loads and all stresses are within allowable limits.

The seismic analysis of the external piping connected to the RPV, such as recirculation piping, core spray piping, mainsteam piping, and feedwater piping, with the shroud repair hardware installed have been evaluated and found acceptable.

The effect of the shroud repair hardware on the RPV internal piping, such as the core spray piping and the feedwater sparger piping, have been evaluated and found acceptable.

A seismic analysis of the jet pumps movement was performed. The evaluation shows the jet pumps movement is less than 0.005-inches. Flow induced vibration movement is less than 0.010-inches. The total movement of the jet pumps will be less than 0.015-inches. There is a 1.5-inch clearance between the shroud repair hardware and the jet pumps. The shroud repair hardware will not come in contact with the jet pumps and will not interfere with jet pump operation.

An evaluation of the seismic loads on the reactor fuel has been performed with the core shroud repair hardware in place. The fuel load is below allowable loading and has been found acceptable.

The effect of the shroud repair hardware on displaced core cooling water was evaluated and considered insignificant. The small water loss will not adversely affect the ECCS as described in the UFSAR or any accident as described in the UFSAR.

An evaluation of the core shroud repair's design reliant structures was performed. The integrity of the design reliant structures will be verified by inspection.

Machining and grinding processes have been controlled to reduce the amount of cold work induced on the shroud repair hardware. Components that are not solution annealed after final reduction, sizing and straightening operations shall have metallographic and microhardness evaluations on test samples. The test samples shall be provided from the same material, same fabrication shop, and use the same process variables.

Other major technical functions and design features of the shroud repair are:

- The tie rod with stabilizer assemblies are designed and fabricated as safety related - seismic class 1 components.
- The repair design will not noticeably increase the tensile stresses at any of the core shroud horizontal welds H1 through H7 or the jet pump support plate welds H8 or H9.
- The repair design will not noticeably increase the tensile stresses at any of the core shroud vertical welds.
- Thermal loading effects of the design repair on the core shroud welds and other reactor vessel components are minimal.
- Flow induced vibration (FIV) effects and acoustic vibration effects after the repair hardware is installed will be minimal.
- The material used in the design repair is IGSCC and IASCC resistant.
- The repair design is removable to allow for future in-service inspections (ISI) or in-vessel visual inspection (IVVI) or other maintenance activities.
- The repair design may however, interfere with other outage activities such as installation of the recirculation line plugs, removal of the jet pumps where the shroud hardware is installed or installation of the jet pump plugs where the shroud hardware is installed .
- The repair design has no welded components.
- The design will allow for installation/removal of the core spray elbow clamps without interference from the installed shroud repair hardware, if they are required.

The core shroud repair has been developed in accordance with ASME section XI repair and replacement program requirements. The design accounts for through wall 360 degree circumferential cracks at the H1 through H8 welds. This repair does not remove the existing flaws nor replace the flawed components, but rather structurally replaces the function of the shroud horizontal circumferential welds H1 through H7 and accounts for through wall cracking of the jet pump support plate H8 weld. Thus the repair will be performed as an alternative to ASME section XI code as permitted by 10CFR 50.55a(a)(3). Use of an alternative to the code requires review and approval of this repair by the NRC.

During the installation of the shroud repair hardware, notches must be cut into the shroud head support ring using the electric discharge machining (EDM) process. On Unit 2 one of the notches was cut to deep, and the cut went through the back of the shroud head support ring. At the right angle notch at azimuth 290 a 2" X 1.5" hole completely through separator head support ring exists. The notch in the shroud head support ring was supposed to be EDM'ed into the shroud head support ring leaving 1/2 inch of the shroud head support ring material at the back of the notch. The effect of this deviation was evaluated and found to be acceptable.

During the installation of the shroud head/separator an interference between the shroud head/separator support legs and the shroud repair hardware occurred at two locations. At azimuths 103 and 283, the upper support-long of the core shroud repair hardware was contacted by the lower portion of the shroud head/separator support legs. The lower portion of the shroud head/separator support legs extend 12 inches below the shroud head. The upper support-long part of the shroud repair hardware extends approximately 2.2 inches above the shroud head support ring.

An evaluation of the loads that were placed on the shroud repair hardware and the shroud head support ring during the installation of the shroud head/separator was performed. All stresses are within allowable limits. Hence, the shroud repair hardware and shroud head support ring design functions have not been altered from those used in the original assessments.

6. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding. A list is found in the station specific attachment) described in the SAR where any of the following is true:

- The change alters the initial conditions used in the SAR analysis
- The changed SSC is explicitly or implicitly assumed to function during or after the accident
- Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
Decrease in Reactor Coolant Inventory (LOCA)	15.6

7. **To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident where the answers differ between each accident scenario listed in Step 6. PROVIDE an explanation for all NO answers.**

Affected accident LOCA **SAR Section:** 15.6

a. **May the probability of the accident be increased?** [ ] Yes [X] No

The probability of an accident will not be increased, because the affected plant systems and components will be capable of performing their intended design functions with the shroud repair hardware installed. This modification will structurally replace the core shroud horizontal welds H1 through H7. Since these welds have or are anticipated to show signs of degradation, this repair will ensure that structure integrity of the core shroud is maintained. The core shroud repair has no moving parts and is passive by design. In addition, the core shroud design repair meets the plant's safety-related design requirements. Therefore, the probability of a component failure is not increased.

b. **May the consequences of the accident (off-site dose) be increased?** [ ] Yes [X] No

The core shroud provides a barrier to separate the upward flow of coolant through the core from the downward flow of coolant in the annulus between the outer surface of the shroud and the reactor pressure vessel wall. It also maintains core fuel geometry and provides a floodable volume inside the Reactor Pressure Vessel (RPV), which is necessary in the event of a Loss Of Coolant Accident (LOCA).

All structures , systems and components (SSC) used to mitigate the (radiological) consequences of the accidents in the UFSAR are independent of the stabilizers, and thus, the consequences of accident will not be affected. The abnormal events in the UFSAR that potentially could be affected by the installation of the stabilizers were evaluated, and they remain unchanged.

The stabilizers impose a negligible change to the plant operating conditions, and thus, the ECCS-LOCA and transient analysis remain valid, as discussed in item #4.

LOCA-Radiological analysis is based on the plant's Engineered Safety Features (ESF) functioning within design parameters, and the radioactive material source terms. The stabilizers will not adversely affect any ESF as discussed in items 4 and 5, and thus, the ESF functions will not be affected. The radioactive material source terms are based on the equilibrium core fuel inventory. This modification is outside the core fuel inventory and will not create any new modified release points. The result of the source terms will not be affected or change. Therefore, the consequences of the LOCA-Radiological analysis will not change.

The MSLB analysis release is limited by the capacity of the MSL flow restrictors, and based on Technical Specification allowables for source terms. As the installation of the stabilizers will not affect either, the consequences of the MSLB analysis will not change.

As described in item #5, the seismic analysis shows that the stabilizers will remain functional following an earthquake.

c. **May the probability of a malfunction of equipment important to safety increase?** [ ] Yes [X] No

This modification will structurally replace the core shroud horizontal welds H1 through H7. Since these welds have or are anticipated to show signs of degradation, this repair will ensure the structure integrity of the core shroud is maintained. The shroud is required to provide a two-thirds core height reflooding volume following a LOCA. During normal operation, the shroud provides a barrier to direct core flow. The repair hardware is:

- designed and fabricated as safety related, seismic class 1;
- designed to remain in position under all normal and accident conditions;
- designed for differential pressure loads from 108% increased core flow conditions.

Stress calculations were performed in accordance with the ASME section III subsection NG to assure reliability and adequate margins of safety in the design. Hence, The shroud repair hardware will not impair the function but ensures that the structural integrity of the core shroud is maintained.

d. **May the consequences of a malfunction of equipment important to safety increase?** [ ] Yes [X] No

The installation of stabilizers ensures that the shroud, even if cracked, will perform its safety functions. The function of the spring stabilizers is to provide lateral stability for the core shroud to ensure core fuel geometry and refloodable volume are maintained. The spring stiffness in the stabilizers was optimized to provide the minimum possible adverse effect of the seismic loads to the reactor internals (i.e. maximum horizontal support for the fuel assemblies) while meeting the stress and displacement limits. The middle spring provides an intermediate lateral support to the tie rod and keeps the shroud from moving closer than 0.5-inches to the jet pump riser braces. The tie rod function is to provide rotational stability for the core shroud to ensure core geometry and refloodable volume are maintained. Thus, consequences of a malfunction of equipment important to safety is not increased. The stabilizers perform a passive function that does not interface with any equipment that is used to mitigate the radiological consequences of a malfunction in the UFSAR as noted in items #4 and #5. The effects of the stabilizers on the consequences of potentially affected transients are negligible. Therefore, there is no increase to the consequences of component malfunction.

11. Determine if parameters used to establish the Technical Specification limits are changed or affected. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 10. List the Technical Specification Technical Specification Bases, SER and SAR sections reviewed for this evaluation.

Technical Specification 1.1 Fuel Cladding - Safety Limit Basis
SER Section 4.4 Reactor - Thermal and Hydraulic Design

Determine which of the following is true for the above specifications:

[X] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 12.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to Question 12.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SERs, or other appropriate references. List the limit(s)/margin(s) below.

[ ] The change does not affect any parameters upon which Technical Specifications are based, therefore, there is no reduction in the margin of safety - NA Question 12 and proceed to Question 14.

12. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

Leakage flow to bypass the steam separators due to machining eight circular holes through the jet pump support plate, cracks in the seven horizontal circumferential welds H1 through H7, cracks in the circumferential weld in the jet pump support plate H8, leakage past the jet pump support plate access hole covers, leakage paths through the shroud head flange pockets/notches, and one of the pockects/notches with a hole cut through the back of the shroud head support ring (Unit 2 only) have been evaluated. To assure a bounding estimate, the evaluation of bypass flow leakage is conservatively assumed that each of the shroud welds develops a complete circumferential crack gap of one mil. These leakage flows are based on applicable loss coefficients and reactor internal pressure differences across the applicable shroud components. The performance impact of the total bypass leakage flow for 100% rated power and 87 to 108% rated core flow is discussed below:

CORE MONITORING:
Measured "total core flow" (actually cumulative flow through the pumps) is an input to the core monitoring computer code's power distribution calculation. These are performed at least daily during steady-state operation above 25% power to demonstrate compliance with the core operating limits as required by Technical Specifications. The code adjusts (reduces) this measured total jet pump flow to account for flow that does not pass active fuel rods (i.e. Ex-channel and water rod flow). The ex-channel bypass flow does not account for the new potential leakage paths associated with the shroud. A conservative estimate on the impact from the various shroud leakage paths on these calculations is an indicated active core flow that is about 0.22% higher than actual. This is small compared to the core flow measurement uncertainty of 2.5% for jet pump plants (Reference 1) used in the uncertainty analysis associated with the Minimum Critical Power Ratio (MCPR) Safety Limit. Additionally, the affect of having 0.22% lower core flow than indicated by the core monitoring code is only a 0.1% decrease in MCPR relative to that calculated during these surveillances. Because this small difference only affects operating margin (margin at steady-state compared to the MCPR operating limit), the margin of safety is not affected. The effect on other core surveillance parameters (LHGR and MAPLHGR) would be even smaller and also insignificant.

FUEL THERMAL MARGIN EFFECT - ANTICIPATED ABNORMAL TRANSIENTS:
The code used to evaluate performance under anticipated abnormal transients and determine fuel thermal margin includes carryunder as one of the inputs. The effect of the increased carryunder due to leakage results in greater compressibility of the downcomer region and, hence, a reduced maximum vessel pressure. Since this is a favorable effect, the thermal limits are not impacted.

EMERGENCY CORE COOLING SYSTEM (ECCS):
The leakage flow above the top guide support ring results in slightly increased carryunder that causes the initial core enthalpy to increase slightly, with a corresponding decrease in the core inlet subcooling. However, because the total downcomer carryunder still meets the design value, there is no impact on the ECCS performance from this condition. Another effect of the leakage flows from the repair holes and the weld cracks is to decrease the time to core uncovery slightly and, also to increase the time that the core is uncovered. The combined effect has been assessed to increase the Peak Clad Temperature (PCT) for the limiting LOCA event (Reference 2) by less than 15 degrees F. The current analysis basis yields LOCA PCTs of approximately 1680 degrees F for the diesel generator failure case. Therefore substantial margin exists to the 10CFR50.46 acceptance criterion of 2200 degrees F. Because the maximum potential effect on the design basis LOCA PCT is very small, there is no adverse effect on the margin of safety. This impact is sufficiently small to be judged insignificant, and , hence, the licensing basis PCT for the normal condition with no shroud leakage is applicable. The sequence of events remains essentially unchanged for the LOCA events with the shroud head leakage.

13. **Is a revision to the SAR or Technical Specifications needed?**

[x] YES - The SAR is to be updated to reflect this repair

[ ] NO

14. Check one of the following:

[X] No Unreviewed Safety Question will result (Steps 7, 8, 12) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] An Unreviewed Safety Question was identified in Step 7, Step 8, or Step 12. The proposed change MUST NOT be implemented without NRC approval.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a design change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

15.

Preparer [signature] 15 JUN 95
Signature Date

16. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion. Ensure an updated copy is sent to Reg. Assurance.

Reviewer [signature] 6/16/95
Signature Date

REFERENCES

Quad Cities Unites 1 and 2 10CFR50.59 Safety Evaluation for modification M04-02-94-007

1. General Electric Standard Application for Reactor Fuel, General Electric Company, (NEDE-24011-P-A, current amendment).

2. T.C. Hoang, et al., . Quad Cities Nuclear Power Station Units 1 & 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, General Electric Company, July 1989 (NEDG-31345P Revision 2).

3. GENE-771-68-1094, Rev. 0; " Quad Cities Units 1 & 2 Shroud Stress Analysis Report'

4. GENE # 25A5672, Rev.1; " Quad Cities Units 1 & 2 Pressure Vessel Stress Report"

5. GENE-771-70-1094, Rev.0; " Quad Cities Units 1 & 2 Seismic Analysis Report"

6. GE-NE-523-203-1294, Rev. 0; " Quad Cities Units 1 & 2 Top Ring Plate and Star Truss Stress Analysis"

7. GENE-771-89-0195, Rev. 0; "Quad Cities Units 1 & 2 RPV Skirt Ring Girder Stress Analysis"

8. GENE Letter, From M.D. Potter to Kenneth Hutko; Subject - "Effect of increased enthalpy due to shroud leakage on the RPV Recirculation Pump inlet for Quad Cities Units 1 & 2", Dated January 19, 1195

9. GENE Letter, From M.D. Potter to Kenneth Hutko; Subject - " Evaluation of the Reactor Internals Piping due to the Reactor Shroud Repair for Quad Cities 1 and 2", Dated January 6, 1995

10. GENE Letter, From M.D. Potter to Kenneth Hutko; Subject -" Evaluation of the GENE Fuel with the Shroud Repair Hardware Installed for Quad Cities 1 & 2, Dated January 6, 1995

11. GENE Letter, From M.D. Potter to Kenneth Hutko; Subject - "Clearance Between the Quad Cities 1 & 2 Shroud Repair Hardware and the Jet Pumps", Dated January 6, 1995

12. GENE Letter, From M.D. Potter to Kenneth Hutko; Subject - "Core Shroud Hardware Water Displacement in the RPV for Quad Cities Units 1 & 2", Dated January 18, 1995

13. Sargent & Lundy Letter - External Design Information Transmittal - ComEd Quad Cities Station Units 1 & 2, From V. K. Verma to M. D. Potter - GENE Project Engineer, Subject - "Core Shroud Modification Project: RPV External Piping", Dated January 11, 1995

14. GENE Letter, From S. Wolf to MD Potter, Subject-"Performance Impact of Shroud Repair Leakage for QC Units 1 and 2, Dated January 4, 1995

15. GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Effect of the Shroud Stabilizer Hardware on the RPV Annulus Region Flow Characteristics for Quad Cities Units 1 and 2, Dated March 13, 1995 (B13-01740, MDP-9514)

16. GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Evaluation of the Reactor RPV Ring Girder, Anchor Bolts, and High Strength Bolts Due to the Reactor Shroud Repair for Quad Cities 1 and 2, Dated January 11, 1995 (B13-01740, MDP-9504)

17. GENE-771-110-0595, Revision 0, "Evaluation of the Acceptability of FDDR No. 1E6AR-FDDR-001 for the Shroud Repair Program at Quad Cities Unit 2, (DRF B13-01740)

18. GENE 771-111-0695, Rev. 0; Shroud Head Contact on Upper Support - Backup calculations for FDDR #EE2-0505

19. GENE 771-113-0695, Rev. 0; Shroud Head Contact on Upper Support - Backup calculations with Impact factor for FDDR #EE2-0505

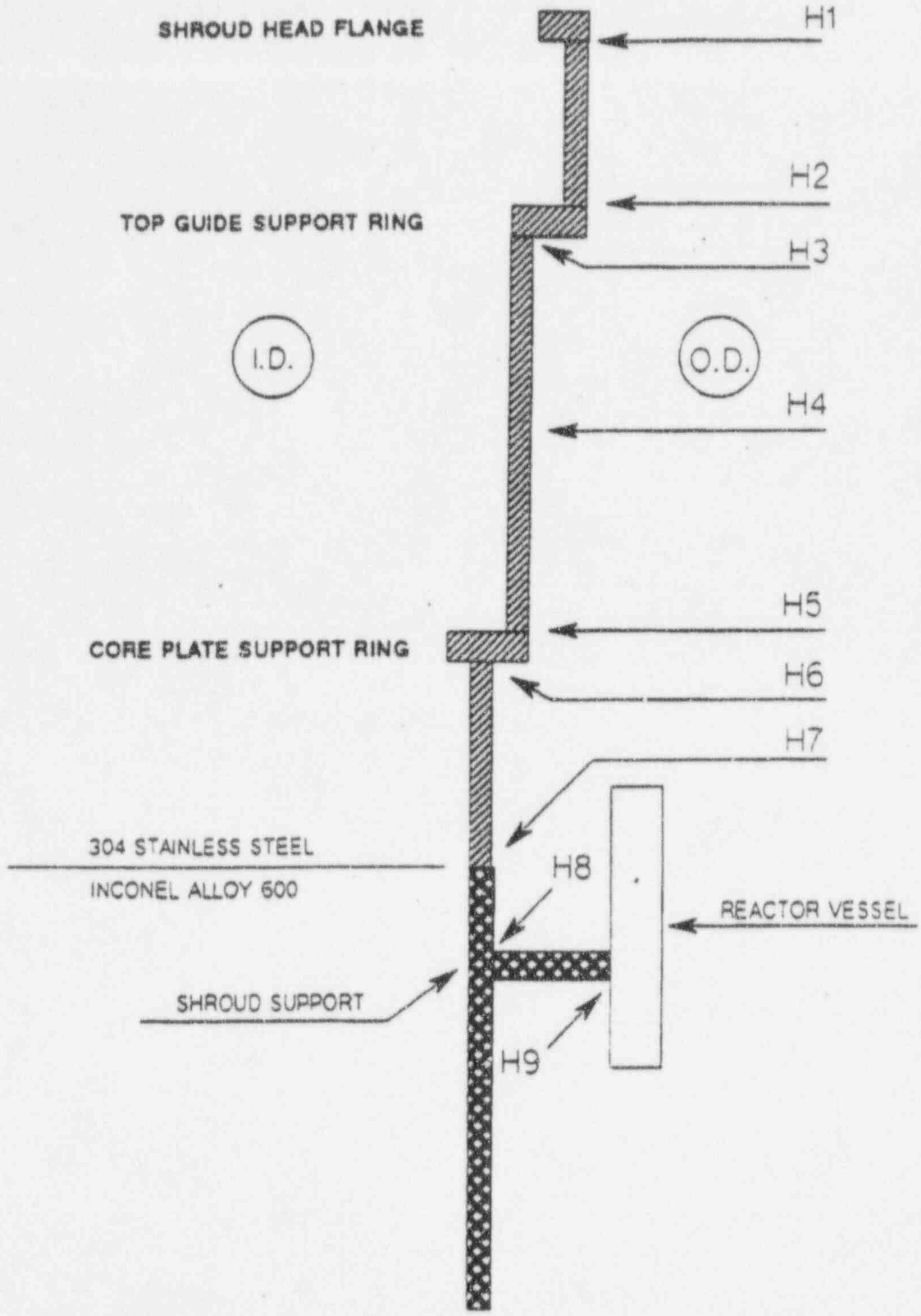

FIGURE 1. HORIZONTAL WELD LOCATIONS

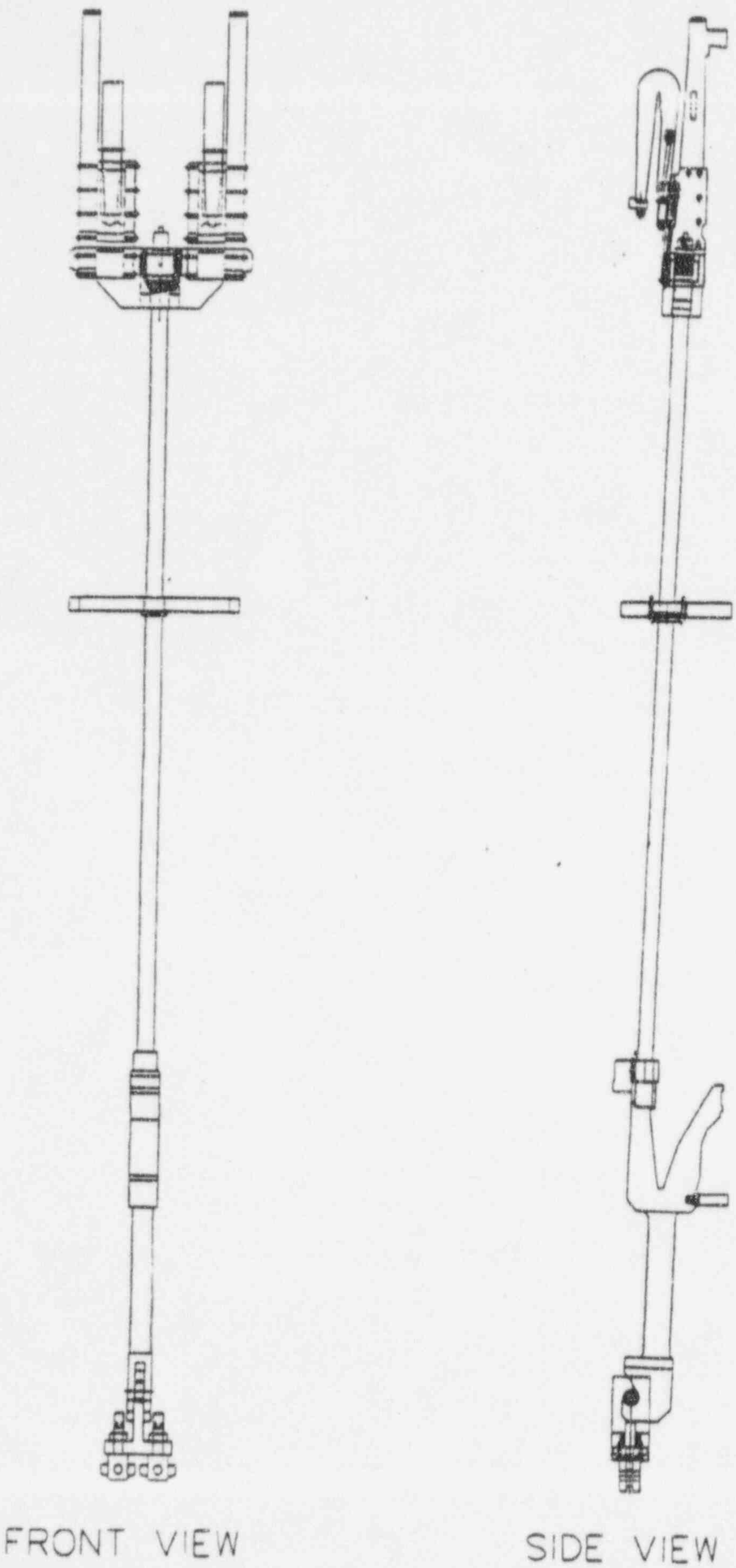

FIGURE 2. SHROUD STABALIZER