Enclosure 17

# 10 CFR 50.59 Safety Evaluation for Dresden Units 2 and 3 Core Shroud Repair

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ATTACHMENT B

**10CFR50.59 SAFETY EVALUATION, REVISION 0**

1. **Procedure/test/change** M12-2(3)-94-004;

**Station / Unit** Dresden / 2 & 3 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 in surface in the heat-affected zone (HAZ) of weld H3 and extended 3 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 H4 (lower shroud cylinder). Brunswick performed additional visual testing (VT) and ultrasonic testing (UT) of the shroud and removed boat samples at welds H2, H3, and H4 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 from the shroud structures at each of the two plant. The samples taken at Quad Cities Unit 1 were at azimuths 154 and 342 The size of each sample was 3"x 2"x 1.5". The samples taken at Dresden Unit 3 were at azimuths 153 and 324. The size of each boat sample was 3"x 2"x 1.35". The purpose of the bout samples was 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 other units at each station, 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 core plate wedge assemblies will be installed between the core plate and the core shroud.

The function of the core plate wedge assemblies is to transmit seismic loads from the core plate to the core shroud. 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 other unite at each station, 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 H1 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 paths through the shroud head flange pockets/notches, and from leakage past the jet pump support plate access hole covers have been evaluated. The performance impact of the total bypass leakage flow for 100% rated power and 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 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 through the shroud head flange pockets/notches, and the jet pump support plate access hole covers, for 100% rated power and 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, reactor recirculation 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 through welds cracks H1 and H2 occurs above the top guide support ring includes steam flow, which effectively increases the total carryunder in the downcomer by about 0.03% at rated conditions. 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.

- 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.

- 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 0.21% 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 0.21% 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 30 degrees F. The current analysis basis yields LOCA PCTs of approximately 2045 degrees F for the design basis LOCA with LPCI injection failure case. Therefore 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.

An evaluation was performed to determine the downcomer flow characteristics inside the annulus region of the RPV with the four stabilizers installed . 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 is discussed below:

- The closest distance between the jet pump suction nozzle inlet (at elevation 317.6 inches from vessel zero, 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 function of the core plate wedge assemblies is to transmit seismic loads from the core plate to the core shroud. 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 motion was the north-south component of the 1940 El Centro earthquake record which is the licensing commitment in the UFSAR, section 3.7.1. For the case where are welds were postulated as cracked, a synthetic time history matching the Housner spectrum curve was also used, to assure conservatism where low frequencies may exist. 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 Safe Shutdown Basis Earthquake (SSE) 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 2.0-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 GE 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 processes have been controlled to reduce the amount of cold work induced on the shroud repair hardware. Machined components that are not solution annealed after final machining shall have metallographic and microhardness evaluations on test samples. Samples shall be provided from the same material, same fabrication shop, and use the same process variables as the components which are being fabricated for the repair. In addition, all austenitic 300 series stainless steel shall have sensitization testing performed for each heat and heat treat lot.

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 materials used in the design repair are IGSCC and IASCC resistant.
- The material lots used in the design repair will be solution annealed after final reduction, sizing and straightening operations.
- 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 repair design does not permit grinding on the shroud repair hardware.
- 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.

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 resultant from 100% 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 [ ] Yes [X] No **important to safety increase?**

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.

8. **Based on your answers to Questions 4 and 5, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?**

[ ] Yes [X] No

The seismic analyses were based on the time history method of analysis. The input motion was the north-south component of the 1940 El Centro earthquake record which is the licensing commitment in the UFSAR, section 3.7.1. For the case where are welds were postulated as cracked, a synthetic time history matching the Housner spectrum curve was also used, to assure conservatism where low frequencies may exist. 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 Safe Shutdown Earthquake (SSE) 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 validated the continued structural integrity of the core support structure and the RPV internals.

All the loads and load combinations that are relevant to the core shroud, have been evaluated and are within design allowables with the core shroud hardware in place. The stabilizers do not add any new operational/failure mode or create any new challenge to safety-related equipment or other equipment whose failure could cause a new type of accident. In addition, the stabilizers do not create any new component/system interactions or sequence of events that lead to a new type of accident.

9. **To determine the factors affecting the specification, it is necessary to review the SAR and SER where the Bases Section of the Technical Specifications does not explicitly state the bases. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillance, or bases may be affected.**

No Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) is affected by this modification.

10. **Will the change involve a Technical Specification revision?**

**[ ] Yes [X] No**

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 paths through the shroud head flange pockets/notches, and leakage past the jet pump support plate access hole covers 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 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 0.21% 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 0.21% 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 30 degrees F. The current analysis basis yields LOCA PCTs of approximately 2045 degrees F for the design basis LOCA with LPCI injection 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 ______________ 22 MAY 95
Signature Date

Ronald R Skoglund 5/23/95

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 ______________ 5/23/95
Signature Date

## REFERENCES

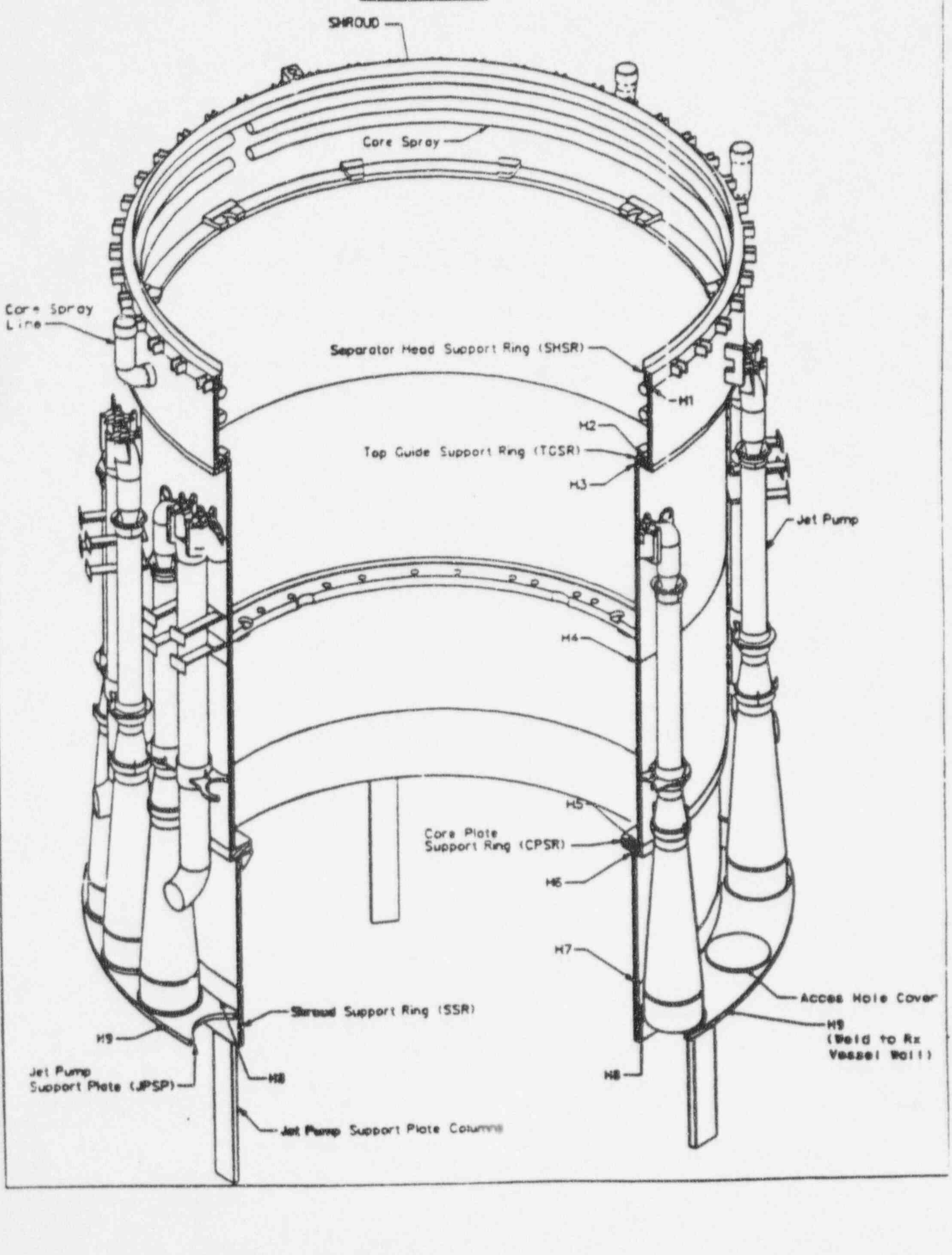
**Dresden Units 2 and 3 10CFR50.59 Safety Evaluation for modification M04-02-94-007**

1. General Electric Standard Application for Reactor Fuel, General Electric Company, (NEDE-24011-P-A, current amendment).
2. GENE Fabrication Specification, 25A5690, Revision 2, "Dresden 2 and 3 - Fabrication of Shroud Stabilizer".
3. GENE 771-81-1194, Revision 1, "Commonwealth Edison Company Dresden Nuclear Power Plant Units 2 & 3, Shroud and Shroud Repair Hardware Analysis, Volume I, Shroud Repair Hardware".
4. GENE 771-81-1194, Revision 1, "Commonwealth Edison Company Dresden Nuclear Power Plant Units 2 & 3, Shroud and Shroud Repair Hardware Analysis, Volume II, Shroud".
5. GENE 771-82-1194, Revision 1, "Backup Calculation for Dresden Shroud repair Shroud Stress Report for Dresdeb Nuclear Power Station Units 2 & 3".
6. GENE-771-83-1194, Revision 1, "Commonwealth Edison Company Dresden Nuclear Power Plant Units 2 & 3, Shroud and Shroud Repair Hardware Analysis, Shroud Repair Hardware Backup Calculation". (Proprietary information)
7. GENE-771-84-1194, Revision 2, "Dresden Units 2 & 3, Shroud Repair Seismic Analysis". (Proprietary information)
8. GENE 771-85-1194, Revision 2, "Dresden Units 2 & 3, Shroud Repair Seismic Analysis Backup Calculations". (Proprietary information)
9. GENE Stress Report, 25A5691, Revision 2, "Pressure Vessel - Dresden Units 2 & 3".
10. GENE 771-77-1194, Revision 2, "Shroud Repairs Program for Dresden Units 2 & 3 - Back-up Calculations for RPV Stress Report No. 25A5691". (Proprietary information)
11. GENE-771-95-0195, Revision 1, "Dresden Units 2 & 3 - Top Ring Plate and Star Truss Stress Analysis".
12. GENE-771-96-0195, Revision 1, "Dresden Units 2 & 3, Top Ring Plate and Star Truss Analysis Backup Calculations". (Proprietary information)
13. GENE-523-A181-1294, Revision 0, "Dresden Units 2 & 3 - Primary Structure Seismic Models".
14. GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Evaluation of the Reactor Internal Piping Due to the Reactor Shroud Repair for Dresden Units 2 & 3, dated March 13, 1995 (B13-01749, MDP-9510)
15. GENE Letter, M. D. Potter - GE Shroud Project Engineer, To Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Evaluation of the GENE Fuel with the Shroud Repair Hardware Installed for Dresden Units 2 & 3, dated March 13, 1995 (B13-01749, MDP-9512)
16. GENE Letter, M. D. Potter - GE Shroud Project Engineer , To Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Clearance Between the Dresden Units 2 & 3 Shroud Repair Hardware and the Jet pumps, dated March 13, 1995 (B13-01749, MDP-9513)
17 GENE Letter, M. D. Potter - GE Shroud Project Engineer, To Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Evaluation of the Reactor RPV skirt Ring Girder, Anchor

Bolts and High Strength Bolts Due to the Reactor Shroud Repair for Dresden Units 2 & 3, dated March 13, 1995 (B13-01749, MDP-9511)

18. GENE Letter, M. D. Potter - GE Shroud Project Engineer, To Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Core Shroud Hardware Water Displacement in the RPV for Dresden Units 2 & 3, dated March 13, 1995 (B13-01749, MDP-9508)
19. GENE Letter, M. D. Potter - GE Shroud Project Engineer, To Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Effect of Increased Enthalpy Due to Shroud Leakage on the RPV Recirculation Pump Inlet for Dresden Units 2 & 3, dated March 13,1995 (B13-01749 MDP-9509).
20. 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 Dresden Units 2 & 3, dated March 13, 1995, (B13-01749, MPD-9515)
21. GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Performance impact of shroud repair leakage for Dresden Units 2 & 3, dated May 18, 1995 (B13-01749, MDP-9536)

FIGURE - 1



SHROUD
Core Spray
Core Spray
Line
Separator Head Support Ring (SHSR)
H1
H2
Top Guide Support Ring (TGSR)
H3
Jet Pump
H4
H5
Core Plate
Support Ring (CPSR)
H6
H7
Acces Hole Cover
Shroud Support Ring (SSR)
H9
H9
(Weld to Rx
Vessel Wall)
Jet Pump
Support Plate (JPSP)
H8
H8
Jet Pump Support Plate Columns

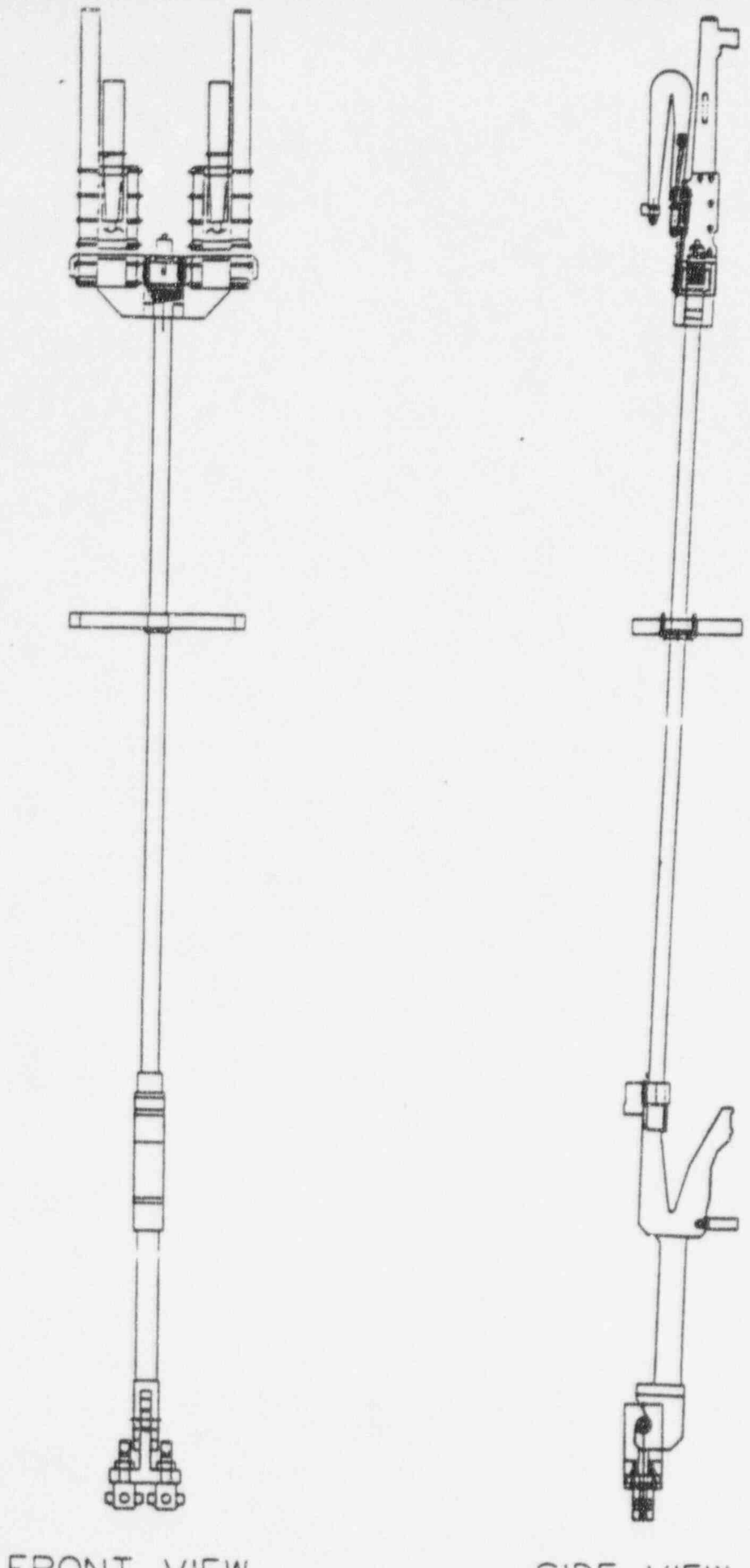
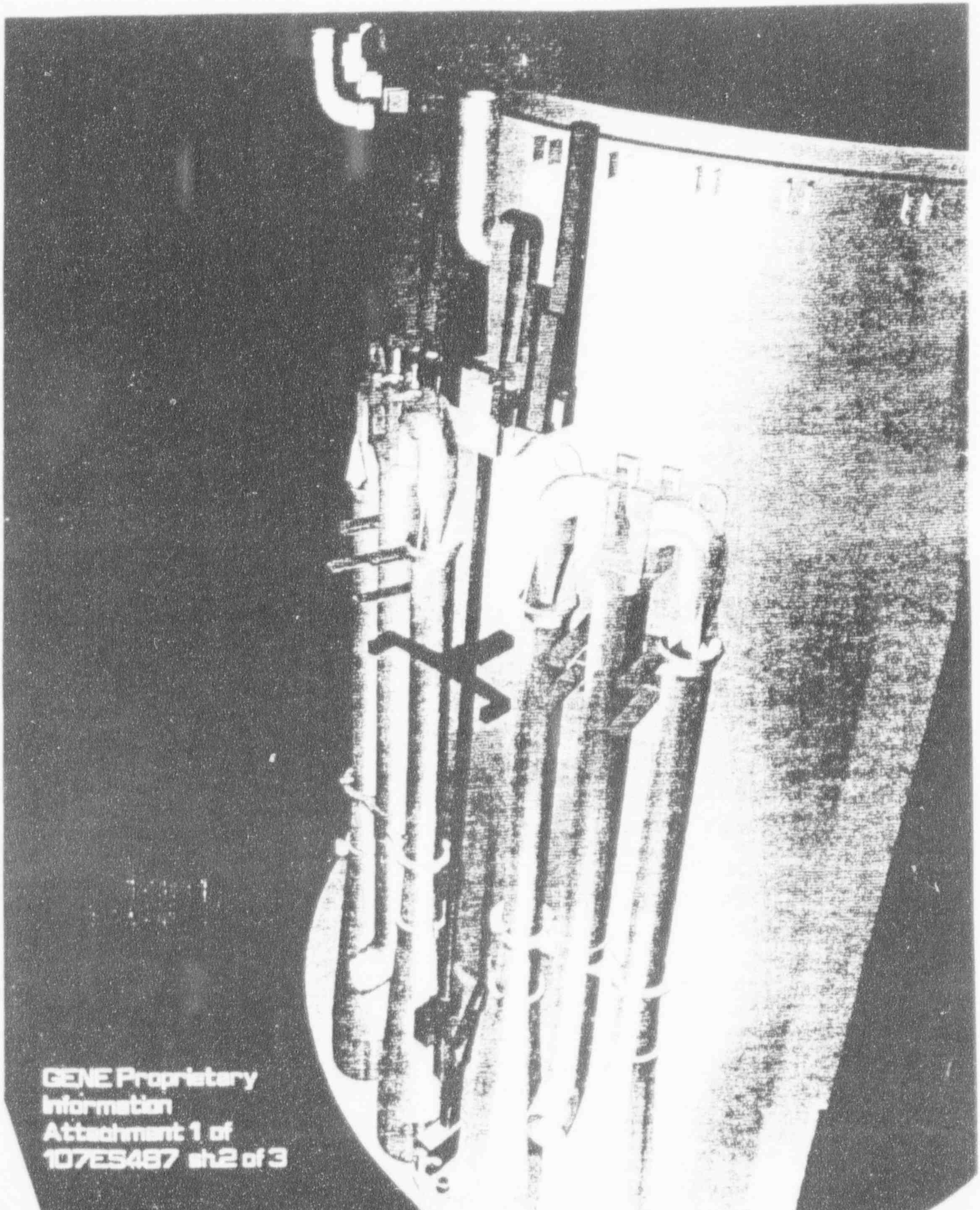
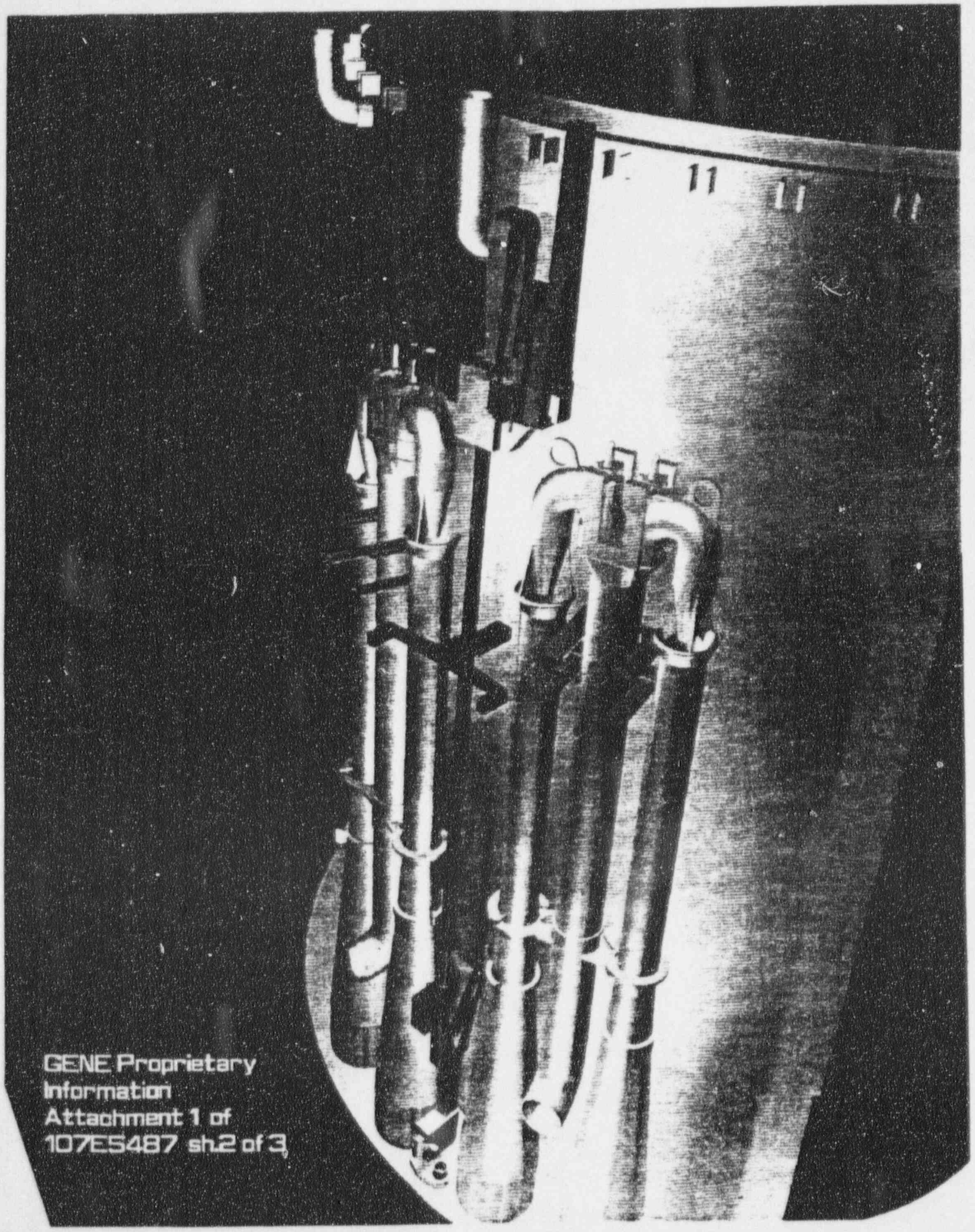



FIGURE 2. SHROUD STABALIZER

Enclosure 19

Color picture
Computer model
Core Shroud Repair Installed at Quad Cities


GENE Proprietary
Information
Attachment 1 of
107E5487 sh.2 of 3


GENE Proprietary
Information
Attachment 1 of
107E5487 sh.2 of 3

GE-NE-L12-00819-05
DRF L12-00819 (26)
AUGUST 1994

# Core Shroud Blowdown Load Calculation During Recirculation Suction Line Break by TRACG Analysis for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2

Y.K. Cheung
J.C. Shaug
F.D. Shum
J.A. Villalta



Approved by: Hwang Choe
H Choe, Project Manager

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~~9409140155~~

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## IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT



GE proprietary information is indicated by "bars" drawn in the margin of the text of this report.

## 3.0 Analytical Model Description

An input base deck was prepared for use with TRACG from an existing base deck for LaSalle.

The nodalization used was a result of sensitivity studies carried out to evaluate the effect of azimuthal and axial nodalization on the computed blowdown load. One sensitivity study consisted of adding an additional axial level in the break region. The result of the study was a negligible difference in the calculated force, which can be seen in Figure 3-2 for a case with twelve azimuthal cells, and 100% break flow area and friction

The increased nodalization provided a more accurate representation of the pressure distribution near the break region, as can be seen in Figure 3-3.

Plane view - 2 rings, 14 azimuthal segments

Elevational view - 15 axial levels

Figure 3-1 TRACG Nodalization of RPV for Quad Cities/Dresden

Figure 3-3. Vessel Pressure for axial level 4 at $t = 2.0$ seconds (normalized with respect to pressure at 0°).

Thus, the use of 14 azimuthal sectors provided a more accurate assesment of the blowdown force.

In addition to the nodalization, the base deck included proportional simulation of jet pumps and feedwater flow in the azimuthal sectors. Other RPV components, such as the steam separators, guidetubes, and external recirculation loops, were also modeled along with the jet pumps in the TRACG base deck.

The TRACG code uses a multi-dimensional two-fluid model for the reactor thermal hydraulics; it solves the conservation equations for mass, momentum, and energy for the gas and liquid phases. The code closes the conservation equations with an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface and the wall; these correlations are based on a single flow regime map used throughout the code. A three-dimensional formulation is used for the vessel component, while the rest of the components (e.g. pipes, tees, valves) are modeled in one dimension. The code also includes a control system model capable of simulating the major BWR control systems (e.g. recirculation flow, pressure).[3-1]

---

[3-1] J.G.M. Andersen, et.al., "TRACG Model Description - Licensing Topical Report," NEDE-32176P, February 1993.

## 4.0 Analytical Model Qualification

The TRACG analytical model has been used for many plant applications including LOCA, BWR transients, and ATWS events, and has been systematically qualified. The qualification process included comparisons to separate effects tests, BWR component performance tests, several integral system effects tests, and several BWR plant tests. Therefore, the overall TRACG analytical model is already well qualified. A sensitivity study has been performed on the portion of the model that has been implemeted to calculate the lateral blowdown load, and is already described in Section 3. The two primary parameters that significantly affect the calculation of the lateral blowdown load on the core shroud are the critical flow rate through the broken suction line and the circumferential flow resistance of the jet pumps; they are discussed below.

### 4.1 Critical Flow

The critical flow model in TRACG has been compared with the data from the Marviken reactor vessel, PSTF test facility, and Edwards test. All these comparisons show an excellent agreement between the TRACG prediction and the test data. Figure 4-1 shows the comparison between the TRACG prediction and Marviken Test 15. The Marviken Test 15 had 31 K subcooling. During the subcooled blowdown period, the TRACG overpredicts the test results. Overall the deviation from the test data is less than 20 % for all periods.

Figure 4-1. Comparison of TRACG Blowdown Flow Rate to Marviken Test 15

## 5.0 Results

The results of the TRACG analysis are provided in this section for the blowdown load, moment, and moment arm acting in the 0°-180° and 90°- 270° planes on the shroud section above the H5 weld. In addition, the results are also provided for the force, moment, and moment arm acting on the complete shroud assembly.

For the force and moment, the critical time period is that below five seconds, when subcooled blowdown occurs, and when the highest load is placed on the shroud. Once two-phase blowdown begins, the load decreases significantly. It should also be noted that the acoustic wave response is not accurately modeled in this analysis and the initial 0.5 to 1.0 second in each of the figures should be ignored. Although TRACG can assess the acoustic response, it would require a calculation time step on the order of one microsecond, the use of which was not feasible for the present analysis. In addition, the TRACG code has not been qualified extensively for calculation of the acoustic wave response due to a recirculation line break.

Figure 5-1 shows the pressure distribution as a function of elevation at 0° and 180°. This shows that depressurization near the suction nozzle causes the force imbalance accross the shroud.

Figure 5-1. Variation of Pressure with Vessel Elevation at t = [illegible].0 seconds

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

July 10, 1995

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

Attn: Document Control Desk

Subject: Dresden Nuclear Station Units 2 and 3
**Additional Information - Dresden Station Core Shroud Repair**
NRC Docket Nos. 50-237 and 50-249

Reference: J.L. Schrage to USNRC letter, dated May 24, 1995.

In the referenced letter, Commonwealth Edison (ComEd) submitted the Design Documents for the proposed repair of the Dresden Station Unit 2 and 3 core shrouds. Upon further review, ComEd has identified a typographical error in Enclosure 16 of the referenced letter (GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Performance impact of shroud repair leakage for Dresden Units 2 & 3, dated May 18, 1995).

The Enclosure to this letter transmits the corrected document (GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Performance impact of shroud repair leakage for Dresden Units 2 & 3, dated June 21, 1995). The revised part of the document is marked by a vertical bar in the right hand margin. This revised document supercedes the original in its entirety. ComEd apologizes for any inconvenience that this typographical error may have caused.

To the best of my knowledge and belief, the statements contained in this response are true and correct. In some respects, these statements are not based on my personal knowledge, but obtained information furnished by other ComEd employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please direct any questions you may have concerning this response to this office.

Sincerely,

John L. Schrage
Nuclear Licensing Administrator

OFFICIAL SEAL
JACQUELINE T EVANS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 12/16/97

Jacqueline T Evans 7/10/95

Enclosure

cc: H.J. Miller, Regional Administrator - RIII
M. N. Leach, Senior Resident Inspector - Dresden
J. F. Stang, Project Manager - NRR
Office of Nuclear Facility Safety - IDNS



A001
1/1

Enclosure

GENE Letter, M. D. Potter - GE Shroud Project Engineer to Kenneth Hutko - ComEd Shroud Project Engineer, Subject - Performance impact of shroud repair leakage for Dresden Units 2 & 3, June 21, 1995

GE Nuclear Energy

June 21, 1995

cc: R. Svarney
E. R. Mohtashemi
B13-01749
MDP-9542

To: Kenneth Hutko
ComEd Shroud Project Engineer

From: M. D. Potter *M. D. Potter*
GE Shroud Project Engineer

SUBJECT: PERFORMANCE IMPACT OF SHROUD REPAIR LEAKAGE FOR DRESDEN UNITS 2 AND 3

Reference: DRF No. B13-01749.

**1. Introduction**

The hardware designed to repair the shroud with identified cracks for Dresden Units 2 and 3 requires the machining of eight holes through the shroud support plate. Each of these holes will have some clearance, which will allow leakage flow to bypass the steam separation system. In addition, potential leakage through the weld cracks (H1 through H8) and the replacement access hole cover is also considered. This letter reports the leakage flow for 100% rated power and core flow.

**2. Evaluation**

2.1 Leakage Flow Evaluation

The most restrictive flow area for leakage through the holes in the shroud support plate is based on a conservative gap between the adjacent surfaces of the shroud support plate and the lower support bracket. In addition, there are a total of eight circumferential shroud welds (H1 - H8) that are considered as potential leakage paths - two above the top guide support ring, three on the upper shroud between the core support ring and the top guide support ring, and three on the lower shroud below the core support ring. It is conservatively assumed that each of these welds develops a complete circumferential crack that opens to 0.001 inches.

The leakage flows for 100% rated power and core flow are summarized in Table 1. These leakage flows are based on applicable loss coefficients and reactor internal pressure differences (RIPD's) across the applicable shroud components. The replacement access hole cover leakage is based on information in the referenced DRF. Leakage from the weld cracks above the top guide support ring is assumed to be two-phase fluid at the core exit quality. Leakage from the remaining paths below the top guide support ring is considered single-phase liquid. All of the leakage flows bypass the steam separators and dryers. The leakage flows below the shroud support ring also bypass the core. The results show that the leakage flows from the repair holes, weld cracks and the access hole cover result in a combined leakage of about 0.44% of core flow.

**Table 1. Summary of Leakage Flows at Rated Power and Flow**

| | |
|---|---|
| Leakage flow (gpm) | |
| Shroud head flange pockets | 1600 |
| Weld cracks | 140 |
| Repair holes in support plate | 325 |
| Access hole covers | 180 |
| Leakage-to-core Mass flow (%) | |
| Shroud head flange pockets | 0.21 |
| Weld cracks | 0.04 |
| Repair holes in support plate | 0.12 |
| Access hole covers | 0.07 |

The steam portion of the leakage flows will contribute to increasing the total carryunder 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 as summarized in the following subsections.

### 2.2 Steam Separation System

The leakage flow through weld cracks H1 and H2 occurs above the top guide support ring and includes steam flow, which effectively increases the total carryunder in the downcomer by about 0.03% at rated conditions. 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 carryunder from the separators and the shroud head leakage is about 0.18% and is bounded by the design value.

### 2.3 Jet Pumps

The increased total carryunder will decrease the subcooling of the flow in the downcomer. This in turn reduces the margin to jet pump cavitation. However, because the total carryunder meets the design-condition carryunder value, there is no impact on jet pump performance compared with the design condition.

### 2.4 Core Monitoring

The impact of the leakage results in an overprediction of core flow by about 0.21% of core flow. This overprediction is small compared with the core flow measurement uncertainty of 2.5% for jet pump plants used in the MCPR Safety Limit evaluations. Additionally, the decrease in core flow resulting from the overprediction results in only a 0.1% decrease in calculated MCPR. Therefore, it is concluded that the impact is not significant.

### 2.5 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.

2.6 Emergency Core Cooling System

Leakage through weld cracks H1 and H2 results in slightly increased carryunder that causes the initial core inlet 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 emergency core cooling system (ECCS) performance from this effect compared with the design conditions. 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 cladding temperature (PCT) for the limiting LOCA event by less than 30°F. The current analysis basis yields a LOCA PCT of about 2045°F for the design basis LOCA with LPCI injection failure. The 10CFR50.46 regulatory limit PCT is 2200°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.

2.7 Fuel Cycle Length

The increased carryunder due to leakage flow above the top guide support ring 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 the leakage results in a minor effect (~0.8 days) on fuel cycle length and is considered negligible.

**3. Conclusions**

The impact of the leakage flows through the shroud repair holes and the potential weld cracks in the shroud have been evaluated. The results show that at rated power and core flow, the leakage flows from the repair holes and the weld cracks are predicted equal to a combined leakage of about 0.44% of core flow (including potential replacement access hole cover leakage). These leakage flows are sufficiently small so that the steam separation system performance, jet pump performance, core monitoring, fuel thermal margin and fuel cycle length remain adequate. Also, the impact on ECCS performance is sufficiently small to be judged insignificant, and hence, the licensing basis PCT for the normal condition with no shroud leakage is applicable.

M. D. Potter