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October 19, 1994

Mr. Ron Hernan
U.S. Nuclear Regulatory Commission
Mail Stop 14-C-7
1 White Flint North Building
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: Non-Proprietary Version of GPUN's September 28, 1994 Oyster Creek Shroud Repair

Dear Mr. Hernan:

Per instructions from Ron Zak of GPUN, attached is the non-proprietary version of GPUN's September 28, 1994 submittal concerning the Oyster Creek Shroud Repair.

If you have any questions, please do not hesitate to call.

Sincerely yours,

Noman M. Cole

Noman M. Cole

Attachment

cc: R. Zak, GPUN

080083

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A001 /1

GPU Nuclear

GPU Nuclear Corporation
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201-316-7000
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Writer's Direct Dial Number:

September 28, 1994
5000-94-0040
C321-94-2157

U.S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
Facility Operating License No. DPR-16
Core Shroud - 15R Outage Contingency Repair

In support of the NRC Staff's expedited review of the subject repair, enclosed is GPU Nuclear's interim safety evaluation report, and repair design specifications. The safety evaluation report is considered interim in that some evaluations supporting the conclusions reached in the report are currently being finalized but are not yet complete. It is GPU Nuclear's intention to submit a final safety evaluation report upon completion of all supporting evaluations.

The enclosed interim report and repair design drawings contain information determined to be proprietary, and an affidavit as required by 10CFR2.790(b)(1) is attached.

If you have any questions or comments on this submittal, please contact Mr. Michael Laggart, Manager, Corporate Nuclear Licensing at (201) 316-7968.

Sincerely,

R W Keaten

R. W. Keaten
Vice President and Director
Technical Functions

c: Administrator, Region 1
Senior Resident Inspector
Oyster Creek NRC Project Manager

9410270270

MPR
ASSOCIATES INC.
ENGINEERS

Attachment to
GPUN Letter
C321-94-2157

Safety Evaluation for Oyster Creek Core Shroud Contingency Repair

Prepared by

MPR Associates, Inc.
320 King Street
Alexandria, VA 22314

September 28, 1994

Non-Proprietary Version

Prepared for

GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station

9410270277

CONTENTS

LIST OF FIGURES

1.0 PURPOSE

The purpose of this report is to describe the design of the contingency repair to address possible cracking of any and all combinations of circumferential welds in the Oyster Creek core shroud, and to provide an evaluation of the impact of the repair on the existing reactor vessel and reactor internals in their intact condition and for assumed failures of shroud welds. The present condition of the Oyster Creek core shroud and the status of inspections of the shroud are described in Reference 1, GPUN's response to NRC Generic Letter 94-03.

The report includes the following main areas:

- Description and drawings of the planned repair.
- Summary of main design features.
- Design, material, analysis, and installation criteria. These criteria are provided as a paragraph-by-paragraph comparison of the Oyster Creek criteria with the BWR Owners Group Vessel Internals Project (VIP) specification, Reference 2.
- Description of seismic design and analyses.
- Description of the installation sequence.

2.0 BACKGROUND

Before Oyster Creek was originally placed into service in the late 1960 time frame, the issue of furnace sensitized stainless shroud parts was addressed (see Reference 3). This resulted in special brackets being installed to cover potential failure of welds H_7 and H_8. See Figures 1, 2, 3, and 4 which show the current configuration of Oyster Creek. These existing 36 brackets provide redundant structural support for the H_7 and H_8 shroud welds. It is planned to use these existing brackets for the lower attachment for the proposed shroud repair.

3.0 SUMMARY

GPUN has developed a contingency repair for shroud weld cracking for possible implementation during the Fall 1994 outage at Oyster Creek. See Reference 1 for more detailed information on GPUN plans for the September 1994 outage.

The overall purpose of this repair is to structurally replace the circumferential stainless steel shroud welds which are subject to inservice cracking.

The design of the Oyster Creek shroud repair consists of a series of stainless steel tie-rod assemblies which are installed in the shroud/reactor vessel annulus, between attachment points near the top of the shroud and the lower shroud support cone. The tie-rod assemblies incorporate radial seismic supports which provide lateral stability and stiffness

to the shroud assembly. The tie-rod/radial support assemblies provide tensile (i.e., vertical) and lateral support to the cylindrical part of the core shroud, including its circumferential welds, for both vertical and overturning loads resulting from normal operation and design accident loads, including seismic and postulated pipe ruptures. This design protects against potential through-wall cracking in any and all of the circumferential welds from the top of the shroud to the bottom (i.e., welds H_1 through H_{6B}). As indicated previously, welds H_7 and H_8 are already protected by a previous repair discussed in Reference 3.

Figures 5 through 10 and 15 show the configuration and features of the proposed shroud repair for Oyster Creek. The design complies with all requirements specified in Reference 2.

Other important features of the repair design are as follows:

A. The repair has no significant effect on the structural or seismic response of the intact (i.e., uncracked) shroud, reactor vessel, or other internals.

B. The repair does not affect reactor performance or previous safety analyses.

C. Design and accident loads are consistent with the Oyster Creek licensing basis, and load combinations and acceptance criteria for stresses and shroud replacements also meet BWROG VIP specification requirements.

D. The repair is considered permanent.

These paragraphs contain proprietary information and are not for public disclosure. They are intentionally left blank.

G. The tie-rod assemblies are preloaded so that during normal operation there is no separation or floating of shroud sections that have 360° failed welds. The preload is accomplished by straight forward means utilizing stud tensioners similar to those used for reactor coolant system closures. **This sentence contains proprietary information and is not for public disclosure.**

H. The radial seismic supports used in conjunction with the tie-rods allow significantly lower preload on the shroud than a tie-rod system without lateral restraints and thus keeps the additional stress low in the shroud and in the

conical support plate. These radial seismic supports are simple in configuration and fabricated of standard BWR materials. If the radial seismic supports were not used, the preload would have to be significantly increased (e.g., factor of about 3 at Oyster Creek) and this would result in significantly higher stresses, especially in the H_2/H_3 weld area and in the conical support plate. Radial supports currently are located at the upper and lower core supports and at two intermediate locations; i.e., at the H_4 and H_5 welds (see Figure 4). Final analysis may indicate that the shroud section between the H_5 and H_{6A} welds is adequately restrained by the lower radial restraints at H_{6A} and if this is confirmed, the radial restraint at H_5 weld will be eliminated.

I. Special features have been incorporated into the design of the tie-rod to accommodate the cold feedwater thermal transient which results in a differential metal temperature between the shroud and tie-rod of about 130°F. Without such features, the cold feedwater transient would cause an additional 1,000,000 lb load to be applied to the shroud.

The remainder of this paragraph contains proprietary information and is not for public disclosure. This is intentionally left blank.

J. Lateral movement of the shroud at the core top guide and lower support plate locations is positively restricted to less than a ½ inch to assure control rod insertion under all accident and seismic conditions.

K. The tie-rod assemblies and other parts are all appropriately heat treated austenitic stainless steel, thereby providing high resistance to IGSCC. These materials were also tested per ASTM 262 practice E. Further, no welding or heat treating is allowed during the manufacture of the tie-rod assemblies.

L. No modifications or cutting of the existing reactor internals are required to install the proposed tie-rod repair (with the exception that several small brackets used to support start-up FIV instrumentation may have to be removed). A spacer ring, between the top of the shroud and its head, allows attachment of the top of the tie-rod via a flanged bracket. Longer shroud head bolts procured earlier in connection with a contingency overhead core spray sparger will be used to accommodate the spacer ring. A hook device on the tie-rods allows attachment of the bottom of the tie-rod to the existing brackets on the conical support plate. The design is such that the upper brackets are positively restrained between the shroud and shroud head, and the hooks at the bottom are positively restrained between the reactor vessel and the existing brackets attached to the conical support plate at the base of the shroud. All components are positively locked using proven mechanical locking devices.

M. The tie-rod assemblies are designed to be installed and readily removable, if required, with straight forward long handled tools (i.e., no robotic tooling is

required) and no non-removable parts are permitted. Tooling and installation procedures will be in accordance with NUREG 612.

N. The design of the tie-rod takes no credit for the coefficient of friction or shearing of interlocking surfaces on any assumed weld failures. (This possibility is covered, however, in the seismic analyses.)

O. Nonlinear, time history analyses demonstrate that the tie-rod/lateral restraint repair does not increase seismic loads on the existing shroud support or reactor fuel, and the stresses due to seismic loads on the reactor vessel and shroud meet established allowable values for all combinations of shroud circumferential weld failures. The same is true for original licensing basis static seismic loads.

P. Installation of the proposed shroud repair is expected to provide a technical basis for substantially reducing the need and therefore extent of future shroud inspections.

4.0 DISCUSSION

In this section of the report, a number of the parameters should be considered as preliminary at this time since final calculations for Oyster Creek have not been completed and verified. These calculations have been completed for another plant with essentially the same tie-rod design as Oyster Creek; and, the Oyster Creek calculations will be completed prior to restart (if this repair is utilized).

4.1 BWR VIP Design Specification (Reference 2)

The subject repair design meets all requirements in Reference 2 as indicated in the paragraph-by-paragraph comparison which follows. Buoyancy loads are appropriately evaluated and no cold reduced XM-19 material is used in the design. Torsional loads will be considered after resolution by the BWR Owners Group. There are no torsional load requirements in the Licensing Basis for Oyster Creek. To aid in technical review, this Safety Evaluation Report uses the same paragraph numbering system and requirements as stated in Reference 2.

Requirement Per Reference 2	Oyster Creek Design
3.2 <u>Safety Design Bases</u> The reactor internals, of which the core shroud is a part, have three basic safety functions:	
1. To provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel, except for BWR/2's. BWR/2's do not require a floodable volume to provide adequate core cooling.	Oyster Creek is a BWR/2, does not have jet pumps, has redundant core sprays and does not require the shroud to maintain a floodable volume.
2. To limit deflections and deformation to assure that the control rods and the Emergency Core Cooling Systems (ECCS) can perform their safety functions during anticipated operational occurrences and accidents.	The tie-rod radial restraints will positively limit the lateral displacement to less than 1/2 inch. The tie-rods and radial restraints will limit deformation of the core spray internal piping to a value which will assure no loss of function of the core spray system.
3. To assure that the safety design basis (1) and (2) above are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.	This is covered above.
Additionally, the reactor internals are designed to meet power generation objectives to:	
1. Maintain partitions between regions within the reactor vessel to provide correct coolant distribution for all normal plant operating modes.	The original configuration of the shroud will be maintained; however, small leakage paths can be opened due to shroud weld cracking. With suitable tie-rod preload, the effect of this leakage is small. It will be confirmed that this leakage is within acceptable limits.
2. Provide positioning and support for the fuel assemblies, control rods, incore flux monitors, and other vessel internals and to ensure that normal control rod movement is not impaired.	Suitable preload will prevent separation of cracked shroud welds during normal operation even if the welds were cracked 100% though-wall. The combination of tie-rods plus radial seismic restraints will maintain the needed alignment even under worst-case assumptions. Specifically, lateral alignment will be maintained less than 1/2 inch.

<table>
<tr><th>Requirement Per Reference 2</th><th>Oyster Creek Design</th></tr>
<tr><td>3.3.1 <u>Normal Operation</u>

The repair design should consider loads existing during periods of reactor startup, shutdown, and power operation. This includes dead weight of the shroud and RPV internals (including bouyancy effects), differential pressure, and thermal-hydraulic loads.</td><td>Design basis loads per Reference 3 and as modified by Reference 9, used in the tie-rod design are as follows:

• Dead weight of shroud including head and separators—188,000 lbs.

• Differential pressure across conical support plate—20.5 psia.

• Hydraulic lift force on shroud—304,000 lbs, due to:

Δp = 16.1 psi core plate
Δp = 4.34 psi separators</td></tr>
<tr><td>3.3.2 <u>Anticipated Operational Occurrences (Upset Conditions)</u>

Loads due to anticipated operational occurrences which have the potential to increase shroud loads above normal operation should be considered. Typical events include: maximum system pressure, pressure regulator failure (open), recirculation flow control failure (max. demand), loss of feedwater with feedwater restart without feedwater heating, and inadvertent activation of a safety relief valve. This category of events also includes the combination of normal loads plus operating basis earthquake (OBE) loads.</td><td>Design basis loads per Reference 3 and as modified by Reference 9, used in the tie-rod design are as follows:

• Differential pressure across conical support plate—20.5 psi.

• Hydraulic lift force on shroud—304,000 lbs.

• OBE loads on shroud:

- Vertical—30,000 lbs
- Horizontal shear at conical support—195,000 lbs
- Overturning moment at conical support—23.8 x 10^6 in.lbs

• Temperature difference of 130°F between shroud and tie-rods for 10 cycles of feedwater transient.</td></tr>
</table>

Requirement Per Reference 2	Oyster Creek Design
3.3.3 <u>Design Basis Accidents (Emergency/Faulted Conditions)</u>	
Loads associated with a design basis earthquake in conjunction with a recirculation discharge line break and/or a main steamline break should be considered. All components of these loads should be considered.	Design basis loads per Reference 3 and as modified by Reference 9, used in the tie-rod design are as follows: • DBE loads on shroud: - Vertical--60,000 lbs - Horizontal shear at conical support--390,000 lbs - Overturning moment at conical support--47.6 x 10^6 in.lbs • Recirculation line break - Downward vertical--1,570,000 lbs - Asymmetric loads - to be determined and evaluated • Steam line break inside flow limiter - Up load on shroud at conical support--917,000 lbs
In analyzing accidents and transients, the seismic analysis must include the shroud repair. The seismic inputs used shall be those which form the current licensing basis. alternatively, new seismic analysis performed in support of repair design may be used.	The design analyses include dynamic analyses of the shroud with all horizonal welds failed as well as an analysis to demonstrate compliance with licensing basis seismic loads given above.
The treatment of the combined accident and seismic loads should be consistent with the current plant licensing basis.	Combined accident and seismic loads are consistent with the current licensing basis. (See Item 5.2.1.1 later in this report.)
4.0 <u>SCOPE</u>	
The shroud repair will address potential cracking in the 304 stainless steel horizontal shroud welds which may be sensitized (shown on Figures 3-3A-F). typically, these are welds H-1 down through the bimetallic weld where the shroud was welded to the shroud support.	The repair, in conjunction with the previous repair per Reference 3, covers all pertinent welds in accordance with VIP spec.
This design criteria is applicable for repair of individual welds or groups of welds up to and including a comprehensive repair including welds H-1 down through the shroud to shroud support weld.	The repair, in conjunction with the previous repair per Reference 3, covers all pertinent welds in accordance with VIP spec.

Requirement Per Reference 2	Oyster Creek Design
5.1.1 <u>Repair Design Life</u> The design life of the repair will normally be for the remaining life of the plant plus life extension beyond the current operating license. Alternatively, the repair may be designed to allow inspection, replacement or renewal of components at the end of their intended life.	The repair covers the life of the plant, including possible life extension of 40 years.
5.1.2 <u>Safety Design Bases</u> The repair shall be designed such that the safety bases described in Section 3.2 of this document is demonstrated.	The safety bases in Section 3.2 are demonstrated.
5.1.3 <u>Safety Analysis Events</u> Safety analysis event scenarios described in individual plant FSARs remain valid and unaltered by the criteria contained in this document.	The safety event scenarios are unaltered and remain valid.
5.1.4 <u>Load Combinations</u> The repair shall be designed for all load combinations required by Section 3.3.	The repair is designed for all loads required by Section 3.3. Load combinations are consistent with the original licensing basis (see Section 5.2.1.1)
5.1.5 <u>Flow Partition</u> Repairs to the core shroud are not required to totally prevent leakage from the core region into the downcomer annulus. However, the design shall ensure that cracked welds do not separate under normal operation as a minimum. Design will account for leakage from the region inside the shroud into the annulus region during normal operation. This leakage should not exceed the minimum subcooling required for proper jet pump and/or recirculation pump operation and the core bypass flow leakage requirements assumed in the reload fuel safety analysis shall be maintained. Designs will also verify acceptable leakage through the flow partition resulting from weld separation during accident and transient events that meet the normal operational requirements for recirculation system performance and core bypass flow.	The tie-rods will be installed with sufficient preload to ensure cracked welds will not separate under normal operation. **The remainder of this paragraph contains proprietary information and is not for public disclosure. This is intentionally left blank.** The proposed shroud repair does not require any openings/holes in the shroud, shroud separator, or shroud support cone; thus no leakage paths are created by the repair between the volumes inside the shroud and the shroud/reactor vessel annulus area.

Requirement Per Reference 2	Oyster Creek Design
5.1.6 <u>Flow Induced Vibration</u> The repair shall be designed to address the potential for vibration, and to keep vibration to a minimum. The natural frequency of the repaired shroud, including the repair hardware, shall be determined. The vibratory stresses shall be shown to be less than the allowable stresses of the repair materials. Forcing functions to be considered include the coolant flow and the vibratory forces transmitted via the end point attachments for the repair. Testing may be used as an alternative or to supplement the vibration analysis.	An analysis based on worst-case assumptions is used to ensure vibration adequacy as follows: • Maximum cross-flow velocity. • Flow force assumed applied at: - Natural frequency of tie-rod - In-phase • Conservative damping coefficient of 5%. Testing is in progress to confirm • Stresses less than endurance limit
5.1.7.1 The repair shall be designed so as to produce acceptable loading on the original structure of the shroud, consistent with the criteria provided herein.	Shroud and conical support plate (and its bracket) stresses are within design basis allowables and are minimized by the use of up to 10 tie-rod assemblies.
5.1.7.2 The repair should minimize stresses introduced into the shroud consistent with the criteria provided so as to minimize aggravating further shroud cracking.	**This paragraph contains proprietary information and is not for public disclosure. This is intentionally left blank.**
5.1.7.3 The repair should minimize the loading on the supporting structures of the shroud, such as the shroud support plate and the RPV wall, to stay within the original design allowable stresses of these structures.	These loads/stresses are minimized as discussed above. Stresses in vessel interfaces meet specified allowables for all accident (emergency/faulted) conditions.
5.1.8 <u>Annulus Flow Distribution</u> The design shall not adversely affect the normal flow of water in the jet pump region or restrict the flow in any way that would affect normal balance of flow in this region. The design shall not restrict the flow of water into the recirculation suction inlet.	The tie-rods are located well away from the recirculation suction inlet and the effect on downcomer flow is negligible as well. For example, the additional pressure drop due to the addition of the tie-rods is less than 1% of the total loop flow pressure drop.

Requirement Per Reference 2	Oyster Creek Design
5.1.9 <u>Core Bypass Flow Distribution</u> For repair designs that incorporate structures inside the shroud, the design shall not adversely affect the core bypass flow distribution.	No structures are located inside the shroud.
5.1.10 <u>Emergency Operating Procedure (EOP) Calculations</u> Inputs to the EOP calculations, such as bulk steel residual heat capacity and reduction of reactor water inventory shall be addressed based on repair hardware mass and water displacement.	The total mass added by the repair and resulting displacement of water is negligible and will be evaluated regarding EOP calculations. For example, the tie-rods increase the weight of the shroud assembly by less than 6%.
5.1.11 <u>Power Uprate</u> For those units currently undergoing a power uprate program, the resulting increased loadings must be considered in the repair design. If a power uprate program is implemented after the installation of a shroud repair, the uprate program must address the increased loads imposed on the repaired shroud.	If a power uprate is evaluated for Oyster Creek, the subject repair design basis will be reviewed accordingly.
5.1.12 <u>Radiation Effects on Repair Design</u> The design of the repair shall account for the affects of irradiation relaxation utilizing end-of-life fluence on the materials.	The effects of radiation and thermal induced relaxation are covered in the design. Preload is conservatively calculated to be reduced about 5% over the life of the plant.
5.2.1 <u>Structural</u> The repair hardware shall be designed to provide structural integrity for a complete circumferential through-wall cracking of the shroud welds covered by this criteria (see Section 4.0) for all analyzed loading conditions. Loads due to fluid hydraulic differential pressure forces acting on the shroud for normal, upset, emergency and faulted conditions as well as seismic loads shall be considered. The pressure differences used for these events shall be those associated with the current plant licensing basis documents. The current plant licensing basis documents may include power uprate and extended operating domain conditions. Load combinations shall be determined in accordance with the current licensing basis documents and applicable codes. The specific requirements are provided below.	The requirements are met as discussed in the following.

Requirement Per Reference 2	Oyster Creek Design
5.2.1.1 <u>Load Combinations</u> Unless otherwise specified in the current plant licensing basis, the following load combinations cases should be considered: <u>Case</u> <u>Conditions</u> (1) Normal and operating loads plus dead weight (2) Case (1) loads plus upset operational transients (3) Case (1) loads plus OBE (4) Case (1) loads plus design basis accident loads, including • Main steam line break • Recirc. line break (5) DBE or SSE The combination of Case (4) plus DBE or SSE loads shall be considered only if required by the current plant licensing basis or if desired to demonstrate plant capability.	The current licensing basis (per Reference 2) is met including the combination of Case (4) plus OBE. In addition, for conservatism (but not to change the licensing basis) steam line break loads will be combined with DBE Case (5).
5.2.1.2 <u>Allowable Stresses</u> Allowable stresses under the above conditions should be consistent with the current plant licensing basis. Unless otherwise specified, the following allowables apply: • Normal and upset loads--Normal code allowables (Case (1) thru (3)). • Accident loads--Faulted code allowables (Case (4) and (5) and Case (4) + SSE, if required). The plant specific submittal shall identify the specific sections and subsections of the ASME Code utilized to designate allowable limits.	These requirements will be met. Applicable Code is ASME Section III, Subsection NB, 1989 Edition, no addenda.
5.2.1.3 <u>Seismic Loads</u> Seismic loads shall include OBE and SSE loadings specified in the current licensing basis.	These requirements will be met including the combinations per Item 5.2.1.1 above.

<table>
<tr><th>Requirement Per Reference 2</th><th>Oyster Creek Design</th></tr>
<tr><td>
5.2.2 <u>Shroud Pressure Drop</u>

The pressure drop across the shroud is composed of two main drops. The first is the drop across the core support plate and the second is across the shroud head. Both of these drops tend to lift the shroud. The magnitude of the pressure drop is a function of the original reactor design, changes to the original design such as power uprate or increased core flow, and of the event that is under consideration. Typically, the current plant license basis documents give pressure drops for three conditions, which are normal operation, main steam line break LOCA, and recirculation line break LOCA.

Since several plants have implemented design changes, each plant should provide their specific values to the designer.

<table>
<tr><th>Event</th><th>At Core Plate and Below</th><th>Above Core Plate</th></tr>
<tr><td>Normal</td><td></td><td></td></tr>
<tr><td>Upset</td><td></td><td></td></tr>
<tr><td>Emergency</td><td></td><td></td></tr>
<tr><td>Fault</td><td></td><td></td></tr>
</table>
</td><td>
Specific design basis values for this repair at Oyster Creek are tabulated as follows:

<table>
<tr><th>Event</th><th>Across Core Plate and Below</th><th>Across Shroud Head</th></tr>
<tr><td>Normal/ Upset</td><td>16.1 psi</td><td>4.34 psi</td></tr>
<tr><td>Faulted</td><td>54.0 psi</td><td>19.0 psi</td></tr>
</table>
</td></tr>
<tr><td>
5.2.3.1 As a minimum, design analyses shall consider the effect of the repair assuming all circumferential welds intact and all circumferential welds completely failed through wall. In addition, other potential limiting failure configurations should be analyzed.
</td><td>
The analyses consider the pertinent cases as follows:

• Analysis for all welds intact (with and without repair).

• Analyses for all circumferential welds failed.

• Other cases as follows to demonstrate worst cases covered:

- H_5 (above bottom plate) failed
- H_{6B} (below bottom plate) failed
- Both H_5 and H_{6B} cases for (1) pinned and (2) sliding assumptions
</td></tr>
</table>

Requirement Per Reference 2	Oyster Creek Design
5.2.3.2 A sufficient number of structural load cases shall be performed to insure that: (1) Installation of the design repair does not adversely affect the existing structural integrity assuming no defective welds present. (2) Structural integrity is demonstrated assuming the horizontal welds covered by this repair are 360° through wall cracked. (3) An enveloping combination of cracked/uncracked welds is bounded by load cases (1) and (2) above, or specific analysis of the enveloping combination shall be performed.	Structural load cases are analyzed covering the required cases indicated in this item.
5.2.3.3 Modeling of assumed 360° through-wall cracks shall be consistent with the repair method utilized. For example, where assumed cracks are designed to separate under design loads, they may be modeled as roller joints (no shear/moment capability). Failed weld joints which are prevented from separation are modeled in a manner consistent with the design condition. The use of friction factors to model a cracked weld shall use a value of 0.2 or other value technically justified. Credit shall be given to lateral supports, where provided, in restraining shroud motion at assumed weld failures.	Modeling is consistent with this item; no credit has been taken for friction in assumed weld failures. Cases in which failed weld is assumed to carry shear loads are also included (see 5.2.3.1, above - the pinned condition).
5.2.3.4 Interfaces of the shroud repair within the reactor vessel, reactor vessel supports, shroud support structure and other internals, including fuel, shall be analyzed to demonstrate that interface loads and stresses are acceptable. This can be accomplished by demonstrating that original design basis interface loads are bounding, or by explicit analysis which demonstrates that original design basis allowable stresses are met.	Interfaces are analyzed including stresses on existing shroud/support structures. All interface stresses meet specified allowables. Use of 10 tie-rods minimizes the interface loads on the tie-rods, the shroud and the conical shroud support cone.

Requirement Per Reference 2	Oyster Creek Design
5.2.3.5 The effect of temperature differences between the shroud and the repair components shall be analyzed. Temperature differences which result in shroud repair elements being cooler than the shroud shall be evaluated. For example, a postulated loss of feedwater followed by restoration of feedwater without feedwater heating into the shroud annulus. In that case, annulus water is substantially cooler than the water within the shroud and this can cause differential shrinkage between the shroud and the repair structure. The limiting safety analysis event shall be used to establish the maximum temperature differences. For this event, analyses shall be performed to demonstrate that allowable stresses are met and that undue loads are not imposed on the shroud assembly. Potential loosening of repair elements during and/or after the event shall be addressed.	Pertinent temperature differences are analyzed including potential loosening of repair elements. The Oyster Creek repair is designed to elastically withstand a water temperature difference inside versus outside the shroud of 260°F (which is computed to be the maximum required temperature transient for Oyster Creek). This 260°F water temperature difference equates to an average effective metal temperature difference (shroud versus tie-rod) of 130°F. Special features are incorporated in the design to accommodate this 130°F temperature difference as discussed for Item 5.1.7.2. **These paragraphs contain proprietary information and are not for public disclosure. They are intentionally left blank.**
5.2.3.6 Analyses shall be performed to substantiate design preloads, where used, and any other loads introduced by the repair, and to demonstrate that shroud displacement limits specified in 5.3.1 are met for both normal, upset, and faulted load combinations for the postulated crack scenarios described above. Where preload is applied to the shroud, the analysis shall show that stresses imposed on the shroud are acceptable.	Design preloads are analyzed to appropriately limit displacements per Item 5.3 and stresses are shown to meet specified allowables.

Requirement Per Reference 2	Oyster Creek Design
5.2.3.7 The effect of the repair on the seismic design of the shroud, reactor vessel, reactor vessel supports and other internals shall be considered for the intact shroud as well as the assumed weld failures described above. Demonstration of seismic adequacy may be accomplished by seismic analysis methods consistent with the current plant licensing basis or by demonstrating that seismic interface loads are bounded by original seismic design basis loads. The effect of any significant gaps in shroud support elements shall be addressed.	The effects of the repair on the seismic analysis of the shroud, reactor vessel and supports and other internals, including fuel, are analyzed using a dynamic non-linear system which includes any effects of gaps at shroud support elements. The following is a summary of these results: • The response of an intact shroud is essentially unchanged because stiffness and mass changes are small. Accordingly, for an intact shroud the seismic analysis of the shroud, reactor vessel and supports and other internals, including fuel, are unchanged. • The response of a shroud even with all pertinent horizontal welds failed is such that the following results are obtained, even considering the effects of gaps at the tie-rod shroud radial supports: - Loads on the fuel are not increased (as compared to an intact and non-repaired shroud). - Stresses on the shroud, the reactor vessel and the repair components are all acceptable.
5.2.3.8 All thermal-hydraulic and structural codes utilized in the design analysis shall be appropriately benchmarked.	All requirements will be met.
5.2.3.9 New or improved calculational methods may be utilized by the designer. For these techniques, appropriate benchmark information to demonstrate that the method is conservative and bounding for the application, should be provided.	All requirements will be met.

Requirement Per Reference 2	Oyster Creek Design
5.3 <u>Functional Requirements</u> The designed repair shall ensure the displacements of any weld that experiences 360° through-wall cracking under all normal, upset, and faulted loading conditions does not exceed the values listed in 5.3.1. The values listed in 5.3.1 are established to: (1) ensure bypass leakage is limited such that there is no adverse impact on core power output and jet pump/recirculation pump NPSH, and (2) limit the deflection and deformation of internals to ensure components maintain their configuration to the extent that design control rod drive scram capability and design ECCS functions are not affected.	 These requirements will be met. See Item 3.2.
5.3.1 <u>Allowable Displacement of Shroud</u>	
5.3.1.1 <u>General Requirements</u>	
• The shroud repair shall be designed so that there is no separation of 360° thru wall cracking of the shroud welds during normal operation, as a minimum.	This requirement has been met.
• The design of shroud repair shall ensure that vertical, horizontal and rotational movement of a shroud with 360° thru-wall cracked welds does not impair the ECCS functions during the conditions covered by Section 5.2.	This requirement will be met.
• For jet pump plants, the design of shroud repair shall limit the vertical and lateral displacement of 360° thru-wall cracked shroud welds which are located at elevations below the jet pump inlets so that leakage from the shroud is within the capacity of the ECSS pumps to maintain the floodable volume in which the core can be adequately cooled in the event of a LOCA in the recirculation piping system.	Not applicable to Oyster Creek, which does not have jet pumps.
5.3.1.2 <u>Plant Specific Requirements</u> The maximum allowable shroud vertical and horizontal displacements, for currently licensed domestic BWR's are provided in GE report number GENE-771-44-0894 Rev. 1, "Justification for Allowable Displacements of the Core Plate and Top Guide - Shroud Repair." The basis for these allowable displacements is also provided in the referenced report.	 This requirement will be met.

Requirement Per Reference 2	Oyster Creek Design
5.4 Qualification of Critical Design Parameters Critical design parameters shall be identified and shall be qualified and documented to ensure that the parameters meet the design basis. Appropriate mockups shall be utilized and shall be designed to represent the configuration of the actual installation being mocked up as closely as possible. Differences between the mockup and actual installation shall be evaluated and the affect on the qualification shall be documented. Measuring devices used during qualification shall be calibrated and traceable to National Institute of Standards and Testing (NIST). As a minimum, qualification of critical design parameters shall include: • Preloaded or tensioned members • Critical dimensions or tolerances • EDM, Rotobroach or other machining process	These requirements will be met. No EDM, Rotobroach or other machining processes to be used on reactor internals at Oyster Creek as a part of the repair installation. (A limited amount of EDM cutting may be used to remove several instrumentation brackets which were used to support start-up FIV tests. These cuts will use a qualified process and will not introduce any significant debris into the vessel.)
5.5 Thermal Cycles The repair hardware shall consider the effects of thermal cycling for the remaining life of the plant. Analysis shall use original plant RPV thermal cycle diagrams. The design shall assume a number of thermal cycles equal to or greater than the number assumed in the original RPV design. Alternately, thermal cycles defined by actual plant operating data may be employed if technically justified. Using this thermal cycle information repair components and the repaired shroud shall be evaluated for fatigue loading along with any other design vibratory loads.	This requirement will be met.
5.6 Chemistry/Flux The design shall recognize the use of existing and anticipated water chemistry control measures for BWR's and shall consider the affects of neutron flux on any materials used in the repair.	This requirement will be met.

Requirement Per Reference 2	Oyster Creek Design
5.7 <u>Loose Parts Considerations</u> Repair hardware mechanical components shall be designed to minimize the potential for loose parts inside the vessel. All parts shall be captured and held in place by a method that will last for the design life of the repair	This requirement is met.
5.8 <u>Inspection Access</u>	
5.8.1 The repair design shall be such that inspection of reactor internals, reactor vessel, ECCS components and repair hardware is facilitated. The installed repair hardware shall not interfere with refueling operations and shall permit servicing of internal components.	The locations of tie-rods have been selected to avoid reactor vessel welds which will be inspected as part of vessel ISI. Figure 10 shows the placement of tie-rods relative to vessel welds. Repair hardware is accessible for visual inspection and/or for periodic preload checks.
5.8.2 All parts of the shroud repair shall be designed so that they can be readily removed and replaced without requiring destructive removal (i.e., no permanently installed pieces). This is to provide full access to the annulus area for other possible future inspections and/or maintenance/repair activities that may prove necessary in the future.	This requirement is met.
5.9 <u>Crevices</u> The repair design shall be reviewed for crevices between repair components and between repair components and original structures to assure that criteria for crevices immune to stress corrosion cracking acceleration are satisfied.	This requirement is met, including suitable venting of crevices to ensure refreshed water flow.
5.10 <u>Material</u>	
5.10.1 The proposed design shall use materials which are highly resistant to Intergranular Stress Corrosion Cracking (IGSCC) and Irradiation Assisted Stress Corrosion Cracking (IASCC) and be suitable for the reactor environmental conditions. Previous NRC staff positions on materials used in BWR reactor environments should be considered. Justification is to be provided by vendor for all materials used.	This requirement will be met.

Requirement Per Reference 2	Oyster Creek Design
5.10.2 Materials shall be manufactured in accordance with ASTM or ASME specifications using allowable stresses given in ASME Section III Class 1 Stress appendices (Section II Part D in 1992 and later editions). Other materials approved by ASME Code Cases and approved for use by Reg. Guide 1.85, "Code Case Acceptability, ASME Section III Materials," may also be used. Alternative materials not specified by ASTM or ASME are also permitted, provided they have demonstrated to be acceptable in the BWR environment. Design stress intensity and allowable stress values shall be established for the limiting design conditions consistent with the methodology of ASME Section III, Appendix III.	This requirement will be met.
5.10.3 Austenitic Stainless Steel which will be welded, shall meet the requirements of EPRI document #84-MG-18, "Nuclear Grade Stainless Steel, Procurement, Manufacturing and Fabrication Guidelines" for chemistry and heat treatment.	No welding used in the tie-rod design proposed.
5.10.4 Austenitic Stainless Steels that are used in repair components which do not require welding should meet the requirements specified in 5.10.3 above. Standard grades or "L" grades which do not meet the requirements of 5.10.3 above may be used if: a) No welding is performed on the material (does not include tack welding. b) The material is given a solution heat treatment followed by a water quench. c) The material's SCC resistance is verified by ASTM A252 practice A or E.	This requirement is met.
5.10.5 The use of austenitic stainless steel shall meet the requirements of Reg. Guide 1.44, "Control of the Use of Sensitized Stainless Steel."	This requirement will be met.
5.10.6 Alloy X-750 shall meet all the requirements specified in EPRI Document NP-7032, "Material Specification for Alloy X-750 for Use in LWR Internal Components, Rev. 1." Alloy X-750 shall not be used in the welded condition, unless SCC resistance has been demonstrated by slow-strain-rate testing under BWR environmental conditions. Extreme caution shall be used in specifying the use of spring-temper grade, due to the difficulty in preventing non-uniform cold work below the minimum specified level.	Inconel X-750 is used for a positioning spring for the upper radiant restraint. This positioning spring is not required for the design basis loads; however, the pertinent requirements of NP-7032 have been complied with.

Requirement Per Reference 2	Oyster Creek Design
5.10.7 Type XM-19 material may be used if: a) The material is solution annealed and water quenched. b) The materials SCC resistance is verified by ASTM A262 practice A or E. c) Materials in the hot-rolled or cold-reduced condition are demonstrated to be SCC resistant by slow-strain-rate testing under BWR environmental conditions.	These requirements will be met. No cold worked XM-19 is used. CERT testing will be performed on hot rolled material from the material used in the fabrication.
5.10.8 Austenitic stainless steel must conform to either Practice A or E of ASTM A262 for IGSCC susceptibility.	This requirement has been met; Practice E was met.
5.10.9 Materials which are not covered by the scope of ASME Section III Material Requirements or the original Design Code of Record will be considered by the NRC staff as alternatives to the Code and evaluated on a case-by-case basis.	No such cases involved in the design or manufacturing of the proposed tie-rods for the Oyster Creek shroud repair.
5.11 <u>Welding/Fabrication</u>	No welding used.
5.12 <u>Pre-Installation As-Built Inspection</u> The repair design shall specify the as-built dimensional tolerance that the repair will accommodate. For critical measurements a pre-installation dimensional check will be performed and reconciled with design tolerances.	This requirement has been met.

Requirement Per Reference 2	Oyster Creek Design
5.13 <u>Installation Cleanliness</u>	
The design shall minimize the in-vessel debris generation. Debris recovery methods shall be incorporated into the installation process consistent with amount and characteristics of debris generated.	This requirement will be met.
If any debris will remain in the vessel after installation, the debris shall be identified and its affect on plant components and fuel shall be evaluated. This evaluation will include, as a minimum: a) material identification of the debris b) effect of the debris on RCS chemistry c) ability of the debris to pass through the reactor core without depositing on zircalloy fuel rods (deposition will affect heat transfer in the core) d) probability and effect of debris blocking essential systems interfacing with the RCSs (pumps, valves, etc.) e) effect of the debris on other essential components in the plant and an assessment of whether or not the debris will increase the probability of short-term or long-term material degradation of essential components in the plant.	This requirement will be met.
5.14 <u>Pre- and Post-Installation Inspection</u>	
5.14.1 Welds that are structurally replaced by the repair will not require pre-installation or post-installation inspection.	This requirement will be met.
5.14.2 Existing reactor internal components utilized for repair anchorage shall be inspected prior to repair installation, as specified by the designer, to ensure the structural integrity of the anchorage.	This requirement will be met. For example, the bracket lug welds at the conical support and bottom of the shroud will be inspected to ensure structural adequacy of the tie-rod bottom end attachment via the hook. However, the design has margin in case any of these welds fail in service. Specifically, if either of these welds are intact, the attachment is adequate.
5.14.3 Each designer shall specify the inspections required for the entire repaired shroud assembly commensurate with design considerations and Code requirements applicable to the specific design.	This requirement will be met.

Requirement Per Reference 2	Oyster Creek Design
5.15 ALARA	
5.15.1 The design should utilize construction and installation techniques that minimize the radiation exposure to the workers using ALARA practices in all steps.	This requirement will be met.
5.15.2 The repair should be able to be installed remotely from the refuel floor with the vessel flooded to normal refueling levels. The repair hardware should include features which facilitate handling during installation and during subsequent inspection, adjustment and removal/replacement of components.	This requirement will be met.
5.15.3 The repair should minimize the amount of radwaste generated.	This requirement has been met.
5.15.4 The repair should minimize the radiation exposure to the plant workers in future repair and inspection operations involving the newly installed structures or activities that are affected by the new structures.	This requirement has been met.
6.0 CODES AND STANDARDS	
6.1 The repairs to the Core Shroud are to be performed as an alternative to ASME Code Section XI as permitted by 10 CFR 50.55a(a)(3). Use of an alternative to the code requires review and approval by NRC.	This requirement will be met.

Requirement Per Reference 2	Oyster Creek Design
6.2 Repair designs shall meet the individual plant FSAR and other NRC commitments for RPV internals mechanical design. Where commitments exist to meet the "intent" of ASME Section III, the following shall apply: • Allowable material properties shall meet ASME III Class 1 stress appendices (Section II Part D in 1992 and later editions). • Fabrication shall meet NG-4000 and welding shall meet ASME Section IX. • Examinations shall meet NG-5000 and NDE shall meet ASME Section V. • Materials shall meet ASME Section II or ASTM specifications. Alternatively materials listed in Code Cases approved by the NRC in Reg. Guide 1.85 may be used.	These requirements will be met.
6.3 The use of ASME Code editions and addenda not yet specifically endorsed by NRC (in 10 CFR 50.55a) will be evaluated by NRC on a case-by-case basis.	No such editions or addenda used in proposed shroud repair design.
7.0 FUNCTIONAL TESTING The design should be passive in nature and not subject to post-installation functional testing.	This requirement will be met.
8.0 QUALITY ASSURANCE PROGRAM Repair design fabrication and installation activities shall be conducted under a quality assurance program meeting the requirements of 10CFR50, Appendix B.	This requirement will be met.

Requirement Per Reference 2	Oyster Creek Design
9.0 DOCUMENTATION The following documentation shall be prepared and maintained as permanent records: • Design Specification. • Design Report (certified, if required by the original design specification or code). • 10CFR50.59 Evaluation. • Installation documentation package per existing administrative procedures. • Seismic analysis of the repaired shroud and affected internals (including fuel). • Safety analysis of the repaired shroud and affected internals (including fuel). • Suitability Evaluation as required by ASME Section XI.	 This requirement will be met.

4.2 Seismic Design and Analyses

4.2.1 Purpose. The purpose of the seismic design analyses is to demonstrate that in the event of a design basis earthquake, the shroud repair modification will (1) assure that the original shroud functional requirements are met for assumed 360°, through-wall cracks in any or all of the cylindrical welds in the stainless steel portion of the shroud, (2) assure that loads imparted to the reactor vessel, fuel and other affected internals are within acceptance criteria and (3) demonstrate that the addition of the shroud repair components does not significantly affect the seismic response of the intact (i.e., uncracked) shroud assembly, including the fuel.

4.2.2 Design Approach. The seismic design of the shroud repair utilizes the pre-loaded tie-rod assemblies to react vertical seismic (and other) loads and lateral seismic restraints (bumpers) to react horizontal and overturning loads, and to limit the displacement of the core at the core top guide and core support plate elevations to acceptable values. Lateral seismic restraints are also included at intermediate locations to prevent unacceptable lateral displacement of shroud shell sections between the top guide and core support plate in the event of multiple, assumed circumferential weld failures. Other features of the seismic design include the following:

- The lateral seismic supports transmit seismic reactions directly from the shroud to the reactor vessel wall at the elevations where the core loads are transmitted (at the top guide and core support plate) and at shell sections.

- The lateral supports are installed in multiple circumferential locations to assure acceptable loads on the vessel and shroud.

These paragraphs contain proprietary information and are not for public disclosure. They are intentionally left blank.

4.2.3 **Seismic Analyses**. Seismic analyses are based on two analysis methods:

A. Original Licensing Basis

The sizing of shroud repair components was based on the Oyster Creek seismic accelerations specified in Reference 3. The accelerations are identified as being applied at the center of gravity of the shroud, separator, fuel assembly. For licensing basis DBE (0.22g, Housner spectra), the accelerations are:

Vertical - 0.32G
Horizontal - 0.48G

Given the weight of the shroud, separator and fuel, these accelerations were used to determine equivalent static inertial loads and the resulting force reaction at the seismic supports for assumed failures of each shroud circumferential weld H1 through H7. The radial restraints were sized for these reaction loads. In addition, structural evaluations of the reactor vessel were performed for these seismic reaction loads. These evaluations showed that the vessel stresses met the allowables.

B. Upgrade Seismic Design Basis

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These paragraphs contain proprietary information and are not for public disclosure. They are intentionally left blank.

These paragraphs contain proprietary information and are not for public disclosure. They are intentionally left blank.

4.2.5 **Independent Design Review**. The overall seismic analysis approach and model described above, including the treatment of hydrodynamic mass, damping, and effective stiffness of supports, has been reviewed by two independent experts in this field who have concurred in the seismic analysis methodology. The final results of the Oyster Creek analyses will also be reviewed.

4.3 Stress Report

A stress report is being prepared and will be completed prior to restart. Loads and allowable stresses, including fatigue evaluation will be in Section A of the Stress Report. Also see appropriate section of the comparison to the BWR-VIP Design Repair Criteria in Section 4.1 of this SER.

4.4 Installation

The process for the installation of one tie-rod assembly is outlined below (see Figures 16 and 17). All of the activities which take place inside the reactor vessel will be monitored with an underwater video system.

- Perform gap measurements between the shroud and vessel at the location of each radial seismic support and between the existing brackets on the conical shroud support and the reactor vessel where the lower end of the tie-rod will be attached.

- Machine radial support members and lower hook of the tie-rods using input from gap measurements.

- Install tie-rod and rotate/translate to engage hook.

- Verify proper installation with video.

- Install outer sleeves with radial support members.

- Install the upper bracket at top of shroud.

- Install nut.

- Tension rod.

- Torque nut to approximately ~25 ft-lbs.

- Crimp locking cup.

- Disconnect handling tools.

- Visually inspect all fitups and crimps.

5.0 REFERENCES

1. GPU Nuclear letter to U.S. Nuclear Regulatory Commission dated August 24, 1994, Response to NRC Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors."

2. BWRVIP letter (J. T. Beckham)) to U.S. Nuclear Regulatory Commission dated September 13, 1994, forwarding "BWR Core Shroud Repair Design Criteria," Rev. 1, September 12, 1994.

3. Oyster Creek Final Design and Safety Analysis Report (FDSAR), Amendment No. 40.

4. GPUN submittal C321-91-2345 dated December 30, 1991 to NRC on new Site Specific Ground Motion Spectra.

5. GPUN submittal C321-93-2321 dated December 23, 1993 to NRC of new In-Structure Response Spectra developed by EQE.

6. NRC SER transmitted by NRC letter dated March 18, 1992 which approved new O.C. Ground Motion Spectra.

7. ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard", American Society of Civil Engineers, September, 1986.

8. Y. Aida, H. Niwa, Y. Sasaki and H. Katayama, M. Nakajma and T. Taira, "Vibration Analysis of BWR Core Fuel Assemblies", <u>Seismic Engineering - 1989</u>, Volume 182, page 171.

9. GPU Nuclear letter to MPR Associates dated June 7, 1994, "Design Criteria for OC Core Shroud Repair Clamps."

OYSTER CREEK REACTOR VESSEL

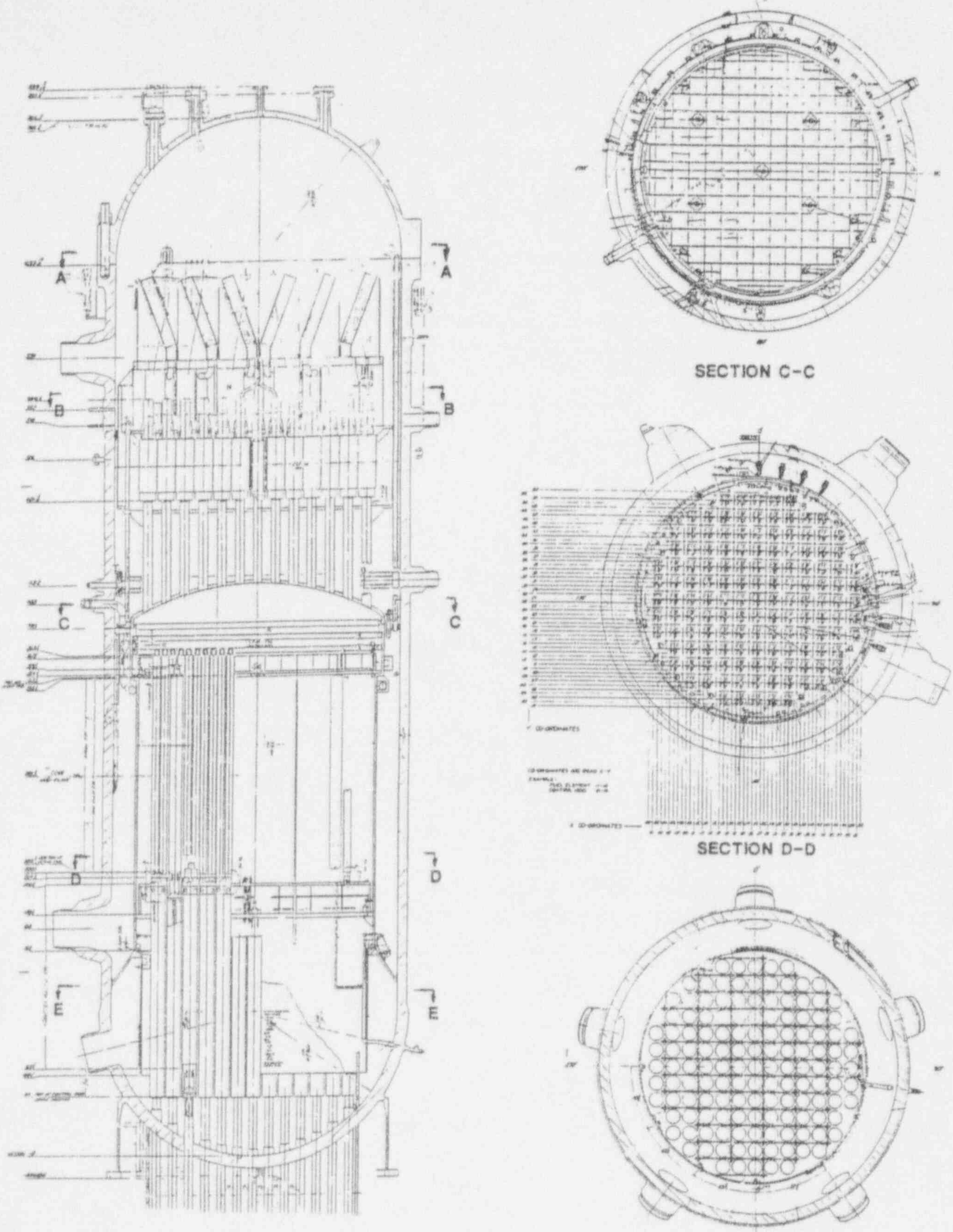

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FIGURE 1

OYSTER CREEK SHROUD

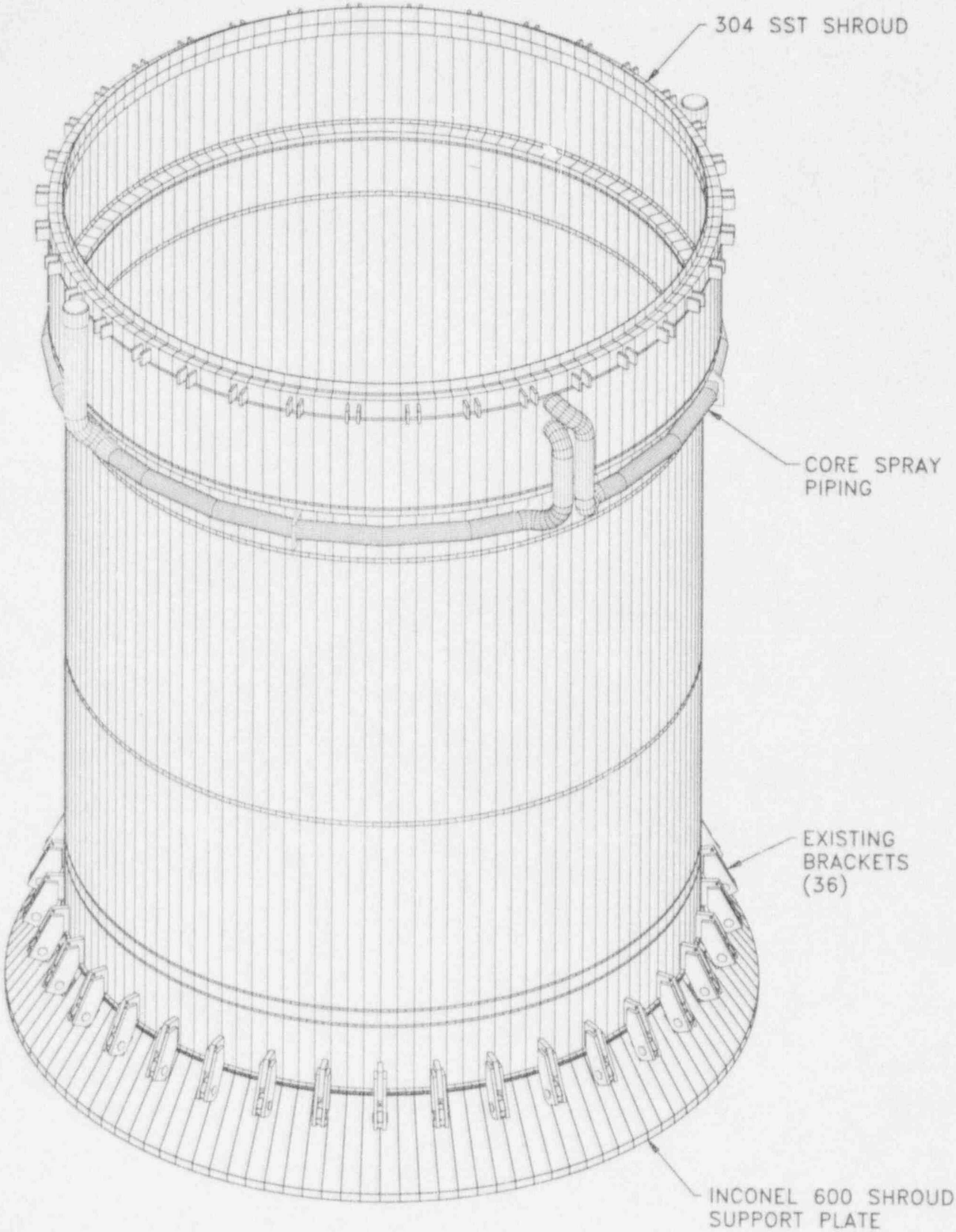

FIGURE 2

MPR ASSOCIATES

BRACKET ASSEMBLY
SHROUD TO CONICAL SUPPORT
OYSTER CREEK

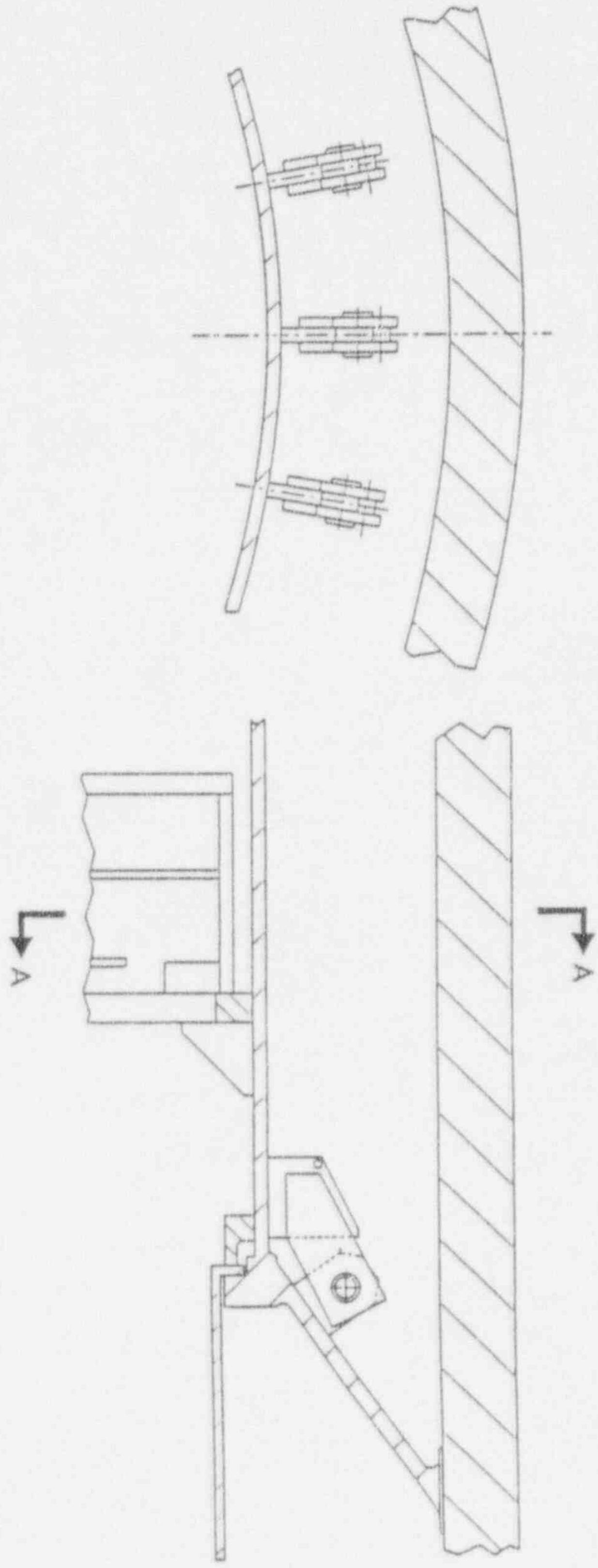

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FIGURE 3

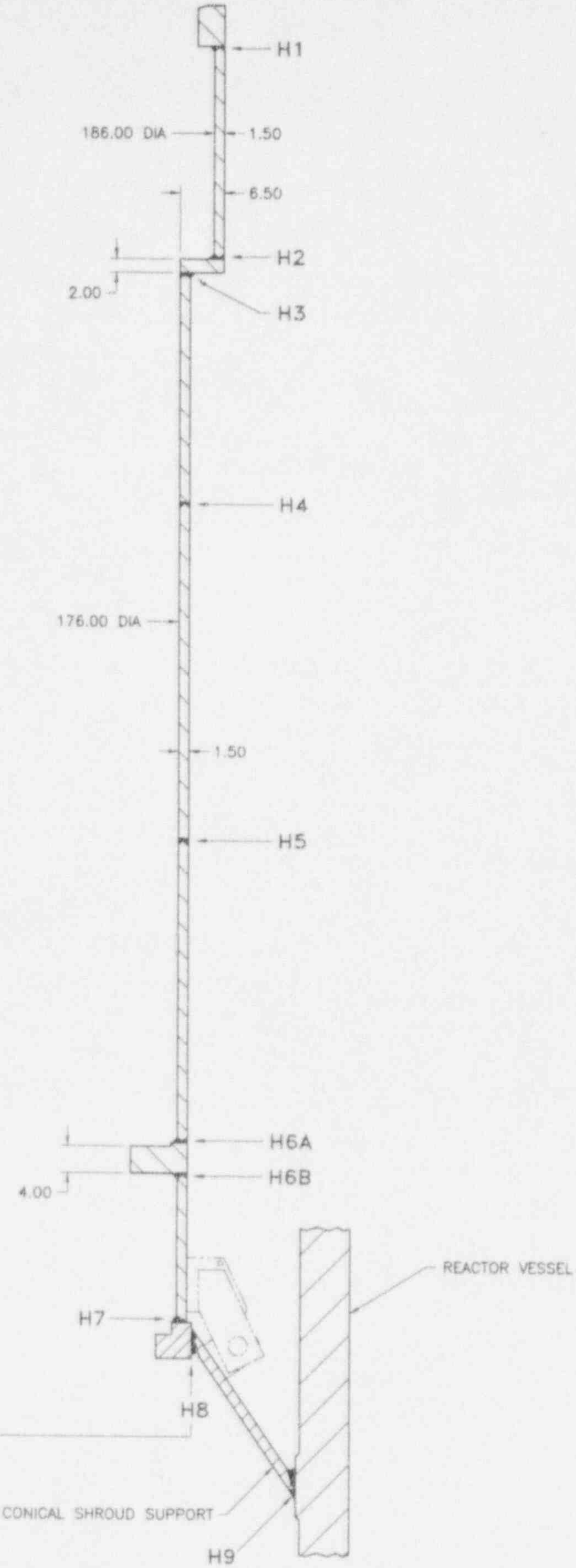

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FIGURE 4. OYSTER CREEK SHROUD HORIZONTAL WELDS

OYSTER CREEK SHROUD REPAIR

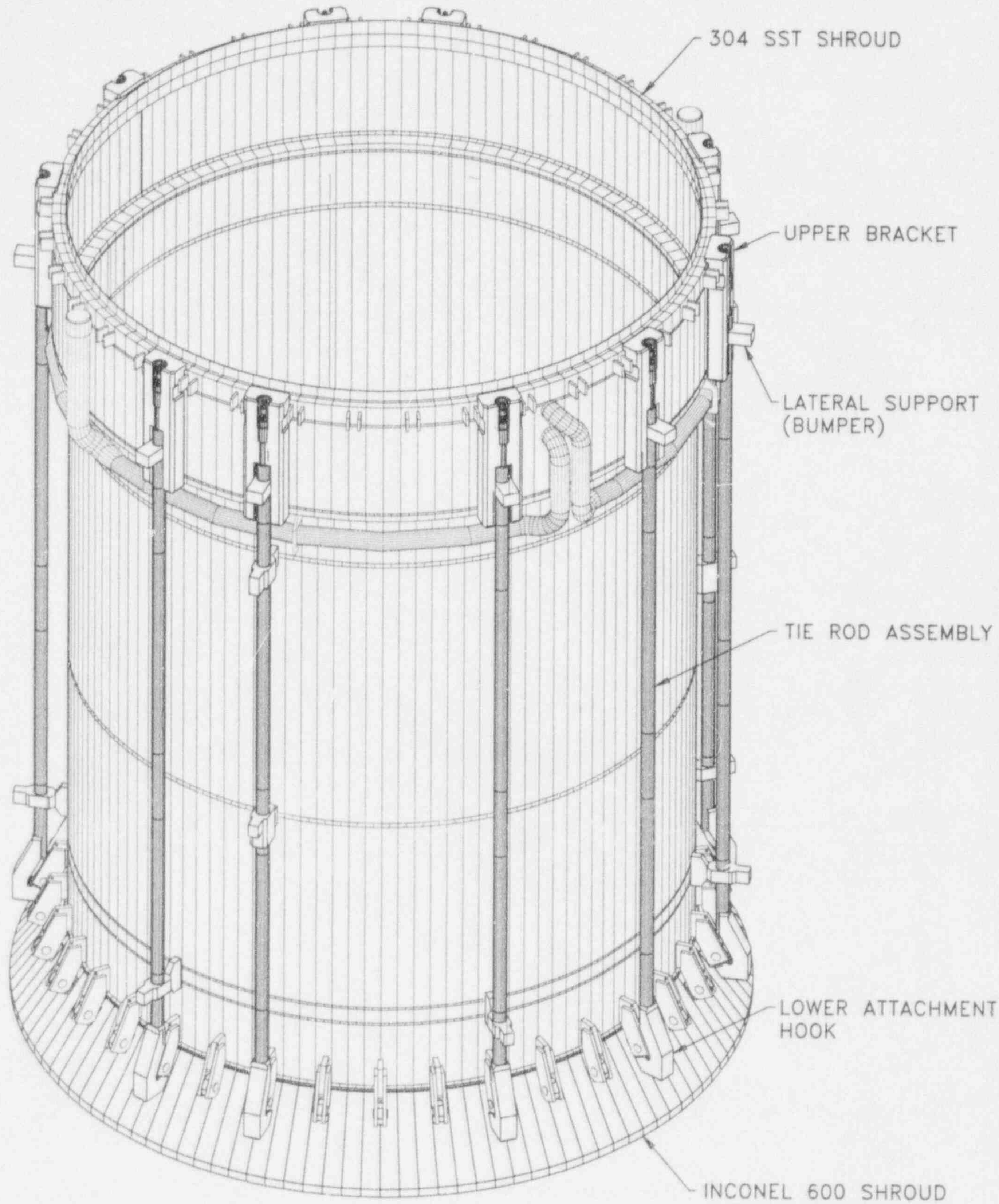

FIGURE 5

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FIGURE 6

SHROUD RESTRAINT ASSEMBLY
UPPER BRACKET

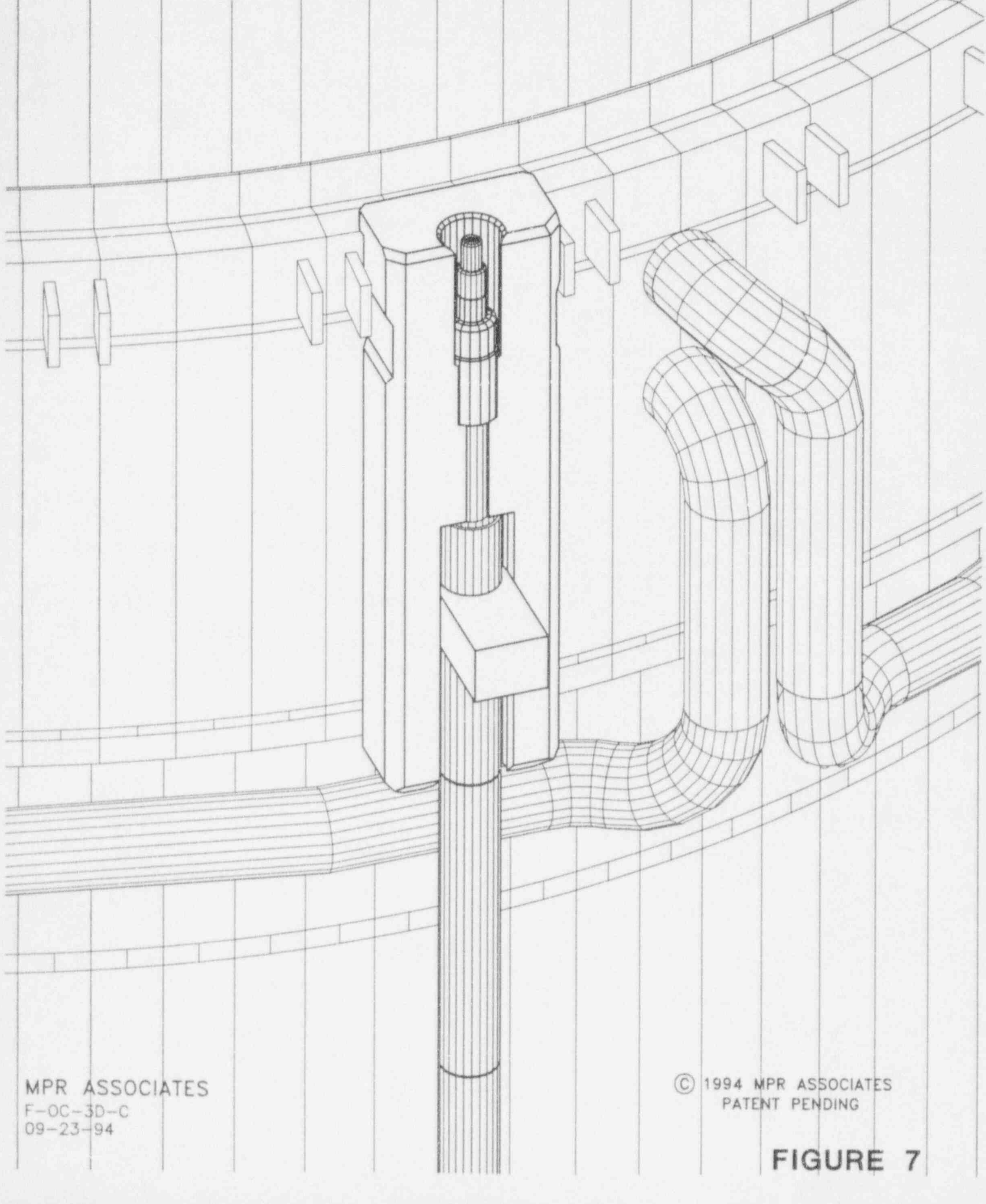

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FIGURE 7

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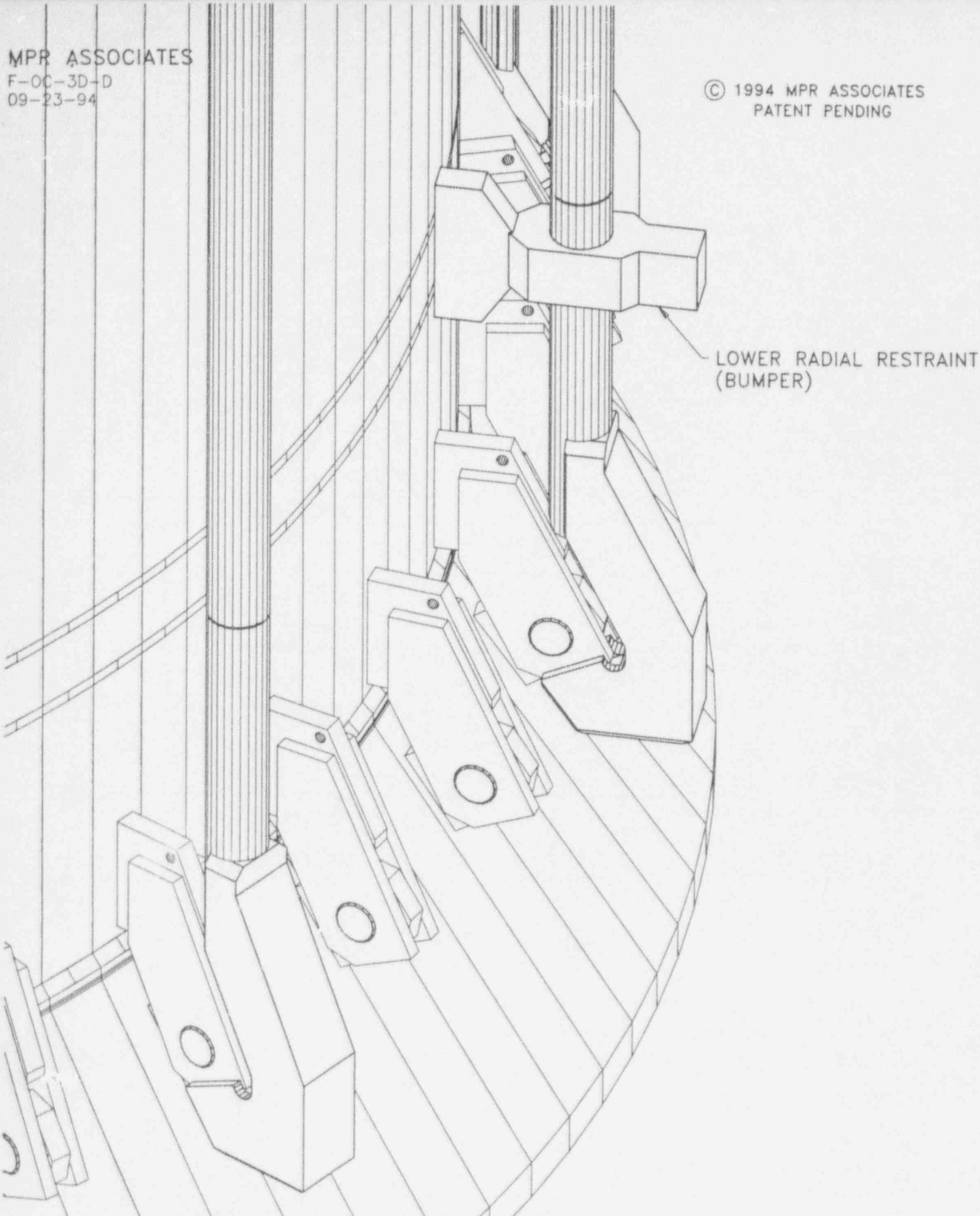

SHROUD RESTRAINT ASSEMBLY
LOWER HOOK

FIGURE 8

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FIGURE 9

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FIGURE 10

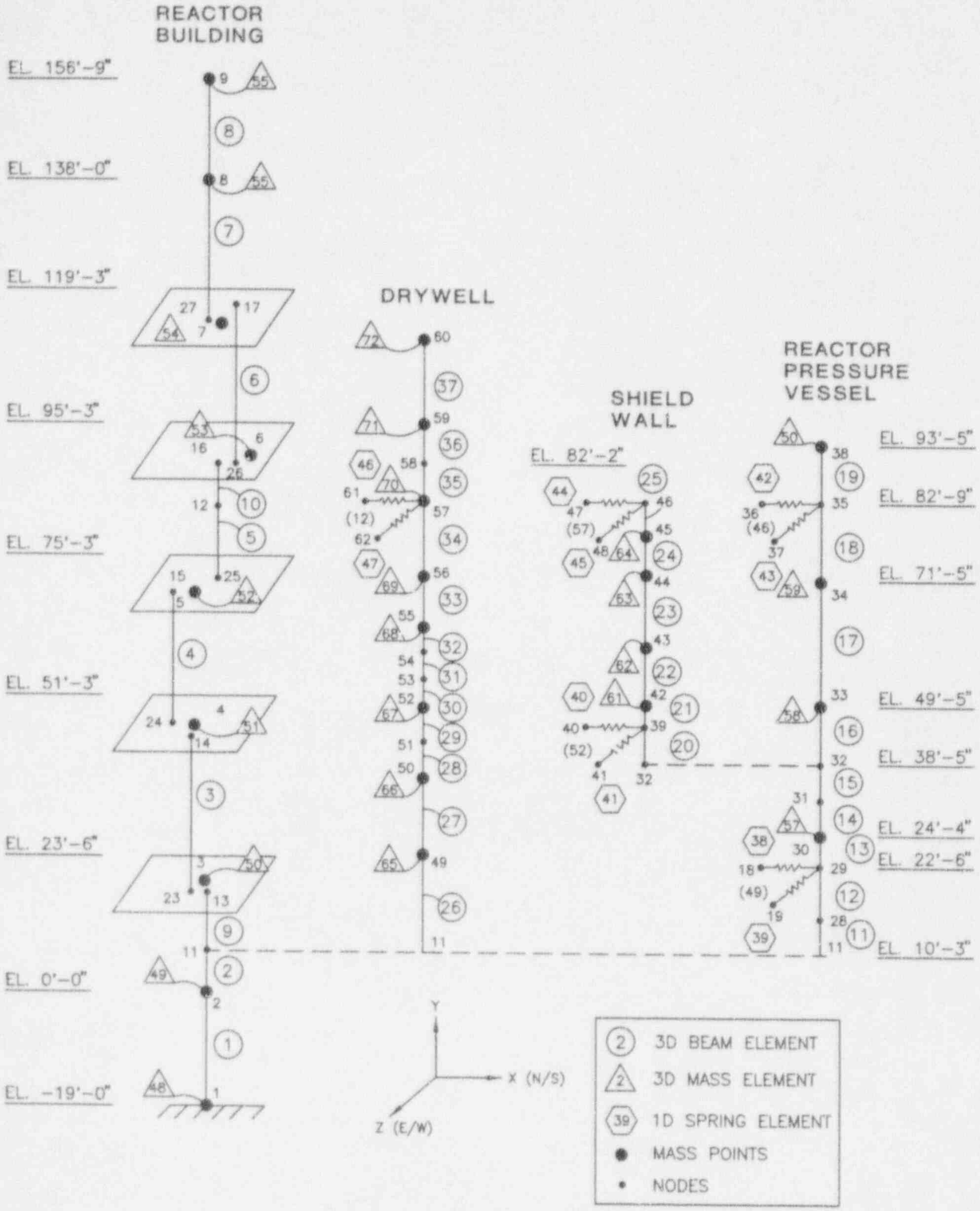

Figure 11. 3-D Coupled Model - Oyster Creek Reactor Building

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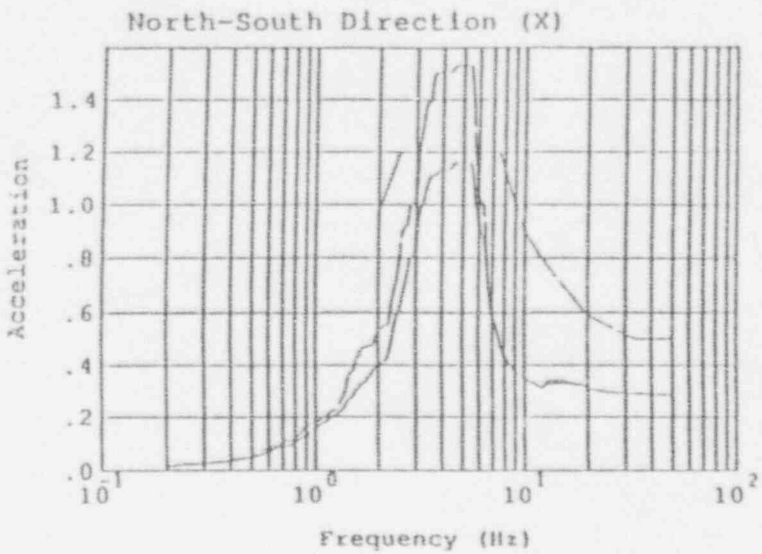

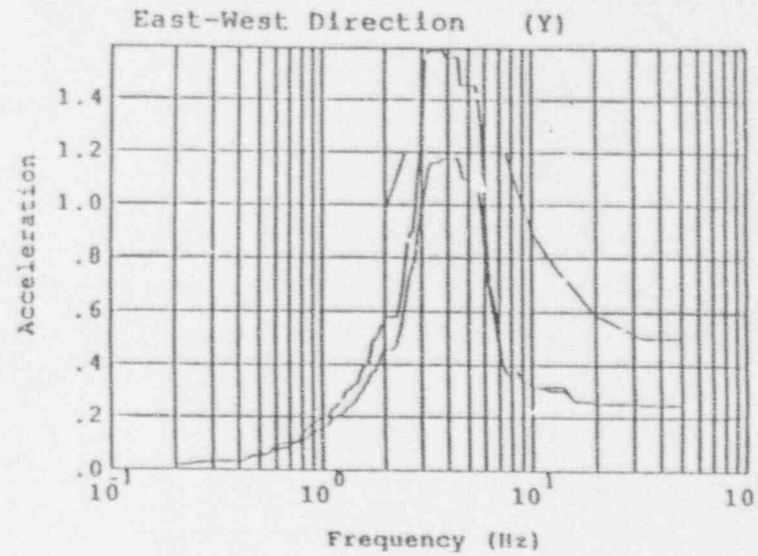

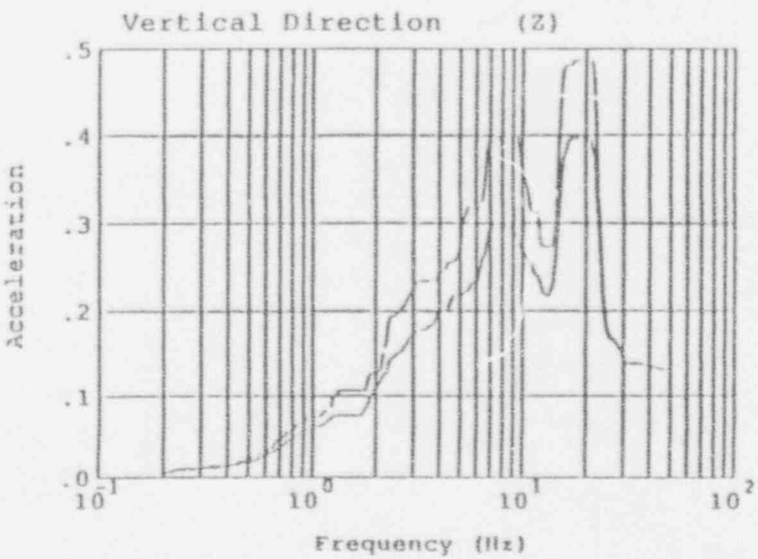

Legend:

Broadened 5% Envel. (1 Loc's-LB, BE & UB) ________________

Broadened 3% Envel. (1 Loc's-LB, BE & UB) ____ ____ ____

1.5 x SQUG Bounding Spectrum ___ ___ ___ ___

Notes:

1 SSE Level

Accelerations in g's

BE Case Broadened 15%

LB & UB Cases Broadened 10%

50124-C-213: GPU Oyster Creek Nuclear Generating Station, Design Basis
Reactor Pressure Vessel, Elevation 71'-5", Node 34

FIGURE 12

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FIGURE 13

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FIGURE 14

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FIGURE 15

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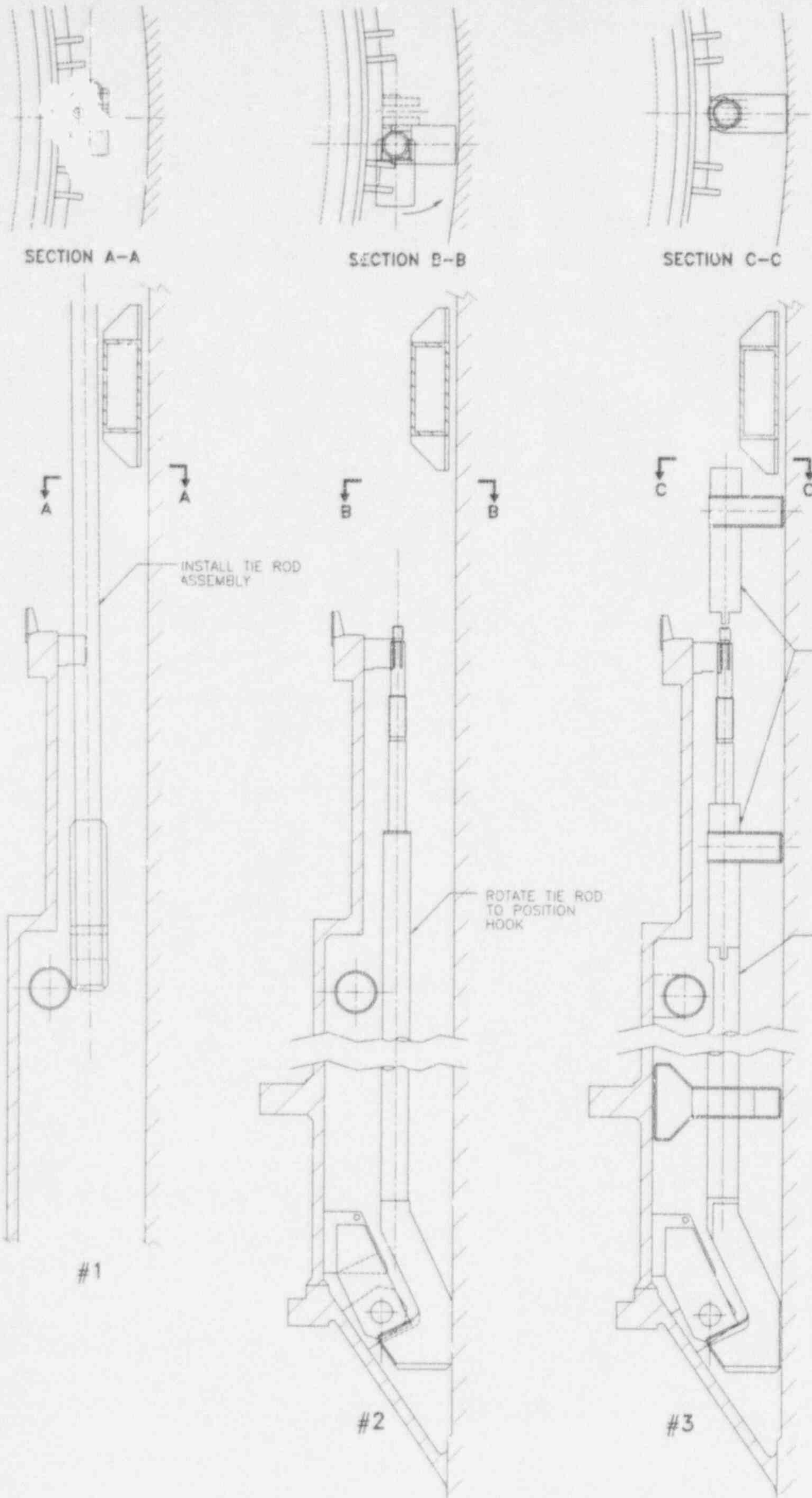

TIE ROD & OUTER SLEEVE ASSY INSTALLATION
FIGURE 16

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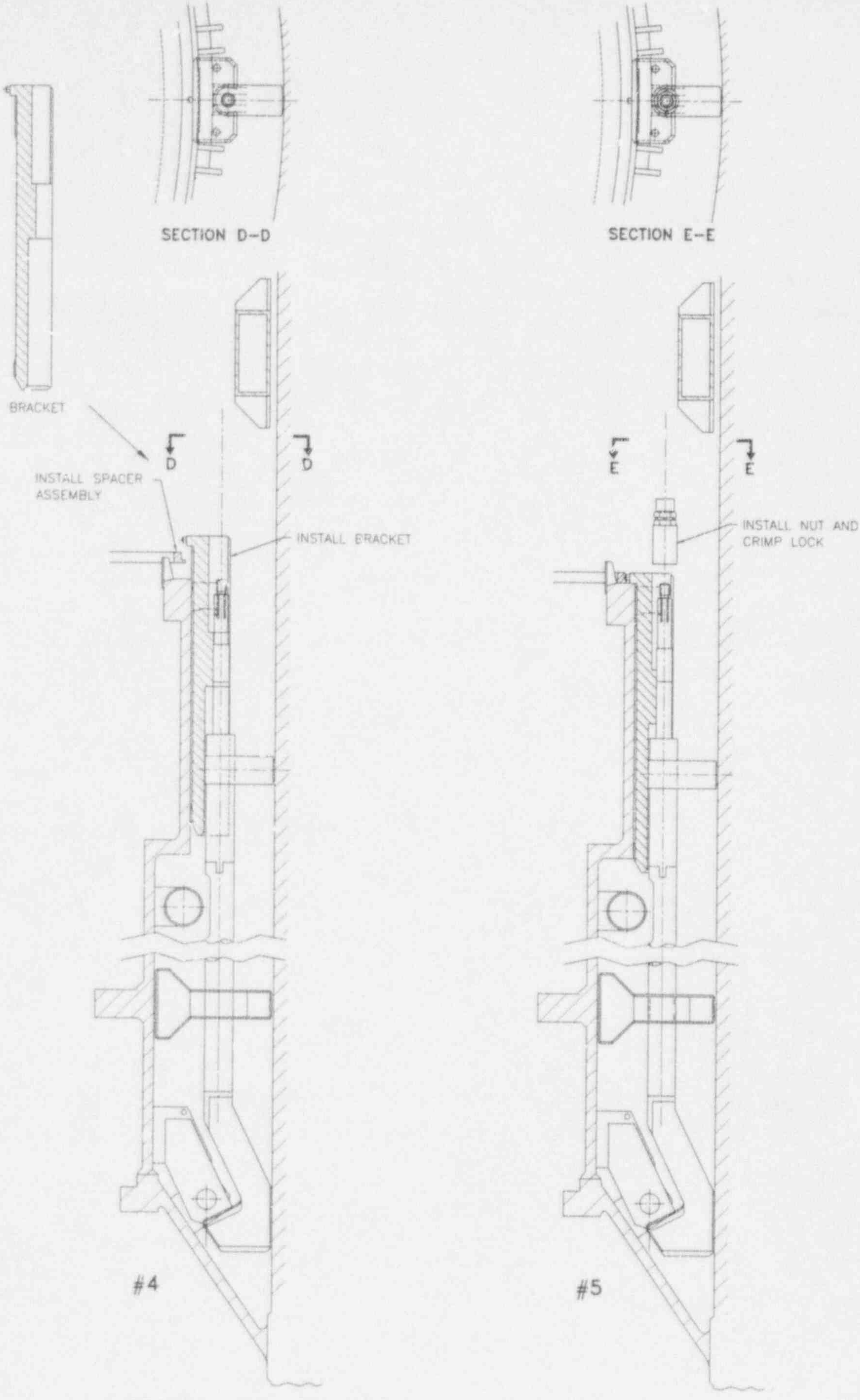


SPACER ASSY, BRACKET AND NUT INSTALLATION
FIGURE 17

Non-Proprietary Version

Attachment to
GPUN Letter
C321-94-2157

Design Specification For Oyster Creek Nuclear Generating Station (OC) Core Shroud Repair

Specification No. 083-9403-001
Revision 0

September 28,1994

Prepared By: Thomas Walker
Thomas Walker

Reviewed By: William McCurdy
William McCurdy

Approved By: Norman Cole
Norman Cole

Accepted By: Julien Abramovici
GPU Nuclear JULIEN ABRAMOVICI

9410270283

RECORD OF REVISION

Revision Number	Date	Reason For Revision
0	9/28/94	Initial Issue

This page is provided in lieu of a List of Effective Pages since all pages are re-issued with each revision.

TABLE OF CONTENTS

Section 1

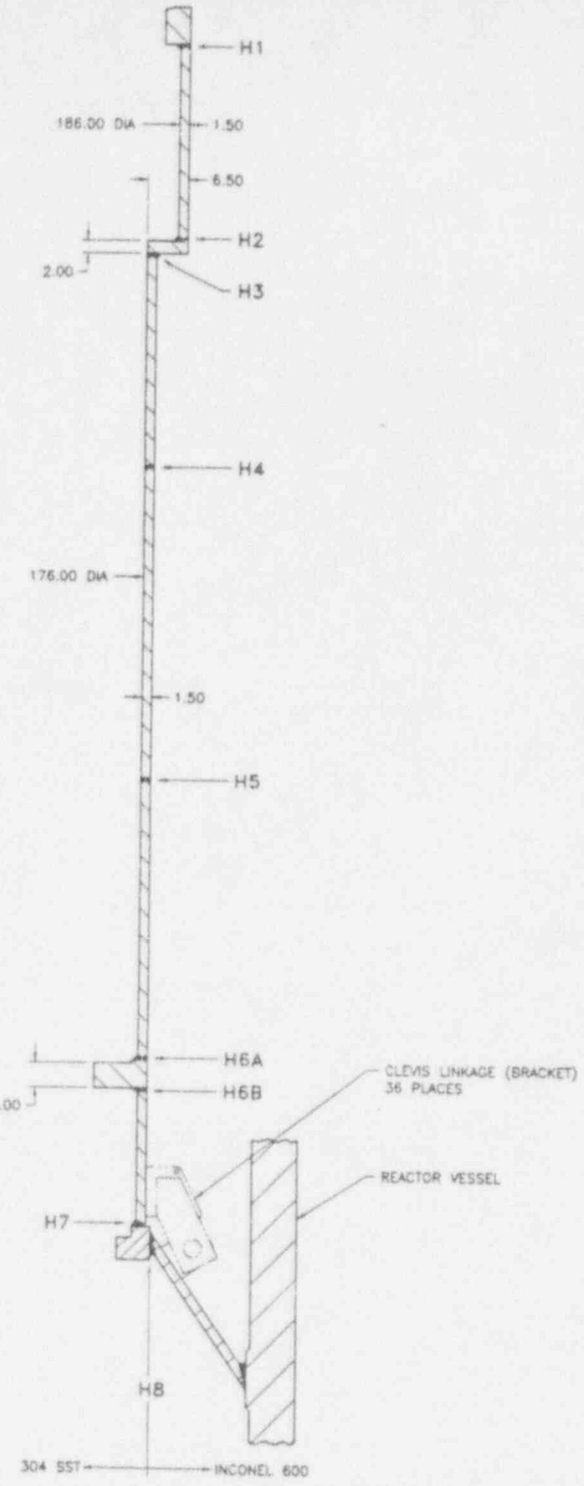
SCOPE

1.1 BACKGROUND

In response to General Electric (GE) SIL No. 572, Rev. 1, GPU Nuclear (GPUN) reviewed the fabrication and operational histories of the Oyster Creek (OC) Nuclear Generating Station to determine the potential for cracking in-core shroud welds. The review concluded that the OC plant had a fairly high risk of having core shroud cracking similar to that previously found by GE at the Brunswick Nuclear Plant. If core shroud cracks are identified during inspections at OC, then GPUN may elect to repair the shroud.

1.2 PURPOSE

The purpose of this specification is to define the functional and design requirements for modifications to the OC core shroud which would address shroud cracking in horizontal welds beyond acceptable limits. The purpose of the modification is to provide a repair which will allow the shroud to perform its operational and safety functions in accordance with the original design and licensing basis requirements.



OC CORE SHROUD WELDS
FIGURE 1

Section 2

QUALITY ASSURANCE REQUIREMENTS

The design of the core shroud repairs is classified as Nuclear Safety Related. Accordingly, the design shall be developed and documented per the requirements of 10CFR50, Appendix B and 10CRF21.

Section 3

CODES, STANDARDS, AND OTHER REFERENCES

The following codes, standards, and references apply to the design of the core shroud repair as specified in this specification.

3.1 ASME BOILER AND PRESSURE VESSEL (B&PV) CODE

1. ASME B&PV Code, Section III, 1989 Edition, no Addenda

3.2 MATERIAL SPECIFICATIONS/STANDARDS

1.	ASTM A262-91	Recommended Practices for Detecting Susceptibility to Intergranular Stress Corrosion Attack in Stainless Steel
2.	ASTM A380-88	Recommended Practices for Cleaning and Descaling Stainless Steel Parts, Equipment, and Systems
3.	ASTM A240-92b	Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels
4.	ASTM A312/A312M-92a	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
5.	ASTM A336/A336M-92a	Specification for Steel Forgings, Alloy, for Pressure and High-Temperature Parts
6.	ASTM A376/A376M-91	Specification for Seamless Austenitic Pipe for High-Temperature Central-Stration Service
7.	ASTM A479/A479M-92	Specification for Stainless and Heat-Resisting Steel Bars and Shapes for Use with Boilers and Other Pressure Vessels

8. ASTM B168-93 — Specification for Nickel-Chromium-Iron Alloys (UNS NO6600, NO6601, and NO6690) and Nickel-Chromium-Cobalt-Molybdenum Alloy (UNS NO6617) Plate, Sheet, and Strip

9. ASTM B637-93 — Specification for Precipitation Hardening Nickel Alloy Bars, Forgings and Forging stock for High-Temperature Service

3.3 GPUN - OYSTER CREEK NUCLEAR GENERATING STATION DOCUMENTS

1. Oyster Creek Nuclear Generating Station, Final Safety Analysis Report (FSAR)

2. Amendment 40 to Oyster Creek Application for Reactor Construction Permit and Operating License.

3. GPUN Technical Data Report No. 1138, "OC Core Shroud Evaluation," Revision 0, dated August 23, 1994.

4. Letter from A. Collado (GPUN) to S.J. Weems (MPR) dated June 7, 1994, forwarding "Design Criteria for OC Core Shroud Repair Clamps."

5. GPUN Memo #5411-93-0124 dated December 14, 1993, J. H. Segar to J. D. Abramovici - Core Shroud Fluence.

3.4 CODE OF FEDERAL REGULATIONS

1. 10CFR50, Appendix B — Quality Assurance Requirements for Nuclear Power Plants & Fuel Processing Facilities

2. 10CFR21 — Reporting of Defects and Noncompliance

3.5 NUCLEAR REGULATORY COMMISSION DOCUMENTS

1. Info Notice 93-79 — Core Shroud Cracking at Beltline Region Welds in Boiling-Water Reactors

2. GL 94-03 — Intergranular Stress Corrosion Cracking of Core Shroud in Boiling Water Reactors

3.6 GENERAL ELECTRIC DOCUMENTS

1. GE-SIL 5´ `. Rev. 1 — Core Shroud Cracks

2. GE Spec. 21A1105AF, REV. 0 — Reactor Pressure Vessel Specification

3. GENE-771-44-0894 — Justification of Allowable Deflections of the Core Plate and Top Guide - Shroud Repair

4. GE Spec 21A-5369 — Rev. 0 - Specification for Core Structure

3.7 COMBUSTION ENGINEERING DOCUMENTS

1. CENC-1143 — Analytical Report for Jersey Central Reactor Vessel

3.8 DRAWINGS

Drawing No.	Sht.	Rev.	Title
GENERAL ELECTRIC			
104R858	1	7	Reactor
104R858	2	8	Reactor
105E1413B	-	1	Shroud Data
107E6480	-	0	Shroud Head Bolt
112C2302	-	6	Bolt-Shroud Head Bracket
112C2303	-	6	Sleeve-Shroud Head Bolt
129B2777	-	1	Collar-Shroud Head Bolt
129B2778	-	3	Tee Bar-Shroud Head Bolt
129B2779	-	3	Retainer-Shroud Head Bolt
129B2782	-	2	Base-Shroud Head Bolt (Old)
129B3389	-	1	Shroud Head & Dryer Guide Rop
137C7868	-	1	Base-Shroud Head Bolt (New)
137C8174	-	2	Spacer-Shroud Head Bolt
158B7254	-	3	Clevis
167B3781	-	0	Retainer-Shroud Head Bolt
167B3782	-	1	Tee-Bar-Shroud Head Bolt

Drawing No.	Sht.	Rev.	Title
167B3784	-	1	Pipe-Shroud Head Bolt
167B3786	-	0	Head-Shroud Head Bolt
167B3788	-	1	Nut-Shroud Head Bolt
167B3789	-	1	Stop-Shroud Head Bolt
167B3791	-	1	Alignment Plate
213A8203	-	0	Pin-Shroud Head Bolt
213A8204	-	1	Spacer-Shroud Head Bolt
213A8205	-	1	Spring-Shroud Head Bolt
213A8206	-	2	Rod-Shroud Head Bolt
209A4759	-	2	Lug-Reactor
209A4760	1	2	Pin-Reactor
237E437	1	9	Spec Control Reactor Vessel
237E437	2	7	Spec Control Reactor Vessel
237E437	3	6	Spec Control Reactor Vessel
237E437	4	3	Spec Control Reactor Vessel
237E438	1	7	Loadings-Reactor Vessel
237E438	2	1	Loadings-Reactor Vessel
706E222		5	Shroud Head & Separators Assembly
706E230	-	1	Core Structure
706E231	1	7	Shroud--Core Structure
706E231	1	7	Shroud--Core Structure
706E231	2	3	Shroud--Core Structure
706E234	1	7	Top Guide--Core Structure
706E235	-	1	Core Support--Core Structure
706E247	-	2	Flow Baffle--Core Structure
795E766	1	2	Support Structure--Core Spray Sparger
795E766	2	1	Support Structure--Core Spray Sparger
846D690	1	3	Shroud Head Bolt
846D690	2	6	Shroud Head Bolt
885D647	-	0	Shield--Core Structure

Drawing No.	Sht.	Rev.	Title
885D672	-	0	Feedwater Sparger
COMBUSTION ENGINEERING			
231-583	-	3	Core Support Assembly & Details
232-560	-	1	Reactor Elevation
232-561	-	1	Reactor Plans
232-562	-	6	Bottom Head Forming & Welding
232-563	-	8	Vessel Forming & Welding--Upper
232-564	-	8	Vessel Forming & Welding--Upper
232-565	-	5	Nozzle Details
232-568	-	3	Bottom Head Penetrations
232-570	-	5	Vessel Final Machining
232-583	-	3	Core Support Assembly Details
232-587	-	4	Internal Attachments
232-595	-	1	As-Built Dimensions

3.9 BWROG DOCUMENTS

1. BWROG ---, Rev. 1 VIP-Core Shroud Repair Criteria Document

Section 4

GENERAL REQUIREMENTS

The functional requirements for the core shroud repair are based on the design bases that the core shroud provides as outlined in Reference 3.3.1. These requirements are as follows:

> The internal components of the reactor vessel (in conjunction with the reactor vessel) allow adequate core cooling to be maintained during normal operation and accident conditions and will not fail during normal operation and accident conditions. The reactor internals are designed to withstand a design basis earthquake.

> To meet the above, the internal components of the reactor are designed to:

a. Provide support for the fuel, steam separators, dryers, etc., during normal operation and accident condition.

b. Maintain required fuel configurations and clearances during normal operation and accident conditions.

c. Circulate reactor coolant to cool the fuel.

d. Provide adequate separation of steam from water.

e. Maintain reactor internal geometry (e.g., separate downcomer and core regions) to permit suitable control of reactivity.

Section 5

FUNCTIONAL DESIGN REQUIREMENTS

5.1 GENERAL DESIGN REQUIREMENTS

The design of the core shroud repair shall comply with the following general requirements.

1. The design life of the repair shall be 40 years.

2. The design shall allow for future maintenance of reactor vessel and internals components.

3. The repair design shall have features which ensure that all parts are captured so as to prevent parts from becoming loose.

4. The long-term impact of the repair on maintenance and inspection activities shall be considered as follows:

   a. The repair design shall be removable without damage to the repair components and/or existing vessel internals.

   b. The repair design shall be remotely installable and removable by long-handled tools—i.e., no robotic tooling shall be required.

   c. The repair design shall be such that if removal is required, all parts of the repair shall be readily removable and replaceable.

   d. The repair design shall allow for all refueling operations without removal or modification.

   e. The repair design shall minimize the impact on other in-vessel maintenance activities to the maximum extent possible (e.g., IVVI, Reactor Vessel ID Belt Line Weld Inspection, Shroud Inspection).

f. The repair design shall minimize future inspections and maintenance of the shroud repair components.

g. The repair design shall minimize the requirement for future inspections of the affected shroud joints.

5. The design of the repair shall satisty the following installation requirements:

a. The design of the repair shall provide installation tolerance to accommodate the as-found condition of the reactor vessel and internals. As-built measurements of critical features shall be taken at installation as needed.

b. The repair design shall be capable of being installed remotely from the refuel floor/bridge with the vessel/cavity flooded up to normal refueling levels.

c. The repair design shall minimize the amount of radwaste generated from the installation process(es).

d. Installation methods/processes shall minimize the potential for increasing the susceptibility of existing in-vessel components to cracking. The design shall consider known cracking phenomena (e.g., IGSCC, IASCC).

e. The design shall be such as to minimize work related radiation exposure of personnel at the plant site.

5.2 DETAILED DESIGN REQUIREMENTS

The design of the core shroud repairs shall comply with the following detailed design requirements.

1. The repair design shall address the failure of any circumferential shroud weld between one or more potentially sensitized 304 stainless steel shroud sections. As a result, the repair shall address the 100%, thruwall failure of welds H1 through H6B (see Figure 1), or any combination of failures of these welds, including all horizontal cracked welds. The repair design shall minimize the amount of material to be added to the internals to minimize impact on core cooling.

2. The repair design shall not use welded components.

3. The repair design shall have a negligible effect on the required flow characteristics in the downcomer annulus.

4. Accounting for friction and mechanical interlocking across cracked weld surfaces shall be consistent with the VIP Criteria Document (Reference 3.9).

5. The repair component(s) design/configuration shall minimize their susceptibility to cracking (e.g., crevice cracking).

6. The repair design shall include the capability of applying the necessary preload to prevent shroud separation during normal operation. The actual preload to be applied shall be based on OC loading conditions including differential pressure and dead-weight loads.

7. The repair design shall account for the effects of relaxation on preload, including the effects of end-of-life fluence on the materials.

8. The repair design shall consider the effects of thermal expansion on the structural integrity of the repair and existing components. Thermal expansion shall also be considered when evaluating the interface between the repair components and the existing vessel internals.

9. The repair design shall consider the effects of the repair on the seismic response of the reactor vessel and internals both in the intact or degraded conditions. Changes in response which would affect loads on the shroud, vessel or other internal components shall be addressed.

10. The qualification of the repair design shall include the following:

    a. Structural Analyses

        - Calculation of design loads for operating conditions upset condition, seismic, main steam line break, and recirculation line break.

        - Calculation of stresses in repair assembly and comparison with allowables.

        - Calculation of stresses in the core shroud and comparison with allowables.

        - Calculation of stresses in the reactor vessel and attachments and comparison with allowables.

        - Evaluation of fatigue resistance shall be performed for the repair components including impact of FIV (Flow Induced Vibration).

    b. Deflection Analyses

- Calculations to show that core spray system function and control blade function are preserved.

5.3 MATERIAL REQUIREMENTS

The repair design shall use corrosion resistant materials which are highly resistant to Intergranular Stress Corrosion Cracking (IGSCC) and Irradiation Assisted Stress Corrosion Cracking (IASCC) and which do not require the addition of a corrosion allowance on part dimensions. In addition, the materials selected (and their fabrication and installation) shall be suitable for use with the existing reactor water chemistry control measures (i.e., Hydrogen Water Chemistry).

1. Materials shall be in accordance with ASTM approved specifications. Materials may also be in accordance with ASME approved specifications.

2. Certified material test reports and traceability shall be required for all materials. In addition, material shall be controlled within the fabrication shops under a quality assurance program which has been determined to meet material traceability and nuclear safety grade manufacturing practices as required by the Code of Federal Regulations, Title 10 Part 50, Appendix B (Reference 3.4.1). Fabrication shall be in accordance with the requirements of the fabrication specification.

3. Materials shall be of the type indicated below. Other materials may be used with written prior approval by GPUN.

   a. Plate

      ASTM A240 Type 304 with the following additional requirements:

      - Tests for resistance to intergranular corrosion per ASTM A262, practice E.

      - Chemical product analysis[1]

   b. Pipe

      Seamless - ASTM A376 or A312, Type 304 with the following additional requirements:

      - Chemical product analysis[1]

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[1]This requirement can be omitted if a suitably reliable supplier is used.

- Tests for resistance to intergranular corrosion per ASTM A262, practice E.

c. Bar and Fasteners

ASTM A479, Type 304 with the following additional requirements:

- Tests for resistance to intergranular corrosion per ASTM A262, practice E.
- Chemical product analysis[1]
- S5 material for optimum corrosion

ASTM A479, Type XM-19, hot rolled with the following additional requirements:

- Tests for resistance to intergranular corrosion per ASTM A262, practice E.
- Chemical product analysis[1]
- CERT Testing

d. Disk Springs

UNS N07750 (X-750) to the requirements of ASTM B637 (or equivalent if plate is used) and the pertinent requirements of EPRI NP-7032 with specified chemistry and temper.

5.4 DESIGN INPUTS AND LOADS

1. The repair design shall allow for the operating environmental conditions listed below (see Reference 3.6.2):

a.	Design Temperature	575°F
b.	Operating Temperature	546°F

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[1]This requirement can be omitted if a suitable reliable supplier is used.

c. The end-of-life fluence levels (40 Effective Full Power Years) at the repair assembly radial position per GPU memo (Reference 3.3.5).

2. The following load cases shall be considered in the design of the repair, with the design meeting the loads for all the cases defined. These load cases correspond to ASME B&PV Code Section III Service Limits.

a. Normal (level A) — This includes normal deadweight, normal operating differential pressure, preloads and thermal loads (due to differential expansion of the repair assemblies)

b. Upset (level B) — This includes a combination of either of the following:

1. Normal Loads (not including thermal transient loads) + Upset Pressure (maximum upset differential operating pressure from Reference 3.3.4)

2. Normal Loads (not including thermal transient loads) + Operating Basis Earthquake (seismic moments from Reference 3.3.4)

c. Faulted (level D) — This includes a combination of the following:

Normal Loads (not including thermal transient loads) + Design Basis Earthquake (seismic moments from Reference 3.3.4) + Pipe Rupture (worst case transient loads as a result of any pipe break including maximum faulted differential operating pressure from Reference 3.3.4). This includes both steam-line and recirculation-line breaks. (Recirculation line break loads will be defined later).

d. Other — This includes a combination of normal/upset differential pressure and thermal transient (maximum shroud to repair assembly temperature difference) levels. Stresses in all repair assembly components shall be less than yield.

3. The following loads shall be considered in the structural evaluations:

a. Deadweight Loads - Based on the deadweights of vessel internals as determined from applicable drawings listed in Section 3.8 (see Appendix A).

b. <u>Differential Pressure Loads</u> - Differential pressures for normal, upset and faulted conditions, as specified in Reference 3.3.4 (see Appendix A).

c. <u>Seismic Loads</u> - The loads on the core shroud due to seismic loads, as specified in Reference 3.3.4 (see Appendix A).

d. <u>Thermal Transients</u> - The thermal transients to be considered in the structural evaluations of the repair design, as specified in References 3.3.3 and 3.3.4 (see Appendix A).

4. The material properties of repair component materials shall be as specified in ASME B&PV Code, Section III, (Reference 3.1.1) for Class 1 components. The material properties of the existing components shall be as specified in the original design specifications.

Section 6

ACCEPTANCE CRITERIA

6.1 STRUCTURAL ACCEPTANCE CRITERIA

The OC Core Shroud is not an ASME B&PV Code component. The following criteria shall be applied to the design of the shroud repairs.

1. Repair designs are not considered a ASME B&PV Code repair, however the design requirements provided for Class 1 components shall be used as a guide to the maximum extent practicable (Reference 3.1.1).

2. The stress intensity limits for the designed repair shall be as specified in the ASME B&PV Code, Section III, (Reference 3.1.1).

3. Material allowable mechanical properties shall be as specified in ASME B&PV Code, for Section III, Class 1 components (Reference 3.1.1). For code materials used in conditions not covered in Section II, Part D (e.g., XM-19 material in the hot-rolled or cold-worked conditions), allowable stresses shall be determined based on the specified room temperature material properties and the requirements in Reference 3.1.1. It shall be assumed that the variations of yield and ultimate strengths with temperature are the same as given in the pertinent tables in Reference 3.1.1 for a similar material (e.g., annealed XM-19).

4. The allowable lateral deflection (at normal operating temperatures) for the various load cases at the upper core guide location, and the core plate location shall be determined from Reference 3.6.3.

5. The allowable vertical deflections (and separation of shroud sections) at normal operating temperatures for the various load cases shall be as defined in Reference 3.6.3.

Section 7

DESIGN DOCUMENTS

The design of the core shroud repair shall be documented in the following design output documents.

1. Detailed design drawings, which show the arrangement of the installed repair as well as fabrication details of the repair components.
2. Repair component Bill of Materials (BOM).
3. A Design Report that documents the design methodology used, the parts of the ASME B&PV Code applied to the design, other specifications invoked and compiles all calculations, evaluations, and drawings applicable to the design.

Appendix A

CORE SHROUD DESIGN LOADS

1. Pressure Drops

	Pressure Drop (psid)	
Location	Normal/Upset	Faulted (MSLB)
Across Core Support Plate	16.09	54.0
Across Separator Head	4.34	19.0

2. Weights

Component	Weight (lbs)
Shroud Including Separator Head	188,000
Separater Head	110,000

3. Seismic Acceleration

	Acceleration (G)	
Seismic Event	Vertical	Horizontal
OBE	0.16	0.24
SSE	0.32	0.48

4. Average Metal Temperature Differences - Core Shroud and Repair Assembly

Operating Condition	Average Temperature Difference Core Shroud - Repair Components (°F)
Normal	20
Upset (limiting transient)	130