ATTACHMENT 1 to JPN-94-053

JAMES A. FITZPATRICK CORE SHROUD

REPAIR SUMMARY REPORT

Non-Proprietary Version

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

MPR
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James A. FitzPatrick Core Shroud Repair Summary Report

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Non-Proprietary Version

Prepared for

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CONTENTS

LIST OF FIGURES

1.0 PURPOSE AND SCOPE

New York Power Authority (NYPA) has developed a preemptive repair for shroud weld cracking for installation during the Winter outage at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) scheduled to begin on November 29, 1994. Inspection plans for this preemptive repair are described in Reference 15.

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

The purpose of this report is to describe, and to provide technical justification for, the design of the shroud repair in accordance with 10 CFR 50.55a(a)(3). This report provides an evaluation of the impact of the repair on the reactor vessel and reactor internals, including fuel, for both the intact shroud condition and for assumed shroud weld failures. The repair addresses possible cracking of any and all combinations of circumferential welds in the FitzPatrick core shroud which are subject to cracking.

New York Power Authority has contracted with B&W Nuclear Technologies (BWNT) to provide a tie-rod system to address potential cracking of circumferential welds in the FitzPatrick shroud. BWNT has contracted MPR Associates, Inc. who is responsible for the detailed design and analysis associated with the tie-rod system, as well as the tie-rod design specification, fabrication drawings, and licensing analysis associated with the tie-rod system. BWNT is responsible for material procurement, manufacturing of the tie rod system, installation tooling, mockup testing, and site installation.

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 FitzPatrick criteria with the BWR Owners Group Vessel Internals Project (VIP) specification, Reference 2.

- Description of seismic design and analyses.

- Description of Stress Report.

- Description of the installation sequence, tooling qualifications, training, ALARA, and schedule considerations.

2.0 DESCRIPTION

The existing FitzPatrick reactor vessel and shroud are shown in Figures 1 through 3.

The design of the FitzPatrick 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 plate/gusset structure. 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 vertical, lateral shear 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 stainless steel circumferential welds subject to cracking from the top of the shroud to the bottom (i.e., welds H_1 through H_7). Weld H_7 is considered subject to cracking, even though it is an Inconel weld, because it has a stainless steel HAZ. The Inconel welds (e.g., H_8 and others in the shroud support plate/gusset area) are not considered subject to cracking.

Figures 4 through 12 show the configuration and features of the proposed shroud repair for FitzPatrick. The design complies with the requirements specified in the core shroud repair design criteria prepared by the BWROG VIP committee (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 FitzPatrick licensing basis, and the load combinations and acceptance criteria for stresses meet BWROG VIP specification requirements.

D. The repair is considered permanent.

This paragraph contains proprietary information and is not for public disclosure. This is intentionally left blank.

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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 compared to a tie-rod system without lateral restraints and thus keeps the additional stress low in the shroud and in the support plate/gusset structure. These radial seismic supports are simple in configuration and fabricated from austenitic stainless steel material resistant to stress corrosion cracking. If the radial seismic supports were not used, the preload would have to be significantly increased (e.g., by a factor of about 15 at FitzPatrick) and this would result in significantly higher stresses, especially in the H_2/H_3 weld area and in the support plate/gusset structure.

This sentence contains proprietary information and is not for public disclosure.

I. Features have been incorporated in 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. These features allow the tie rod system to handle such transients without causing excessive loads on the shroud and the tie-rod attachment points.

J. Lateral movement of the shroud at the core top guide and lower support plate locations is positively restricted to less than 0.39 inches to assure control rod insertion under all accident and seismic conditions. The basis for this limit is discussed in Reference 16.

K. The tie-rod assemblies and other parts are all appropriately heat treated, low carbon, austenitic stainless steel (less than 0.03% carbon for 300 series stainless steel and less than 0.04% carbon for XM-19 material). These materials were also tested per ASTM 262 practice E and the material for key parts made from XM-19 material was further qualified by Constant Extension Rate Tests. In addition, no welding or heat treating is allowed during the manufacture of the tie-rod assemblies and special machining controls were employed to minimize the effects of cold work. As a result, the tie rod assemblies are highly resistant to IGSCC.

L. Only a minimum of modifications of the existing reactor internals are required to install the proposed tie-rod repair. Specifically, holes are EDM machined at the top of the shroud and through gusset plates at the bottom. The EDM process will be qualified to eliminate microcracking, minimize the recast layer and the heat affected zone.

M. A hook device on the tie-rods allows attachment of the bottom of the tie-rod to the gusset plate at the bottom, and a bracket device secures the tie rods to the top of the shroud.

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

N. The repair does not require any openings or through holes in the shroud, shroud separator head or shroud support plate. Thus, no leakage paths are created by the repair between the water volumes inside and outside the shroud.

O. 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-0612.

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

Q. 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.

R. 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.

3.0 REVIEW AND ANALYSIS

The BWROG Vessel Internals Project specification (Reference 2) covers the pertinent issues involved in connection with the subject repair. The VIP specification has been submitted for NRC review and is used as the generic basis for the determination that the repair is acceptable. Plant specific issues are covered in the repair design specification (Reference 3).

NYPA will perform a Nuclear Safety Evaluation in accordance with 10 CFR 50.59 prior to installation of this repair. No system design bases are affected by this repair because there are no significant changes to the reactor internals. In essence, the subject repair only increases the weight of the shroud plus fuel by less than 3% which is insignificant. The repair provides sufficient structural strength and rigidity such that a repaired shroud will meet the functional requirements specified for the original shroud, including seismic capability, control rod alignment and core spray pipe integrity. The effects of shroud leakage (through an assumed cracked weld) on Emergency Core Cooling and Reactor Shutdown have been evaluated and are concluded to be acceptable per Reference 5.

As indicated herein, the repaired shroud meets the pertinent requirements of the original shroud.

3.1 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. To aid in technical review, this report uses the same paragraph numbering system and requirements as stated in Reference 2.

Requirement Per Reference 2	FitzPatrick 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.	FitzPatrick is a BWR-4 which utilizes the shroud (to the elevation of the jet pump suction inlet) to provide 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 0.39 inches. 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. 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 0.39 inches.

Requirement Per Reference 2	FitzPatrick Design
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 buoyancy effects), differential pressure, and thermal-hydraulic loads.	Design loads per Reference 3 used in the tie-rod design are as follows: • Dead weight of shroud including head and separators--266,000 lbs. • Differential pressure across shroud support plate (See Item 5.2.2) • Hydraulic lift force on shroud--479,000 lbs,, (based on the pressure drops across the core plate and separators for uprated flow conditions, see Item 5.2.2)
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.	Design basis loads per Reference 3 used in the tie-rod design are as follows: • Differential pressure across shroud support plate--(see Item 5.2.2) • Hydraulic lift force on shroud--555,000 lbs., (based on the pressure drops across the core plate and separators for uprated flow conditions, see Item 5.2.2) • OBE loads on shroud: - Vertical--66,500 lbs - Horizontal shear at shroud support--758,000 lbs - Overturning moment at shroud support--1.75 x 10^8 in.lbs • Temperature difference of 130°F between shroud and tie-rods for 10 cold feedwater transients (3 cycles per event).

Requirement Per Reference 2	FitzPatrick 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 References 5 and 9, used in the tie-rod design are as follows: • DBE loads on shroud (Reference 3): - Vertical—133,000 lbs - Horizontal shear at shroud support—1.01×10^6 lbs - Overturning moment at shroud support—2.33×10^8 in.lbs • Recirculation pump suction line break (Reference 5) - Asymmetric load—205,000 lbs • Steam line break inside flow limiter - Up load on shroud at shroud support—990,000 lbs (based on uprated flow conditions, see Item 5.2.2)
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 structural evaluation of the repair considered both the design basis seismic loads given above and seismic loads calculated from dynamic analyses of the shroud. These evaluations addressed various shroud configurations including an intact shroud, the failure of all horizontal welds and the worst combination of failed welds. The current seismic design basis analysis (Reference 9) was also considered in the development of the seismic design loads.
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)

Section 3.1 (Continued)
BWR VIP Design Specification Comparison

Requirement Per Reference 2	FitzPatrick Design
4.0 SCOPE	
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 covers all pertinent welds in accordance with VIP spec. (See Figure 3.)
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 covers all pertinent welds in accordance with VIP spec. (See Figure 3.)
5.1.1 Repair Design Life	
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 remaining life of the plant—25 years (20 effective full power years).
5.1.2 Safety Design Bases	
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 Safety Analysis Events	
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 Load Combinations	
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)

Requirement Per Reference 2	FitzPatrick Design
5.1.5 Flow Partition 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 can be installed with sufficient preload to ensure cracked welds will not separate under normal operation. 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.
5.1.6 Flow Induced Vibration 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%. Calculated stresses are less than the 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 shroud support plate stresses are within design basis allowables and are minimized by the use of up to 10 tie-rod assemblies.

Requirement Per Reference 2	FitzPatrick Design
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 recirculation loop flow pressure drop.
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 fuel plus shroud assembly by less than 3%.

Requirement Per Reference 2	FitzPatrick Design
5.1.11 Power Uprate 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.	Margin is provided in the design of the repair assembly to accommodate uprated power and increased flow conditions. However, tie rod preloads are based on current design conditions.
5.1.12 Radiation Effects on Repair Design 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 Structural 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 Items 5.2.1.1 through 5.2.1.3.

Requirement Per Reference 2	FitzPatrick Design
5.2.1.1 Load Combinations Unless otherwise specified in the current plant licensing basis, the following load combinations cases should be considered: Case — Conditions (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 is met including the combination of Case (4) plus DBE.
5.2.1.2 Allowable Stresses 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 NG 1983 Edition with addenda through Summer 1984. In addition, the combination of normal loads plus DBE (Case 5) is conservatively designated as an emergency load combination.
5.2.1.3 Seismic Loads 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>FitzPatrick Design</th></tr>
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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.

Event	At Core Plate and Below	Above Core Plate
Normal		
Upset		
Emergency		
Fault		

</td><td>

The following conservative pressure drops based on uprated power and flow conditions were used in the design of the tie rod repair assemblies.

Event	Across Core Plate and Below	Across Shroud Head
Normal (Ref 3)	26.1 psi	7.8 psi
Upset (Ref 3)	24.6 psi	11.7 psi
Faulted (Ref 3)	26.5 psi	29.0 psi

The following pressure drops based on the current design power and flow conditions were used to establish tie rod preload requirements.

Event	Across Core Plate and Below	Across Shroud Head
Normal (Ref 5)	24.2 psi	7.2 psi

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Section 3.1 (Continued)
BWR VIP Design Specification Comparison

Requirement Per Reference 2	**FitzPatrick Design**
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.	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_7 (below bottom plate) failed
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).

Requirement Per Reference 2	FitzPatrick Design
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 shroud support plate.
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 FitzPatrick 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 FitzPatrick). 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 features minimize the load imposed on the shroud and assure elastic response of the tie-rods so that no loosening will occur as a result of this transient. The tie-rods have been designed to accommodate the transient without placing undue load on the shroud. (See Item 5.1.7.2)
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	FitzPatrick 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 horizontal seismic response 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 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.	These requirements have been met. Thermal hydraulic analyses were performed by General Electric using TRACG (Reference 5). ANSYS was used for some structural evaluations and for the seismic evaluation of the repaired shroud.
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.	These requirements have been met. No new methods were used.

Requirement Per Reference 2	FitzPatrick 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 are 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 (See Item 5.1.5).
• 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 has been met. As discussed, in Reference 14 the integrity of the core spray pipe is maintained during all service loadings, including a steam line break combined with a design basis earthquake.
• 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.	This requirement has been met. As discussed in Reference 5, Emergency Core Cooling would not be impaired by leakage through cracked welds during a recirculation line break.

Requirement Per Reference 2	FitzPatrick Design
5.3.1.2 Plant Specific Requirements 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.	Shroud horizontal displacements will be less than 0.39 inches. The basis for this requirement is discussed in Reference 16.
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	All of the processes used during the FitzPatrick shroud repair will be qualified on full scale mockups (see Section 3.4.1). Additionally, the measurement and tensioning equipment and processes will be qualified on the full scale mockup and will be traceable to NIST.
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 has been met. Specifically, the analysis used the original RPV thermal cycle diagrams (revised to address the plant power uprate condition) and the number of thermal cycles specified in the diagrams. The repair components and affected regions of the repaired shroud were evaluated for fatigue loading as well as design vibratory loads.

Section 3.1 (Continued)
BWR VIP Design Specification Comparison

Requirement Per Reference 2	FitzPatrick Design
5.6 <u>Chemistry/Flux</u>	
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 has been met. Specifically, the materials used in the repair are suitable for use with the existing and anticipated reactor water chemistry control measures. The effects of neutron flux on the stress relaxation of the repair materials has been addressed in the analyses.
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. Specifically, all parts of the tie-rod assemblies are captured and retained by appropriate locking devices. The locking device designs have been used successfully for many years in reactor internals. To minimize the possibility of debris falling into the core region, a core cover is used during tie-rod installation activities (see Section 3.4.2 of this report).
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. Specifically, the repair assembly can be removed in part or in total using the installation tooling. No permanently installed parts are used.

Requirement Per Reference 2	FitzPatrick Design
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. In the region where the hook is attached to the reactor vessel gusset plate, the loads and stresses are low such that there is no concern with stress corrosion cracking.
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 is met. All materials used in the design are highly resistant to IGSCC and IASCC. For a list of materials, see Section 5.3 of Reference 3 and Appendix B of Reference 14. For a further discussion of material use, see Section 2.0(K) of this report.
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 has been met. All materials are in accordance with ASME or ASTM specifications and allowable stresses are determined per ASME Section II Part D. For the hot-rolled XM-19 material, allowable stresses are determined based on the ASME Section III methodology. Design stress intensity and allowable stress are established for the design conditions consistent with ASME Section III methodology.
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 is used in the tie-rod design.

Requirement Per Reference 2	FitzPatrick Design
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 A262 practice A or E.	This requirement is met. Specifically, no welding is performed, the 300 series austenitic stainless steel material is used in the solution annealed condition and meets ASTM A262 practice A or E requirements.
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 is met. All austenitic stainless steel materials meets the requirements of Reg. Guide 1.44.
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 not used in the repair 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 are met. XM-19 material will be used in the solution annealed or hot rolled condition; no cold worked XM-19 is used. All XM-19 material meets ASTM A262 practice E requirements. In addition, the resistance to SCC of the hot rolled XM-19 material will be demonstrated by Constant Extension Rate Testing (CERT). The 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.

Section 3.1 (Continued)
BWR VIP Design Specification Comparison

Requirement Per Reference 2	FitzPatrick Design
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.	Material used in the repair is covered by the scope of ASME Section III material requirements except for hot rolled XM-19 and A336 annealed XM-19. The requirements for A182, which is a Section III material, and A336 are similar. Accordingly, A336 annealed XM-19 material is considered acceptable. See Items 5.10.2 and 5.10.7 for additional information on hot rolled XM-19 material requirements.
5.11 Welding/Fabrication	No welding is used.
5.12 Pre-Installation As-Built Inspection 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. The critical dimensions identified are the shroud to vessel annulus at each radial support location and the height of the shroud at each tie rod location. These as-found dimensions will be measured early in the repair process so that radial supports and tie-rod outer sleeve can be custom sized to accommodate the as-built shroud/vessel dimensions.

Requirement Per Reference 2	FitzPatrick Design
5.13 Installation Cleanliness 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 paragraph contains proprietary information and is not for public disclosure. This is intentionally left blank.** See Section 3.4.2 of this report for a discussion of the Core Cover Plate to be used to prevent debris from getting into the core region.
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. As discussed above, no debris will beleft in the vessel.
5.14 Pre- and Post-Installation Inspection	
5.14.1 Welds that are structurally replaced by the repair will not require pre-installation or post-installation inspection.	See Reference 15 for a discussion of the shroud weld inspections to be performed.

Requirement Per Reference 2	FitzPatrick Design
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 gusset welds at the shroud support plate and reactor vessel will be inspected with enhanced VT-1 examinations to ensure structural adequacy of the tie-rod bottom end attachment via the hook.
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.
5.15 <u>ALARA</u>	
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 has been met. ALARA practices have been used during the design of the repair components, tooling development and qualification, traveler development, and training. All installation activities will be performed from the elevation of the refuel bridge with the refuel pool full. All debris generated during the repair will be maintained under water at all times during the repair. **This sentence contains proprietary information and is not for public disclosure.**
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 has been met. The repair assembly components have been equipped with handling features which permit the tie rod installation (and removal) operations to take place from bridges over the refueling pool, with the water level at full height, using long handled tools.
5.15.3 The repair should minimize the amount of radwaste generated.	This requirement has been met. The minimization of radwaste was considered during the design phase of the project. ALARA practices will be used to minimize radwaste during the site installation.
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. Radiation exposure to the plant workers during future repairs and inspections has been considered in the design of the hardware and installation tools and in the development of the installation procedures. As discussed, above the tie rods may be removed from bridges over the refueling pool, with the water level at full height, using long handled tools.

Requirement Per Reference 2	FitzPatrick Design
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. Review and approval by NRC per 10 CFR 50.55a (a) (3) is the approach which will be pursued.
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 are met. As discussed in Items 5.10.2 and 5.10.7, all materials meet ASME Section II or ASTM specifications and material allowables are based on ASME Section II Part D. The parts are procured, fabricated and inspected in accordance with ASME Section III, Subsection NG.
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.	Except for the use of 1992 edition of ASME Section II, Part D for material allowable stresses (see Item 6.2) no such editions or addenda are used in the shroud repair design.
7.0 FUNCTIONAL TESTING The design should be passive in nature and not subject to post-installation functional testing.	This requirement has been met. No post installation testing of the repair assemblies is required.

Requirement Per Reference 2	FitzPatrick Design
8.0 QUALITY ASSURANCE PROGRAM Repair design fabrication and installation activities shall be conducted under a quality assurance program meeting the requirements of 10 CFR 50, Appendix B.	 This requirement will be met. All design, fabrication and installation activities are being conducted under a quality assurance program meeting the requirements of 10 CFR 50 Appendix B.
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). • 10 CFR 50.59 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. The specified documents have either been prepared or are in the process of preparation.

3.2 Seismic Design and Analyses

3.2.1 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.

3.2.2 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 bottom plate elevations to acceptable values. Lateral seismic restraints are also included at an intermediate location (at weld H4) 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 bottom plate) and where the shroud is stiffened.

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

- The supports are relatively rigid and provide positive restraint to limit the maximum possible displacement of the core, top and bottom, to values less than 0.39 inches, which meet the requirements of the BWR VIP design specification, Reference 2.

- Radial gaps are provided between the lateral supports and the vessel wall. These gaps are set based on as-built measurements taken on assembly and are small (3/8") relative to the shroud diameter (about 15'). The small gaps limit the seismic response of the shroud and core assembly and prevent resonant amplification of shroud motion. The effect of the gaps on shroud response, fuel loads and reaction loads are evaluated explicitly by means of non-linear dynamic analyses described in 3.2.3, below.

3.2.3 Seismic Analyses. Seismic analyses were performed for two sets of seismic inputs:

A. Equivalent Static Loads/Moments

Equivalent static seismic loads for the shroud were developed by GE for NYPA as part of BWROG activities in support of the core shroud cracking assessment. (Reference 6) These loads include horizontal shears and overturning moments at shroud welds H1 through H8 for the intact (uncracked) Fitzpatrick shroud for both

OBE and DBE cases. Vertical seismic loads were specified as 0.25g for the OBE and 0.5g for the DBE. The horizontal and vertical seismic loads were used to calculate seismic reactions at the lateral seismic supports and on the tie-rods for postulated failures of each shroud circumferential weld from H1 through H7. Vertical and horizontal components were combined on an absolute sum basis.

B. Nonlinear Dynamic Analysis

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3.2.5 Seismic Design Loads. Initial sizing of repair components was based on the equivalent static loads specified by GE and described in 3.2.3, above. The final stress report is based on the vertical loads specified in 3.2.3 in combination with the horizontal loads determined from the dynamic analysis of the failed weld cases.

3.2.6 Independent Design Review. The overall seismic analysis approach, modeling and results including the treatment of hydrodynamic mass, damping, and effective stiffness of supports, has been reviewed by an independent seismic expert, Dr. R. P. Kennedy, who has concurred in the seismic analysis methodology.

3.3 Stress Report

A stress report has been prepared (Reference 14) which provides structural evaluations of all repair assembly and affected shroud and reactor vessel components in accordance with the requirements of the design specification (Reference 3). A summary of results is provided in the following.

3.3.1 Load Combinations. The structural evaluations of the shroud repair considered the loads and load combinations specified in the repair design specification (Ref. 3). These load combinations, which are consistent with the current licensing basis, are summarized in Table 3.

As shown in Table 3, the shroud repair structural evaluations considered the loads due to a main steam line break or recirculation line break in combination with a design basis earthquake. There are no torsional load requirements in the Licensing Basis for FitzPatrick.

Table 3

Design Loads and Load Combinations

Load Case	Service Level	Load Combination
Normal: Operation	A	Normal Loads (including deadweight, normal operating differential pressure, tie rod preloads and normal thermal loads (due to differential expansion of the tie rods and shroud at normal operating temperatures)
Normal: Thermal Transient	A	Normal Loads + Thermal Transient Load
Upset: Pressure	B	Normal Loads + Upset Pressure
Upset: Operating Basis Earthquake	B	Normal Loads + Operating Basis Earthquake
Emergency: Design Basis Earthquake	C	Normal Loads + Design Basis Earthquake
Faulted: Steam Line Break	D	Normal Load + Design Basis Earthquake + Steam Line Break
Faulted: Recirculation Line Break	D	Normal Load + Design Basis Earthquake + Recirc Line Break

3.3.2 Repair Assembly Structural Evaluation. The tie rod assembly shown on the drawings listed in Appendix B of Reference 14 and installed with the planned preload satisfies the structural criteria for the repair specified in the repair design specification (Reference 3). Specifically,

- The Design by Analysis criteria of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG are satisfied.

- Stresses in the tie rod components will be less than yield during all normal and upset operating conditions, including anticipated thermal transients.

- The maximum fatigue usage in the tie rod assembly due to OBE and thermal expansion, (including startup and shutdown) loads occurs in the threaded section of the spring rods. The fatigue usage at this location is less than 8%.

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- The stresses in the tie rod resulting from vibration of the shroud during normal operation is less than 500 psi. As a result, the fatigue usage from shroud vibration is negligible.

3.3.3 Shroud Structural Evaluation. As specified in the shroud repair design specification (Reference 3), the stresses in the core shroud were evaluated to the stress criteria of the ASME B&PV Code, Section III, Subsection NG. The results of these evaluations show that the stresses are within code allowables for all design loads.

3.3.4 Reactor Vessel Structural Evaluation. The stresses in the existing shroud support plate, gusset and reactor vessel have been evaluated in accordance with the original design code. These evaluations are documented in Reference 14 as an addendum to the vessel stress report.

3.4 Installation

3.4.1 Mockups, Qualification, and Training. All tooling used for tie rod installation will be extensively tested and qualified underwater on full scale mockups prior to use in the field. Many of the personnel used during qualification will also serve on the site crew for tie rod assembly installation.

Several mockups will be used to test and qualify tooling and to qualify and train personnel for the tie rod assembly installation task. A training and qualification mockup has been constructed which is representative of a segment of the shroud and vessel in the area of one tie rod and upper bracket. The mockup is such that the complete installation and loading of a tie rod can be demonstrated underwater at full scale. This mockup represents an approximately 30 degree segment of the shroud and vessel. All pertinent local interferences (jet pumps, gussets, headers, etc) in the segment are included in the mockup. The mockup is transportable and when combined with a work platform of sufficient height, will be used for full height (approximately 80') training. The qualification mockup will be transported to site prior to the installation.

The Electrical Discharge Machining (EDM) process will be used to machine the tie rod attachment points in the 304 SST top shroud flange and in the Inconel 600 gussets at the lower shroud support plate.

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In addition to the qualification mockup described above, the EDM system will be tested in a pressure tank to simulate the water depth in the field. All process parameters are being qualified under full pressure conditions. The EDM process is being qualified to produce an EDM surface condition that results in no micro cracking, a minimum heat effected zone, and a minimal re-cast layer depth. These examinations will be performed under a magnification of 200X. The mockup inserts will be dimensionally inspected after EDM machining to verify anticipated results. Following the dimensional inspection process, the EDM mockup inserts for both the Inconel gussets and the 304 SST top shroud flange will be sectioned, polished, and inspected to verify required surface conditions.

A formal training plan is being developed for approval by NYPA that establishes the training requirements for the personnel designated to perform the actual field work activities. NYPA representatives will participate in or witness training activities. All personnel will be required to perform multiple cycles of hands on training with the actual field tooling and equipment prior to deploying to the site. During this training process, the field procedures will be used to ensure that personnel are familiar with the field procedures and equipment operating instructions and that the controlling field documents are thoroughly checked out. Satisfactory completion of training will be documented and provided to NYPA.

3.4.2 **Site Work Plan**. Extensive closed circuit video coverage will be used during the shroud repair task. This will include in air general overview camera systems and the capability to monitor all underwater processes/inspection remotely. Audio communications will also be provided at the actual work locations.

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Prerequisites

- The steam dryer and steam separator have been removed from the vessel and stored in a suitable location.

- Inspections of the shroud have been completed.

- Refueling water is at normal refueling height.
- The cavity filtration system (normal refueling cavity purification system) is in use and water clarity at the bottom of the shroud is acceptable for viewing.

Work Plan

- Install the auxiliary work bridge. All over-the-vessel work will be performed from two work bridges (the JAF refuel bridge and an auxiliary bridge) to allow parallel operations.
- Install the core cover (ventilated for core cooling). This device covers the entire top of the core shroud to prevent foreign materials from inadvertently entering the inside of the shroud. It will be marked at the tie rod locations to assist the crews in identifying the different installation locations.
- Setup equipment (EDM, video, debris system).
- A measurement tool will be used to determine the gap between the shroud and the vessel at each radial support member location such that the radial supports can be custom machined for each azimuthal and axial location. This will be done in parallel with the EDM of the gusset hole.
- At each tie rod assembly location, EDM a tie-rod attachment hole in the lower gusset plate and two holes at each location in the upper flange of the shroud. Two EDM stations will be operated in parallel, one from each work bridge.
- Machine radial support members (upper and lower) using input from gap measurements. Radial support member machining will be performed off of the job critical path.
- Video will be used to verify correct installation after each step of the installation process.
- Install the rod and engage the hook in tie rod attachment hole. Two installation stations will be operated in parallel. Installation through pre-loading will be completed on each tie rod assembly before beginning installation of the next one from that work bridge.
- Install outer pipes with bumpers (as applicable).
- Install bracket on shroud and tie rod assembly.
- Install nut, tension the tie-rod with a preload.
- Disconnect handling tools from tie rod and upper bracket.

- Crimp locking cup between nut and tie rod.

- Visual inspect all fitups and crimps. Visual inspect entire area between shroud and vessel for debris (closeout).

- Remove equipment.

- Remove core cover.

- Remove work bridges.

3.4.3 Schedule/Staffing/ALARA.

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Dose on this task will be minimized using proper ALARA techniques. It is assumed that the general area dose rate on the refuel floor will be 2 to 4 mrem per hour. The expected dose for this task will be 2 to 4 man rem. More refined dose estimates will be made as surveys are performed and information is updated.

4.0 SUMMARY AND CONCLUSIONS

NYPA has developed a contingency repair to cover the possibility of weld/heat affected zone (HAZ) cracking of the stainless steel circumferential shroud welds at the FitzPatrick plant. Based on experience of others, these welds/HAZ may crack at FitzPatrick.

The repair design is evaluated herein and is concluded to be adequate to restore the shroud to its original design basis, irrespective of any expected circumferential weld/HAZ cracking including any combination of such cracked welds. These evaluations have considered pertinent requirements based on the following:

- Outage inspection plans for cracking.

- BWR Owners Group generic design repair criteria, submitted to NRC (Reference 2).

- Design specification developed to cover generic and plant specific requirements.

Accordingly, the subject repair is considered adequate.

5.0 REFERENCES

1. New York Power Authority 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 BWRs."

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. MPR Associates "Design Specification for James A. FitzPatrick Nuclear Power Plant (JAF) Core Shroud Repairs," Specification No. 291001-001, Revision 0, dated September 22, 1994. (Also included in Reference 14.)

4. James A. FitzPatrick Updated Final Safety Analysis Report.

5. FitzPatrick Nuclear Station Shroud Safety Assessment, GENE-523-A136-0994, dated October 11, 1994.

6. General Electric Reports GENE-523-154-1093 and General Electric Letter GLS 94-04, Screening Criteria.

7. GE Analysis 383 HA 876, Rev. 0, "Fitzpatrick Seismic Analysis of Reactor Pressure Vessel and Internals," August 1972.

8. Stone and Webster Report JPN-92-053, Attachment 2, "Procedures and Criteria for Generation of In-Structure Response Spectra, and James A. Fitzpatrick FSAR Update," Revision 0, July 1982.

9. GE Analysis 383 HA 876, Rev. 1, "Seismic Analysis of Reactor Pressure Vessel and Internals," June 1977.

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

11. GE Report DAR-139, "Seismic Analysis of Edwin Hatch-I Nuclear Plant."

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

13. Hatch Unit 1 FSAR update, Rev. 3, Section 12.6.2.1.3, 1985.

14. MPR Associates, "James A. FitzPatrick Nuclear Plant Core Shroud Repair Design Report," Rev. 0, October 1994.

15. Invessel Visual Inspection Program - Design Basis Document, JAF-RPT-NBS-01848, Rev. 0.

16. Core Shroud - Core Plate Displacement Report, JAF-RPT-NBS-01849, Rev. 0.

FITZPATRICK REACTOR VESSEL

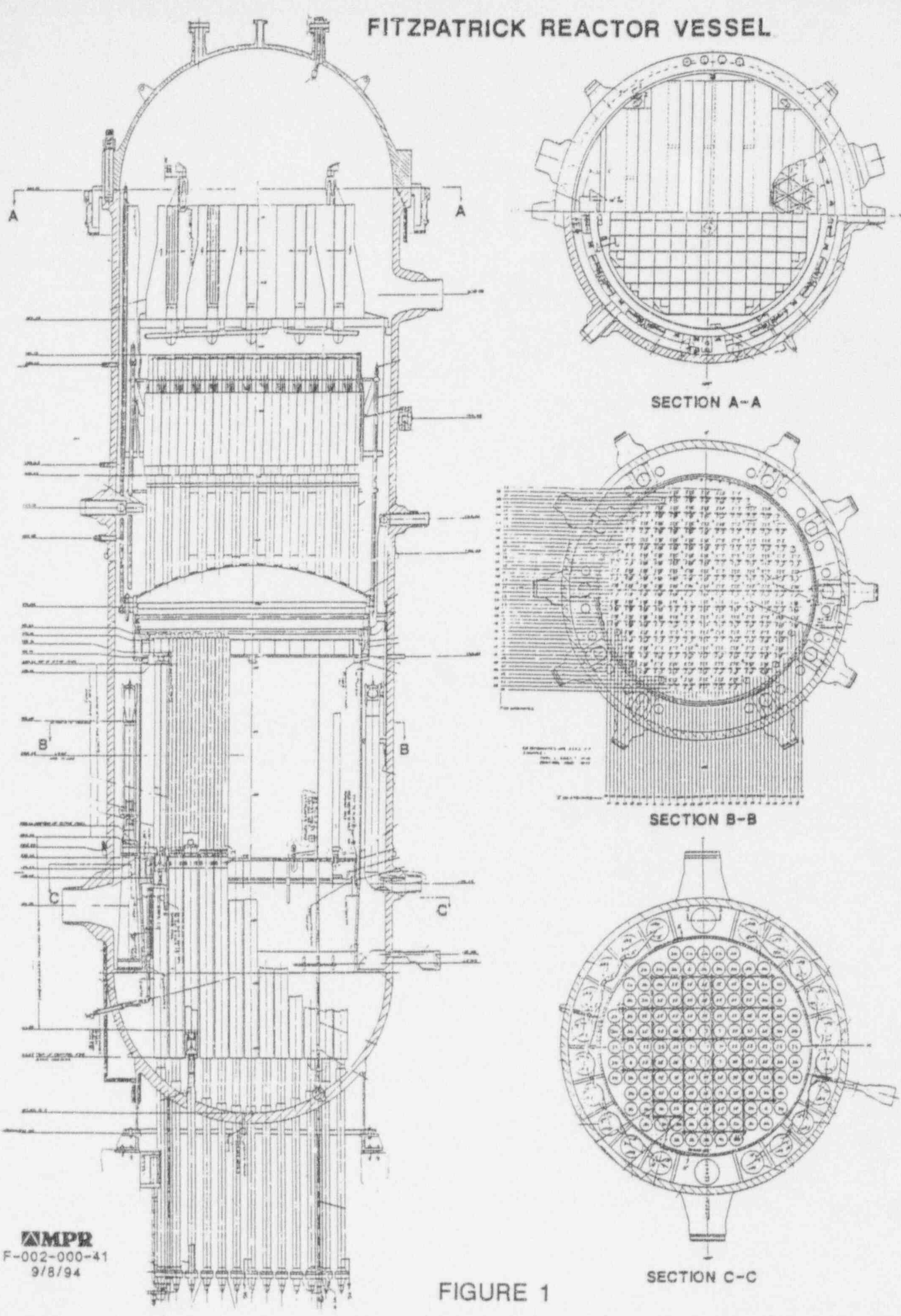

FIGURE 1

FITZPATRICK SHROUD

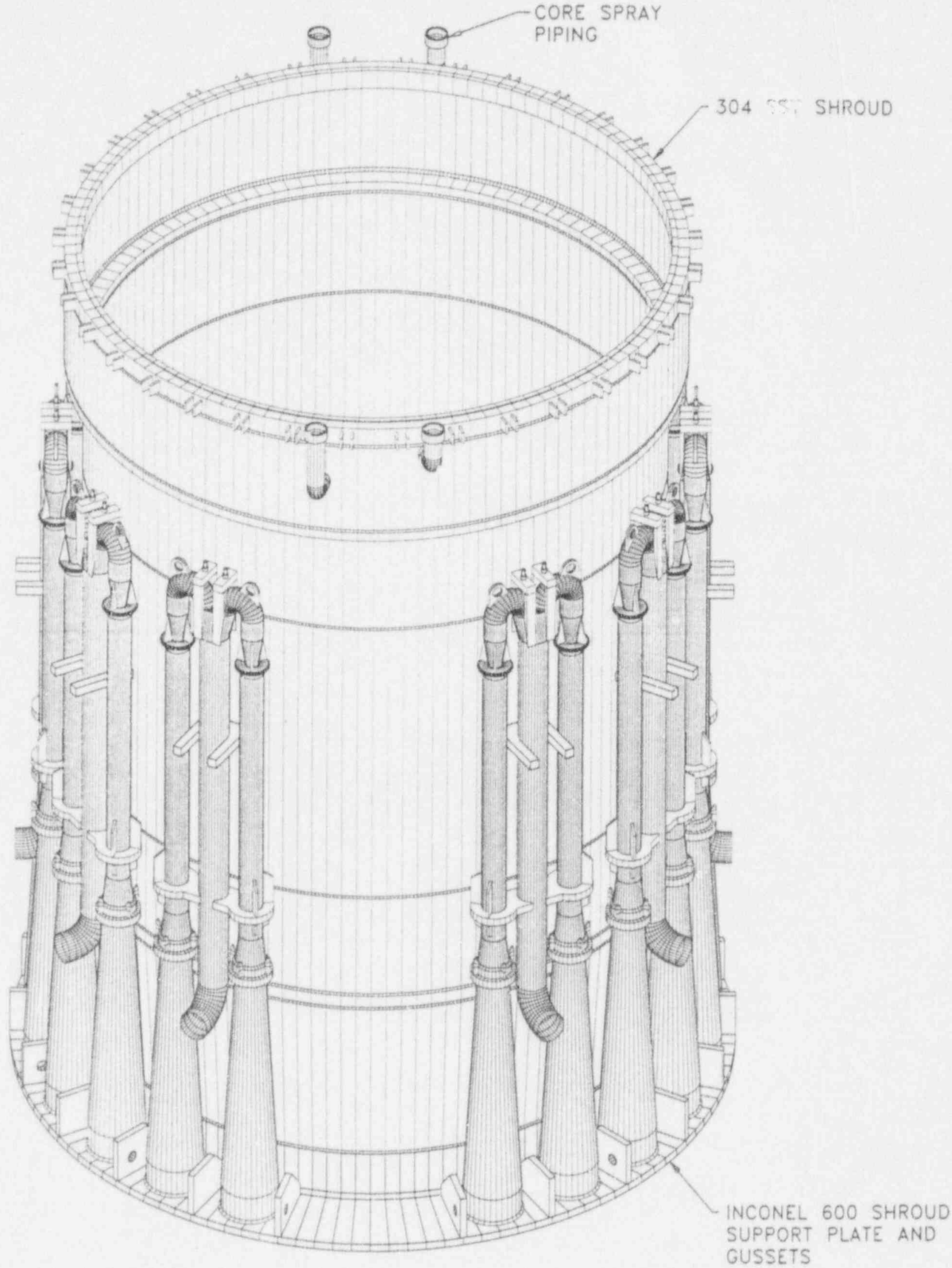

FIGURE 2

MPR ASSOCIATES
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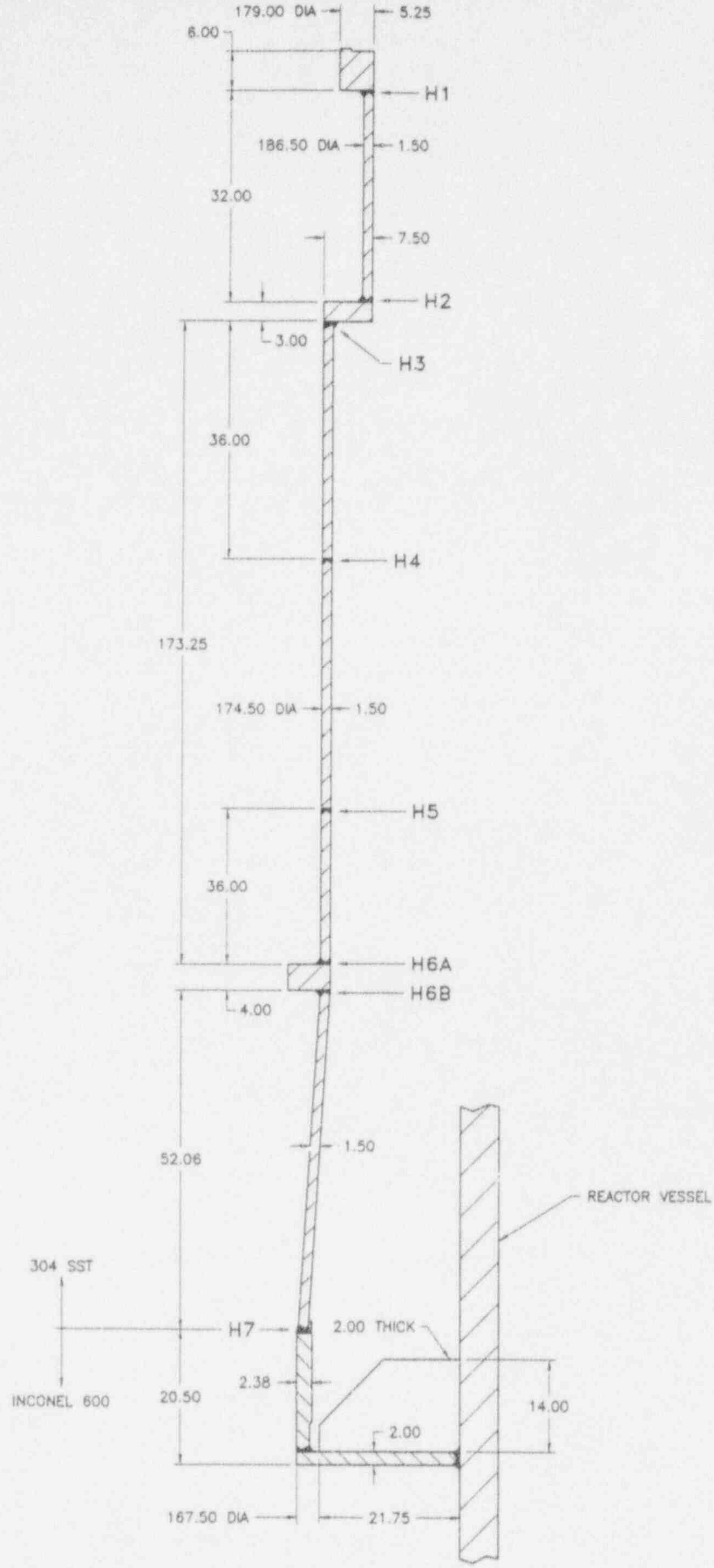

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10/14/94 (118)

FITZPATRICK SHROUD - HORIZONTAL WELDS

FIGURE 3

FITZPATRICK SHROUD REPAIR

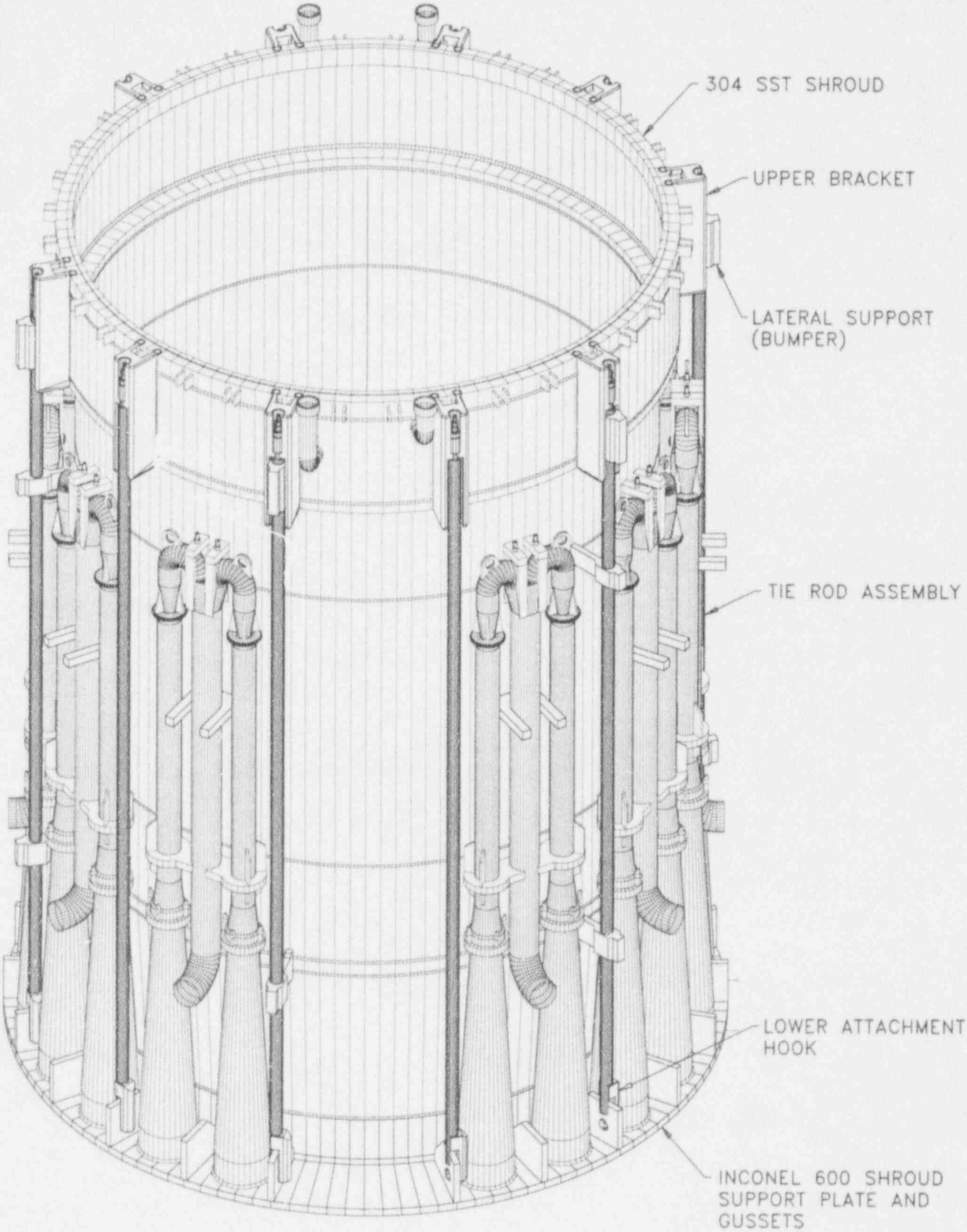

FIGURE 4

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FIGURE 5

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FIGURE 6

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FIGURE 7

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FIGURE 8

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FIGURE 9

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FIGURE 10

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FIGURE 11

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FIGURE 12

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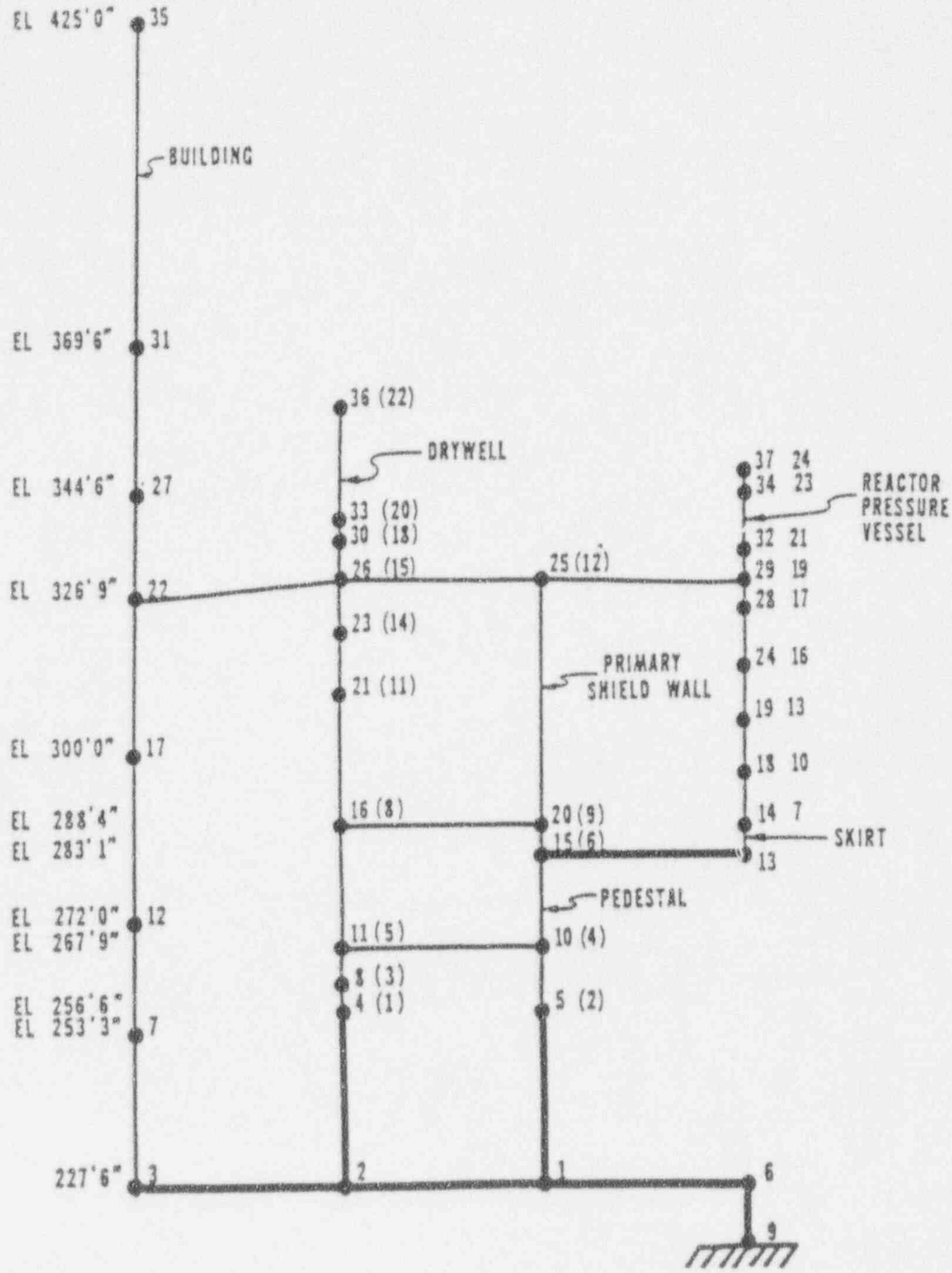

JAMES A. FITZPATRICK		FSAR UPDATE
MODEL FOR DYNAMIC ANALYSIS OF REACTOR BUILDING		
REV. 0	JULY, 1982	FIGURE NO. 12.5-1

REACTOR BUILDING MATHEMATICAL MODEL

FIGURE 13

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FIGURE 14

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FIGURE 15

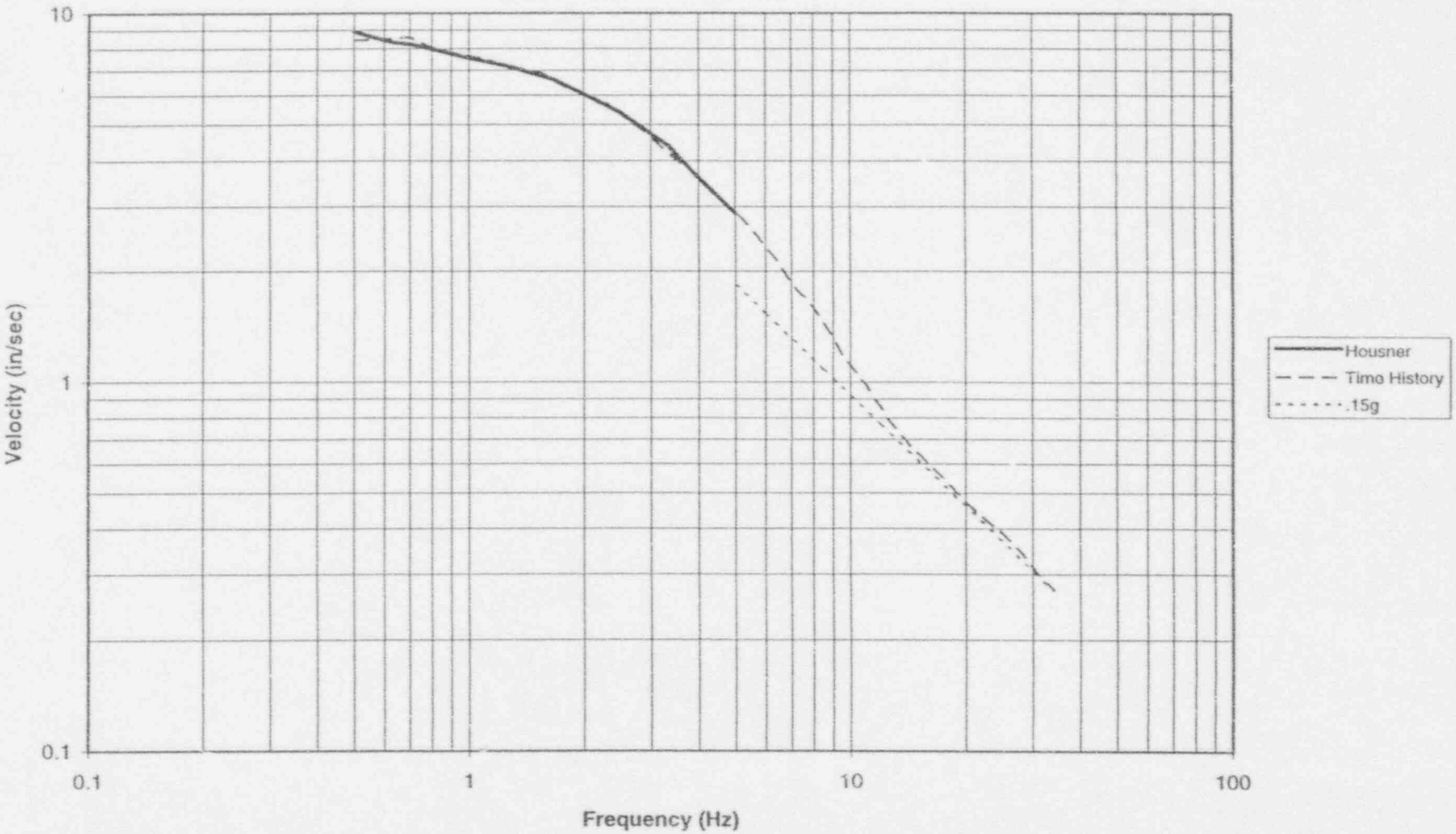

Figure 16. Comparison of the Ground Response Spectrum From the Artificial Time History to the Ground Response Spectrum for the Design Basis Earthquake at 5% Damping

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FIGURE 17

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FIGURE 18

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FIGURE 19

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Attachment 2 to JPN-94-053

Summary of New York Power Authority's
Core Shroud Inspection Plan

The inspection plan associated with the core shroud repair:

1. 100% accessible enhanced VT-1 of the 10 gusset plate welds used in the repair.
2. VT-1 top side of the reactor vessel attachment weld (H9); two inches on both sides of the 10 gusset plates inspected in item 1 above.
3. 100% UT of accessible H2 and H3 welds including creep wave or eddy current.
4. 100% visual examination of the accessible radial welds (six) on the top guide support ring.
5. Visual inspection (VT-1) of the SV2A and SV2B welds for a minimum of 6 inches above the H2 weld.
6. UT with creep wave or eddy current for a minimum of 6 inches of four vertical welds above and below weld H4.
7. 100% UT with creep wave or eddy current of accessible weld H6B.

Note: Enhanced VT-1 inspections may be substituted for ultrasonic examinations.

Attachment 3 to JPN-94-053

Summary of Commitments

Commitment Number	Commitment	Due Date
JPN-94-053-01	Evaluate the impact of the additional mass and water displacement on the EOP calculations, associated with the tie-rod assemblies. (Repair Report, Section 5.1.10, pg 11)	Prior to Startup after Winter 1994 Refueling Outage

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