VERMONT YANKEE
NUCLEAR POWER CORPORATION

Ferry Road, Brattleboro, VT 05301-7002

REPLY TO
ENGINEERING OFFICE
580 MAIN STREET
BOLTON, MA 01740
(508) 779-6711

August 7, 1996
BVY 96-96

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References: (a) License No. DPR-28 (Docket No. 50-271)
(b) Letter, VYNPC to USNRC, BVY 96-48, dated April 15, 1996
(c) Letter, USNRC to VYNPC, NVY 96-120, dated July 9, 1996

Subject: Response to Request for Additional Information Regarding Vermont Yankee Core Shroud Modification

In Reference (b) Vermont Yankee submitted its plans for modification of the core shroud during the Fall, 1996 refueling outage. In Reference (c) you requested additional information needed to complete your review of the modification plans. The requested information is included as Attachment 1.

Additionally, we have provided a copy of the 10CFR50.59 safety evaluation of our core shroud modifications. This evaluation is provided as Attachment 2.

Attachment 1 is considered proprietary information by MPR Associates. In accordance with 10CFR2.790(b)(1), an affidavit attesting to the proprietary nature of the enclosed information is attached. As such, MPR Associates requests that Attachment 1 be withheld from public disclosure.

We trust that the information provided is acceptable; however, should you have any questions, please contact this office.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Robert E. Sojka
Operations Support Manager

9608130183 960807
PDR ADOCK 05000271
P PDR

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS

WITHHOLD ATTACHMENT 1 FROM PUBLIC DISCLOSURE PER 10CFR2.790

A001

Change Distribution: PDR 1 INP LTR ENCL

MPR
ASSOCIATES INC.
ENGINEERS

August 5, 1996

Affidavit Pursuant to 10 CFR 2.790
Relative to Core Shroud Repair for
Vermont Yankee Nuclear Power Station

MPR Associates, Inc.
The Commonwealth of Virginia
City of Alexandria

I, Noman M. Cole, depose and say that I am a Principal of MPR Associates, Inc. duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations in conjunction with Vermont Yankee Nuclear Power Corporation.

The information for which proprietary treatment is sought is contained in the attached document titled, "Vermont Yankee Nuclear Power Station - Response to Request for Additional Information from the NRC Office of Nuclear Reactor Regulation." This document contains information on the design of the shroud repair system for the Vermont Yankee Nuclear Power Station.

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by MPR Associates in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure, which is owned and has been held in confidence by MPR Associates, is the design of the shroud repair system for the Vermont Yankee Nuclear Power Station.

2. The information consists of design information or other similar data concerning a repair system, method or component, the application of which results in substantial competitive advantage to MPR Associates. MPR has patent applications pending for this shroud repair system.

3. The information is of a type customarily held in confidence by MPR Associates and not customarily disclosed to the public. MPR Associates has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. This system was applied in determining that the subject document herein is proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of MPR Associates because:

    a. Other repairs for similar purposes are performed and sold by major light water reactor competitors of MPR Associate

    b. Development of these repair designs by MPR Associates required thousands of manhours and hundreds of thousands of dollars. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

    c. In order to acquire such information, a competitor would also require considerable time and inconvenience to develop these repair designs.

    d. The information consists of information related to repair of cracked shrouds in the Vermont Yankee Nuclear Power Station and other BWRs as well. The application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their designs to better compete with MPR Associates, take marketing or other actions to improve their position or impair the position of MPR Associates' design, and avoid developing similar data and analyses in support of their design methods or shroud repair system.

    e. In pricing MPR Associates products and services, significant research, development, engineering, analytical, manufacturing, quality assurance and other costs and expenses must be included. The ability of MPR Associates' competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

f. Use of the information by competitors in the international marketplace would increase their ability to market such repair designs by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on MPR Associates' potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

Noman M. Cole
Noman M. Cole
A Principal

Sworn to before me
this 5 day of August, 1996

Dorothy C. Adams
Notary Public

My commission expires: March 31, 2000

Attachment 2

Vermont Yankee Core Shroud Modification

10CFR50.59 Safety Evaluation

Vermont Yankee Safety Evaluation 96-26

Safety Evaluation for Vermont Yankee Core Shroud Modification

Summary

As a result of ultrasonic inspections performed on the Vermont Yankee core shroud during the 1995 refueling outage the USNRC issued a Safety Evaluation Report (SER) that required Vermont Yankee to either reinspect or repair the core shroud prior to startup from the 1996 refueling outage. Based on technical and economic studies Vermont Yankee has elected to modify the core shroud.

The modification selected, which is to install a mechanical system of tie rods and lateral bumpers, is not currently recognized as a repair method in the ASME Boiler and Pressure Vessel Code, Section XI. As such, USNRC regulations (10CFR50.55a(a)(3)) require that USNRC approval be obtained for an alternate repair method. A request for approval of the core shroud modification has been submitted to the USNRC.

In addition, USNRC has stated that they wish to review utility 10CFR50.59 safety evaluations prior to issuing an SER for shroud modifications.

Appendix A to this safety evaluation contains the supporting technical information for the evaluation, either directly or by reference.

Description of Change to Facility as Described in the FSAR

The FSAR describes the core shroud as a welded cylindrical assembly cantilevered off the shroud support plate and supported by the shroud support legs. See Figures 3.3-1, 3.3-2, 3.3-7, 3.3-9 and 3.3-11 of the FSAR and FSAR Chapters 3.3.4.1.1, 3.3.5 and 4.2.4.2.

The modification design replaces the structural load path of the welded cylinder with tie rods and lateral bumpers (see Figure 1). The design will function even if one or more of the circumferential welds fail following installation of the modification. The vertical load carrying capacity of the shroud, which was originally provided by tension in the cylinder, is replaced by the tie rods. The lateral load carrying capacity of the cylinder, which was provided by shear and moment stresses in the shroud cylinder

that were transmitted into the shroud support plate and shroud support legs, is replaced by the lateral bumpers.

Refer to Appendix A for a complete description of the modification design and how it functions.

Reasons for Change

The reason for the change is the development of intergranular stress corrosion cracking in several core shroud welds. Based upon the possibility of future crack growth it cannot be assured that the core shroud can operate without failure for the remainder of licensed plant life without modification.

Engineering Evaluation of Changes

The core shroud has the following safety and operational design bases:

1. Provide proper coolant distribution during all anticipated normal operating conditions to allow proper operation of the core without fuel damage. (FSAR Chapter 3.3.2.1)

2. Facilitate refueling operations and provide adequate working space for inspections. (FSAR Chapters 3.3.2.2 and 3.3.2.4)

3. 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. (FSAR Chapter 3.3.3.1)

4. Limit deflections and deformation to assure that the control rods and the core standby cooling systems can perform their safety functions during abnormal operating transients and accidents. (FSAR Chapter 3.3.3.2)

5. Assure that items 3 and 4 are satisfied in accordance with specified loading criteria so that the safe shutdown of the plant and removal of decay heat are not impaired. (FSAR Chapter 3.3.3.3)

The following discussion will demonstrate how the core shroud repair

satisfies these operational and safety design bases.

1. Provide proper coolant distribution during all anticipated normal operating conditions to allow proper operation of the core without fuel damage. (FSAR Chapter 3.3.2.1)

The core shroud modification is designed to ensure that no vertical or horizontal weld separation develops during any normal or upset operating conditions, defined as normal plant operation and the abnormal operating transients described in Chapter 14.5 of the FSAR. All materials used in the fabrication of the core shroud modification are compatible with the BWR environment. See Section 7 of Appendix A. It is conservatively assumed that small fluid leakage may occur through cracks that may develop in the welds, as well as leakage past the attachment points where the modification is connected to the shroud support plate. This small amount of leakage (conservatively calculated to be 101 gpm or less) is concluded to be insignificant. See Sections 6.3 and 6.4 of Appendix A.

The added flow resistance in the vessel annulus caused by the addition of the modification components has been evaluated and determined to be insignificant. See Section 6.3.2 of Attachment A.

Thus it is concluded that the core shroud modification does not adversely affect this operational design basis for the core shroud.

2. Facilitate refueling operations and provide adequate working space. (FSAR Chapter 3.3.2.2 and 3.3.2.4)

The core shroud modification is installed outside the core shroud and does not extend above the top of the core shroud steam dam. Therefore it provides no impediment to the conduct of refueling operations. If required for future inspection access the core shroud modification is removable.

Thus it is concluded that the core shroud modification does not adversely affect these operational design bases for the core shroud.

3. 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. (FSAR Chapter 3.3.3.1)

The tie rods and lateral bumpers prevent any permanent displacement of the core shroud that would allow bypass leakage paths from inside the core shroud into the annulus region of the reactor vessel. As stated above, it is conservatively calculated that less than 101 gpm of leakage could occur through any cracks that develop in the shroud welds. This small amount of leakage has been evaluated and shown to have no effect the ability of the core standby cooling systems to maintain the core level at or above two-thirds core height following a design basis loss of coolant accident. See Section 6.5 of Appendix A.

Thus it is concluded that the core shroud modification does not adversely affect this safety design basis for the core shroud.

4. Limit deflections and deformation to assure that the control rods and the core standby cooling systems can perform their safety functions during abnormal operating transients and accidents.

The system of tie rods and lateral bumpers in the modification are designed to limit the amount of displacement that the core shroud can experience during all abnormal operating transients and accidents. The calculated displacements are significantly smaller than allowable. The effect of core shroud displacement on the internal core spray piping has been evaluated and determined to be acceptable. This requirement, in conjunction with 3 above, ensures that the core shroud modification will maintain a coolable core geometry and limit deflections that could affect insertion of the control rods. See Sections 6.1 and 6.5 of Appendix A.

Thus it is concluded that the core shroud repair does not adversely affect this safety design basis for the core shroud.

5. Assure that items 3 and 4 are satisfied in accordance with specified loading criteria so that the safe shutdown of the plant and removal of decay heat are not impaired.

The core shroud original design was based on the guidance of Section III to the ASME Boiler and Pressure Vessel Code, even though the core shroud was not specified as a Code component (the ASME Code did not develop rules for the design of components other than pressure vessels until 1971) (Appendix C.2.5.1 of the FSAR).

The core shroud modification is designed to the 1989 Edition of Section III of the ASME Boiler and Pressure Vessel Code, following the rules for core support structures.

All existing plant components affected by the shroud modification were evaluated to their design criteria as specified in the Vermont Yankee FSAR. See Sections 4.5 and 5.4 of Appendix A.

Thus it is concluded that the core shroud modification does not adversely affect this safety design basis for the core shroud.

Installation Process

In addition to considering plant operation following the installation of the core shroud modification the engineering evaluation considered the actual installation process.

All equipment handling will be performed in accordance with plant procedures and in accordance with the heavy load requirements of NUREG-0612.

In order to prevent foreign material from entering the core and to assist in tool and equipment handling, a perforated core cover will be installed during the repair process. The added pressure drop due to the cover is insignificant at normal shutdown cooling flows (less than 0.1 psi drop at 7000 gpm). The weight of the cover is sufficient to prevent uplift in the unlikely event of an inadvertent start of all six CSCS pumps. See Appendix B

Foreign material exclusion controls will be in place during any evolutions that could introduce loose parts into the reactor coolant system.

The electrical discharge machining process employed for a portion of the repair has been evaluated and determined to result in no adverse consequences. See Section 4.8 of Appendix A and Appendix C.

10CFR50.59 Safety Evaluation

1. May the proposed activity directly/indirectly increase the chance of a:

Chapter 14.6.2 - Control Rod Drop Accident
Chapter 14.6.3 - Loss of Coolant Accident
Chapter 14.6.4 - Refueling Accident
Chapter 14.6.5 - Main Steam Line Break Accident

NO

Applicable FSAR Sections: Chapter 3.3, Chapter 3.5, Chapter 4.2, Appendix C

Explain: The core shroud is not an accident initiator for any of the four design basis accidents. The installation of the core shroud modification does not alter the function of the core shroud or adversely affect any components that could be considered accident initiators. Therefore, the core shroud modification will not increase the probability of any of these four design basis accidents.

2. May the proposed activity directly/indirectly increase the radioactivity material release from a:

Chapter 14.6.2 - Control Rod Drop Accident
Chapter 14.6.3 - Loss of Coolant Accident
Chapter 14.6.4 - Refueling Accident
Chapter 14.6.5 - Main Steam Line Break Accident

NO

Applicable FSAR Sections: Chapter 3.3, Chapter 3.5, Chapter 4.2, Appendix C

Explain: The core shroud plays no role in the mitigation of a control rod drop accident. The core shroud plays an indirect role in mitigating a refueling accident since the refueling accident assumes only two fuel

assemblies are damaged in the fuel drop. The core shroud modification does not adversely affect the ability of the top guide assembly from supporting fuel assemblies; thus the conclusions of the refueling accident analysis will be unchanged by the installation of the modification. The core shroud repair plays a direct role in mitigating the consequences of a main steam line break or a loss of coolant accident by ensuring that a coolable core geometry and control rod insertion is maintained. As discussed in the Engineering Evaluation above, the core shroud modification does not adversely affect the ability of the core shroud to maintain a coolable core geometry or achieve control rod insertion during a main steam line break or a loss of coolant accident. The assumptions in the analyses for these two accidents will remain unchanged and the installation of the modification will have no effect on any plant systems or equipment which prevent or mitigate failures of the four radioactive material barriers. Therefore there will be no possibility of any increase in radioactive material release from any design basis accident.

3. May the proposed activity directly/indirectly increase the chance of a malfunction occurring which:

Initiates a FSAR 14.5 abnormal operational transient and causes:

Nuclear system pressure increases
Reactor vessel moderator temperature decreases
Positive reactivity increases
Reactor vessel coolant inventory decreases
Reactor core coolant flow decreases
Reactor core coolant flow increases
Core coolant temperature increases
An excess of coolant inventory

OR - Impacts FSAR 1.6.2/FSAR 1.6.3 equipment performance (as described in FSAR system/safety evaluation chapters)

OR - Impacts station blackout, anticipated transients without scram, Appendix R or Alternate Shutdown procedures or equipment and causes or threatens failure of any of the four

radioactive material barriers or nuclear safety/engineered safeguard systems. **NO**

Applicable FSAR Sections: Chapter 3.3, Chapter 3.5, Chapter 4.2, Appendix C

Explain: Except for a reactor core flow decrease, the core shroud cannot initiate any of the Chapter 14.5 abnormal operating transients. As discussed in the Engineering Evaluation, failure of one or more of the core shroud welds could result in a small core flow decrease above the elevation of the failed weld(s). The core shroud modification is designed to prevent shroud separation during any normal or upset conditions, thus ensuring that the Chapter 14.5 analyses remain unchanged.

By ensuring that core shroud safety and operational design bases are satisfied, and by ensuring that no existing plant components are adversely affected by the core shroud modification, the core shroud modification cannot adversely affect any FSAR Chapter 1.6.2 or 1.6.3 equipment.

The core shroud, and the installation of the core shroud modification, play no role in the initiation or mitigation of station blackout, anticipated transients without scram, Appendix R or Alternate Shutdown procedures or equipment.

By ensuring that all appropriate structural loading criteria are satisfied, the core shroud modification will not cause or threaten failure of any of the four radioactive material barriers or nuclear safety/engineered safeguard systems.

4. May the proposed activity directly/indirectly increase the radioactive material release from any of the four radioactive material barriers, as a result of a malfunction which:

Initiates a FSAR 14.5 abnormal operational transient and causes:

Nuclear system pressure increases
Reactor vessel moderator temperature decreases
Positive reactivity increases

Reactor vessel coolant inventory decreases
Reactor core coolant flow decreases
Reactor core coolant flow increases
Core coolant temperature increases
An excess of coolant inventory

OR - Impacts FSAR 1.6.2/FSAR 1.6.3 equipment performance (as described in FSAR system/safety evaluation chapters)

OR - Impacts station blackout, anticipated transients without scram, Appendix R or Alternate Shutdown.

$\boxed{\text{NO}}$

Applicable FSAR Sections: Chapter 3.3, Chapter 3.5, Chapter 4.2, Appendix C

Explain: As discussed in the Engineering Evaluation above, the core shroud modification is designed to ensure that proper control rod insertion is maintained and a coolable core geometry is maintained for all normal, upset, emergency and faulted conditions specified in the Vermont Yankee FSAR. This ensures that the core shroud modification will not adversely affect any analyses for the above specified transients. The core shroud modification will not adversely affect the performance of any Chapter 1.6.2 or 1.6.3 equipment or systems.

By maintaining the operational and safety design bases of the core shroud, the core shroud modification will not adversely affect the mitigation of station blackout, anticipated transients without scram, Appendix R or Alternate Shutdown.

By ensuring that all appropriate structural loading criteria are satisfied the core shroud modification will not increase the radioactive material release from any of the four radioactive material barriers.

5. May the proposed activity directly/indirectly create the possibility of an accident occurring which is different from:

FSAR 14.4.3 - Mechanical failure leading to a radioactive material boundary breach

FSAR 14.4 3 - Overheating of the fuel barrier
FSAR 14.4.3 - Arbitrary single pipe rupture **NO**

Applicable FSAR Sections: Chapter 3.3, Chapter 3.5, Chapter 4.2, Appendix C

Explain: As discussed above, the core shroud modification satisfies all of the safety and operational design bases for the core shroud and does not impose any unacceptable loadings on existing plant equipment. Therefore, the core shroud repair cannot create any different accident than the existing core shroud configuration.

6. May the proposed activity create the possibility of a malfunction which is different than one that causes:

Nuclear system pressure increases
Reactor vessel moderator temperature decreases
Positive reactivity increases
Reactor vessel coolant inventory decreases
Reactor core coolant flow decreases
Reactor core coolant flow increases
Core coolant temperature increases
An excess of coolant inventory

and/or could cause or threaten failure of any of the four radioactive material barriers or nuclear safety/engineered safeguard systems.

NO

Applicable FSAR Sections: Chapter 3.3, Chapter 3.5, Chapter 4.2, Appendix C

Explain: As discussed above, the cracked core shroud with the modification installed is functionally equivalent to the core shroud. In addition, the design ensures that all existing plant equipment is not adversely affected by the installation of the repair. Therefore, the core shroud modification cannot create any different malfunction than the existing core shroud configuration.

7. Does the proposed activity reduce the difference between a system failure point and accepted safety limit or in any way affect the margin of safety provided, as defined in the basis for any technical specification.

$\boxed{\text{NO}}$

Applicable Technical Specification Section: None

Explain: There are no Technical Specification sections that involve the core shroud.

<u>Thus it is concluded that installation of the core shroud modification as described in Appendix A does not present an Unreviewed Safety Question as defined by 10CFR50.59</u>

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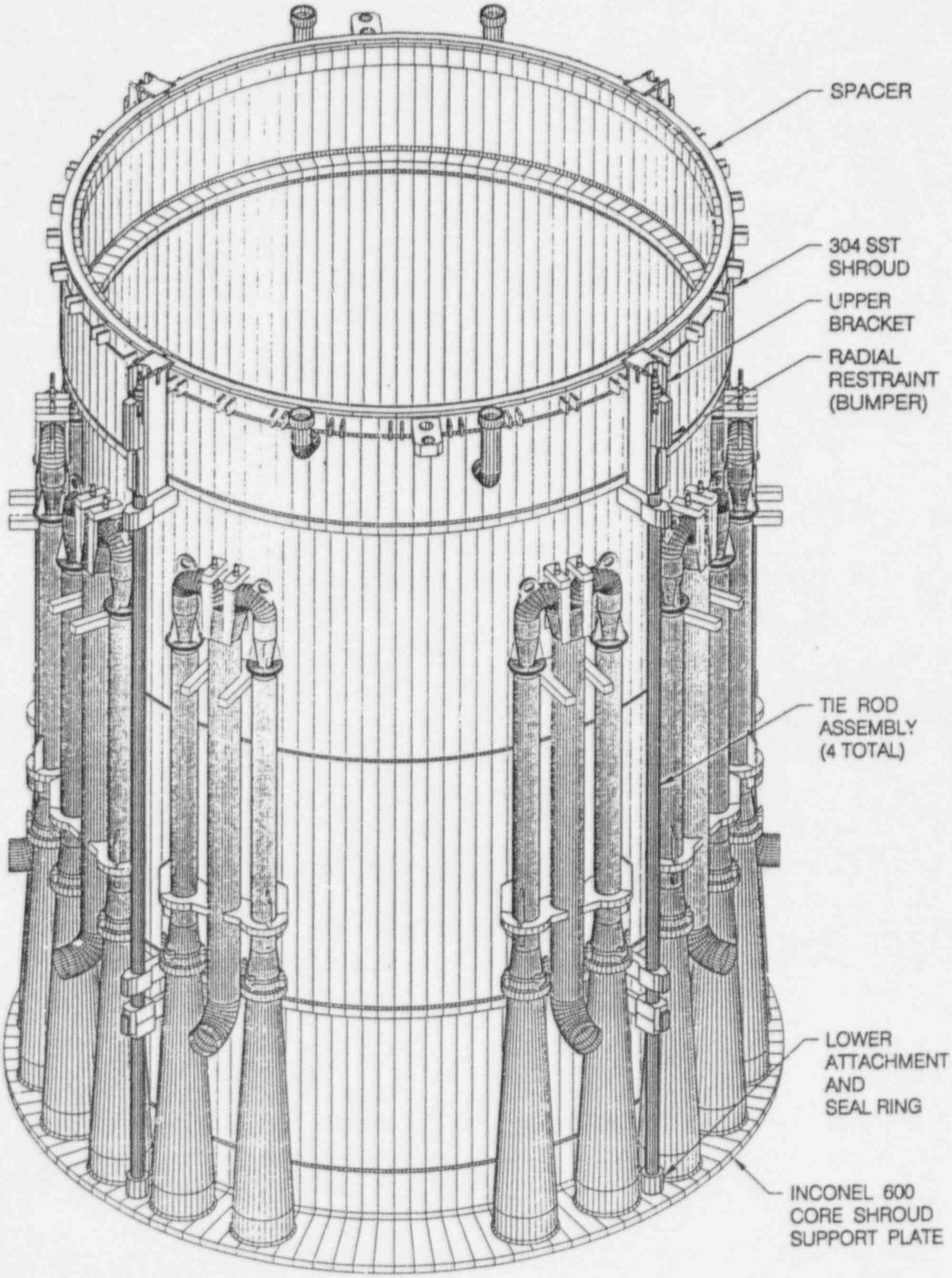

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VERMONT YANKEE SHROUD REPAIR
FIGURE 1

Appendix A

Vermont Yankee Core Shroud Modification Design Summary

Vermont Yankee Nuclear Power Station
Core Shroud Repair Summary

Revision 0

April 1996

Vermont Yankee Nuclear Power Corporation
Governor Hunt Road
Vernon, Vermont 05354

9604180371

CONTENTS

CONTENTS

Section 1

INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report summarizes the design of the core shroud repair for the Vermont Yankee (VY) Nuclear Power Station. The report follows the guidelines for the format and content for core shroud repair design submittals prepared by the BWR Vessel and Internals Project (EPRI Report TR-105692).

1.2 SUMMARY

The VY core shroud repair addresses the potential through wall cracking of any combination of the potentially sensitized 304 stainless steel circumferential core shroud welds, i.e. H1 through H7 (See Figure 1-1). The repair is not included under the ASME Boiler and Pressure Vessel Code Section XI definition for repair or replacement. Rather, the repair is developed as an alternative repair pursuant to 10 CFR 50.55a(a)(3).

The detailed design of the repair is documented in Reference 1. As summarized below, the repair satisfies the requirements specified in the Vermont Yankee specification for the repair and the BWR Vessel and Internals Project "BWR Core Shroud Repair Design Criteria" (References 2, 3 and 4). The repair is consistent with the current plant licensing basis and ensures that the shroud will satisfy its operational and safety functions even if welds H1 through H7 fail. The repair can also accommodate a complete failure of H8 with the shroud legs intact.

1.2.1 Repair Overview

As shown in Figures 1-2 and 1-3 the repair consists of a set of four tie rod assemblies which hold the shroud together. Radial restraints are provided at four elevations to limit the lateral movement of the shroud sections. The repair design specification is provided in Reference 4.

1.2.2 Structural and Design Evaluations

The shroud repair hardware limits the displacement of the shroud such that the shroud will maintain its basic as-designed configuration during all identified operating, transient and accident conditions. In particular, the load carrying capability of the repair assemblies is sufficient to prevent separation of shroud segments during normal operating conditions for any combination of circumferential weld failures. The repair hardware radial restraints

maintain the required shroud capabilities with respect to positioning and support of the fuel assemblies, and other vessel internals, and core alignment for control rod drive insertion. See Section 6.1 of this report for additional details on shroud displacement evaluations.

As summarized below, the repair satisfies the structural requirements specified in References 2, 3 and 4.

- Repair Assembly - The tie rod assembly satisfies the structural criteria for the repair hardware. In particular:

  - Although the repair is not considered an ASME B&PV Code repair, the repair satisfies the Design By Analysis stress and fatigue criteria of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG (Reference 5).

  - Stresses in the repair hardware vertical load path will be less than yield during all normal and upset operating conditions, including anticipated thermal transients. As a result, tie rod preload will not be lost inservice.

  See Section 4.3 of this report for additional information on the repair assembly structural evaluation.

- Shroud - The stresses in the shroud resulting from the repair will be within the stress allowables of Section III, Subsection NG of the ASME Boiler & Pressure Vessel Code.

  See Section 4.4 of this report for additional information on the shroud structural evaluation.

- Reactor Vessel - The stresses in the reactor vessel resulting from the repair will be within the stress allowables of the ASME Boiler & Pressure Vessel Code, Section III, Class A, 1965 with Summer 1966 addenda. In addition, the response of the reactor vessel to seismic accelerations is not affected by the repair.

  See Section 4.5 of this report for additional information on the reactor vessel structural evaluation.

- Reactor Internals - The fuel shear loads during a seismic event result in stresses in the top guide and core support plate which are less than the allowable stress.

  See Section 4.5 of this report for additional information on the evaluation of loads on reactor internals.

- Fuel - The maximum fuel acceleration is less than the design acceleration for the fuel.

See Section 5.4 of this report for additional information on the evaluation of fuel loads.

- Core Spray Pipe - The shroud repair assembly will limit the vertical and lateral displacement of the shroud during all normal, upset, emergency and faulted service loading conditions such that the core spray pipe is not over stressed.

  See Section 6.5 of this report for additional information on the impact on the core spray system due to the repair.

1.2.3 System Evaluations

The impact on plant operations of postulated 360° through wall cracking of the shroud circumferential welds with the repair assemblies installed was evaluated. These evaluations showed that there would be no impact on normal plant operations. The overall Core Standby Cooling Systems (CSCS) performance would not be affected and control rod drive insertion capability would be maintained. The parameters considered in the evaluations include core shroud weld crack leakage, leakage at the repair assembly attachment points, and lateral and vertical displacement of the core shroud. See Section 6 of this report for additional information on these evaluations.

1.2.4 Material and Fabrication

The materials specified for use in the repair assemblies are resistant to stress corrosion cracking and have been used successfully in the BWR reactor coolant system environment. The repair assemblies are fabricated from solution annealed Type 304 or 304L stainless steel, solution annealed Type XM-19 stainless steel and alloy X-750 per EPRI NP-7032. No welding is permitted in the fabrication or installation of the repair and special controls and process qualifications are imposed in the fabrication of the repair to assure acceptable material surface conditions after machining. See Section 7 of this report for additional information on repair hardware materials and fabrication.

1.2.5 Pre-Modification and Post-Modification Inspection

The inspections to be performed to support the repair are summarized below.

- Pre-Modification Inspection - Prior to installation of the shroud repair, Vermont Yankee will perform ultrasonic inspections of design reliant welds. These inspections will cover portions of the vertical welds in the H3/H4, H4/H5 and H6/H7 shroud segments, the welds in the core support ring and welds H8 and H9.

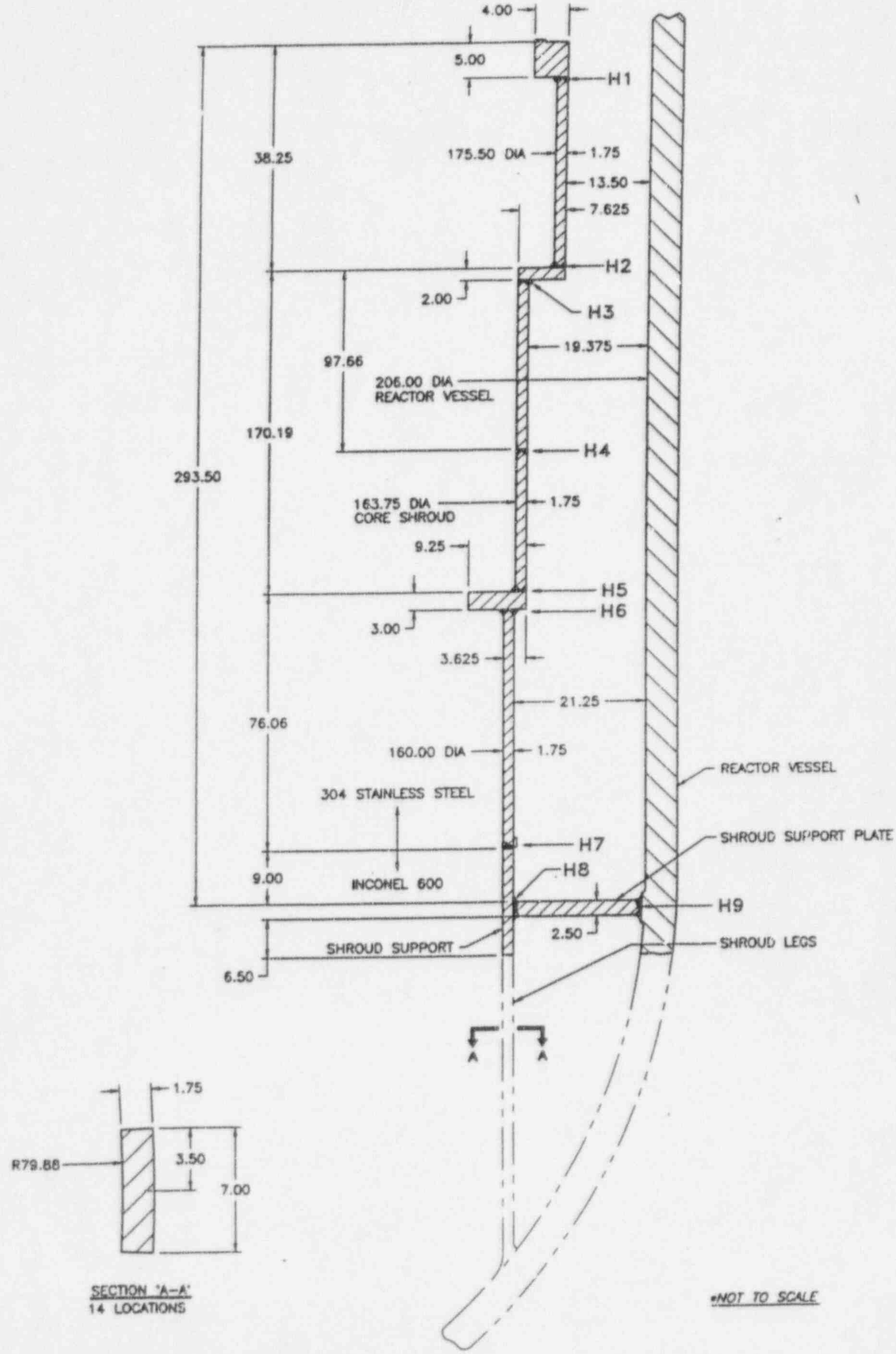
  The repair relies on portions of the vertical welds in the H1/H2 shroud segment to be intact. However, due to tooling limitations, it is not practical to ultrasonically inspect the vertical welds in the H1/H2 shroud segment. Therefore, rather than inspect these

vertical welds, portions of circumferential welds H1 and H2 are designated as design reliant welds; these circumferential welds provide an alternate path for the loads carried by the vertical welds. Circumferential welds H1 and H2 were ultrasonically inspected in 1995. The results of these inspections will be used to demonstrate that sufficient design reliant weld length exists. It should be noted that Vermont Yankee is considering H1 and H2 as design reliant welds only for inspection reasons and that the repair is designed as a repair to H1 and H2.

The specific scope of the pre-modification inspections is discussed in Section 8.1.

- Post Modification Inspection - Prior to reactor pressure vessel reassembly, visual inspections will be performed to verify the proper installation of repair. The scope of these inspections is discussed in Section 8.2.

Inspection of the shroud and the repair in future refueling outages will be based on the "Guidelines for Reinspection of Core Shrouds" recently developed by the BWRVIP. The actual inspection scope will be submitted to USNRC at least 90 days prior to the start of the 1998 refueling outage.



VERMONT YANKEE - SHROUD WELDS
FIGURE 1-1

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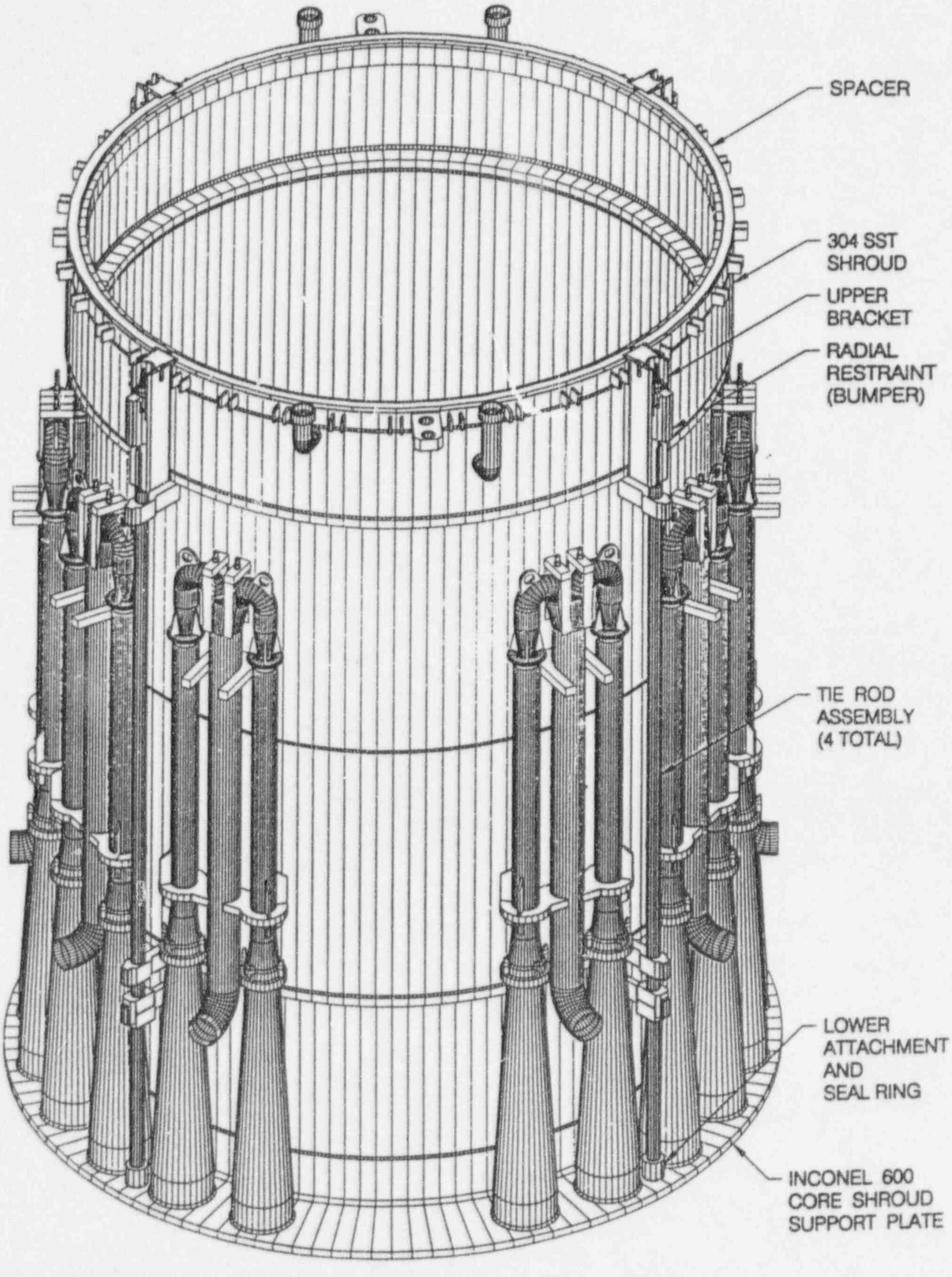
U.S. PATENT 5,402,570



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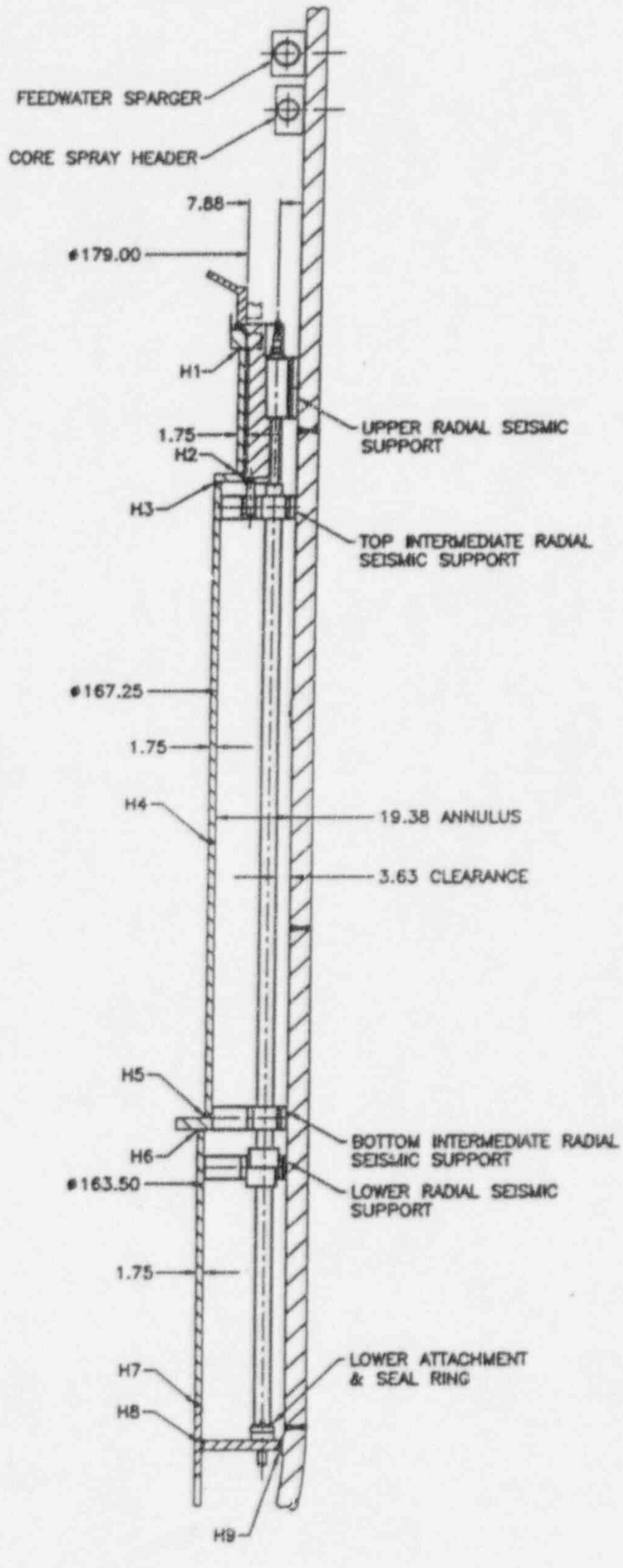
VERMONT YANKEE SHROUD REPAIR
FIGURE 1-2



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VERMONT YANKEE - SHROUD REPAIR
FIGURE 1-3

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Section 2

BACKGROUND

2.1 SHROUD OPERATION AND SAFETY FUNCTIONS

The core shroud operational and safety functions are provided in the Vermont Yankee Nuclear Power Station FSAR and are reproduced below:

3.3 Reactor Vessel Internals Mechanical Design

3.3.1 Power Generation Objectives

Reactor vessel internals (exclusive of fuel, control rods, and incore flux monitors) are provided to achieve the following power generation objectives:

a. Maintain partitions between regions within the reactor vessel to provide proper coolant distribution, thereby allowing power operation without fuel damage due to inadequate cooling.

b. Provide positioning and support for the fuel assemblies, control rods, incore flux monitors, and other vessel internals to assure that normal control rod movement is not impaired.

3.3.2 Power Generation Design Basis

1. The reactor vessel internals shall be designed to provide proper coolant distribution during all anticipated normal operating conditions to allow proper operation of the core without fuel damage.

2. The reactor vessel internals shall be arranged to facilitate refueling operations.

3. The reactor vessel internals shall include devices that permit assessment of the core reactivity condition during periods of low power and subcritical operations.

4. Adequate working space and access shall be provided to permit adequate inspection of reactor vessel internals.

3.3.3 Safety Design Basis

1. The reactor vessel internals shall be arranged 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.

2. Deflections and deformation of reactor vessel internals shall be limited to assure that the control rods and the core standby cooling systems can perform their safety functions during abnormal operational transients and accidents.

3. The reactor vessel internals' mechanical design shall assure that Safety Design Bases (1) and (2) are satisfied in accordance with the loading criteria of Appendix C, so that the safe shutdown of the plant and removal of decay heat are not impaired.

2.2 VYNPS RESPONSE TO GL 94-03

USNRC Generic Letter 94-03 requested BWR licensees to (1) inspect the core shrouds in their BWR plants at the next scheduled refueling outage and (2) perform a safety analysis supporting continued operation until inspections were conducted.

Vermont Yankee replied to the generic letter in References 6, 7 and 8.

The USNRC issued References 9, 10 and 11.

Vermont Yankee inspected the core shroud during the 1995 refueling outage. Flaw indications were identified in several welds.

The USNRC evaluated the results of the inspections and the associated engineering analyses and issued a Safety Evaluation Report that required Vermont Yankee to either reinspect or repair the core shroud prior to startup from the 1996 refueling outage (Reference 11).

Section 3

DESCRIPTION OF REPAIR

3.1 DESIGN OBJECTIVES

The function of the core shroud repair is to structurally replace all potentially sensitized 304 stainless steel circumferential core shroud welds, i.e. H1 through H7 (See Figure 1-1). In addition, the repair can accommodate a complete failure of the H8 shroud weld with the shroud support legs intact. The design life of the repair is 40 years.

3.2 DESIGN CRITERIA

The repair is developed as an alternative repair pursuant to 10 CFR 50.55a(a)(3). The repair is consistent with and meets the criteria developed by the Boiling Water Reactor Vessel and Internals Project, "BWR Core Shroud Repair Design Criteria" (Reference 3). The design specification for the repair is provided in Reference 4.

The vessel internals were originally designed in accordance with the "intent" of Section III of the ASME B&PV Code. Accordingly, the repair is designed to satisfy the requirements of Section III, Subsection NG, "Core Support Structures", of the ASME Boiler & Pressure Vessel Code (Reference 5). In addition, stresses in the vertical load paths shall be less than yield for normal and upset operating conditions. As a result, preload will not be lost inservice.

The repair prevents vertical separation of the shroud during normal operating conditions at any postulated failed circumferential weld(s).

The repair is designed for the current plant operating conditions. However, margin is provided to allow for a potential future increase in core flow and/or a power uprate. In particular, the core shroud pressure differentials considered in the design analyses have been increased by 15% over those for the current operating conditions.

3.3 DESCRIPTION OF REPAIR COMPONENTS AND DESIGN FEATURES

The core shroud repair design consists of four tie rod assemblies installed 90° apart in the core shroud/reactor vessel annulus. Each assembly consists of a tie rod, upper bracket, lower T-head and seal assembly, and four lateral restraints (See Figure 1-2 and 1-3). The assemblies, which are designed and fabricated as safety-related components, are used to maintain the alignment of the core shroud assuming all circumferential welds are cracked 360° through wall.

A spacer ring is provided between the top shroud flange and shroud head. Cut outs are provided in the ring which allow the top bracket to be hung from the top shroud flange. The bracket is captured by the shroud head and the top lateral restraint. The bracket extends from the top flange to just above the H3 weld and provides support for the top lateral restraint. The tie rod passes through a hole in the top lateral restraint and bracket and is held by a nut. The tie rod extends down to the T-head at the shroud support plate. The T-head is connected to the plate through a hole which is machined in the shroud support plate. The hole in the shroud support plate is sealed with a seal ring which is preloaded against the support plate. The seal preload is independent of the preload in the tie rod.

The radial restraints are solid stainless steel spacers which provide positive rather than spring type lateral restraint of the core shroud. The restraints are integral with the tie rod assemblies. The restraints are installed based on field measurements to provide a small effective gap relative to the vessel wall. At the shroud elevations which support the top guide and core support plate the effective gap is about 1/8 inch (which equates to less than 0.0007 inches of radial clearance per inch of shroud diameter). At the upper intermediate and bottom radial restraint locations a slightly larger effective gap of ½ inch is provided. As discussed in Section 6.1, these gaps result in acceptable shroud displacements during all loading conditions.

Together the tie rods and radial restraints resist both vertical and lateral loads resulting from normal operation and design accident loads, including seismic loads and postulated pipe ruptures. The tie rods provide vertical load carrying capability from the upper bracket on the top shroud flange to the lower T-head connected to the shroud support plate. The tie rod installation preload is selected such that the installation preload plus the thermal expansion load generated by plant heatup results in a total load on the tie rods sufficient to ensure that shroud segments cannot vertically separate during normal plant operation, even in the event that welds H2 and H3 fail after installation of the repair.

The tie rod design incorporates an additional structural member which can assist in carrying the large primary loads associated with accident and safe shutdown earthquake events. However, at Vermont Yankee all vertical tie rod loads are carried by the inner tie rod member.

Each cylindrical section of the shroud is prevented from unacceptable lateral motion by the radial supports even if it is assumed that the welds contain 360° through wall cracks. The motion of the top flange and the shroud sections above H3 are restrained by the top bracket and the upper radial support. The shroud sections between H3 and H4 are restrained by the top intermediate radial support. The shroud sections between H4 and H6 are restrained by the bottom intermediate radial support. The shroud section between H6 and H7 is restrained by the lower radial support. All horizontal support for the fuel assemblies is provided by the top guide and the core support plate. Lateral restraint of the shroud at these elevations is provided by the upper radial and the bottom intermediate radial supports.

By restraining the vertical and lateral displacement of the shroud cylinders the repair assembly effectively replaces the potentially sensitized 304 stainless steel circumferential welds, i.e. H1 through H7. In order to restrain the shroud cylinders, the repair relies to various extents on the following existing welds being intact:

- Vertical welds in the shroud cylinders

- Radial welds in the shroud flange and the top guide and core plate support rings

- Top guide support plate welds

- Shroud support plate to reactor vessel weld (H9)

The design does not rely on the entire length of each of these welds being intact.

The repair relies on portions of the vertical welds in the H1/H2 shroud segment to be intact. However, due to tooling limitations, it is not currently practical to ultrasonically inspect the vertical welds in the H1/H2 shroud segment. Therefore, rather than inspect these vertical welds, portions of circumferential welds H1 and H2 are designated as design reliant welds; these circumferential welds provide an alternate path for the loads carried by the vertical welds. It should be noted that Vermont Yankee is considering H1 and H2 as design reliant welds only for inspection reasons and that the repair is designed as a repair to H1 and H2.

Section 4

STRUCTURAL AND DESIGN EVALUATION

4.1 DESIGN LOADS AND LOAD COMBINATIONS

The loads and load combinations required by the VYNPS Final Safety Analysis Report (FSAR) are evaluated in the design of the shroud repair. As required by the specification for the repair (Reference 2) a SSE earthquake during refueling was evaluated as an additional emergency service level loading. These loads and load combinations are summarized in Table 4-1.

A combination of hand calculations and finite element analyses were used to define the design loads. Hand calculations were used to determine the loads on the repair hardware and shroud due to deadweight (including buoyancy effects), core pressure differential, differential thermal expansion effects, and a recirculation line break. These calculations used existing component weight, differential pressure and LOCA design basis information as design inputs.

The core shroud pressure differentials specified in Section 3.3.5 of the Vermont Yankee FSAR are used as the basis for the pressure differentials used in the design of the repair; the data in the FSAR is based on a power output of 1665 Mwt and the licensed plant power is 1593 Mwt. Additional margin has been provided in the core differential pressures used in the design of the repair to allow for a potential future increase in core flow and/or a power uprate. In particular, the core shroud pressure differentials considered in the design analyses have been increased by 15% over those specified in the FSAR.

Seismic loads on the shroud and repair hardware were determined by dynamic time history analyses. The analyses were performed using the ANSYS computer program and the current FSAR seismic model modified to include the shroud repair components. The seismic analysis models and inputs are discussed in Section 5 of this report.

The original design approach for Vermont Yankee was based on a two-dimensional load combination of one horizontal direction and the vertical direction. Absolute summation was utilized and the greater of the north-south/vertical and east-west/vertical combinations was selected. In the analyses for the shroud repair, the loads determined in the analysis of the vertical, North/South and East/West seismic loadings were combined by SRSS as described in USNRC Regulatory Guide 1.92.

The recirculation line break LOCA produces a spatial and time varying lateral pressure in the shroud/reactor vessel annulus. The initial acoustic phase of the transient is very abrupt relative to the shroud inertia and frequencies, and does not have a significant effect on the shroud. The remainder of the transient extends over a relatively long period of time and as such, is considered

a static pressure. This load was combined with normal operating loads and design basis earthquake loads in the evaluation of a postulated recirculation line break.

Loads during normal operation are a combination of the tie rod installation preload, differential thermal expansion between the shroud and repair hardware, gravity and pressure loads. All combinations of potential weld failures were considered. The largest operating loads are obtained if the shroud is uncracked when the installation preload is applied and remains uncracked during operation. Due to the change in shroud flexibility associated with some weld failures (e.g., failure of H2 and H3), tie rod and shroud loads are generally smaller if the shroud cracks after the repair hardware is installed. The larger loads were considered in the structural evaluations, while the smaller loads were considered in the evaluations performed to ensure that shroud segments do not separate during normal operation.

4.2 ANALYSIS MODELS AND METHODOLOGY

Analysis models and methods used to evaluate the repair hardware and existing structures are discussed below. The models and methods used to develop the seismic loads on the components are discussed in Section 5.

4.2.1 Structural Analysis Models and Methods

A combination of hand calculations and finite element analyses were used to evaluate the repair hardware and existing structures. Three-dimensional finite element analyses using the ANSYS code were used to determine the structural response of the shroud, and shroud support plate. Hand calculations were used in the evaluations of the repair hardware and tie rod preload. Hand calculations were also used to evaluate vessel stresses due to loads from the radial restraints and shroud support plate.

The finite element models used to determine the effective shroud spring constants at each of the radial restraints were also used to evaluate the shroud stresses at these locations. As a result, a consistent set of assumptions was used to generate the lateral seismic loads and to evaluate the resulting stresses. As discussed above, to bound the potential response of the repaired shroud, seismic analyses were performed assuming that the cracked welds would not carry any shear load (i.e., sliding) and assuming that the welds would remain sufficiently interlocked to carry shear (i.e., pinned). The spring constants and corresponding stresses were evaluated for both of these boundary condition assumptions.

4.2.2 Weld Crack Model

As discussed in Section 6.1, no separation of the shroud occurs during normal operation. The shroud will only separate completely during a main steam line break, and then only for a few seconds. For load combinations other than the main steam line break local areas of the shroud may temporarily separate under seismic loading. However, the majority of the shroud remains in contact with a net compressive load across the postulated failed welds.

The seismic loads on the tie rod and shroud are conservatively evaluated considering both pinned and sliding models of postulated circumferential weld failures. The vertical stiffness of the shroud for compressive loads is calculated with the welds at H2 and H3 modeled as pinned. In estimating the reduction in tie rod load resulting from the postulated failure of H2 and H3, no credit is taken for the fillet welds at H2 and H3.

4.3 REPAIR HARDWARE EVALUATION

4.3.1 Repair Hardware Structural Evaluation

As discussed in Reference 1, the repair hardware satisfies the structural criteria for the repair specified in the repair design specification. In particular:

- The Design By Analysis stress and fatigue criteria of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG are satisfied.
- Tie rod preload will not be lost inservice; stresses in the repair hardware vertical load path 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 12%.
- The fatigue usage from shroud and flow induced vibration is negligible.

For a given loading, the limiting loads on the tie rod assemblies and radial restraints occur with different assumed shroud cracks. However, since the vertical and lateral load paths are essentially independent, the bounding vertical loads for all break cases were considered with the bounding radial loads for all break cases. The limiting stress in the repair assembly during normal operation is the bearing stress between the bracket ledge and the shroud flange. The stress is less than 80% of allowable. The inner sleeve is not loaded during normal operation.

4.3.2 Flow Induced Vibration

The tie-rods were analyzed to ensure that reactor coolant flow would not induce unacceptable vibration. The basic approach to obtain resistance to flow-induced vibration of the tie rod assembly is to provide features that conservatively assure a high degree of structural damping and thereby minimize the response to flow-induced vibratory excitation. Accordingly, flow induced vibration effects are conservatively calculated assuming that the tie rods are excited at their natural frequency and with a conservatively low damping factor. As discussed above, the evaluations show that stresses resulting from flow induced vibration are small and pose no fatigue concern.

4.3.3 Radiation Effects

The effects of radiation were considered in the selection of the repair materials and fabrication processes. As discussed in Section 6, all materials used in the repair have been used successfully for years in the BWR environment.

As discussed in Section 4.6, the potential relaxation of the tie rods due to radiation and temperature effects was considered in the design of the repair and the evaluation of tie rod preload.

4.4 SHROUD EVALUATION

The stresses in the core shroud were evaluated to the stress criteria of the ASME B&PV Code, Section III, Subsection NG (Reference 5). For a given service loading, the limiting loads on the shroud occur with different assumed shroud cracks. For example, the stresses due to lateral loads are dependent upon the condition of the welds in the vicinity of the radial restraint and the stresses in the H2/H3 ring are dependent upon the condition of both H2 and H3. All combinations of potential weld cracking were considered in the analyses.

During normal operation the maximum stress in the shroud is less than 22% of allowable. Similarly, the tie rod load on the shroud support plate is less than 80% of allowable. The lateral displacement of the shroud sections is discussed in Section 6.1.

4.5 REACTOR PRESSURE VESSEL AND INTERNALS

The response of the reactor vessel to seismic accelerations is not affected by the repair. In particular, a comparison of the reactor vessel accelerations due to seismic loadings for an intact, unrepaired shroud to those for a repaired shroud with a break at the location which results in the largest radial support loads, shows that the vessel accelerations for the two cases are nearly identical.

The stresses in the reactor pressure vessel due to the loads on the radial restraints were evaluated to the original vessel design code (ASME B&PV Code, Section III, Class A vessel, 1965 Edition, including Summer 1966 addenda). These evaluations show that the stresses in the vessel due to the radial restraints are small and well within the code stress allowables for all defined loadings.

As discussed in Section 5.4 the stresses in the top guide and core plate due to lateral seismic fuel loads for the repaired shroud are less than allowable. The lateral displacement of these structures is discussed in Section 6.1.

The evaluation of the seismic loads on the fuel is discussed in Section 5.4.

4.6 LOSS OF PRELOAD

The preload in the tie rods during normal operation is a function of the installation preload, the differential thermal expansion of the tie rods and shroud during heat up to operating temperatures, and the relative stiffness of the tie rods and shroud. As a result, the maximum tie rod load will occur when the tie rods are installed with the shroud welds intact and the welds remain intact during operation. The minimum tie rod load will occur when the tie rods are installed with the shroud welds intact and the welds fail after installation. The entire range of tie rod loads is considered in the structural and displacement evaluations. In addition, the installation preload is selected to provide margin for any loss of preload due to thermal and radiation relaxation effects.

As discussed in Section 6.1, the installation preload is selected so that the shroud will not separate during normal operation even if welds H2 and H3 fail after installation of the repair. The stresses in the tie rods are less than yield during operation. In addition, the tie rods are loaded, unloaded and loaded again during installation to ensure that all components are properly seated prior to tightening and crimping the load nut in place. As a result, a significant reduction in installation preload during operation is very unlikely.

In the unlikely event that installation preload is lost on one of the tie rods, the remaining load is sufficient to prevent the shroud from separating at failed welds under current operating conditions.

4.7 LOOSE PARTS CONSIDERATIONS

The various pieces that make up the repair assemblies are captured and restrained by appropriate locking devices such as locking cups and crimping. These locking device designs have been used successfully for many years in reactor internals. Loose pieces cannot occur without failure of the locking devices or repair assembly components. Such locking devices and the stresses in the pieces which make up the tie-rod/radial restraint system are well within allowable limits for normal plant operation. In addition, the design includes suitable features to prevent detachment of the tie-rods even if preload were lost.

The repair assemblies are fabricated from stress corrosion-resistant material. Therefore it is unlikely that a component will fail. However, in the unlikely event that a tie-rod becomes detached from its attachment point during normal plant operation, there are no nuclear safety consequences to the shroud or to the other tie-rods. If individual components should somehow break off the repair assembly, they would fall to the bottom of the downcomer annulus or if small enough, could be transported into the recirculation loop and its pump. The consequences of a loose component are no different than that postulated from other loose parts from the reactor internals within the recirculation system.

4.8 INSTALLATION CLEANLINESS

A temporary core cover will be utilized to preclude foreign object entry into the core area. All tooling used for installation will be inventoried and subjected to foreign material exclusion procedures when in the reactor vessel area. Furthermore, the tooling will be extensively field hardened prior to site deployment to reduce the possibility of tool failures and/or breaks which could potentially result in loose parts remaining in the vessel. If failures occur most likely the parts could be retrieved from the temporary core cover or from the top of the shroud support.

Four oblong through thickness slots will be machined in the shroud support using the EDM process. The process is such that no slug results from the slot formation. This process will result in a very fine debris (swarf) being generated. This debris will be primarily comprised of carbon, nickel, iron, chromium, etc. which are the primary elements contained in the shroud support ring and EDM electrode material. This swarf will be flushed and vacuumed from the cut during the machining operation. The swarf and reactor water will be filtered prior to discharge back into the cavity. Since the area under the shroud ring is inaccessible, the electrode is designed to assure that predominantly fine swarf is released when the electrode breaks through the underside surface of the shroud support plate to minimize swarf entry into the reactor vessel. However, due to the nature of the process and configuration required, there is the likelihood that some larger particles will remain in the reactor vessel.

Subsequent to the EDM operations, the surfaces of the slots will be honed (approximately 5 mils surface removal depth) sufficient to remove the portion of the recast layer, resulting from the EDM process, which may contain fissures. During honing operations, the swarf generated will be vacuumed from the area. Some may fall below the shroud support plate. The small amount of debris not collected is not detrimental to the BWR system.

Subsequent to completion of the tie rod hardware installation activities, a final video inspection in the reactor vessel and cavity will be performed to verify no foreign object entry during the repair.

Table 4-1

VYNPS Core Shroud Repair
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))
Upset : Thermal Transient	B	Normal Loads + Thermal Transient Load
Upset : Pressure	B	Normal Loads + Upset Pressure Transients
Upset : Operating Basis Earthquake	B	Normal Loads + OBE
Emergency : Safe Shutdown Earthquake	C	Normal Loads + SSE
Emergency : Safe Shutdown Earthquake (During Refueling)	C	Shutdown Loads + SSE (Shutdown Loads include deadweight, tie rod preloads)
Faulted : Steam Line Break	D	Normal Loads + SSE + Steam Line Break
Faulted: Recirculation Line Break	D	Normal Load + SSE + Recirculation Line Break

Section 5

SEISMIC ANALYSES

This section describes the analyses performed to calculate the seismic loads on the reactor internals of the Vermont Yankee Nuclear Power Station with intact and failed core-shroud welds and the MPR core-shroud modification installed. It summarizes the seismic models, the seismic inputs used, and the results obtained. The loads from these analyses are inputs to the design stress analyses discussed elsewhere in this report.

5.1 DESCRIPTION OF THE SEISMIC MODELS

The seismic dynamic models are two-dimensional finite-element beam models. Their scopes include the reactor building, the drywell, the biological shield wall, the reactor pedestal, the reactor pressure vessel, the reactor internals and the core. Figures 5-1, 5-2 and 5-3 illustrate the scope and structure of the seismic models. Time history seismic ground excitation is applied at the base of the reactor building. Three models were utilized: two horizontal models (East-West and North-South) and a vertical model. The models are based on existing seismic models of the primary structures of Vermont Yankee prepared for the replacement of the reactor recirculation system piping (Reference 16).

The geometry, masses, stiffness coefficients, etc. of the existing models documented in Reference 15 were retained except for the addition of the modification hardware and the addition of the weld failures in the individual load cases. The previous seismic models were converted from the original format into ANSYS 5.2 format for use in the present analysis. The conversions of the models were verified by comparing the natural frequencies and seismic forces and moments of the intact models without the core shroud modification installed to those of the previous analysis (Reference 16).

The converted seismic models were modified to add the mass and stiffness coefficients for the MPR core-shroud modification. The modified models include non-linear gap elements to accurately represent the effect of the small gaps between the lateral supports and the vessel wall. Linear elastic elements are used to model all other components. Figures 5-2 and 5-3 show the modifications to the models. The tie rods provide restraint against axial and rotational displacement of the core shroud. The springs K_{aTR} and $K_{\gamma TR}$ represent the axial and rotational stiffness coefficients of the modification, respectively.

The lateral restraints of the core shroud modification prevent excessive displacement of the core shroud components in case of failure of the horizontal welds in the core shroud. Small, radial clearances are provided between the restraints and the reactor vessel. The restraints are modeled with gaps and springs. The upper restraint and the lower-intermediate restraint have radial clearances of about 1/8 inch. These restraints assure that the alignment of the core is maintained within limits. The upper-intermediate restraint and the lower restraint have radial clearances of

about ½ inch. These restraints assure that, in the event that multiple, complete failures of circumferential shroud welds, the shell sections maintain sufficient radial alignment that the walls of the shell sections overlap preventing a fluid flow path from being opened.

The stiffness coefficients of the restraint springs, K_{BR}, K_{B11}, K_{B12}, K_{B2}, K_{B3}, and K_{B4}, represent the resistance of the core shroud to local displacement due to contact of the restraint with the reactor vessel after closure of the radial clearance. The stiffness of the top bracket, which spans from the shroud flange to the H2/H3 transition ring, is represented by the K_{BR} spring. The local stiffness coefficients used at the lateral restraints are calculated for each restraint location using a three-dimensional finite-element model of the core shroud. The stiffness coefficients vary depending on the assumed condition of the core shroud (i.e., depending on the weld break case) being considered in the analysis. The three-dimensional model is shown in Figure 5-4.

The damping values of the seismic models were obtained from the Vermont Yankee Final Safety Analysis Report (Reference 16). These damping values are according to Regulatory Guide 1.61 (Reference 17). The damping varies depending on whether the horizontal or vertical model is being considered and whether an OBE or an SSE is being considered. Table 5-1 summarizes the damping values used.

5.2 SEISMIC INPUTS

The design basis seismic input for Vermont Yankee is the 1952 Taft earthquake anchored at 0.07g for the operating basis earthquake and 0.14g for the safe shutdown earthquake (Reference 12). To provide added conservatism in the shroud repair design, Vermont Yankee specified the use of a USNRC Regulatory Guide 1.60 response spectrum input for the repair seismic analysis. As a result input to the seismic analysis is a time history ground motion at the base of the reactor building which satisfies Regulatory Guide 1.60 requirements for ground motion spectra. The USNRC has previously reviewed and accepted this alternative design approach for Vermont Yankee (Reference 13).

The original seismic design basis for Vermont Yankee assumed no vertical amplification, and applied a vertical seismic load of 0.10g. The analyses of the shroud repair explicitly evaluate the vertical seismic response of the repaired shroud.

Three ground acceleration time histories: two independent horizontal time histories and a vertical time history, plotted in Figures 5-5, 5-6 and 5-7 scaled to a peak ground acceleration of 0.14g, are defined. These time histories are the time histories used in the current FSAR analyses for Vermont Yankee Nuclear Power Station. As shown in Figures 5-8, 5-9 and 5-10, the time histories are independent, synthetic earthquake time histories developed to match a Regulatory Guide 1.60 (Reference 18) acceleration response spectrum. For specific analyses, the time histories are scaled to the appropriate peak ground acceleration based on the seismic event (i.e., OBE or SSE) being considered.

When applying the horizontal time histories to the uncoupled, two-dimensional horizontal seismic models, the time histories are scaled to account for torsional interaction between the East-West and North-South responses of the structure. The scale factor for the East-West horizontal seismic model is 1.15. The scale factor for the North South horizontal seismic model is 1.05. These factors are applied to the time histories after they are scaled to the appropriate peak ground acceleration for the seismic event being considered. When applying the vertical time history to the vertical model, the time history was multiplied by a factor of 1.1 in accordance with the FSAR for Vermont Yankee.

5.3 CORE SHROUD CONFIGURATIONS ANALYZED

The MPR tie-rod modification to the core shroud is designed to accommodate failure of one or more of the horizontal welds in the core shroud. It is also acceptable for installation on an intact core shroud as a preemptive measure. In order to ensure that the limiting seismic loads were evaluated, a large number of assumed core shroud configurations were analyzed. These configurations bound the range of possible configurations. Over 60 seismic analysis runs were performed. The specific configurations evaluated in the seismic analyses are summarized below.

5.3.1 Horizontal Earthquakes - Shroud Configurations Analyzed

The shroud configurations discussed below were analyzed for 1) both the north/south and east/west earthquake loads and 2) both the operating and safe shutdown earthquakes.

Three intact core shroud cases were analyzed. The first intact case is an intact shroud without the tie-rod modification installed. This case, solved by modal superposition, is the verification case discussed earlier, which was used to verify the conversion of the model to an ANSYS model. The second intact case is an intact shroud without the tie-rod modification installed and solved by the direct integration method used to solve the repair break cases. This case is a reference for comparison to the broken-weld configurations analyzed. The third intact case considered is an intact core shroud with the tie-rod modification installed. This case evaluates the preemptive installation of the tie-rod modification on an intact core shroud.

A range of potential single weld and multiple weld failures was considered. The single weld failures analyzed included the failure of H7, H4 and H3. The H7 weld is the lowest-elevation circumferential weld and has the largest mass above it. In addition, breaks below the core support plate result in both lateral core supports (top guide and core support plate) being above the break.

For single breaks between the top guide and the core support plate, the core is laterally supported partly by the shroud and partly by the repair. The H4 weld is an unsupported weld between the core plate and the top guide. The H3 weld is between the top guide and the core support plate and has the shortest moment arm to the top restraint, which carries the lateral load due to the top of the core and the overhanging steam separators.

For single breaks above the top guide, the core is laterally supported by the shroud. For this case the repair assembly is only loaded by the overhanging steam separators. As a result, the loads on

Table 5-1

Damping Values Used in the Seismic Analyses of the Core Shroud Repair[1].

Component Description	Horizontal Directions		Vertical Direction	
	OBE	SSE	OBE	SSE
CRD guide tubes and housings	1%	2%	1%	2%
Reactor pressure vessel and other internals, stabilizers, star truss, etc.	2%	4%	2%	4%
Drywell	2%	4%	2%	4%
Reactor building, biological shield, and reactor-vessel pedestal	4%	7%	4%	7%
Reactor fuel assemblies	6%	6%	4%	6%

[1]Damping values were obtained from the Vermont Yankee FSAR, Section A.10.2.7

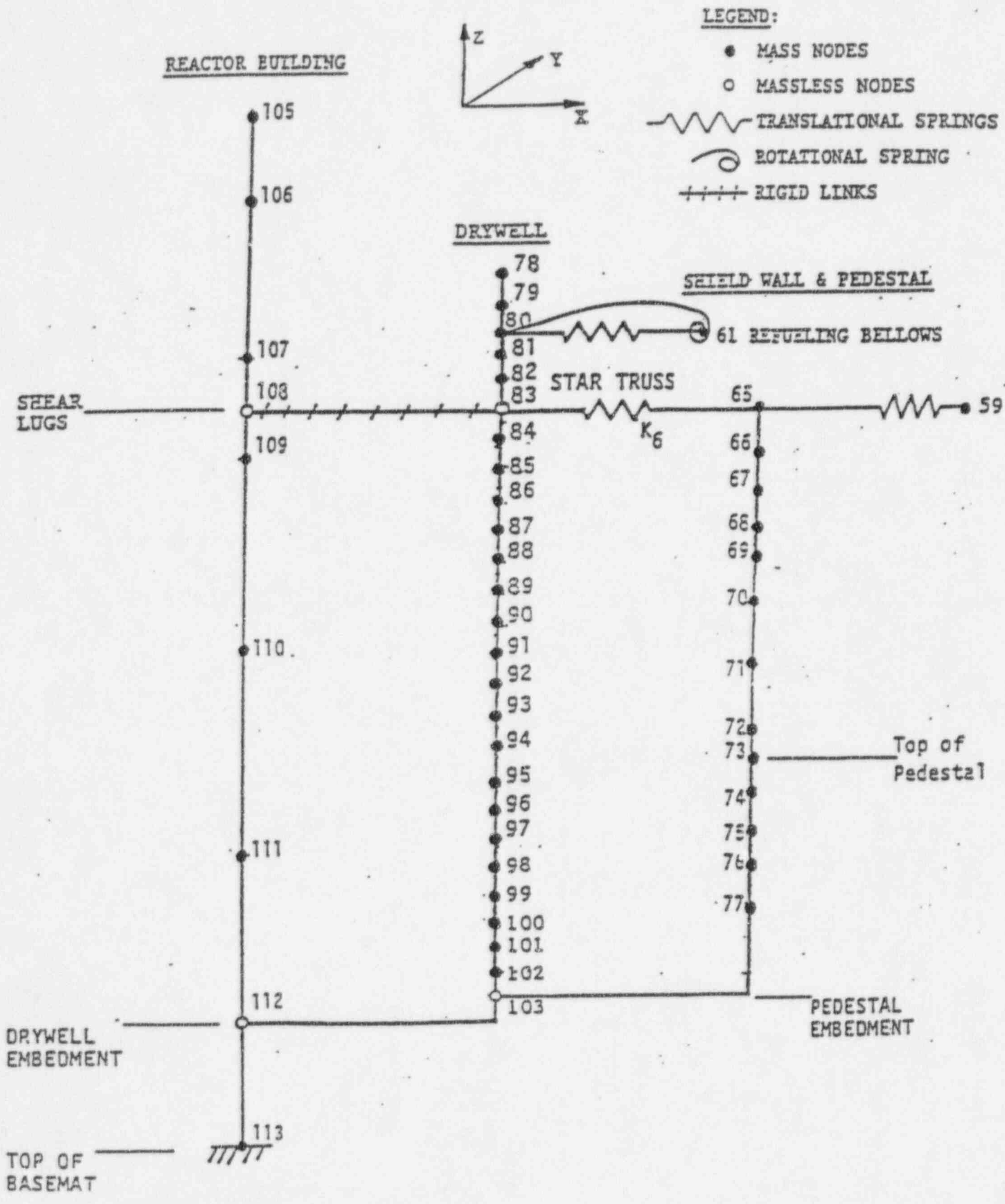

Figure 5-1. Seismic Model of Vermont Yankee Primary Structures — Reactor Building.

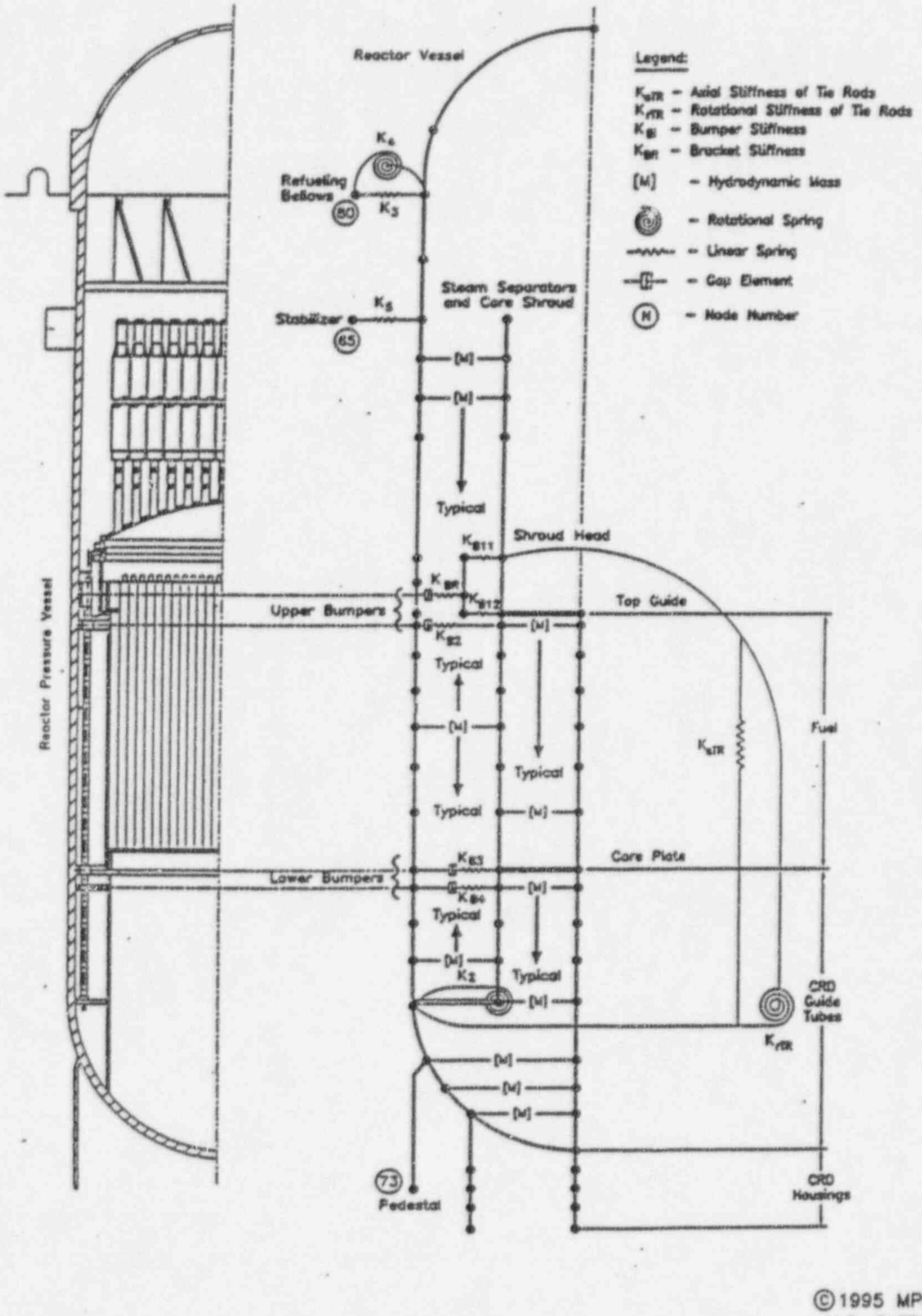

Figure 5-2. Horizontal Seismic Model of Vermont Yankee — Reactor Vessel and Internals.

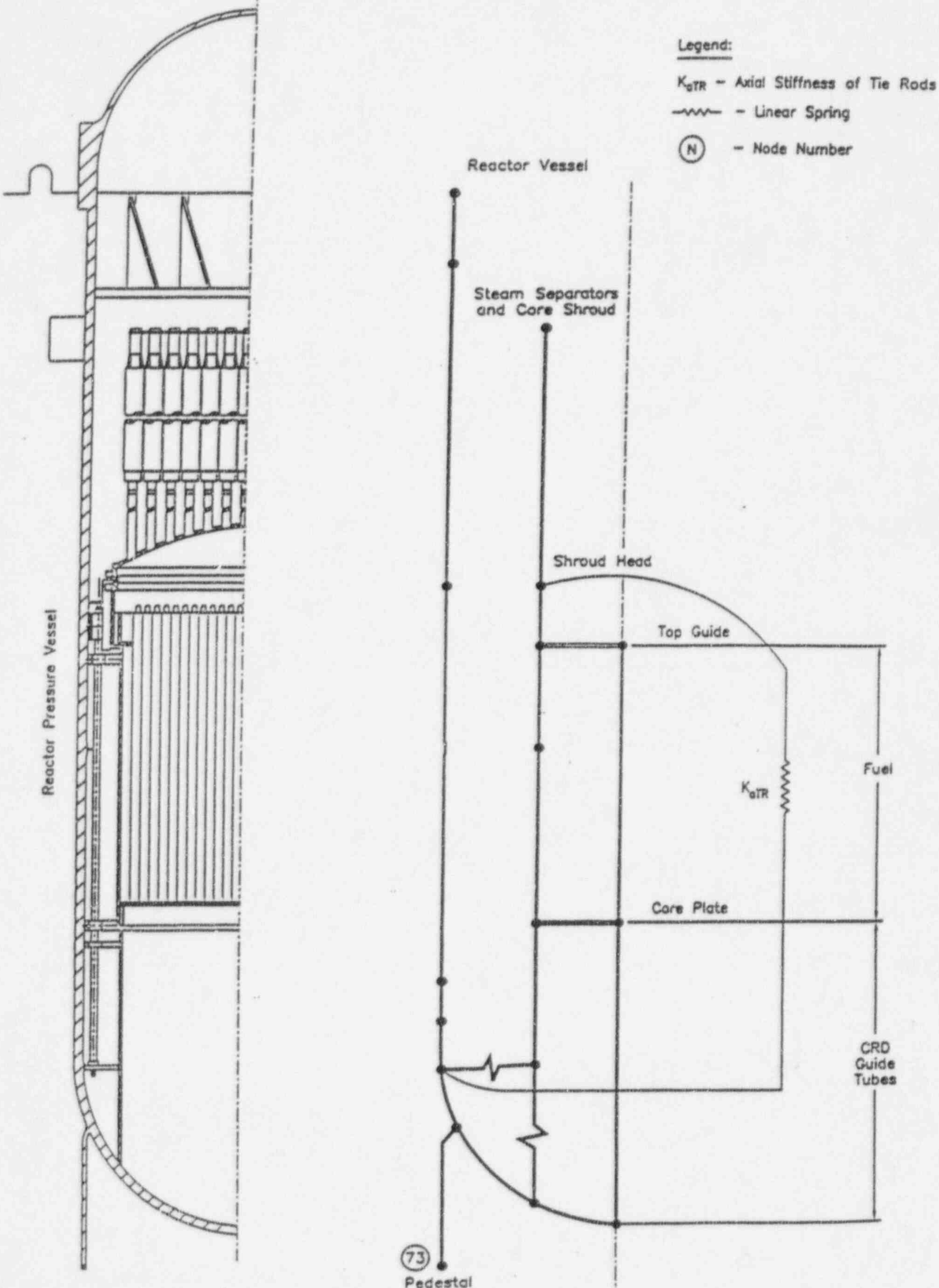

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Figure 5-3. Vermont Yankee Vertical Seismic Model
Reactor Vessel and Internals With Shroud Repair Installed

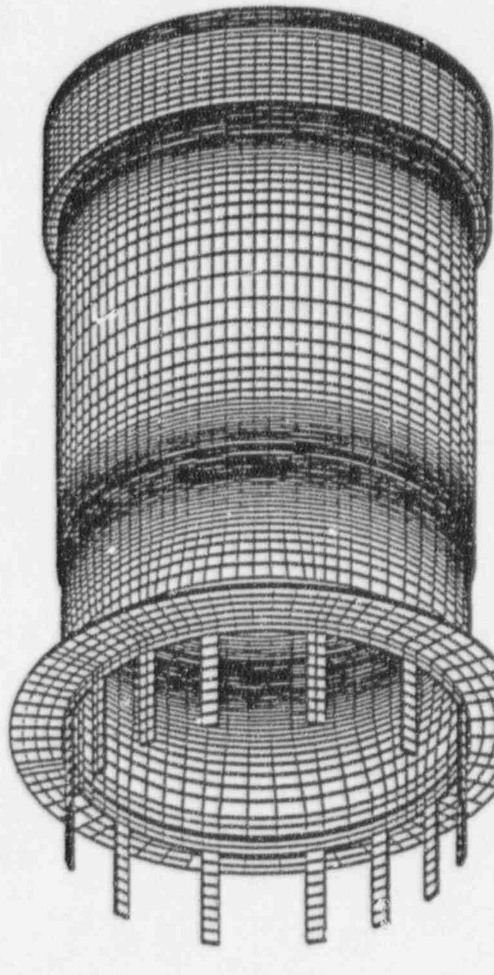

Figure 5-4. Three-dimensional model of Vermont Yankee core shroud.

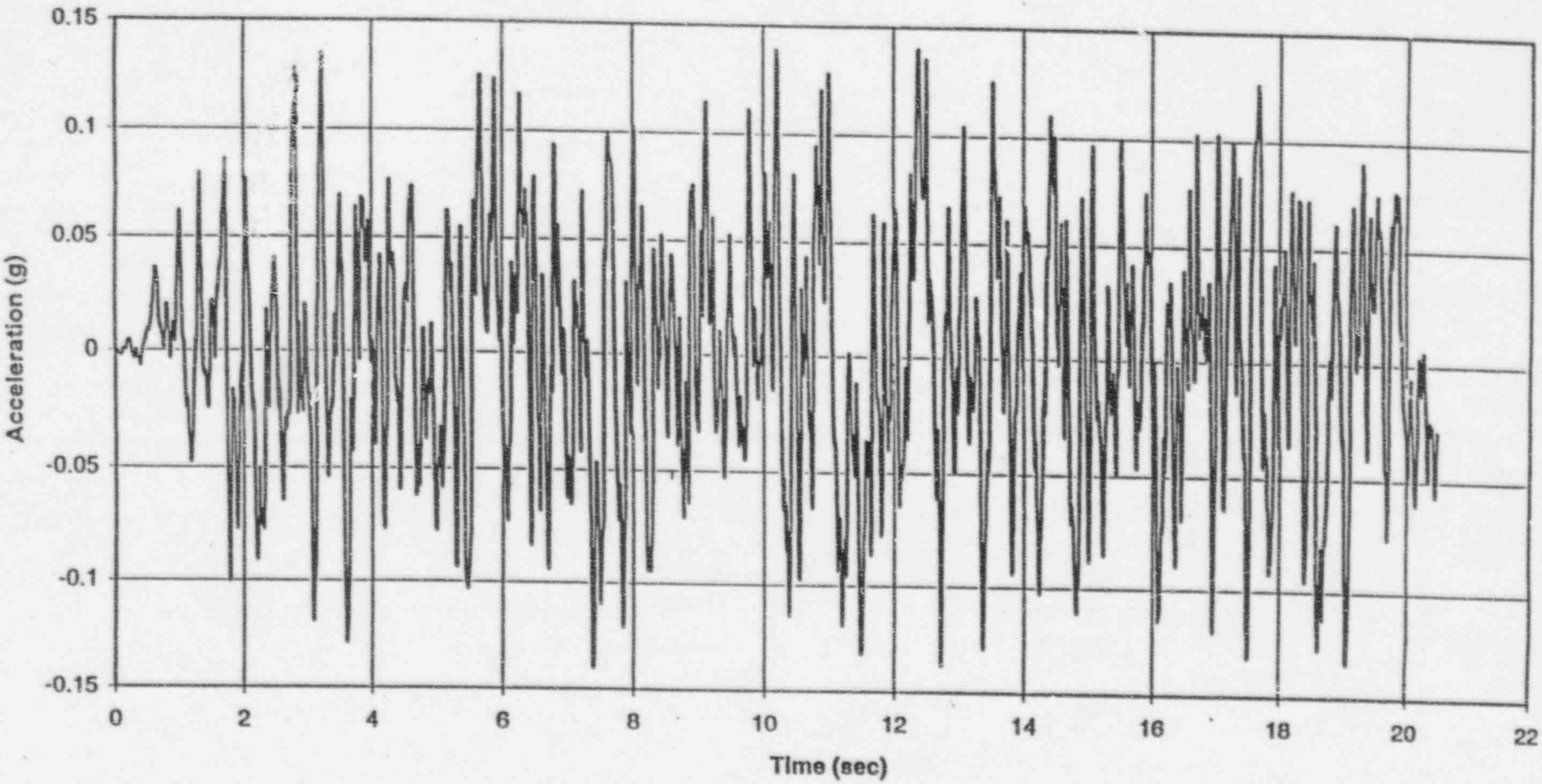

Figure 5-5. Acceleration Time History For The North-South Seismic Ground Motion (H1X)

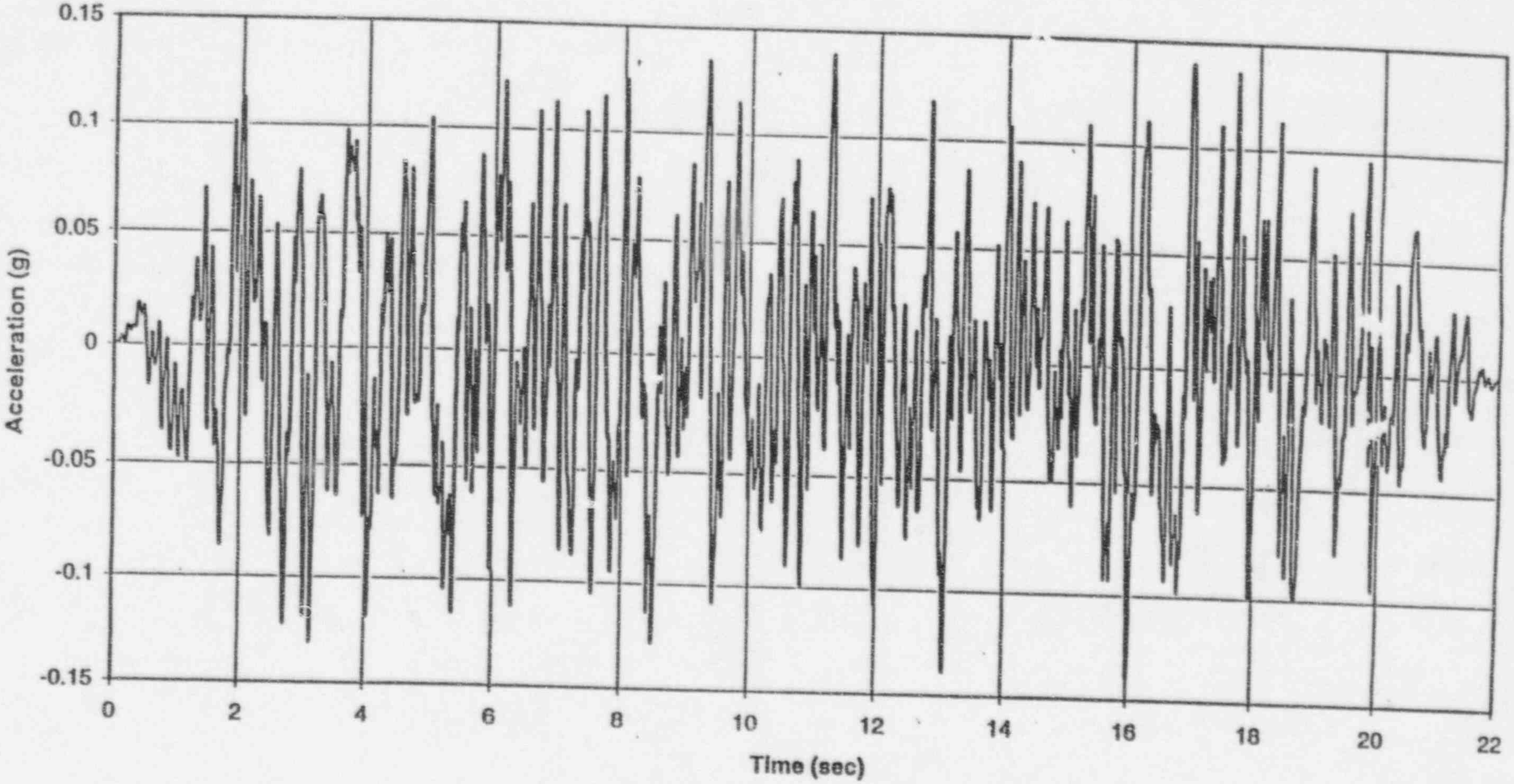

Figure 5-6. Acceleration Time History For The East-West Seismic Ground Motion (H2Y)

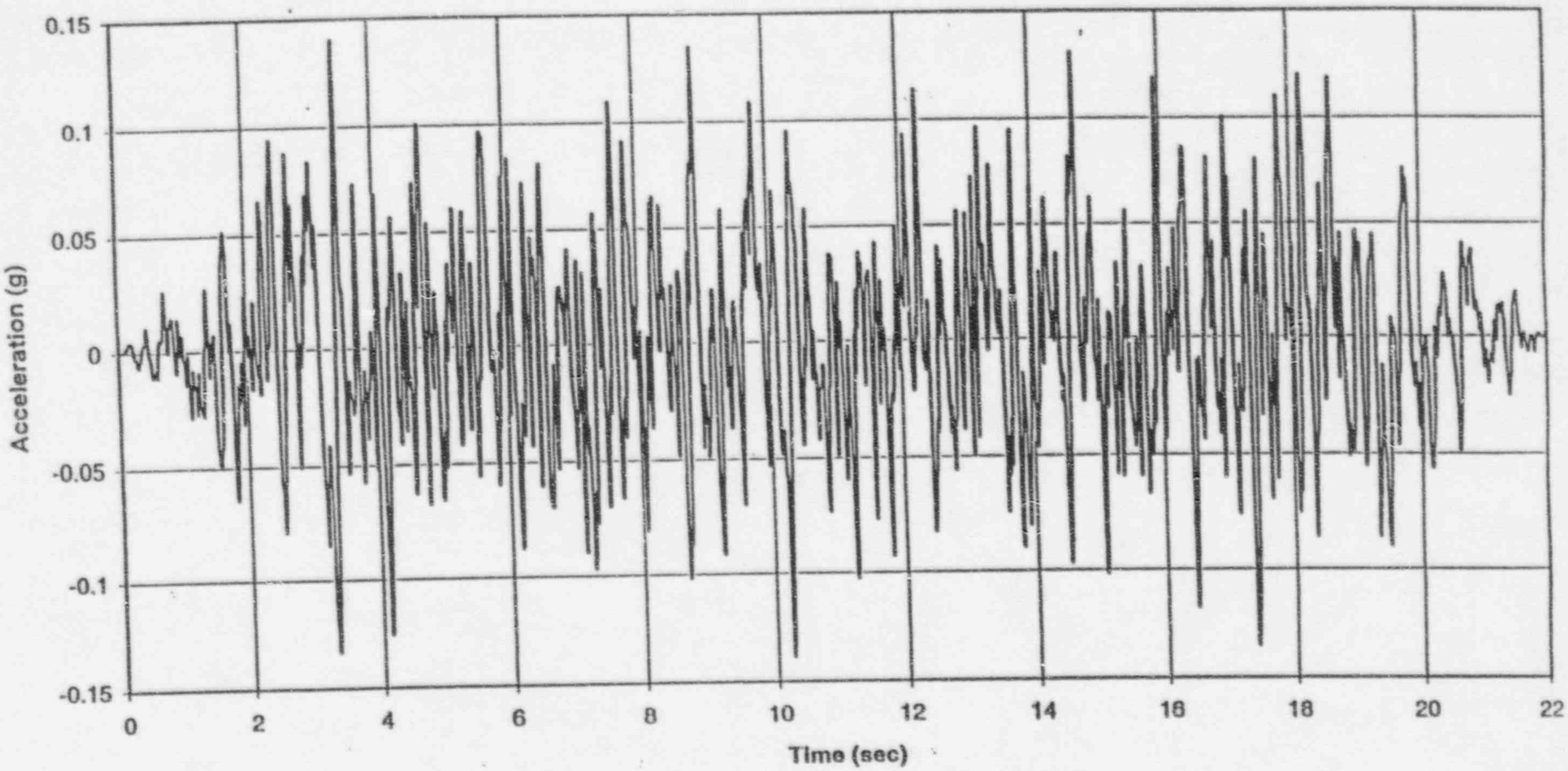

Figure 5-7. Acceleration time history for vertical seismic model.

the tie rod and top bumper for this case are bounded by those for the multiple break case discussed below.

For each of these single break cases, two different weld configurations were evaluated. The two weld configurations use different assumptions regarding the load-bearing capacity of the failed welds. One configuration (the "pinned" configuration) assumes that shear, but no moment, can be carried across the weld. The other configuration (the "sliding" configuration) assumes that neither shear nor a moment can be carried across the failed weld. In this way, both "rough" and "smooth" crack-surface conditions are covered.

The horizontal earthquakes were also analyzed for a multiple weld break case. For this case it was assumed that all of the welds H1 through H7 are broken and sliding. Here, the core shroud is broken into many pieces and the tie rods and lateral restraints support the reactor core and steam separators. This case puts the minimum reliance on the core shroud and the maximum reliance on the tie-rod repair.

Two additional configurations were considered to assess the impact of the failure of H8. Since the maximum loading of the H8 weld occurs when it must carry the entire cantilevered mass of the core, shroud and separators, the other weld break cases (i.e. failure of H7, H4 and H3) were not repeated with H8 broken.

5.3.2 Vertical Earthquakes - Shroud Configurations Analyzed

In the evaluation of vertical earthquakes three intact core shroud cases were analyzed. These cases are analogous to the intact shroud cases performed for the horizontal earthquakes discussed above. These cases were used to verify the conversion of the model to an ANSYS model and to demonstrate that the installation of the repair has no impact on the response of an intact shroud.

A failure of the weld at H7 was considered in the analyses for the vertical earthquakes. A break at this location maximizes the mass restrained by the tie rods. In addition, since this weld is below the core plate, a break at this location would also result in the largest upload due to differential pressure.

Multiple weld failures were also considered in the analyses. In particular, the failure of H2, H3 and H7 was evaluated. As discussed above, the failure of H7 maximizes the mass restrained by the tie rods, while the failure of H2 and H3 would minimize the compressive load across the failed welds due to deadweight and tie rod load.

Finally, two additional configurations were evaluated to assess the impact of the failure of H8.

These shroud configurations were analyzed for both the operating and safe shutdown earthquake. In addition, the single and multiple break configurations were analyzed for a safe shutdown earthquake with a main steam line break; the large pressure drop across the core plate and shroud head during a main steam line break result in separation of the shroud at the failed weld.

5.4 SEISMIC ANALYSIS RESULTS

Seismic forces and moments were calculated for each of the seismic load cases described in the previous section of the report. The response to vertical, North/South and East/West seismic analyses were combined by SRSS for use in the evaluation of the repair hardware and the core shroud. The stress analyses are described elsewhere in this report.

The analyses show that the loads for the fuel are not substantially changed by the repair for both intact and failed core shrouds. A comparison of the maximum fuel acceleration (obtained by SRSS summation of the accelerations calculated for the north-south and east-west earthquakes) to the fuel vendor's proprietary value of maximum allowable acceleration show the fuel accelerations to be acceptable with substantial margin (Reference 19).

As shown in Appendix C of Reference 2, for the current loads the ratio of calculated to allowable stress in the top guide and core plate is less than 0.86 and 0.62 respectively. With the repair installed and a break at H3 there is a small increase in the seismic fuel loads. However, for this limiting break case, the top guide stresses are still less than 91% of allowable and the core plate stresses are less than 70% of allowable.

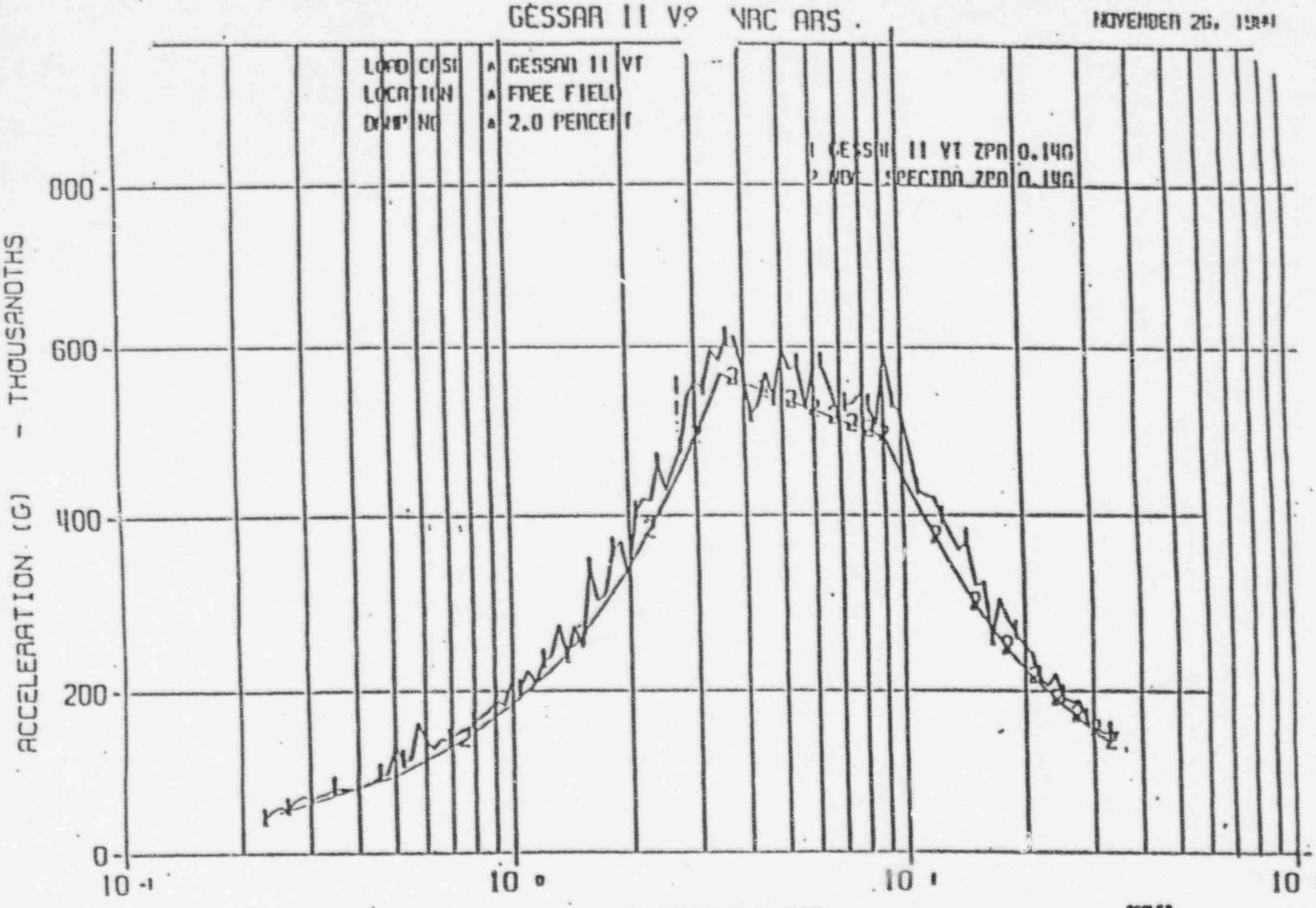

Figure 5-8. Acceleration Response Spectrum For The East-West Seismic Ground Motion

SCALES USED

VERMONT YANKEE
NUCLEAR POWER STATION

GESSAR II vs. NRC ARS

FIGURE A.10.3-6

REV G

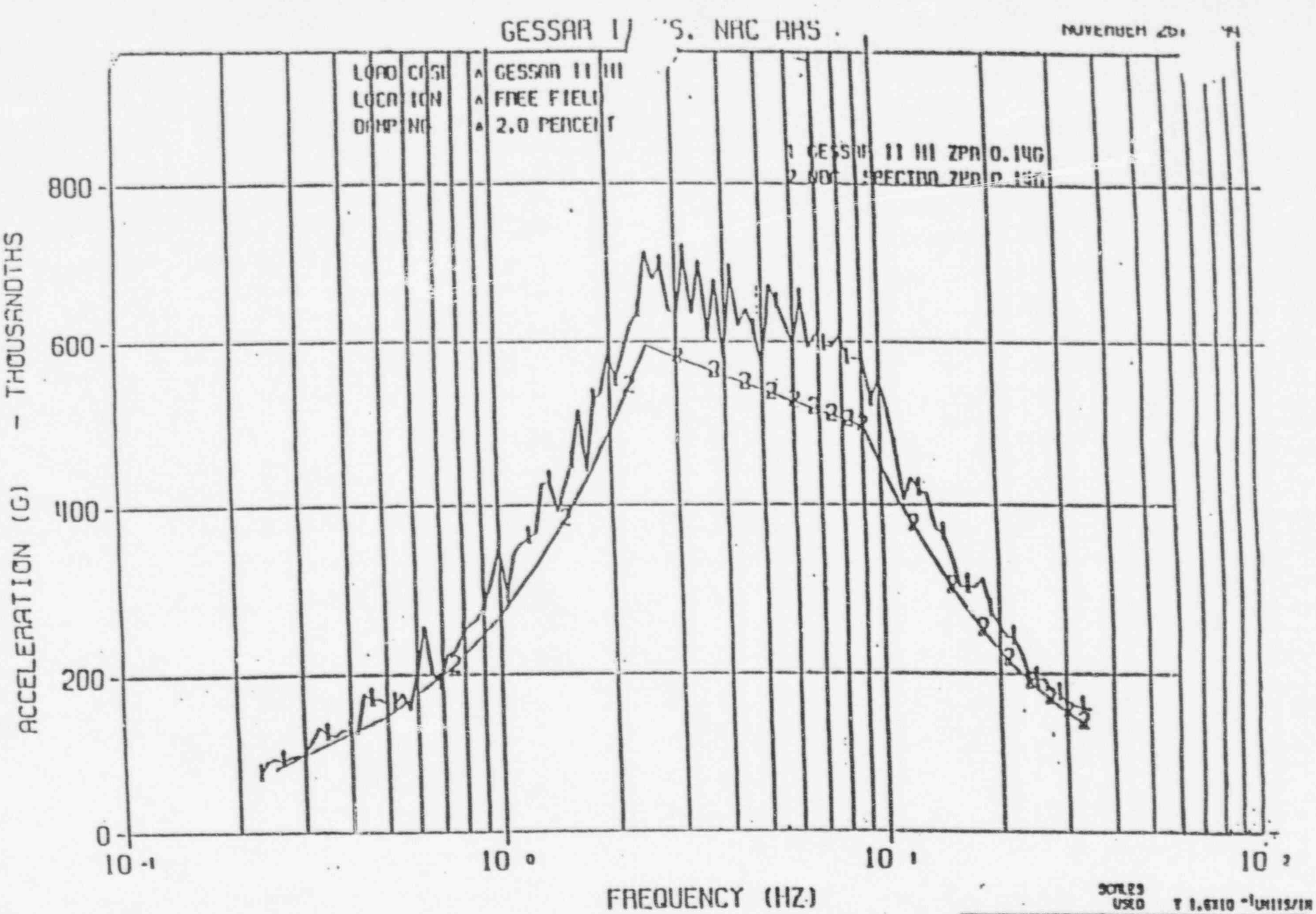

Figure 5-9. Acceleration Response Spectrum For The North-South Seismic Ground Motion

REV 6

VERMONT YANKEE
NUCLEAR POWER STATION

GESSAR II vs. NRC ARS

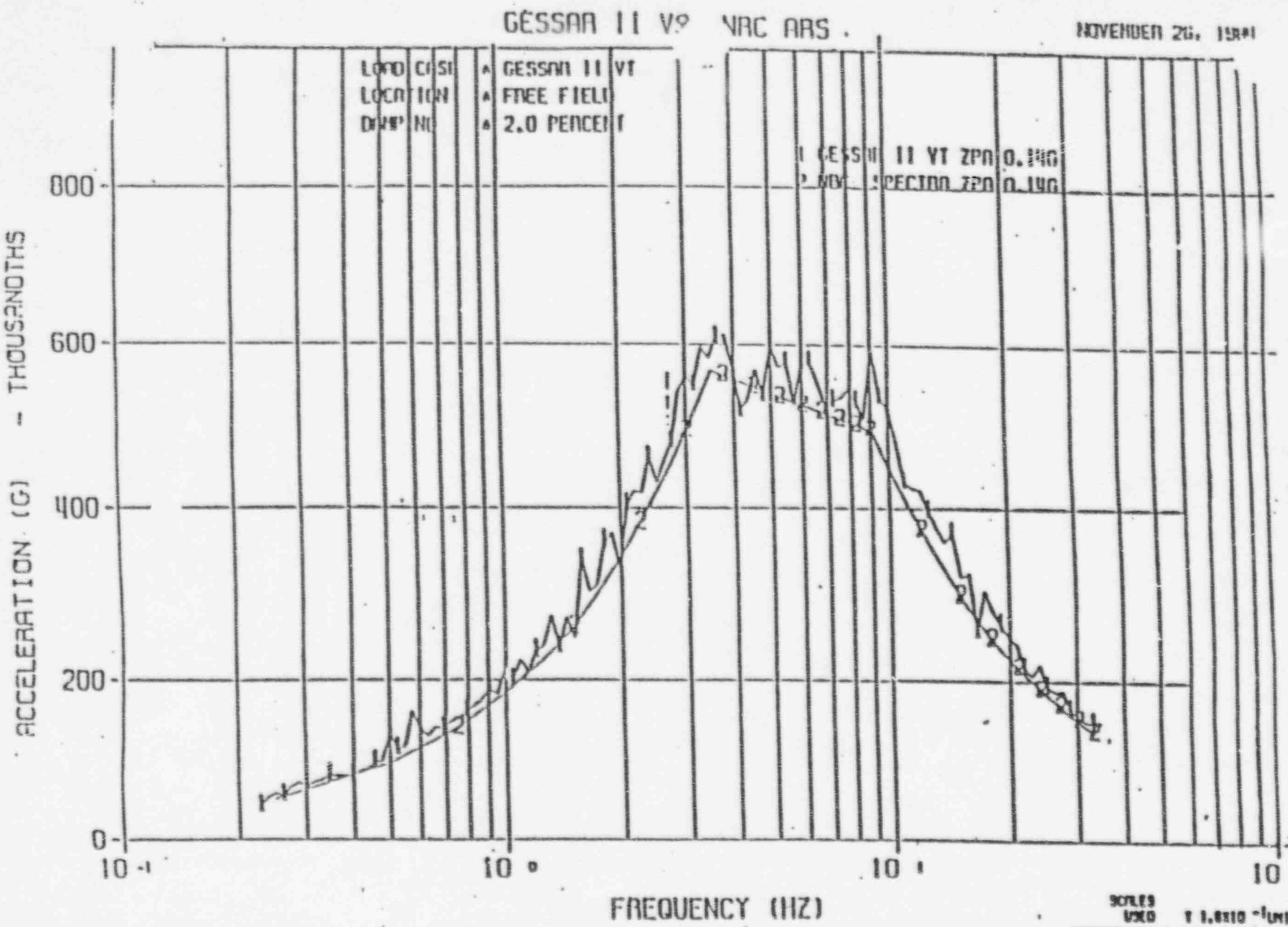

Figure 5-10. Acceleration Response Spectrum For The Vertical Seismic Ground Motion

REV 6

VERMONT YANKEE
NUCLEAR POWER STATION

GESSAR II vs. NRC ARS

FIGURE A.10.3-4

Section 6

SYSTEMS EVALUATION

6.1 SHROUD DISPLACEMENT

The potential for vertical separation and/or lateral displacement of the shroud cylinders was evaluated for all loadings and the range of potential circumferential weld failures. The evaluations show that even during the limiting case (i.e. welds H2 and H3 fail after installation), and with a 15% margin on the shroud pressure differentials, a compressive load is carried in all shroud sections. As a result, there is no separation of the shroud during normal and upset operational transients.

For the current operating conditions no separation of the shroud would occur during an OBE event. However, with a 15% increase in pressure differentials some small temporary separation could occur due to tipping of the core shroud during an OBE. Similarly, a small temporary vertical separation could occur during a SSE or SSE plus main steam line break event. After the temporary separation no significant shroud bypass flow will occur. The small temporary vertical displacement will not affect the Core Standby Cooling Systems.

The lateral displacement of the core shroud is limited by the radial restraints provided on each tie rod assembly. The displacement of the shroud rings at the top guide and core support plate are limited to less than 0.188 inches for all service loadings by the radial restraints at these locations. This small displacement is much less than the allowable lateral displacements of the top guide and core plate of 0.96 and 0.33 inches respectively for Service Level A/B loadings (Reference 20). As a result, these displacements provide a significant margin for Service Level A/B, C and D loadings.

The lateral displacements of the remaining shroud cylindrical sections are also limited by radial restraints. The displacement of the shell sections between H3 and H5 is limited to less than 0.75 inches. This ensures that the 1.75 inch thick shell sections will always be over lapped, preventing the formation of an additional leakage path. Similarly, the shell section between H6 and H7 is limited to a lateral displacement of less than 0.75 inches. This ensures that the shroud sections will overlap.

6.2 BYPASS FLOW

Although the shroud will not separate at failed welds during normal or upset operating conditions, some leakage may occur through the cracked welds (H1 through H8). Some small amount of additional leakage may occur across the seal rings at the four locations at which the repair assemblies are attached to the shroud support. The slots in the spacer ring

between the shroud flange and head are sufficiently shallow to prevent a leakage path between the upper core plenum and the vessel downcomer from being formed.

A summary of the potential shroud leakage flow rates is provided in Table 6-1. The leakage across the shroud was evaluated for shroud pressure differentials 15% greater than the current design values. Bypass flow through cracked circumferential shroud welds is conservatively estimated assuming that each weld develops a complete circumferential crack that opens to 0.001 inches. The seal rings provided at each of the repair assembly-to-shroud support plate attachment points are preloaded against the shroud support plate. This preload is independent of the tie rod load. A conservative estimate of the potential bypass flow across the seal rings was obtained by assuming that a 0.001 inch gap exists between the seal ring and shroud support plate.

The total maximum calculated bypass flow of 101 gpm (0.078% of core flow) is sufficiently small such that the steam separation system performance, jet pump performance, core monitoring, fuel thermal margin and fuel cycle length are not affected by the repair. Similarly, the impact on CSCS performance is insignificant.

6.3 NORMAL OPERATION

As discussed above, the potential leakage across a repaired shroud is negligible and has no significant impact on plant operation.

6.3.1 Steam Separation System

Leakage flow through cracks in welds H1 and H2 occur above the top guide support ring. This flow would slightly increase the total carryunder in the downcomer. The total leakage flow also has the effect of slightly decreasing the flow per separator and slightly increasing the separator inlet quality. However, the leakage flowrates are very small and judged to have a negligible effect on overall steam separator performance.

6.3.2 Recirculation System

The installation of the repair assembly will have a negligible effect on the downcomer flow area. The repair assembly will decrease the available flow area in the downcomer at the top of the shroud by less than 6%. The pressure drop associated with this small restriction is about 0.005 psi. The small leakage flow rates have a negligible effect on the subcooling of the downcomer flow. Accordingly, the overall effect of the repair on recirculation system flow and pressure drop is insignificant.

6.3.3 Core Monitoring System

The small leakage flowrate has a negligible effect on core flow and power relative to the normal instrumentation power uncertainty of 1 to 2% (Reference 14). Therefore it is concluded that the impact of the repair on the core monitoring system is not significant.

6.3.4 Operating and Fuel Cycle Length

The increased carryunder due to leakage flow above the top guide would result in a slight increase in the core inlet enthalpy, compared with the no leakage condition. The combined effect of the slightly increased enthalpy and the slightly reduced core flow due to leakage is judged to have a negligible impact on fuel cycle length.

6.4 ANTICIPATED OPERATIONAL OCCURRENCES AND CORE OPERATING LIMITS

As discussed above, the small leakage rates associated with the repaired shroud will result in a small increase in carryunder and core inlet enthalpy, and a small reduction in core flow. The changes are very small and are judged not to affect core operating limits. As discussed in Reference 14 there is no impact on safety limits even for a cracked and unrepaired shroud.

6.5 LOSS OF COOLANT ACCIDENT ANALYSIS AND CSCS PERFORMANCE

The Core Standby Cooling System (CSCS) includes the Core Spray, Low Pressure Coolant Injection, and High Pressure Coolant Injection Systems. A cracked shroud could potentially affect the performance of the CSCS by affecting the distribution or flowrate of coolant provided by the system.

Since the core spray system penetrates the core shroud between H1 and H2, the potential exists to affect the operation of this system. The maximum calculated displacement of this section of the shroud with the repair installed was determined to occur for a main steam line break concurrent with a SSE. The maximum vertical displacement is less than 1 inch. Analyses show that this displacement would result in acceptable stresses in the core spray piping in the vessel. As a result, the performance of the core spray system is not affected by the repair.

The water level in the core following a recirculation line break is maintained by the CSCS to a level equal to the jet pump suction. The small leakage paths associated with a cracked and repaired shroud will have a very small impact on the CSCS flowrate required to maintain this water level. The leakage rate through cracks during a recirculation line break is estimated to be less than 101 gpm. This leakage is negligible relative to the single-pump CSCS capacity of 2838 gpm for the Core Spray pump, 6570 gpm for the low pressure injection pump, and 4250 gpm for the high pressure injection pump (Reference 2, Appendix E, FSAR Table 6.5-9). Therefore the leakage paths has no impact on CSCS performance during a recirculation line break event.

As a result, the overall CSCS performance is not changed by the repair.

Table 6-1

Summary of Shroud Bypass Leakage Flows

Location	Leakage Flow [1] (gpm)	Leakage-to-Core Mass Flow (%)
Weld Cracks (H1 Through H8)	90.4	0.07
Seal Rings At Repair Assembly-to-Shroud Support Plate	10.3	0.008
TOTAL	100.7	0.078

NOTES:

1. Estimated leakage is for normal operating conditions with a 15% increase in shroud differential pressures.

Section 7

MATERIALS AND FABRICATION

7.1 MATERIALS SELECTION

The materials specified for use in the repair assemblies are resistant to stress corrosion cracking and have been used successfully in the BWR reactor coolant system environment. The repair assemblies are fabricated from solution annealed Type 304 or 304L stainless steel, solution annealed Type XM-19 stainless steel and alloy X-750 per EPRI NP-7032. Type 304 stainless steel is used for the top bracket and radial restraints. X-750 material is used for the spring rod assembly and top adapter. XM-19 is used for the bottom adapter.

As required by the shroud design specification, all materials specified for use in the shroud repair are in accordance with ASME or ASTM approved specifications. All materials have been previously used in the BWR environment similar to that seen by the repair assembly. The materials are not susceptible to general corrosion and are resistant to Intergranular Stress Corrosion Cracking (IGSCC) in a BWR environment. Additional information on material specification, procurement and fabrication requirements implemented to ensure that the repair hardware is highly resistant to IGSCC is provided in Sections 7.2 and 7.3.

Material properties and allowable stresses for repair components are as specified in the ASME B&PV Code, Sections II and III, 1989 Edition for Class 1 components. For Alloy X-750 material, allowable stresses are determined from Code Case N-60-5.

7.2 MATERIAL PROCUREMENT SPECIFICATIONS

All tie rod hardware items are constructed from either austenitic stainless steel or alloy X-750. Welding on these materials is prohibited by the procurement requirements. These materials as procured, are highly resistant to IGSCC. NDE of material used for load bearing members is performed in accordance with ASME Code Section III, Subsection NG-2000. Specific material requirements are summarized below for the material used in the repair.

- Austenitic Stainless Steel

  All stainless steel items are procured in accordance with the applicable ASME or ASTM standards supplemented by the following:

  - All stainless steel alloys are either Type 304, 304L, (F)XM-19. Type 304 alloys have 0.03% maximum carbon. Type (F)XM-19 alloy has 0.04% maximum

carbon. All stainless steel materials are full carbide solution annealed and either water or forced air quenched from the solution annealing temperature, sufficient to suppress chromium carbide precipitation to the grain boundaries in the center of the material cross section.

- Solution annealing of the material is the final process step in material manufacture. For material procured to SA(A)479, Supplementary Requirement S5 is applicable, or the yield strength (0.2% offset) is limited to 52 ksi maximum for the 300 series stainless steel and 84 ksi for the (F)XM-19 material. ASTM A262 Practice E tests are performed on each heat/lot of stainless steel material to verify resistance to intergranular attack and that a non-sensitized microstructure exists (no grain boundary carbide decoration).

- Pickling, passivation or acid cleaning of load bearing members is prohibited after solution annealing unless an additional 0.010 inches material thickness is removed by mechanical methods. For other non load bearing items, metallography at 500X is performed on materials from each heat, similarly processed, to verify excessive intergranular attack has not occurred.

- Controls are also specified in the procurement documents to preclude material contamination from low melting point metals, their alloys and compounds, as well as sulfur and halogens, during material processing and handling.

• Alloy X-750

  Alloy X-750 Condition CIB is also used for some items. This material is in general conformance with EPRI NP-7032, "Material Specification for Alloy X-750 for Use in LWR Internal Components (Revision 1)". One exception is that forced air cooling from the solution annealing temperature instead of water quenching is permitted. The heat treated cross section is sufficiently small to still obtain the desired microstructure throughout the section. The material has either Class A or Class B microstructure and shows acceptable behavior when subjected to the rising load tests. These tests confirm acceptable resistance to IGSCC.

7.3 MATERIALS FABRICATION

No welding or thermal cutting is used in the fabrication and assembly of the items. Cutting fluids and lubricants are approved prior to use. Controls are also specified to preclude material contamination from low melting point metals, their alloys and compounds, as well as sulfur and halogens, during processing and handling. Passivation, pickling or acid cleaning of the items is prohibited. Liquid penetrant testing after final machining or grinding on critical surfaces will be performed.

Abusive machining and grinding practices will be avoided. Machining and grinding process parameters and operations will be controlled. Additionally, machining process parameters in critical load bearing threaded areas will be controlled, based on qualification samples, which have been subjected to macroscopic and metallographic examinations and microhardness testing. Evaluations will include hardness magnitudes and depths, depth of severe metal distortion, depth of visible evidence of slip planes and depth of cold work.

Solution anneal heat treatment will be performed on the load bearing threaded areas on those items constructed of 300 series (i.e. the main load nut) or (F)XM-19 stainless steel (i.e. the bottom adapter). This heat treatment will also be based on qualification samples to verify maintenance of mechanical properties, dimensional stability, grain size and intergranular corrosion resistance per ASTM A262 Practice E. The cold work depth on the Alloy X-750 in the threaded areas will also be limited to a maximum depth of 0.003 inches, to minimize the potential for service related performance degradation.

Section 8

PRE-MODIFICATION AND POST MODIFICATION INSPECTION

8.1 PRE-MODIFICATION INSPECTION

Prior to installation of the shroud repair, Vermont Yankee will perform ultrasonic inspections of design reliant welds. These inspections will cover portions of the vertical welds in the H3/H4, H4/H5 and H6/H7 shroud segments, the welds in the core support ring and welds H8 and H9.

The repair relies on portions of the vertical welds in the H1/H2 shroud segment to be intact. However, due to tooling limitations, it is not practical to ultrasonically inspect the vertical welds in the H1/H2 shroud segment. Therefore, rather than inspect these vertical welds, portions of circumferential welds H1 and H2 are designated as design reliant welds; these circumferential welds provide an alternate path for the loads carried by the vertical welds. Welds H1 and H2 were ultrasonically inspected in 1995. The results of these inspections will be used to demonstrate that sufficient design reliant weld length exists. It should be noted that Vermont Yankee is considering H1 and H2 as design reliant welds only for inspection reasons and that the repair is designed as a repair to H1 and H2.

The specific scope of the pre-modification inspections is as follows:

The specific scope of the pre-modification design reliant weld inspections will be further detailed in the 1996 Outage Core Shroud Inspection Scan Plan (available for review at Vermont Yankee).

In addition to the design reliant weld inspections above, Vermont Yankee (as a minimum) will perform the following pre-modification/installation reviews and inspections.

- Visual inspection of annulus area for tie rod installation interference and annulus cleanliness.

- Review of plant drawings for possible installation interference in the annulus area.

- Review of plant drawings for tooling access into the annulus area.

- Review of plant drawings for equipment access and laydown.

- Review of plant refueling floor area for equipment access and laydown.

8.2 POST MODIFICATION INSPECTION

8.2.1 Prior To RPV Reassembly

Prior to reactor pressure vessel reassembly, visual inspections will be performed by TV to verify the proper installation of repair. The scope of these inspections is summarized as follows:

- Top and both sides of bracket to confirm proper seating,

- Nut to confirm crimping.

- One side of the lower end of bracket and upper outer sleeve to assure the pin of the outer sleeve properly mates with the slot in the lower end of the bracket and that clearance exists between the bottom of the bracket and top of the outer sleeve,

- One side of the tie rod assembly full height to confirm proper assembly of outer sleeves and radial supports, and

- One side of the seal ring to verify the engagement with the slot in the shroud support plate. To verify that the bottom "t" adapter is correctly oriented, check proper engagement of the pin with the lowest outer sleeve.

8.2.2 During Subsequent Refueling Outages

Inspection of the shroud and the repair in future refueling outages will be based on the "Guidelines for Reinspection of Core Shrouds" recently developed by the BWRVIP. The actual inspection scope will be submitted to USNRC at least 90 days prior to the start of the 1998 refueling outage.

Section 9

REFERENCES

1. MPR-1730, "Vermont Yankee Nuclear Power Station Core Shroud Repair - Design Report", Revision 0, April, 1996,

2. VYS-046, "Specification For Design, Fabrication, and Installation Services for Reactor Pressure Vessel Core Shroud Repair at Vermont Yankee Nuclear Power Station," Yankee Atomic, Revision 1.

3. BWRVIP Core Shroud Repair Criteria, Revision 1, September 12, 1994

4. Design Specification For Vermont Yankee Nuclear Power Station (VYNPS) Core Shroud Repair, MPR Specification 249001-001, Revision 1.

5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, "Core Support Structures," 1989.

6. BVY 94-82, VYNPC Letter to USNRC, dated August 12, 1994.

7. BVY 95-45, VYNPC Letter to USNRC, dated April 21, 1995.

8. BVY 95-55, VYNPC Letter to USNRC, dated May 24, 1995.

9. NVY 95-01, USNRC Letter to VYNPC, dated January 5, 1995.

10. NVY 95-52, USNRC Letter to VYNPC, dated April 25, 1995.

11. NVY 95-55, USNRC Letter to VYNPC, dated April 22, 1995.

12. "Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission, in the Matter of Vermont Yankee Nuclear Power Company, Vermont Yankee Nuclear Power Station, Docket No. 50-271", June 1, 1971.

13. NVY 85-168, USNRC Letter to VYNPC, dated August 12, 1985.

14. GENE-523-A018-0295, "Duane Arnold and Vermont Yankee Shroud Safety Assessment," Revision 0, April 18, 1995.

15. GE-NE- 23A4591, "Vermont Yankee Nuclear Power Station, Primary Structure Seismic Analysis, Vermont Yankee Nuclear Power Corporation," Revision 1, General Electric Company.

16. Final Safety Analysis Report, Vermont Yankee Nuclear Power Station, Vermont Yankee Nuclear Power Corporation, Revision 13.

17. Regulatory Guide 1.61 "Damping Values for Seismic Design of Nuclear Power Plants," Revision 0, U. S. Nuclear Regulatory Commission, October 1973.

18. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, U. S. Nuclear Regulatory Commission, December 1973.

19. OPVY 93/96, Yankee Atomic Letter to Framatome Technologies, dated March 29, 1996.

20. GENE-771-44-894, "Justification for Allowable Deflections Of The Core Plate and Top Guide Shroud Repair", Revision 2, November 16, 1994.

Appendix B

Vermont Yankee Core Shroud Modification

Core Cover Evaluation

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1.0 Objective

The objective of this evaluation is to assess the effect of installing a core cover during the repair of the Vermont Yankee core shroud during the 1996 refueling outage.

The purpose of the core cover is to provide a foreign material exclusion barrier over the core and act as an aid to the repair process by providing a method for assisting tool manipulation.

The core cover is an aluminum circular disk manufactured from perforated plate with 1/8 inch holes on 3/16 inch centers. The minimum weight is 1600 lb. See Reference 6.1.

This calculation will assess the added pressure drop of the core cover and the ability of the core cover to remain in place during the inadvertent start of four LPCI pumps and two CS pumps.

2.0 Method of Solution

The pressure drop of the core cover will be calculated.

The weight of the cover will be compared against the flow induced delta-pressure to ensure there is no uplift.

The natural circulation driving head will be compared against the delta-pressure to ensure that core flow is not adversely affected.

3.0 Inputs and Assumptions

Normal shutdown cooling flow is 7000 gpm or less.

Inadvertent start of four LPCI pumps will develop 28000 gpm; the two CS pumps will develop 3000 gpm each.

Refueling water level will be at or above building elevation 343 ft.

Adequate shutdown cooling is achieved as long as the core outlet temperature is below saturation temperature for the existing water level.

4.0 Calculation/Analysis

The core cover plate is perforated aluminum with 1/8 inch diameter holes on 3/16 inch centers. This is the same hole pattern as the RHR suction strainers; see References 6.1 and 6.2. The RHR suction strainers were tested by their vendor and have a measured pressure drop of 0.1 psid at 7000 gpm. From Reference 6.3, the area of one strainer is 3398 square inches. The specific flow rate is 7000 gpm/3398 in^2 or 2.06 gpm/in^2.

The inside diameter of the core shroud at the elevation of the core cover is a minimum 168 inches per Reference 6.3. This provides a minimum flow area of 22,167 in^2; at 34000 gpm the specific flow would be 1.53 gpm/in^2.

This would result in a pressure drop of $(1.53/2.06)^2$ * 0.1, or 0.055 psid.

Conservatively assume that the delta-pressure acts on an area equivalent to the projected area of the core cover (neglecting the fact that the core cover is 34 percent open area due to the holes); at an area of 22,167 in^2 the uplift force would be 22167*0.055 = 1219 pounds. This compares to a core cover weight of 1600 pounds. No uplift would be predicted. This is conservative, since in reality one train of RHR will be out of service for maintenance during the shroud repair.

In order to ensure no boiling occurs in the core during refueling the core cover must allow a flow of at least 7000 gpm. The specific flow of the core cover at 7000 gpm equals (7000/22,167) or 0.316 gpm/in^2. The pressure drop of the core cover at 7000 gpm equals $(0.316/2.06)^2$ * 0.1 or 0.002 psid.

The limiting case would be natural circulation. The driving force for natural circulation is the density difference between the inner fluid and the outer fluid. The inner fluid temperature is taken to be the linearized difference between the core inlet temperature and the core outlet temperature; it is assumed to be at the 2/3 core height, the inlet to the jet pumps.

The core outlet temperature is limited to the saturation temperature assuming the refueling cavity is filled to at least the 343 foot elevation.

The normal elevation of the refueling water level is elevation 343' 6.75" (from Reference 6.6) which is 919.7 inches above vessel zero (which is elevation 266' 11"). The top of active fuel is 351.5 inches above vessel zero. (See Reference 6.4). The height of water is (919.7-351.5) or 568.2 inches or 47.3 feet. At a bulk pool temperature of 110F the pressure at the top of active fuel equals [(1/0.016165) lb/ft^3 * 47.3 ft] or 2926 lb/ft^2 or 20.3 psig. Density taken from Reference 6.5.

Pressure at core outlet equals 20.3 + 14.7 or 35 psia. This is equivalent to a saturation temperature of 259F.

Given a core inlet temperature of 110F the linearized core temperature rise equals 12.4F per foot [(259-110)/12].

The driving force for natural circulation is the density difference between the annulus and the core at the 2/3 core height elevation. The annulus temperature equals 110F; at 2/3 core height the core temperature equals 110 plus 8*12.4 or 209F.

The fluid density at 110F equals (1/0.016165) or 61.86 lb/ft^3. The fluid density at 209F equals (1/0.016705) or 59.86 lb/ft^3.

The density difference equals 61.86-59.86 or 2 lb/ft^3, or 0.1 psid for the 7.5 feet between the jet pump inlet and the core cover. This is fifty times the core cover pressure drop at 7000 gpm; thus it is concluded that the core cover will not adversely affect the natural circulation capability of the reactor.

5.0 Conclusions

The addition of the core cover will not adversely affect the core cooling capability of the shutdown cooling system during the shroud repair evolution.

The core cover will not become dislodged in the event of an inadvertent start of the CSCS pumps.

6.0 References

6.1 FTI Calculation 12-1257392-00, dated 6/10/96

6.2 Drawing 5920-6764, Rev 0

6.3 Drawing 5920-528, Rev 0

6.4 Drawing 5920-3773, Rev 10

6.5 ASME Steam Tables

6.6 Drawing 5920-208, Rev 11

Appendix C

Vermont Yankee Core Shroud Modification

EDM Swarf Evaluation

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efficiency a maximum of 7.25 in^3 of swarf would remain in the reactor vessel. When the EDM electrode breaks through the shroud support plate a small amount of swarf (0.05 lb) will not be captured by the filter; this is included in the calculation of the 95% efficiency. Since this material is not passed through any filters the particle size will be larger than 2 microns. Examination of the debris from the test shows this material to be very fine particles, with a few 1/8 inch in size.

A conservative corrosion rate of stainless steel in the BWR environment is 0.003 inches in 40 years (Reference 6.3).

Considering the stainless steel cladding in the vertical shell section of the reactor vessel there is approximately 340E3 square inches of cladding (Reference 6.4). In one year the cladding alone generates about 25 in^3 of corrosion product, compared to less than 7.5 in^3 from the EDM process. This assessment neglects the reactor internals, the feedwater heaters, the recirculation system piping and the feedwater piping, all of which are additional sources of corrosion product.

5.0 Conclusions

The FTI swarf filtration system is adequate to ensure that any swarf remaining in the reactor vessel will have no adverse affect on the reactor.

6.0 References

6.1 FTI Drawing 1249-006-03

6.2 Drawing 5920-252, Rev 7

6.3 BWRVIP Document "In-vessel Core Spray Piping Repair Design Criteria"

6.4 Drawing 5920-103, Rev 3

6.5 USNRC SER for Hatch Shroud Repair, dated September 25, 1995

6.6 FTI Letter VY-PM-96-086, dated July 24, 1996