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SAFETY EVALUATION OF THE PROPOSED REPAIR

FOR THE VERMONT YANKEE CORE SHROUD

VERMONT YANKEE ATOMIC POWER CORPORATION

DOCKET NO. 50-271

1.0 BACKGROUND

In Boiling Water Reactors (BWRs), the core shroud is a stainless steel cylinder within the reactor pressure vessel (RPV) that provides lateral support to the fuel assembly. The core shroud also serves to partition feedwater in the reactor vessels downcomer annulus region from cooling water flowing through the reactor core. The RPV, core shroud and other RPV internals are designed to accomplish three basic safety functions:

- provide a refloodable coolant volume for the reactor core to assure adequate core cooling in the event of a nuclear process barrier breach
- limit deflections and deformation of internal safety-related RPV components to assure that control rods and Emergency Core Cooling Systems can perform their safety functions during anticipated operational transients and/or design basis accidents
- assure that the safety functions of the core internals are satisfied with respect to safe shutdown of the reactor and proper removal of decay heat

In 1991, cracking of the core shroud was visually observed in a foreign BWR. The crack in this BWR was located in the heat-affected zone of a circumferential weld in the mid-shroud shell. The General Electric Company (GE) reported the cracking found in the foreign reactor in a Rapid Information Communication Services Information Letter (RICSIL) 054. GE identified the cracking mechanism as intergranular stress corrosion cracking (IGSCC).

A number of domestic BWR licensees have recently performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic experience. The combined industry experience from plants which have performed inspections to date indicates that both axial and circumferential cracking can occur in the core shrouds of GE-designed BWRs, and that extensive cracking can occur in circumferential welds located both in the upper and lower portions of BWR core shrouds. The cracking reported in the Brunswick Unit 1 core shroud was particularly significant since it was the first time that extensive 360° shroud cracking had been reported by a licensee in a domestic BWR. The 360° shroud crack at Brunswick Unit 1 was located at weld H3 which joins the top guide support ring to the mid-shroud shell.

Information Notice 93-79 was issued by the NRC on September 30, 1993, in response to the observed cracking at Brunswick Unit 1.

The cracks reported by the Commonwealth Edison Company (ComEd, the licensee for the Dresden, LaSalle, and Quad Cities units) in the Dresden Unit 3 and Quad Cities Unit 1 core shrouds were of major importance, since they signified the first reports of 360° cracking located in lower portions of BWR core shrouds. These 360° cracks are located at shroud weld H5 which joins the core support plate ring to the middle shroud shell in both the Dresden and Quad Cities units. Information Notice 94-42 and its Supplement were issued by the NRC on June 7, and July 19, 1994, to alert other licensees of the shroud cracking discovered at Dresden Unit 3 and Quad Cities Unit 1.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03 (Reference 1) to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in their core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

- inspect the core shrouds no later than the next scheduled refueling outage;
- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;
- develop an inspection plan which addresses inspections of all shroud welds, and which delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject;
- develop plans for evaluation and/or repair of the core shroud; and
- work closely with the Boiling Water Reactors Owners' Group (BWROG) on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking.

Vermont Yankee (VY) inspected its core shroud during the Spring 1995 refueling outage and detected indications of possible IGSCC in several welds that were reported in VY's response to GL 94-03 in May 1995 (Reference 2). Based on its review, (Reference 3) the staff concurred with VY's assessment that there was adequate structural margin to allow operation for at least one additional fuel cycle and indicated that VY should reinspect and/or repair the core shroud prior to start up from the 1996 refueling outage. VY decided to proceed with a modification of the core shroud and in a letter dated April 15, 1996 (Reference 4), VY submitted its plans for modification of the core shroud during the 1996 refueling outage. Based on its review of this submittal, the staff requested additional information to complete its review (Reference 5). The licensee provided the response to the request for additional information on August 7, 1996 (Reference 6). The submittal also included the licensee's 10 CFR 50.59 safety evaluation of the core shroud repair.

2.0 EVALUATION

2.1 Scope of the Modification Design

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

The modification is developed as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code pursuant to 10 CFR 50.55a(a)(3)(i) and is consistent with, and meets, the criteria developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP), "BWR Core Shroud Repair Design Criteria" (Reference 7). The design specifications for the modification are provided in References 8 and 9.

The design-related requirements which were used for the design analysis of the VY shroud modification are contained in Reference 8. All procurement and plant requirements pertinent to the modification are contained in Reference 9.

The modification is designed in accordance with the requirements of Section III, Subsections NB and NG of the 1989 ASME Boiler and Pressure Vessel Code (Reference 10). In addition, the licensee has concluded that stresses in the vertical load paths are less than yield for normal and upset operating conditions. As a result, preload is not projected to be lost inservice. The modification prevents vertical separation of the shroud during normal operating conditions at any postulated failed circumferential weld(s).

The modification 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.

2.2 Shroud Repair Modification Description

The core shroud modification 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 Figures 2 and 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° throughwall.

A spacer ring is provided between the top shroud flange and shroud head. Cutouts 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. At the upper intermediate and bottom radial restraint locations, a slightly larger effective gap of 1/2-inch is provided. As discussed later in this Safety Evaluation (SE), 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, as discussed in Section 2.3 of this SE. 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 would not vertically separate during normal plant operation, even in the event that welds H2 and H3 fail after installation of the repair.

Each cylindrical section of the shroud is prevented from unacceptable lateral motion by radial supports even when the circumferential welds are assumed to 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 and the shroud section between H6 and H7 is restrained by the lower radial support. The 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 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 modification 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 which provide an alternate path for the loads carried by the vertical welds. The welds H1 and H2 are considered as design-reliant welds only for inspection reasons; however, from a design standpoint the modification has been designed as a repair of welds H1 and H2.

2.3 Structural Evaluation

The structural design calculations and related analyses are documented in a proprietary design report (Reference 4) consisting of three volumes. Volume one of the report contains the structural and seismic evaluations of the repair assembly, the shroud and the reactor vessel. It also contains the systems evaluation, materials and fabrication requirements, as well as pre-modification and post-modification inspection considerations. In addition, it contains the shroud repair design specifications and allowable core plate, top guide and shroud deflections in Appendices A and B, respectively. Volumes two and three of the proprietary design report contain calculations and analytical results of other aspects of the design. These are provided in Appendices C through L. They include structural evaluations related to the tie rod assembly (Appendix C); seal ring assembly and spacer ring (Appendix D); core shroud (Appendix E); and reactor vessel (Appendix F). The development of the tie rod shroud loads and deflections are provided in Appendices G and H, respectively. Appendix I contains hydraulic calculations related to the effect of tie rod assemblies or downcomer flow characteristics and leakage through failed welds. Other miscellaneous evaluations such as those related to tie rod stress relaxation, displacement between the core shroud and the reactor vessel, core spray piping stresses and displacements and repair assembly radial expansion are contained in Appendix J. Seismic evaluations and calculations performed by the licensee to determine parameters such as damping values and rotational stiffness in the repair assembly and the shroud under various postulated seismic conditions are contained in Appendix L.

The repair assembly and affected shroud and RPV components are shown to satisfy the Final Safety Analysis Report (FSAR) structural requirements using the FSAR load combinations. The tie rod preload restrains vertical separation of any or all circumferential shroud welds that could develop 360° through-wall cracks and will prevent separation of any or all circumferential welds during normal plant operation. An additional emergency service level loading, a safe shutdown earthquake (SSE) during refueling, was included in the design specifications of the repair, and this loading condition was considered in the structural evaluation.

The limiting loads on the tie rod assemblies and radial restraints vary depending on the assumed shroud cracks. However, since the vertical and lateral load paths are essentially independent, the bounding vertical loads 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. Stresses in the repair hardware vertical load path have been determined to be less than yield during all normal and upset operating conditions, including anticipated thermal transients. Hence, tie rod preload will not be lost inservice. The licensee's tie rod preload strategy, provided in the proprietary design report, addresses all potential crack locations and shroud wall configurations at these locations. The critical weld locations are H2, H3, and H6. The most severe consequences are determined to occur if these welds are postulated to be initially intact but fail subsequently in operation. For this scenario, the licensee's calculations indicate that there is sufficient preload to prevent weld separation during normal operation due to the change in rigidity of the shroud structure. The staff has reviewed these calculations and finds them acceptable.

One of the limiting upset transients is considered to be the design basis cold feedwater transient. During this transient, due to injection of cold feedwater into the shroud annulus, a maximum temperature difference of 130°F between the shroud and the cooler tie rod components could exist. This would cause an increase in the tensile load on the tie rods. Special features are provided in the tie rod design to accommodate this temperature difference. Specifically, a variable spring design was implemented that minimizes rod stiffness during thermal transients and offers increased stiffness during large pressure drops. The staff has reviewed the licensee's calculations in Appendix C of the design report and finds them acceptable. The calculations indicate that stresses in the tie rod components meet ASME Code Section III, Subsection NB, Paragraphs 3222/3223 allowable stresses for this transient.

The effects of radiation and temperature on the tie rod assemblies are also addressed in the design. Specifically, the effects of thermally/radiation-induced relaxation of the rod preload stresses have been considered and taken into account. These effects are calculated to reduce the tie rod preload by less than 5% at end of life. The staff finds this acceptable.

2.3.1 Seismic Evaluation

A number of analyses were performed to calculate the seismic loads on the reactor internals of the VY nuclear plant. The loads from these analyses were used as inputs for designing the repair hardware for the core shroud and determining the adequacy of the vertical and horizontal displacements at postulated severed shroud locations during design basis accidents.

Two-dimensional finite element beam models of the reactor building, pressure vessel, the internals and the core were developed. Three models were utilized for the analysis of the structure in the East-West, North-South and vertical directions. The models are based on existing seismic models of the primary structures of VY prepared for the replacement of the reactor recirculation system piping (Reference 12). The geometry, masses, stiffness coefficients, etc. of the existing models, documented in Reference 13, were retained except for the addition of the modification hardware and the addition of the weld failures in the individual load cases. The seismic models were modified to add the mass and stiffness coefficients for the core-shroud modification. The modified models include non-linear gap elements to 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.

The restraints are modeled with gaps and springs. The upper restraint and the lower-intermediate restraint have radial clearances of about 1/8 inch. In Reference 4, the licensee indicated that these restraints are designed to assure that the alignment of the core is maintained within limits. The upper-intermediate restraint and the lower restraint have radial clearances of about 1/2 inch. The submittal (Reference 4) further indicated that these restraints assure that, in the event of multiple complete failures of circumferential shroud welds, the shell sections shall overlap, preventing a fluid flow path from being opened.

The staff has reviewed the seismic models including the geometry, mass and stiffness coefficients for the core shroud modification and finds them acceptable.

The design basis seismic input for VY is the 1952 Taft earthquake anchored at 0.07g for the operating-basis earthquake (OBE) and 0.14g for the SSE (Reference 12). The input to the seismic analysis is a time history ground motion at the base of the reactor building which is based on a ground spectrum that satisfies Regulatory Guide 1.60 requirements. The staff has previously reviewed and accepted this design approach for VY (Reference 13).

Although the original seismic design basis for VY assumed no vertical amplification of an applied vertical seismic load of 0.10g, the analyses of the shroud repair, however, explicitly evaluates the vertical seismic response of the repaired shroud.

A range of potential single weld and multiple weld failures were 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. Analyses of the structure were also conducted in the horizontal and vertical directions to demonstrate that the installation of the repair hardware has no impact on the response of an intact shroud.

Seismic forces and moments were calculated for each of the seismic load cases described earlier. The response to vertical, North/South and East/West seismic analyses were combined by square root of the sum of the squares for use in the evaluation of the repair hardware and the core shroud.

The analyses show that the resulting loads on the fuel are not substantially changed by the repair for both intact and cracked core shrouds. A comparison of the maximum fuel acceleration to the fuel vendor's proprietary value of maximum allowable acceleration shows the fuel accelerations to be within acceptable values with substantial margin.

Based on its review of the seismic analysis as discussed above, the staff finds the calculation of the seismic forces and moments for the evaluation of the repair hardware acceptable.

2.3.2 Evaluation during design basis accidents

The maximum separation of any circumferential weld with a 360° through wall crack above or below the core support plate during a main steam line break plus an SSE was determined to be 0.5 inches. Vertical separation at failed welds can occur only under the main steam line break condition. This momentary separation is followed by a settling of the shroud as the pressure in the shroud decays over several seconds. The staff has reviewed the results of the pressure decay presented in the GE document GENE-523-A018-0295 (Reference 14) and concurs with the licensee's assessment that the resulting load is not an impact-type load. Therefore, the staff finds the licensee's evaluation of the main steam line break event in the Appendix C of the design report acceptable. The tie rods alone will limit the vertical uplift of the shroud assembly to less than 0.5 inches. The guide tubes will also limit vertical motion of the core support plate to about 0.5 inches, independent of (and without) the tie rod repair. The 0.5 inches displacement will not affect control rod insertion capability as discussed in the GE document GENE-771-44-0894, Revision 2, (Reference 11). This GE document has been previously reviewed and approved by the staff.

The lateral forces for the recirculation line break (RLB) consist of acoustic forces of short duration (several milliseconds) followed by blowdown forces for several seconds. After the first several seconds of RLB blowdown, these blowdown forces decrease significantly when the subcooled flow is completed. For these reasons, the RLB loads can be characterized as a steady load for which the dynamic amplification factor is not significant.

Further, since no separation of the shroud at a cracked weld occurs during an RLB (the tie rod assembly preload exceeds the RLB differential pressure vertical load), the lateral RLB force can be reacted either by the tie rod assembly radial restraints to the reactor vessel or by shear forces across the cracked weld and through the shroud support plate to the vessel. However, as a conservative approach, no credit has been taken for the load being carried by shear through the cracked welds. These resultant stresses in the repair hardware have been determined to be within the design stress allowables. Based on the above discussion, the staff finds that the RLB event has been conservatively evaluated in the shroud repair design report.

The licensee's evaluation of the shroud support plate, including its connection to the reactor vessel and the tie rod attachment for various load combinations, are provided in Appendix F of Reference 4. These load combinations included loss-of-coolant accident (LOCA) (steam or recirculation line break) plus SSE. In addition, Calculation 2499502-601 (Appendix F of Reference 4) provides a fatigue evaluation for the shroud support plate at the slotted hole which will be machined in the plate for the tie rod attachment.

The licensee also performed an evaluation to determine the maximum stress in the reactor vessel wall at the point of contact of the radial supports on the tie rods. The radial loads imposed on the reactor vessel wall due to the radial seismic reactions were treated as primary loads. The resulting membrane and bending stresses were determined by Bijlaard analysis in accordance with Table N-413 of Section III of the ASME Code (1965 edition including Summer 1966 Addenda which is the applicable Code version for the VYNPS reactor vessel). The resulting stresses were combined with other primary stresses in the evaluations to determine the stresses in the vessel due to radial loads. The calculated vessel stresses are less than 50% of the allowable stresses. Shear stresses have a negligible effect on the vessel principal stresses. The staff has reviewed the licensee's analysis in Appendix F of the proprietary design report and determined that the applicable ASME Code stress limits are met. The staff, therefore, finds the design of the lateral seismic supports acceptable.

The maximum lateral displacement of the shroud at both the core support plate and upper guide plate locations of the shroud for combinations of DBE, main steam line and recirculation pipe LOCA, is limited by the bumpers on the tie rods to 0.188 inches. The maximum lateral displacement is within the GE requirements determined on the basis of test data to ensure control rod insertion. GE Report-771-44-0894, Rev. 1, September 2, 1994, contains the justification for allowable displacements of the core plate and the top guide subsequent to a shroud repair. The staff has reviewed this document and finds it acceptable and applicable to VY.

Satisfactory scram with a core plate permanent misalignment of as much as 0.75 inches is indicated according to the GE test data. Limiting shroud permanent displacements to about 0.188 inches is considered satisfactory on the basis that internals of the shroud are structurally adequate for existing design-basis seismic loads and considering that internal loads, including fuel, are not increased due to this repair. The maximum lateral displacement of any potential severed cylindrical section of the shroud with all shroud circumferential welds cracked is also limited to 0.75 inches. This displacement is equal to about 43% of the shroud wall thickness (1.75 inches) and such a displacement will not significantly affect cooling of the core since the cylindrical sections still overlap one another by one inch.

The tie rods were analyzed and tested to ensure that reactor coolant flow would not induce unacceptable vibration. Results of these analyses and tests are presented in Appendix D of the licensee's proprietary design report. The staff has reviewed the licensee's analyses and finds that the stresses resulting from flow-induced vibration are acceptable from a fatigue standpoint.

Based on its review of the stress analysis of repair assembly hardware during design basis accident conditions, as discussed above, the staff finds it acceptable.

2.3.3 Core spray piping evaluation

The core spray piping is anchored to both the reactor vessel and the shroud. If a main steam line break were to occur and 360° through-wall cracking of a circumferential shroud weld were present, upward displacement of the shroud relative to the vessel could result. Analyses were performed to determine the resulting stresses in the core spray piping. The staff has reviewed the licensee's analyses in Appendix L of the design report submittal. The result of the analyses indicate that the maximum stress resulting from the shroud displacement caused by a main steam line break, including seismic loads generated by an SSE and differential thermal expansion, is less than the allowable stress specified in Section III, Subsection NB, Paragraphs 3225/F-1331 of the ASME Code. Accordingly, no pinching or other flow path restrictions are likely to occur.

Special steps were taken in the development and check out of the installation procedures to assure that the tie rod assemblies would not damage the core spray piping in the annulus area during installation. The installation tooling and procedures were checked out in a full-scale mockup. The staff finds the structural analyses of the core spray piping, as discussed above, acceptable.

2.3.4 Loose parts consideration

The various pieces that make up the tie rod/radial restraint system are captured and restrained by appropriate locking devices. The locking device designs have been used successfully for many years in reactor internals. Such locking devices and the stresses in the pieces which make up the tie rod/radial restraint system are within allowable limits for normal plant operation. Further, the design includes suitable features to prevent detachment of the tie rods even if preload were lost. However, in the unlikely event that a tie rod becomes detached from its attachment point during normal plant operation, there are no significant safety consequences to the shroud or to the other tie rods. Shroud crack leakage may increase slightly if one tie rod should fail and a through-wall crack existed, but such leakage would be too small to be detectable. If individual components should somehow break off the tie rod assembly, they will fall into the support plate or, if small enough, could be transported into the recirculation loop and its pump. The consequences of a loose tie rod component are similar to other loose parts from the reactor internals within the recirculation system. The staff finds the locking devices and the consequences of any potential failure of the locking devices acceptable.

2.3.5 Conclusion

In summary, based on the forgoing discussion concerning the licensee's evaluation of the various aspects associated with the core shroud repair, the staff concludes that the modification is acceptable from a structural standpoint.

2.4 SYSTEMS EVALUATION

2.4.1 Introduction

The intent of the licensee's core shroud modification design documents, provided in Reference 4, was to demonstrate that fuel geometry and core cooling would be maintained given the unlikely occurrence of a through-wall failure of any horizontal weld during normal operations and design basis events with the core shroud repair installed. Fuel geometry must be maintained to ensure control rod insertion while core cooling is ensured by proper core standby cooling system (CSCS) performance. The VYNPC submittals provided analyses of the principal effects and issues of operating the plant with postulated circumferential core shroud welds cracked and tie rod assemblies installed. Some of the conditions analyzed by VY included tie rod assembly induced leakage, core shroud weld crack leakage, downcomer flow characteristics, and lateral displacement and vertical separation of the core shroud. The staff has reviewed these portions of the VY submittals and provided an evaluation of VY's findings in the following discussion.

2.4.2 Tie Rod Assembly System Induced Leakage

As discussed above, the installation of the tie rod assemblies requires the machining of four oblong through thickness slots in the core shroud support plate using the Electrical Discharge Machining (EDM) process. VY estimated that a small amount of core flow leakage may occur across the seal rings at the four locations at which the repair assemblies are attached to the shroud support. At these locations, a 0.001-inch gap between the seal ring and the shroud support plate is assumed. The total calculated leakage from the installation of the tie rod assemblies is estimated to be 10.3 gpm for normal operating conditions with a 15% increase in shroud differential pressures (Reference 4). The 15% increase in shroud differential pressures used in the licensee's analyses provides margin for future power uprate. The staff confirmed that this leakage is equivalent to 0.008% of core flow at 105% rated power and 100% rated core flow (Reference 12). The staff does not consider this leakage rate to be significant with regards to total core flow and, therefore, it is acceptable.

At VY, the CSCS consists of the high pressure coolant injection (HPCI) system, the automatic depressurization system (ADS), the core spray (CS) system, and the low pressure coolant injection (LPCI) system. The staff notes that the leakage from the core shroud support plate to the downcomer annulus does not affect the performance of the above systems. Therefore, the CSCS performance is not affected by the physical installation of the tie rod stabilizer assembly system.

2.4.3 Core Shroud Weld Crack Leakage

The tie rod assemblies are installed with a cold preload to ensure that no vertical separation of any or all cracked horizontal welds will occur during normal operations. Vertical separation, if sufficiently large, could compromise fuel geometry and control rod insertion. For Vermont Yankee, a maximum vertical displacement of 11.6 inches is required for the top guide to clear the top of the fuel channels (Reference 14). The staff notes that, with the repair, the estimated vertical separation during normal operations will not affect the fuel geometry, and therefore, control rod insertion is not precluded.

However, a small leakage path could exist due to existing through-wall core shroud weld cracks. VY conservatively modeled the crack to provide a 0.001-inch leakage path per weld, H1 through H8 (Reference 4). VY estimated that the total leakage from all welds, H1 through H8, having postulated 360° through-wall cracks was approximately 90.4 gpm for normal operating conditions with a 15% increase in shroud differential pressures (Reference 4). The staff confirmed that this leakage is equivalent to 0.07% of core flow at 105% rated power and 100% rated core flow (Reference 12). Although core shroud crack leakage is unlikely due to the preload on the tie rod, VYNPC concluded that there are no consequences associated with the repair installed based on these small leakages during normal operations. The staff acknowledges that the total leakage is insignificant and will not affect the performance of the CSCS.

2.4.4 Downcomer Flow Characteristics

VY analyzed the available flow area in the downcomer with the four tie rod assemblies installed. The staff acknowledges that the size of the tie rod assemblies are small compared to the size of the jet pump assemblies and thus, the tie rod stabilizer assemblies are not expected to significantly affect the flow characteristics in the downcomer. However, since the downcomer annulus is smaller at the top of the core shroud with other existing obstructions such as the core spray lines, VY evaluated the flow blockage area of the upper core shroud restraint of the tie rod assembly located between welds H1 and H2. VY's analysis demonstrated that the installation of the tie rod assemblies will decrease the as-built available downcomer flow area by less than 6 percent (Reference 12). The staff reviewed the downcomer flow calculation which accounted for the core spray piping, miscellaneous bolts and lugs, and the upper radial restraint and bracket assembly of the tie rod assemblies.

Additionally, VY provided the corresponding pressure drop to the decrease in downcomer flow area. VY estimated that the pressure drop due to the installation of tie-rod assemblies is about 0.005 psi (Reference 4). Based on this information and information from other reviews of similar core shroud repairs, the staff concluded that the increase in the pressure drop is insignificant. Therefore, the staff agrees with VYNPC that the installation of the tie rod stabilizer assemblies should not affect the recirculation flow of the reactor.

2.4.5 Potential Lateral Displacement of the Core Shroud

VY evaluated the maximum lateral displacement of the core shroud at the core plate and top guide under normal operations and load combinations such as SSE, main steamline break (MSLB), and RLB, assuming 360° through-wall cracks at any weld location. Lateral displacement of the core shroud could damage core spray lines and could produce an opening in the core shroud, inducing core shroud bypass leakage and complicating recovery. Maximum permanent displacements of the core shroud sections between H3 and H5 and between H6 and H7 are limited to less than 0.75 inches by radial restraints. This lateral displacement is less than the 1.75 inch thickness of the shroud, and accordingly, the separated portions of the shroud would remain overlapped during worst case conditions. Additionally, a permanent lateral displacement of the top guide or core support plate is limited to less than 0.188 inches for all loadings by radial restraints at these locations (Reference 4). Displacements of this magnitude will not significantly increase the scram time as demonstrated in Reference 11. Therefore, the staff has concluded that the maximum lateral displacement of the core shroud would not result in significant leakage from the core to the downcomer region following an accident scenario and the ability to reflood the core to 2/3 core height would not be precluded.

2.4.6 Potential Vertical Separation of the Core Shroud

VY evaluated the maximum vertical displacement of the core shroud assuming 360° through-wall cracks at any weld above or below the core plate during a OBE, SSE, MSLB, and an SSE plus MSLB. These postulated events would result in a large upward load on the core shroud which could impact the ability of the control rods to insert and the ability of the core spray system to perform its safety function. As stated above, a maximum vertical separation of 11.6 inches is required for the top guide to clear the top of the fuel channels. With the repair installed, the maximum vertical separation during an SSE or SSE plus MSLB is similar in magnitude to other core shroud repairs reviewed and approved by the staff (Reference 6). This separation is limited by the tie rod stabilizer assemblies and should not impact the core spray system. The licensee stated that a temporary separation due to the tipping of the core shroud could occur during an OBE with the increase of pressure differentials by 15 percent. However, temporary separation during an OBE is not expected with the current differential pressures and operating conditions. Therefore, based on this assessment, the staff concluded that postulated separation during an OBE, SSE or an SSE plus MSLB event would not preclude any of the systems from performing their safety functions.

2.4.7 Conclusion

The staff has evaluated VY's safety evaluation of the consequences of the proposed core shroud repair. The staff has found that the proposed repair should not impact the ability to insert control rods, the performance of the CSCS, particularly the core spray system, or the ability to reflood and cool the core. The staff concluded that the proposed repair does not pose adverse consequences to plant safety, and therefore, plant operation is acceptable with the proposed core shroud repair installed.

2.5 MATERIALS, FABRICATION AND INSPECTION CONSIDERATIONS

2.5.1 Materials and Fabrication

By letter dated April 15, 1996, as supplemented by letter dated August 7, 1996, VY stated that Type 304 or 304L austenitic stainless steel, solution annealed Type (F)XM-19 stainless steel, and nickel-based (Ni-Cr-Fe) alloy

X-750 materials were selected for the fabrication of the core shroud repair assemblies. Specifically, Type 304 or 304L austenitic stainless steel will be used for the top bracket and the radial restraints, the bottom adapter will be made of (F)XM-19 stainless steel, and alloy X-750 will be used for the spring rod assembly and top adapter. These materials have been used for a number of other components in the BWR environment and have demonstrated good resistance to stress corrosion cracking by laboratory testing and long-term service experience. No welding or thermal cutting is permitted in the fabrication, assembly or installation of the core shroud repair assemblies for the purpose of minimizing its susceptibility to IGSCC.

All material was procured to either American Society for Testing and Materials (ASTM) or American Society of Mechanical Engineers (ASME) specifications, except as delineated below. Material properties and allowable stresses for repair components are as specified in the ASME Boiler and Pressure Vessel (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. Non-destructive examination (NDE) of material used for load bearing members is performed in accordance with ASME Code Section III, Subsection NG-2000.

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 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.

VY will avoid any abusive machining and grinding practices and will control machining and grinding process parameters and operations in critical load bearing threaded areas, based on qualification samples, which have been subjected to macroscopic and metallographic examinations and microhardness testing. The licensee's evaluations will include hardness magnitudes and depths, depth of severe metal distortion, depth of visible evidence of slip planes and depth of cold work. Further, VY will perform solution annealing heat treatments 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.

The staff has reviewed VY's submittal regarding the proposed core shroud repair and concludes that the selected materials and fabrication methods for the tie rod repair assemblies are acceptable.

2.5.2 Pre-modification and Post-modification Inspection

Prior to installation of the shroud repair, VY will perform ultrasonic inspections of design reliant welds -- circumferential welds which provide an alternate path for the loads carried by the vertical 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. 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. VY is considering H1 and H2 as design-reliant welds only for inspection reasons; the repair is designed as a repair to H1 and H2.

VY did not provide a detailed pre-modification inspection plan in their submittal to support the proposed core shroud repair installation. The specific scope of the pre-modification design-reliant weld inspections is instead detailed in the 1996 Outage Core Shroud Inspection Scan Plan, which is available for review at VY.

Prior to reactor pressure vessel reassembly, VY will perform visual inspections by TV to verify the proper installation of the 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 dot in the shroud support plate. To verify that the bottom "T" adapter is correctly oriented, proper engagement of the pin is checked with the lowest outer sleeve.

VY has not yet finalized its inspection plan for future inspections of the core shroud and the tie rod assembly components. The staff recommends that VY's inspection plan should consider the following: (1) the plant-specific repair design requirements; (2) the extent and the results of the baseline inspection performed during pre-modification inspection; (3) the threaded areas and the locations of crevices and stress concentration in the tie rod stabilizer assemblies; and (4) BWRVIP reinspection guidelines when they are established. The NRC staff will review VY's inspection plan when submitted. VY is requested to submit its plans for the next inspection of the core shroud and the tie rod assembly components at least 6 months prior to the inspection. This schedule will allow sufficient time for the NRC staff to evaluate the proposed plan and resolve any comments it may have. Since the core shroud and the tie rod stabilizer assemblies are generally classified as ASME Code Class B-N-2 components (core structural support), the inspection plan will be required to be incorporated into the plant in-service inspection plan.

3.0 CONCLUSION

The proposed core shroud repair has been designed as an alternative to the requirements of ASME Boiler and Pressure Vessel Code pursuant to Title 10, Code of Federal Regulations, Part 50.55a(a)(3)(i). Based on a review of the shroud modification hardware from structural, systems, materials, and fabrication considerations, as discussed above, the staff concludes that the proposed modifications of the VY shroud are acceptable, and subject to the submittal of the post-modification inspection program, will not result in any increased risk to the public health and safety.

Attachments: Figures 1, 2, and 3

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Date: October 2, 1996

REFERENCES

(1) Letter, NRC to all Licensees, Generic Letter 94-03, NVY, dated July 25, 1994.

(2) Letter, J. Pelletier, Vermont Yankee Nuclear Power Corporation (VYNPC) to NRC, BVY 95-55, dated May 24, 1995.

(3) Letter, NRC to VYNPC NVY 95-55, dated April 27, 1995.

(4) Letter, VYNPC to USNRC, BVY 96-48 dated April 15, 1996, including the attached Vermont Yankee Core Shroud Repair Design Report MPR-1730.

(5) Letter, USNRC to VYNPC, NVY 96-120 dated July 9, 1996.

(6) Letter, VYNPC to USNRC, BVY 96-96 dated August 7, 1996.

(7) BWRVIP Core Shroud Repair Criteria, Revision 1, September 12, 1994.

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

(9) 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.

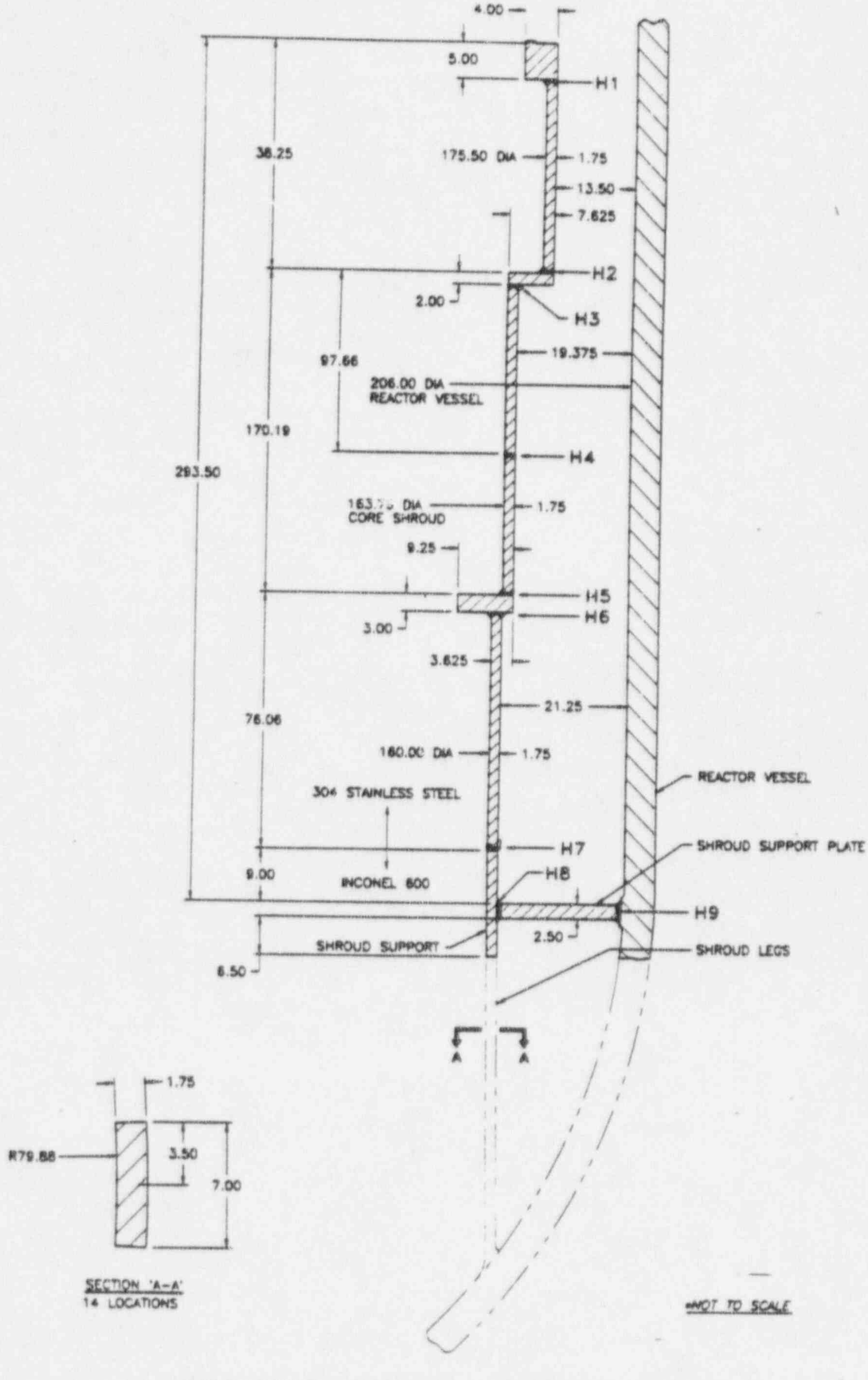
(10) ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, "Core Support Structures," Subsection NB, , "Class 1 Structures," 1989.

(11) GE Report GENE-77-44-0894 Revision 2 - "Justification for allowable Displacements of the Core Plate and Top Guide Shroud Repair."

(12) Final Safety Analysis Report, VYNPS, VYNPC, Revision 13.

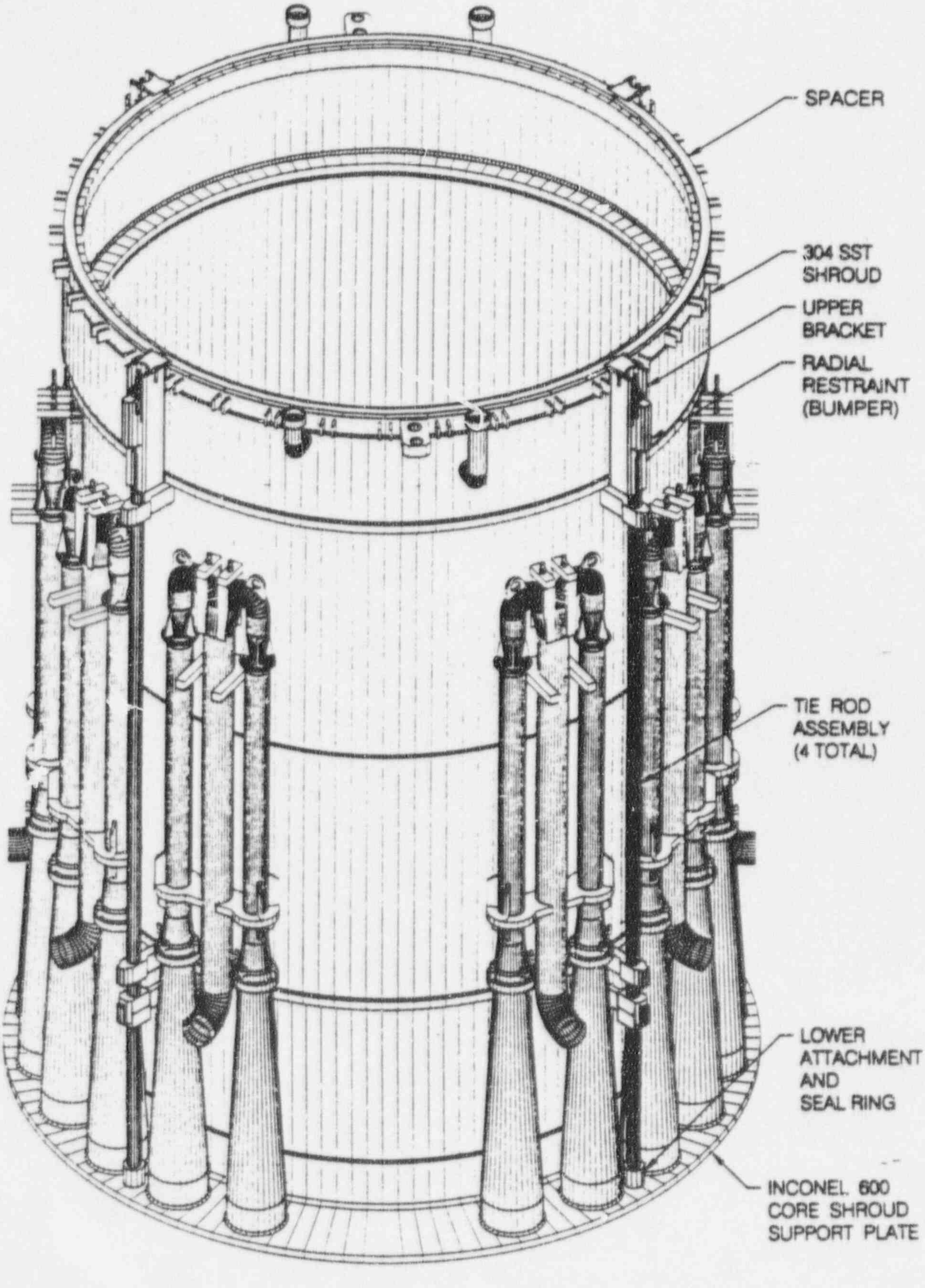
(13) GE-NE-23A4591, "Vermont Yankee Nuclear Power Station, Primary Structure Seismic Analysis, Vermont Yankee Nuclear Power Corporation," Revision 1, General Electric Company.

(14) GE Report GENE-523-A018-295, Revision 0, "Duane Arnold and Vermont Yankee Shroud Safety Assessment," April 18, 1995.



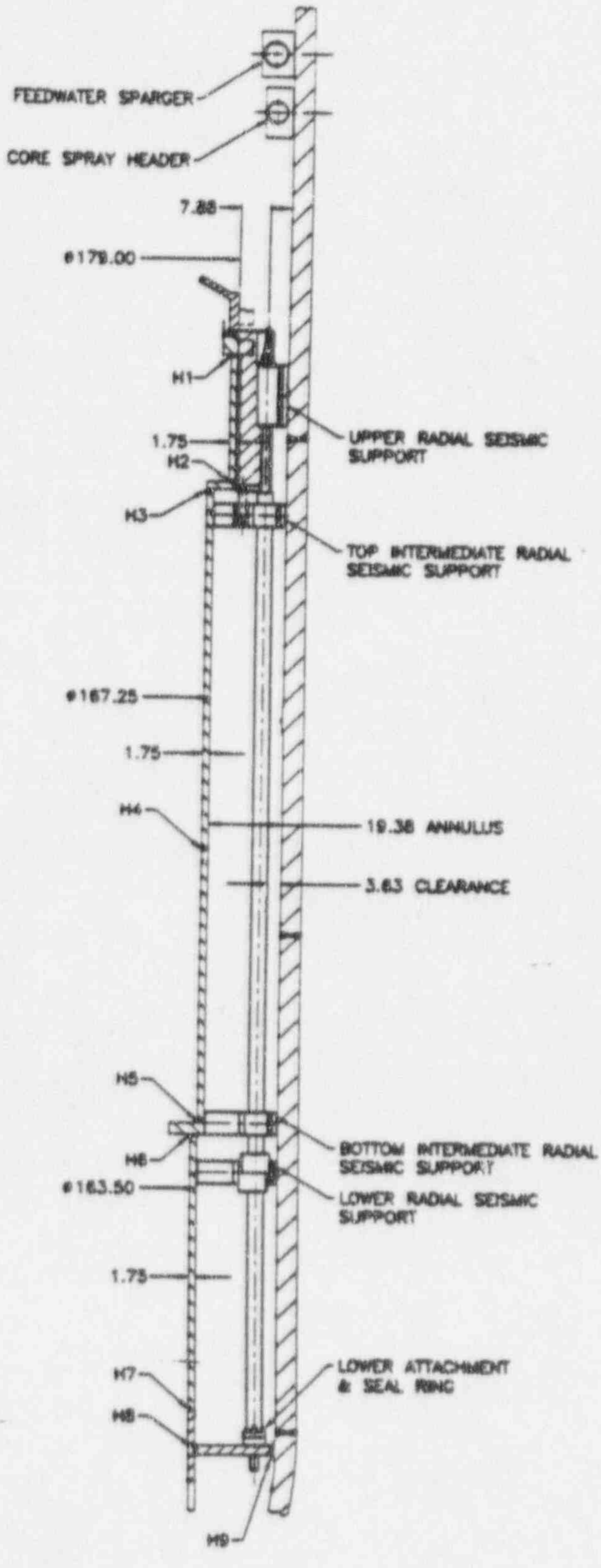
VERMONT YANKEE - SHROUD WELDS
FIGURE 1

MPR
F-002-194-01 (D1)
03/25/96 (GJB)



MPR
F-VERM-3D-B
3/27/96 (TAPE)

VERMONT YANKEE SHROUD REPAIR
FIGURE 2



VERMONT YANKEE - SHROUD REPAIR
FIGURE 3

MPR
F-246-004-13 (236)
03/26/96 (GJB)