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AmerGen Energy Co, LLC
Clinton Power Station BWRVIP-04A
Core Shroud Repair Design Submittal to the Nuclear Regulatory Commission (NRC)

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ABSTRACT

AmerGen at their Clinton Power Station (CPS) is planning to implement a permanent repair of all horizontal circumferential shroud welds that are cracked or subject to cracking. This document provides the information to the NRC for their review and approval of the CPS Core Shroud Repair. This document is prepared in the format, and contains the information as recommended by BWRVIP-04A report "BWR Vessel and Internals Project, Guide for Format and Content of Core Shroud Repair Design Submittals", report number EPRI-TR-1006600.

Table Of Contents

1.0 INTRODUCTION AND SUMMARY

The shroud repair submittal is required for the repair [an alternative replacement per American Society of Mechanical Engineers Boiler and Pressure Vessel Code ASME Code (Reference 2.e) Section XI; even though it structurally replaces all of the shroud horizontal welds] since according to the original Code of Construction the shroud is classified as a Core Support Structure, i.e., a subsection NG component.

AmerGen needs to make this submittal, and it must be in the format suggested by BWRVIP-04A. The purpose of this document is to provide a general roadmap for the contents of the shroud repair submittals to the United States Nuclear Regulatory Commission (USNRC) staff pursuant to 10 CFR 50.55a (a)(3) (i) (Reference 1). This document was prepared in the BWRVIP-04A format suggested in EPRI document EPRI-TR-1006600 (Reference 3) of April 2002; *Guide for Format and Content of Core Shroud Repair Design Submittals (BWRVIP-04A).* This GE shroud repair design submittal document was prepared in the BWRVIP-04A format, and contains information suggested by that EPRI document. This report summarizes the design of the core shroud repair for Clinton.

The submittal to the USNRC for the Clinton Core Shroud Repair project consists of a number of individual documents listed in Section 9 and included as attachments to this document. The following paragraphs provide an overview of major documents.

1) <u>Specifications (References 14, 15, 20, and 21)</u> establish the design criteria (total design loads, load combinations, acceptance criteria etc.) as well as material procurement, and fabrication requirements for the Clinton shroud repair project.
2) <u>Seismic & Dynamic, and Stress Analysis Reports (References 16, 17, 18, 19, and 22).</u> The Seismic & Dynamic analysis report provides, in adequate detail, the Seismic & Dynamic reanalysis results summary of the Clinton reactor building including RPV and internals covering various crack weld configurations with repair hardware installed. This report also provides the Seismic & Dynamic loads on repair hardware as well as existing RPV internals and RPV components. The certified hardware stress report, the certified RPV stress report, and certified Shroud support system stress report summarize the results of the analyses/evaluations performed to show the conformance of the design hardware, the RPV, and shroud support system to the design criteria set forth in the design specifications.
3) <u>Input to 10CFR50.59 Evaluation (Reference 30)</u> documents the GE input to an evaluation per 10 CFR 50.59 for the installation of the stabilizers on the Clinton Core Shroud by CPS.

In summary the installation of the proposed repair satisfies the applicable regulatory requirements and guidance and is consistent with the current plant licensing basis. The repair conforms to the requirements of the "Core Shroud Repair Criteria (BWRVIP-02)" without any alternate approaches or exceptions.

2.0 BACKGROUND

This section summarizes the safety functions of the shroud and identifies generic USNRC and industry actions taken in response to the occurrence of shroud weld cracking.

2.1 Shroud Operation and Safety Functions

Shroud Operation

The typical arrangement of the Boiling Water Reactor vessel internals is shown in Figure 2-1. The core structure surrounds the active core of the reactor and consists of the core shroud, shroud head (not shown) and steam separator assembly, core support plate, and top guide. This structure is used to form partitions within the reactor vessel, sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies, control rod guide tubes, and steam separators. Figure 2-2 shows the typical reactor internal flow paths.

The core shroud is a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a postulated recirculation line break. The floodable inner volume as shown in Figure 2-2 is the volume inside the core shroud up to the level of the jet pump suction inlet.

The boundary of the inner volume consists of the following:

- The jet pumps suction inlet down to the shroud support ring.
- The shroud support ring that forms a barrier between the outside of the shroud and the downcomer annulus inside of the reactor vessel.
- The reactor vessel wall below the shroud support ring.
- The core shroud up to the level of the jet pump suction inlet.

The volume enclosed by the core shroud is composed of three regions, each with a different shroud diameter. The upper shroud has the largest diameter and, surrounds the core discharge plenum that is bounded by the shroud head on top and the top guide grid below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has an intermediate diameter and is bounded at the bottom by the core plate and at the top by the top guide/grid. The lower shroud, surrounding part of the lower plenum, has the smallest diameter and at the bottom is welded to the reactor vessel shroud support cylinder (this is one of the horizontal welds which is assumed cracked and is structurally replaced by the repair hardware). The various cylindrical sections made of 304L stainless steel that comprise the Clinton core shroud arrangement including the top guide/grid, are shown in Figure 2-3. Figure 2-3 also shows location and designation of welds that are structurally replaced by the repair.

Some basic factors that are considered in the repair hardware are:

- Diameter and thickness of the shroud.
- Number of horizontal welds and heights of cylindrical sections.
- Number of vertical welds connecting the cylindrical shells.
- Shroud support plate type.
- Configuration and location of ECCS piping, brackets, etc.
- Jet pump restrainer bracket clearances.
- Differential pressures and Seismic & Dynamic loadings.

Safety Functions

The reactor internals, of which the core shroud is a part, have three basic safety functions:

1. To provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.
2. To limit deflections and deformation to assure that the control rods and the ECCS can perform their safety functions during anticipated operational occurrences and accidents.
3. To assure that the safety design functions (1) and (2), above are satisfied, so that the safe shutdown of the plant and removal of decay heat are not impaired.

Additionally, the reactor internals are designed to meet power generation objectives to:

1. Maintain partitions between regions within the reactor vessel to provide correct distribution, for all normal plant operating modes.
2. Provide positioning and support for the fuel assemblies, control rods, in-core flux monitors, and other vessel internals and to ensure that normal control rod movement is not impaired.

2.2 NRC and Industry Actions

1. In 1990, crack indications were reported at core shroud welds located in the beltline region of an overseas reactor (BWR/4). This reactor had completed approximately 190 months of power operation before the cracks were discovered. As a result of this discovery, General Electric (GE) issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications", on October 3, 1990, (Reference 5) to all owners of GE BWRs. This RICSIL summarized cracking found in the overseas reactor and recommended that, at the next refueling outage, plants with 304L stainless steel shroud should perform a visual examination of the accessible areas of the seam welds and associated heat affected zone on the inside and outside surfaces of the shroud.

   During the 1993 refueling outage of Brunswick Unit 1 (BWR/4), in-vessel visual inspection (IVVI) revealed cracks at weld regions of the core shroud. Brunswick found both circumferential and axial cracks in the shroud, although cracking was predominantly circumferential. Circumferential cracks were located on the shroud inside surface in the heat-affected zone (HAZ) of weld H-3 and extended 360^0 around the circumference on the shroud. Weld H-3 is a horizontal weld that attaches the bottom of the Top Guide Support Ring (TGSR) to the top of the shroud cylinder below the ring. The H-2 weld that joins the upper shroud cylinder to the top of the other side of the TGSR was also cracked extensively although the cracking was shallower than that at weld H3. The first axial crack discovered was located on the outer shroud surface at weld H-4 (lower shroud cylinder). Brunswick performed additional visual testing (VT) and ultrasonic testing (UT) of the shroud and removed boat samples at welds H-2, H-3, and H-4 to evaluate the length and size of the cracks, and to validate ultrasonic inspection sizing test procedures.

2. General Electric issued Revision 1 to RICSIL 054 on July 22, 1993, to update the information on the core support shroud cracks and to provide revised interim recommendations to perform visual examinations of accessible areas of the shroud at all GE BWRs during the next scheduled outage. Since the issuance of RICSIL 054, revision 1, cracking has been identified in several domestic BWR core shrouds. On October 4, 1993, General Electric issued SIL 572, Revision 1 (Reference 6). This document superseded RICSIL 054, R1 and provided updated guidance on inspection and cracking susceptibility.

3. This led the NRC to issue information notice IN 93-79 (Reference 7) in response to the Brunswick cracking information.

4. Several GE BWR owners inspected their core shrouds during Spring 1994 planned outages and observed extensive cracking at the circumferential welds. These inspection findings caused the NRC staff and industry to re-evaluate the significance of this issue. Due to the 360 degrees extent of the cracking, and the location at a lower elevation where extensive cracking had not been previously observed, the inspection and analyses performed for Dresden Unit 3 and Quad Cities Unit 1 were especially noteworthy. Therefore, NRC issued another information notice IN 94-42 (Reference 8) and a Supplement 1 to it on July 19, 1994 outlining the information found at Dresden Unit 3 and Quad Cities Unit 1.

5. NRC issued generic letter GL 94-03 (Reference 9) on July 25, 1994 requesting all BWR operating licenses except for Big Rock Point (which does not have a core shroud), to

- 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 core shroud welds, and which delineates the examination methods to be used for the inspections of the core 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 with the BWR Owners Group (BWROG) on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to Intergranular Stress Corrosion Cracking (IGSCC).

On June 10, 1994, the BWR Vessel Internals Project (VIP) was established to focus industry resources and senior management attention on the resolution of vessel internals cracking issues, with shroud cracking identified as the highest priority. The general repair design criteria, which is the product of the Repair Technical Subcommittee of the VIP, is detailed in Reference 4 and is applied to Clinton core shroud repair hardware design.

Since the BWRVIP was established, several BWR owners have presented submittals to the USNRC to obtain NRC approval of proposed core shroud repair hardware installations. The NRC has noted that differences in the format and contents of these submittals have increased the time and efforts required for review. The need for this document was established during the June 22, 1995, meeting between the BWRVIP and NRC. This document is a product of the repair Technical Committee of the BWRVIP.

2.3 Utility responses to GL 94-03

USNRC generic Letter 94-03 requested BWR licenses 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 are conducted.

On December 29, 2001, AMERGEN submitted the actions requested by GL 94-03 to the USNRC and followed it up with results of the actions committed in December 29, 2001 letter to USNRC via U-603558 on June 6, 2002 (Reference 28). The response of June 6, 2002 addressed the results of the Clinton core shroud inspections per attachment 1 of U-603558 (Reference 28). Inspections performed during the 2002 outage and the subsequent evaluations and actions taken such as introducing noble metal chemistry at the plant have justified continued operation through cycle 10 (Up to 2006 refueling outage). Clinton has committed to repair the core shroud with the modifications that are submitted with this submittal to NRC.

Finally it should be noted that AmerGen through its corporate entity Exelon, is actively participating in the BWRVIP activities to resolve this issue.

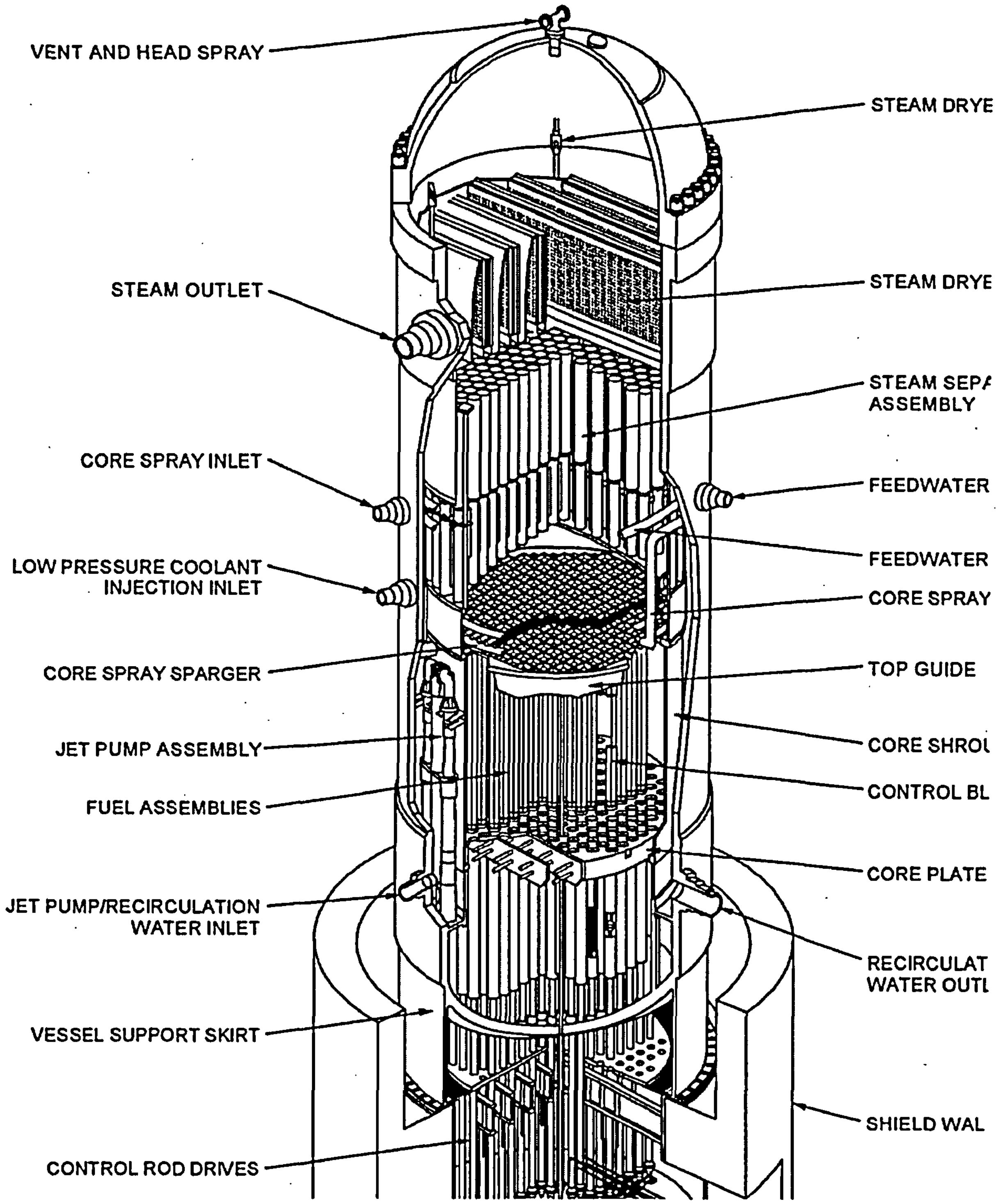

FIGURE 2-1 TYPICAL ARRANGEMENT OF BOILING WATER REACTOR VESSEL INTERNALS

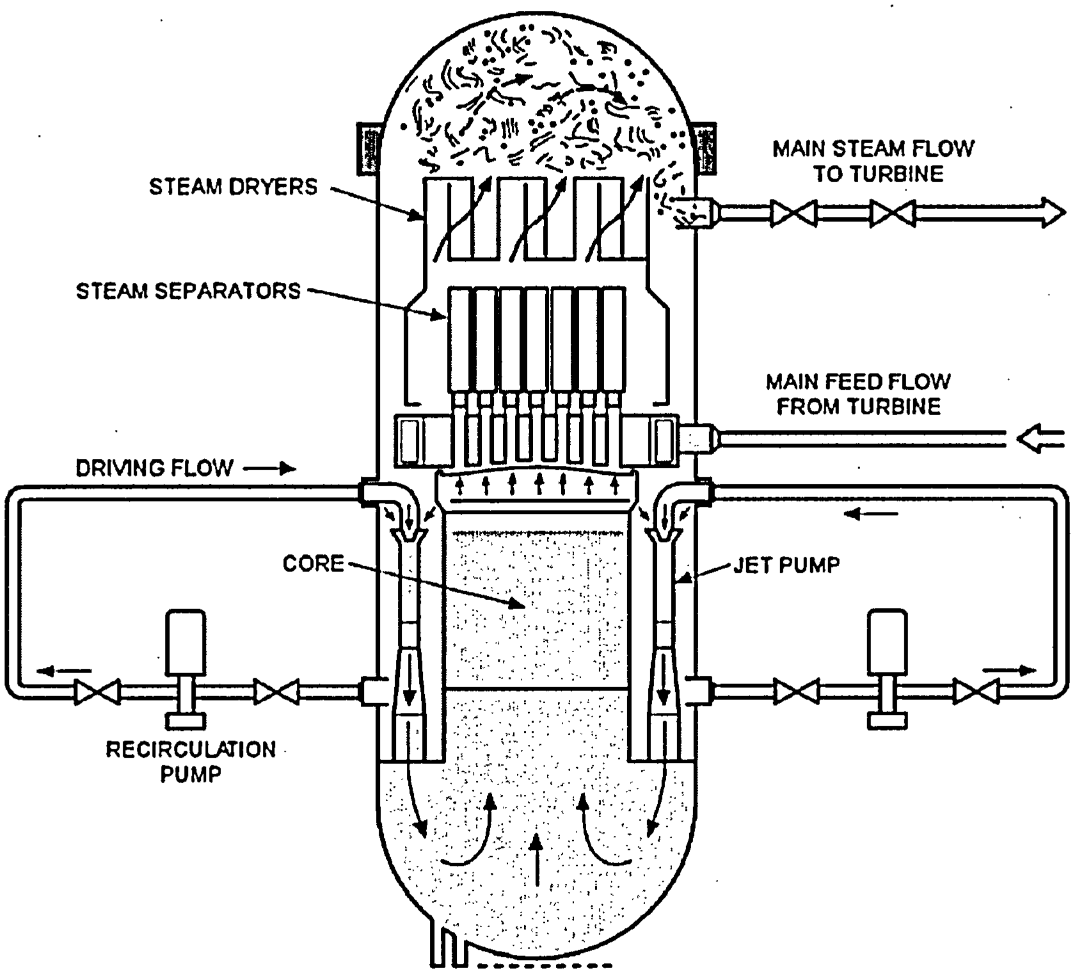

Figure 2-2 Boiling Water Reactor Vessel Internal Flow paths

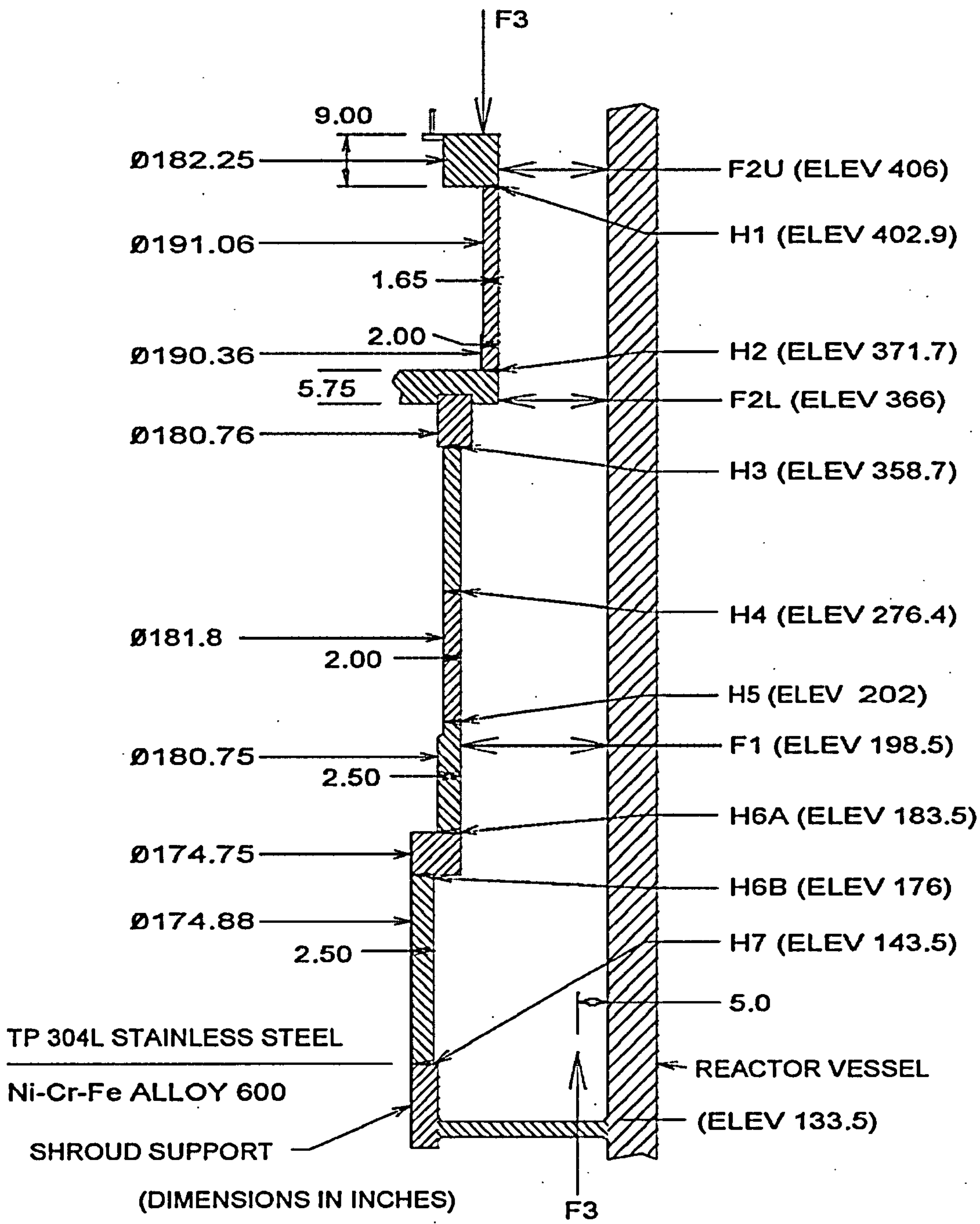

Figure 2-3: Shroud Horizontal Weld Locations

3.0 DESCRIPTION OF THE REPAIR

This section of the document covers the design objectives of the repair hardware, the design criteria applied for the repair hardware, general description of the repair hardware, and finally the design features of the repair hardware.

3.1 Design Objectives

Radially acting stabilizers are mounted on four vertical preloaded tie rods to maintain the alignment of the core shroud to the reactor pressure vessel (RPV), and the originally designed reactor flow partitions. The set of stabilizers replaces the structural functions of the shroud welds, which are postulated to contain cracks. [[

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3.2 Design Criteria

A total of 8 welds, designated H1 through H7, of the Clinton core shroud will be structurally replaced by a set of four stabilizer assemblies. This repair modification is an alternate to the requirements of the ASME Code pursuant to 10CFR50.55a(a)(3)(i). The design criteria applied to the repair meets the criteria developed by the BWRVIP and contained in the "BWR Core Shroud Repair Design Criteria" (Reference 4). The list of plant events for which the repair is designed is shown in section 3.3 of BWRVIP-02 design criteria document (Reference 4).

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3.3 Description of Repair Components and Design Features

Description of Repair Components

A description of the shroud repair design shown in Figures 3-1 through 3-3 is as follows.

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FIGURE 3-1 CLINTON REPAIR HARDWARE ASSEMBLY

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FIGURE 3-2 CLINTON REPAIR HARDWARE UPPER ASSEMBLY

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FIGURE 3-3 CLINTON REPAIR HARDWARE LOWER ASSEMBLY

4.0 STRUCTURAL AND DESIGN EVALUATION

This section of the submittal outlines the structural and design evaluations performed to demonstrate the adequacy of the proposed shroud repair for Clinton. Key information included in this section is: (1) the analysis models and methodology, including the load cases and load combinations, (2) the Seismic & Dynamic and structural analysis results for the repair assemblies, including compliance with the BWRVIP "Core Shroud Repair Design Criteria", (Reference 4), (3) the evaluation of the effects of the repair on the reactor pressure vessel attached piping and reactor internals, (4) evaluation of the effect of the repair on the shroud shell and shroud support structure, (5) the assessment of the potential for flow induced vibration, (6) the evaluation of the radiation effects on the repair components, (7) the evaluation of the potential for loose parts; (8) safety consequences of a postulated loss of preload, and (9) the standards for installation cleanliness.

4.1 Analysis Models and Methodology

This subsection covers the Seismic & Dynamic and structural analysis models, analysis methods used, and the summary of the results of the Seismic & Dynamic and stress analysis of the repair hardware including the compliance with the BWRVIP "Core Shroud Repair Design Criteria", (Reference 4). The models, methods, and the results are summarized wherever possible otherwise for the details, references are listed in adequate form to locate the details.

4.1.1 Description of Structural Models and Analysis

4.1.1.1 Description of Seismic & Dynamic Model

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In order to insure that the installation of the stabilizer design does not adversely affect the existing dynamic qualification of the RPV and internals, assuming no defective welds are present, analyses for the uncracked case were performed with and without the shroud repair in place. It was concluded that seismic & dynamic loads in the RPV and internal structures are decreased, or at least not significantly increased, by the shroud stabilizer installation. It was also shown that loads in the RPV and internals are generally reduced by the inclusion of the most limiting combination of assumed cracks. [[

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The piping input motion response spectra taken at the RPV attachment points are essentially the same for the un cracked benchmark model without the shroud repair hardware and the bounding shroud crack model with the shroud repair hardware. Therefore, no impact on piping occurs as a result of the repair hardware installation.

See Seismic & Dynamic analysis report GENE-0000-0023-6259-01 (Reference 17) for further details of the Seismic & Dynamic modeling and analysis method.

4.1.1.2 Description of Structural Models

This section of the submittal documents the stress analysis performed on the shroud repair hardware components, and the affected, governing existing shroud sections and shroud hardware. Also, the determination of the thermal preload and other operating conditions loads for the tie rod repair hardware system is documented in this report. Material properties, loading analyzed for, the allowable stress limits used for Code compliance are shown in detail in References 16 and 19. The stress analysis methods used are summarized below:

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Applied Loads

Applied loads are shown in the design specification (26A6213) and the Seismic & Dynamic analysis report (Reference 17).

4.1.2 Linear Vs. Non-Linear Analysis Method

There are no non-linear features such as structural gaps, limit stops etc. in the Clinton Seismic & Dynamic model, Seismic & Dynamic analyses or any hardware stress analysis.

4.1.3 Weld Crack Model

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4.1.4 Load Cases and Load Combinations

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4.1.4.1 Normal Operation

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4.1.4.2 Limiting Upset Conditions

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4.1.4.3 Emergency Condition

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4.1.4.4 Faulted Conditions

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4.1.4.5 Bounding Cases of Cracked Welds

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4.1.4.6 Load Combinations

The following governing load combinations and their classification (per Clinton USAR and the BWRVIP-02 load combinations) were considered for the stabilizer design:

Normal:	Weight, normal operating pressure differences and temperatures
Upset 1:	Weight, upset operating pressure differences and normal operating temperature plus OBE plus SRV
Upset 2:	Weight, Upset pressure differences, plus maximum transient temperature
Emergency:	Weight plus upset operating pressure differences plus SRV plus LOCA
Faulted 1:	Weight plus SSE plus SRV plus LOCA plus main steam line LOCA pressure differences
Faulted 2:	Weight plus SSE plus AP plus main steam line LOCA pressure differences
Faulted 3	Weight plus SSE plus Recirculation Line LOCA loads plus upset dynamic operating pressure differences.

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4.1.4.7 Shroud Deflections

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4.2 Reactor Pressure Vessel and Reactor Internals

The acceptability of the structural integrity requirements of the Code Design Specification 26A6214 (Reference 15) are documented in RPV Stress Report 26A6216 (Reference 18). The Clinton original pressure vessel stress report (documents referenced in Reference 18), are used as reference for the existing certified RPV stress report.

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4.3 Evaluation of Shroud Shell, Shroud Head, and Shroud Support System

The stress results were either prorated from the existing stress analyses or reanalyzed to represent the tie rod loads and differential pressures corresponding to the different service levels. The governing, maximum combined stress intensities are compared to the allowables. The details of the evaluation are contained in Reference 19.

4.4 Flow Induced Vibration

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4.5 Radiation Effects

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4.6 Loose Parts Consideration

The repair hardware assemblies are designed to minimize the number of parts and to prevent a loose part. All components are mechanically constrained using pin fasteners, threaded fasteners (all threaded joints are drilled and pinned, and locking springs, which engage into slots and prevent loosening due to vibration. Locking devices are not used as primary load carrying components and are carefully placed to assure that inadvertent overload will not occur. In order to prevent a loose part for the long term, IGSCC is considered by limiting the sustained stress levels, using IGSCC resistant materials, and precluding any welding. In addition, a periodic inspection program is planned by the utility. Sustained tensile stresses in Alloy X-750 components are less than ½ Sy (50% of yield stress at temperature) and thus initiation of IGSCC is highly unlikely.

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4.6.1 Design Features to Preclude Loose Parts

The stabilizer installation involves the following operations that could generate small objects or debris that may remain in the reactor after the repair is completed. It should also be noted that GE has applied this process for previous shroud repairs (more than 10 plants), as well as for access hole cover modifications. There have been no known fuel failures associated with the repair process. As an additional precaution, the area is vacuumed after the process, and a post-job visual inspection is performed to confirm the effectiveness of the cleanup process.

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The shroud support plate is a low flow area. Thus, swarf or particulates from honing which comes to rest on the shroud support plate is not expected to migrate significantly. Any swarf or honing particulate that may be picked up from the shroud support plate is acceptable as discussed below.

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The potential for the particles generated by the installation processes having adverse effects on instrumentation was also reviewed. Because the remaining particles are expected to be dispersed by the flow throughout the reactor, and there is no flow through the instrumentation that would tend to draw in these particles, it is not expected that these particles would be able to migrate into the instrumentation lines in sufficient quantities to cause plugging or other adverse effects. Therefore, it is very unlikely that these particles will have any significant effect on the instrumentation.

<u>Control of Parts and Tooling During the Installation</u>
Parts and tooling are logged and controlled per plant tool control procedures prior to installation in the vessel. Parts and tooling are checked for loose parts and foreign material prior to use and installation. A core cover will be utilized during installation to prevent foreign materials from entering the inside of the shroud. This core cover shall be lightweight and segmented for ease of handling and disposal, and shall be perforated to allow coolant flow. The thermal and hydraulic effects will be evaluated to assure no adverse effects on core cooling while installed.

4.6.2 Effects of Postulated Repair Assembly Failures

All pieces of the stabilizer assemblies are locked in place with mechanical devices that have demonstrated a number of reactor years of acceptable performance in similar BWR applications. Loose pieces cannot occur without the failure of a locking device. The stresses in the stabilizer components during normal plant operation are less than one half of the normal loading combination allowable stresses. The stabilizers are fabricated from stress corrosion resistant material. Therefore, it is unlikely that a stabilizer component will fail. However, if one stabilizer is postulated to fail during normal plant operation, there would be no consequence to the shroud (even if it is cracked) or to the other three stabilizers. The leakage through a cracked shroud may increase slightly, but not enough to be detectable. The plant would continue to operate until the next refueling outage, when the broken stabilizer would be detected during the normal IVVI program, and repaired. The postulated broken component may fall to the shroud support plate or may be sucked into the recirculation pump. The consequences of the postulated loose stabilizer piece would be consistent with the consequence of other postulated loose pieces.

4.7 Loss of Preload

The mechanical preload in the stabilizer system is small (about 5%) compared to the thermal preload and is used to provide initial fit-up tightness and ease of installation. The thermal preload developed in the stabilizer system and reacted onto the shroud is significant. To preclude any loss of preload due to inelasticity in the stabilizer or shroud material, the cross section stresses for all Service Level A and B load combinations are limited to less than the yield strength of the materials. [[

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4.8 Installation Cleanliness

During the stabilizer installation process, cleaning and cleanliness control is in accordance with requirements of GE specification 21A2040 (Reference 23). In addition, no graphite lead pencils are allowed to contact stainless steel and nickel alloys. EDM residue (swarf) is required to be captured to the maximum extent practical at every EDM machining location.

5.0 SYSTEMS EVALUATION

5.1 Bypass Flow

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5.2 Normal Operation

5.2.1 Steam Separation System

The leakage flow has the effect of slightly decreasing the flow per separator and slightly increasing the separator inlet quality. The separator performance is based on the applicable separator test data over the operating water level range. [[

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5.2.2 Recirculation System

The shroud repair leakage has no significant impact on the sub-cooling of the flow in the downcomer. Hence, there is no change in the net positive suction head and the margin to jet pump cavitation remains adequate. There is no impact on jet pump performance compared with the normal design condition.

5.2.3 Core Monitoring System

The impact of the leakage results in an over prediction of core flow by about 0.045% of core flow [[

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5.2.4 Operating and Fuel Cycle Length

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5.3 Anticipated Abnormal Transients

The code used to evaluate performance under anticipated abnormal transients and determine fuel thermal margin includes carryunder as one of the inputs. Since there is no significant change in carryunder due to leakage, the thermal limits are not impacted.

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This modification will have no adverse impact on the Transient analyses, Accident analyses, Abnormal Transient without Scram (ATWS) analyses, Stability, and EOOS currently in place for Clinton operation.

5.4 Loss of Coolant Accident Analysis and ECCS Performance

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]] Therefore, it is also expected that the conclusion drawn for the current analysis below would be applicable for the bounding 120% extended power uprate condition.

The leakage flows through the repair holes result in slightly increased time to core recovery, following core uncovery. The effect has been conservatively assessed to increase the PCT for the limiting LOCA by less than 6° F. Adding a 6 deg. F PCT penalty due to the shroud leakage and the reactor coolant inventory reduction due to this shroud repair results in a PCT of 1601 deg. F (1595 + 6). This PCT is significantly lower than the 10CFR50.46 acceptance criteria of 2200 Deg. F. Furthermore, this shroud repair does not affect the limiting break location, limiting break size nor the limiting single failure. This 6 deg. F PCT increase will be reported to the NRC via the standard 10CFR50.46 submittal.

5.5 Conclusions

The impacts of the leakage flows through the shroud repair holes have been evaluated. Also, the impact of the shroud leakage and the coolant inventory reduction on the ECCS performance were evaluated and the impact was determined to be a 6 Deg. F increase in the PCT. Even with this 6 Deg. F increase in the PCT, the PCT is about 600 Deg. F below the 10CFR50-46 acceptance criteria of 2200 deg. F.

6.0 MATERIALS AND FABRICATION

Materials and fabrication requirements applied to the Clinton repair hardware design are summarized in this section of the report. [[

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6.1 Materials Selection

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Alloy X-750 material meets the requirements of section 5.10.5 of the BWRVIP design criteria document (Reference 4) with specific plant in-reactor service related conditions which are permitted per note at the end of section 5.10 of BWRVIP-02 (Reference 4) in addition to being fully compliant with BWRVIP-84 (Reference 11) criteria.

Type XM-19 stainless steel material meets the requirements of section 5.10.6 of the BWRVIP design criteria document BWRVIP-02 (Reference 4) in addition to being fully compliant with BWRVIP-84 (Reference 11) criteria.

Type 316 stainless steel material meets the requirements of section 5.10.6 of the BWRVIP design criteria document BWRVIP-02 (Reference 4) in addition to being fully compliant with BWRVIP-84 (Reference 11) criteria.

6.2 Material Procurement Specifications

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6.3 Materials Fabrication

The stabilizer assemblies are fabricated from 316 SST, type XM-19 stainless steel, and Alloy X-750. There is no welding permitted during fabrication or installation. The fabrication practices applied meet the requirements of section 5.11 of the BWRVIP design criteria document (Reference 4) in addition to being fully compliant with BWRVIP-84 (Reference 11) criteria. The fabrication requirements are specified by GE specification 26A5734 (Reference 21).

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Specific list of ASME specifications and other GENE documents applicable to material fabrication are noted in sections 3.1 through 3.8 of GE specification 26A5734

(Reference 21). This specification 26A5734 also provides the requirements for the hardware fabrication practices to be applied to the Clinton shroud repair hardware fabrication.

7.0 PRE-MODIFICATION AND POST- MODIFICATION INSPECTION

This section summarizes the pre-modification and post-modification inspection requirements.

7.1 Pre-Modification Inspection

7.1.1 Minimum ligament lengths for weld H9

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7.1.2 Shroud Support Plate

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7.2 Post-Modification Inspection

7.2.1 Prior to RPV Assembly

Recommendations for overall visual inspection at the completion of the installation to both baseline the configuration and to confirm the installation arrangement are in compliance with BWRVIP-02 criteria (Reference 4).

7.2.2 During Subsequent Refueling Outages

Overall post modification reinspections during subsequent refueling outages for repair hardware and vertical welds are in accordance with BWRVIP-76 criteria (Reference 26).

7.2.2.1 Repair hardware Re-inspections

Overall visual reinspection recommendations and the tightness inspection recommendations are in accordance with BWRVIP-76.

7.2.2.2 Vertical Welds Reinspections

The reinspection recommendations for vertical welds are in accordance with BWRVIP-76.

7.2.2.3 H9 Weld Reinspections

The reinspection recommendations for H9 weld are in accordance with BWRVIP-38 and BWRVIP-104.

It is anticipated that existing plant in service inspections program will ensure that these inspections / reinspections are accomplished as required.

8.0 REFERENCES

1. The United States Nuclear Regulatory Commission "Title 10 Code of Federal Regulations".

2. American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel Code:
   a. Section III, Subsection NG, 1998 Edition with 2000 Addenda.
   b. Section II, Parts A, B, and C.
   c. Section III, Subsection NB, 1971 Edition with Summer 1973 Addenda (For RPV only).
   d. Section III, Subsection NG, 1974 Edition with Summer 1976 Addenda (For Shroud and Shroud support structures only).
   e. Section XI, 1989 Edition with no Addenda.

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15. Code Design Specification – Clinton, Reactor Pressure Vessel, GE Document No. 26A6214, Rev. 1.

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17. Clinton - Seismic & Dynamic Analysis - Core Shroud Repair Modification, GE Report No. GENE-0000-0023-6259-01, Rev. 1.

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27. "Clinton Power Station ECCS-LOCA Evaluation for GE-14", GENE reports GE-NE-A22-00110-27-02 plus NEDC-32974P and NEDC-32945P

28. Letter from K. J. Polson of Clinton Power Station to NRC dated June 6, 2002 "Core Shroud Inspection Results and evaluations", and letter from M. Pacillo of AmerGen Energy Co. LLC to USNRC, Number U-603558, dated December 29, 2001 "AmerGen's responses to GL 94-03 for Clinton" including Attachment 1 (Core Shroud Scanning and Inspection Results).

29. "Survey of Nuclear Reactor System Primary Circuit Heat Exchangers", by Nelms, H. A., Report ORNL-4399, April 1969, Oak Ridge National Laboratory, Oak Ridge, Tennessee (This is Reference 1 of GE Internal Analysis Method for Vibration Due To Vortex Shedding).

30. GE Input to 10CRF50.59 Evaluation by CPS for Installation of Stabilizers on the Clinton Core Shroud, GE document 0000-0023-6259-04, Revision 0

9.0 ATTACHMENTS

1. Clinton- Seismic & Dynamic Analysis - Core Shroud Repair Modification, GE Report No. GENE-0000-0023-6259-01, Revision 1 (GE Proprietary).

2. Clinton RPV Internals Evaluation, GE Document No. GENE-0000-0023-6259-03, Revision 2 (GE Proprietary).

3. GE Input to 10CRF50.59 Evaluation by CPS for Installation of Stabilizers on the Clinton Core Shroud, GE document GENE-0000-0023-6259-04, Revision 0 (GE Proprietary)