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TERRY,C.D.   Niagara Mohawk Power Corp.
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See Rpt.

SUBJECT: Forwards proprietary response to request for addl info re Unit 1 core shroud repair & Rev 0 to 383HA718, "Thermal Cycle,Reactor Vessel & Nozzle - Description,Bases & Assumptions." Rev 0 to 383HA718 withheld (ref 10CFR2.790).

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NIAGARA MOHAWK POWER CORPORATION/NINE MILE POINT NUCLEAR STATION, P.O. BOX 63, LYCOMING, N.Y.13093 /TEL. (315) 349-7263
FAX (315) 349-4753

CARL D. TERRY
Vice President
Nuclear Engineering

February 24, 1995
NMP1L 0907

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

RE: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Subject: *Response to Request for Additional Information Regarding the Nine Mile Point Unit 1 Core Shroud Repair (TAC No. M90102)*

Gentlemen:

By letter dated February 23, 1995, the Commission made a request for additional information concerning the Nine Mile Point Unit 1 Core Shroud Repair. Attachment 1 to this letter provides responses to the requested information except for Requests 2, 3, 4, 5, and 7. Responses to these requests will be submitted to the Commission following completion of the H8 weld inspection.

Attachment 2 to this letter provides GE-NE Specification 383 HA 718: "Thermal Cycles, Reactor Vessel and Nozzle - Description Bases and Assumptions." GE-NE Specification 383 HA 718 is considered by its preparer, General Electric, to contain proprietary information exempt from disclosure pursuant to 10CFR2.790. Therefore, on behalf of General Electric, Niagara Mohawk hereby makes application to withhold this document from public disclosure in accordance with 10CFR2.790(b)(1). An affidavit executed by General Electric detailing the reasons for the request to withhold the proprietary information has been included. Niagara Mohawk will provide the Commission a non-proprietary version of the subject document as appropriate by March 31, 1995.

As indicated in previous submittals, the Nine Mile Point Unit 1 Core Shroud Repair will be performed as an alternate to ASME Section XI as permitted by 10CFR50.55a(a)(3). Consequently, Commission approval is required. Niagara Mohawk requests approval of the repair plan prior to the Nine Mile Point Unit 1 Reactor Vessel Hydrostatic Test which is scheduled for March 15, 1995. Approval prior to the scheduled hydrostatic test will preclude potential delays in performing the test and the extension of our current refueling outage.

Very truly yours,

C. D. Terry
Vice President - Nuclear Engineering

CDT/JMT/kab
Enclosure

A001

xc: Regional Administrator, Region I
Mr. L. B. Marsh, Director, Project Directorate I-1, NRR
Mr. D. S. Brinkman, Senior Project Manager, NRR
Mr. B. S. Norris, Senior Resident Inspector
Records Management

ATTACHMENT 1

Request for Additional Information
Regarding Core Shroud Repair
Nine Mile Point Unit 1

Information Request 1

Wedges between the core support and the shroud (also called the clamp/spacer) have been provided at each stabilizer location to prevent motion of the core plate relative to the shroud. In order to verify that these wedges prevent both lateral and vertical motion, we request that you provide the analyses and calculations to demonstrate the adequacy of the wedges to prevent relative motion in the presence of postulated 360° through wall cracks at welds H_{6A} and H_{6B}.

Response 1

The clamp/spacer assemblies are located in the annulus between the shroud and the core support and are intended to restrict horizontal motion. The clamp/spacer assembly is not intended to carry vertical loads. The vertical load path from the core support is through the tie bolts connecting to the core support ring in the shroud. The vertical load puts the H6A weld in compression and does not rely on a structurally sound weld to carry the load. Weld H6B is not in the load path.

At the core support elevation, the clearance between the clamp/spacer and shroud is limited to 0.020 inches. Loads resulting in core plate horizontal movements of greater than 0.020 inches will be transferred directly to the shroud shell and to the lower spring. This load path bypasses the H6A and H6B welds.

The maximum calculated horizontal load at the lower spring during a seismic event is 63,800 lb. which includes the shroud, the core, and the other internal components. The horizontal load at the core plate/shroud contact will be less than the 63,800 lb., but the specific load at this location was not calculated. The 63,800 lb. load is used as a bounding condition. The allowable compressive stress of the 316 stainless steel spacer is 1.5 times the yield strength, or 29,025 psi. The required contact area is:

$$\text{Area} = \text{Load/Allowable Stress} = 63{,}800 \text{ lb.}/29{,}025 \text{ psi} = 2.2\text{in}^2$$

The spacer contact area of 5.89 in^2 exceeds the required contact area of 2.2 in^2 and is therefore sufficient to carry the bearing load.

Information Request 6

An upset condition wherein cold water is introduced into the annulus while the reactor inlet plenum remains at 545°F is considered a bounding upset thermal event for the tie rod

9503090248

assembly. This situation could potentially occur with the loss of feedwater followed by restoring the feedwater flow, but without heating. Provide the reference document "GE-NE Specification 383 HA 718: Thermal Cycles, Reactor Vessel and Nozzle, Description Basis and Assumptions," which contains typical assumptions leading to this event. Identify the conditions assumed for the NMP1 analysis and provide documentation to demonstrate that the stresses in the tie rod assemblies and attachment point remain within allowable limits during this transient for the limiting combination of postulated weld failures.

Response 6

A copy of Specification 383 HA 718 is provided (Attachment 2).

An upset condition wherein cold water is introduced into the annulus while the reactor inlet plenum remains at 545°F. This situation could occur with the loss of feedwater followed by restoring the feedwater flow, but without heating. Typical conservative assumptions leading to this event are described in Specification 383 HA 718, "Thermal Cycles, Reactor Vessel and Nozzle - Description Basis and Assumptions." The scenarios for these events are generic although the specific details may differ depending on the vintage of the plant.

The NMP1 load definition document, 237E434, does not include annulus temperatures for this event, but the temperatures are shown in detail on later plant thermal cycle diagrams. The thermal cycle diagrams for Hatch 1 and Peach Bottom shows the temperature inside the shroud drops by 15°F from operating temperature while the annulus temperature drops as low as 300°F. This event results in the largest temperature difference between the shroud and annulus. These temperature conditions have been determined by GE to be applicable for NMP1. Assuming the temperature inside the NMP1 shroud remains at 545°F while the annulus temperature drops to 300°F adds margin to the temperature difference. This event is compared with other transients described in the Nine Mile Point Unit 1 FSAR, Chapter XV and is considered the bounding upset thermal event for the tie rod assembly. Blowdown events may result in more rapid cooling, but both the shroud and repair hardware are cooling at a similar rate and the net thermal effects are not as severe.

Annulus region assumed at 300°F

Shroud inner surface assumed at 545°F

Tie Rod Temperature = Annulus Temperature = 300°F

Shroud Temperature = Average Temperature = 422°F

The unrestricted differential thermal expansion between the shroud and the stabilizer is calculated at 0.367 inches for the upset transient. This entire thermal differential was assumed to be taken by the stabilizer assembly as a bounding thermal loading condition. The resulting 188,638 lb. load per stabilizer assembly produces secondary stresses that are well within the 3Sm stress limit. The stresses in the tie rod assembly

attachment points for this upset condition are presented in GE-NE-B13-01739-04, Rev. 0, Tables 3-5 and 4-5.

Information Request 8

Provide Figure 3-12 of Report No. GE-NE-B13-01739-04 relating to the longitudinal displacement of the tie rod C-spring. This figure is missing in the submitted document.

Response 8

The tie rod C-spring longitude displacement is shown in Figure 4-12 of GE Report GE-NE-B13-01739-04, Rev. 0, previously submitted to the Commission.

Information Request 9

GE-NE-B13-01739-04, Rev. 0 stated that potential stagnant flow conditions were considered at the H_9 location. Provide information on the stagnant flow conditions and the criteria for acceptability.

Response 9

Information regarding stagnant flow conditions and the criteria for acceptability at the H8 weld repair bracket locations is contained in Reference 9 of GE Report GE-NE-B13-01739-04, Rev. 0. Reference 9 is included in Attachment 3 of this submittal. Reference 9 is a letter from B. M. Gordon (GE) to J. D. Lazarus, et. al., "Suitability of Nine Mile Point 1 H8 Shroud Weld Repair Bracket," January 10, 1995.

Information Request 10

Provide a copy of Reference 7 (letter to Roy Corieri and George Inch from J. A. Villalta and A. Mahadevan, Subject: Displacement of the Nine Mile Unit 1 Shroud due to DBE and Recirculation Line Break, dated September 26, 1994) which discusses the asymmetric loads during a recirculation line break.

Response 10

The referenced memo (Attachment 4) contains the asymmetric loads on the shroud resulting from a recirculation line break. The limiting stabilizer load is its reaction load against the overturning moment. Assuming weld failures below the core support, the lower spring contact provides the pivot point about which the shroud rotates. The moment at the pivot location is 4500 kip-inches. Weld failures above this location move the pivot point up, and the reaction moment is less.

Information Request 11

Provide analysis of the downcomer flow characteristics with the four tie rod assemblies installed. Specifically address the available flow area in the annulus and the associated pressure drop.

Response 11

The calculations of the flow restrictions in the annulus are attached (Attachment 5). The restrictions of the four stabilizers have negligible effects on the pressure drop through the annulus.

Information Request 12

GE-NE-B13-01739-05, Rev. 1, stated that with the stabilizers and H_8 brackets installed, no vertical displacement would occur during normal operations if any or all welds were completely cracked. Provide analysis of the potential leakage through all postulated cracked welds with no displacement. Specifically, address cracks above the core plate, below the core plate, and total leakage from all welds.

Response 12

The preload and/or the reactor pressure keep cracks in compression and minimizes leakage. If more than one weld is cracked, one of the welds may be loaded less than the others. The potential leakage through a postulated crack is 20 gpm if the crack is below the core plate and 10 gpm if above the core plate. The postulated cracks are assumed to be smooth walled cracks, 0.001 inch wide through the shroud wall. This leakage is negligible when compared to the total shroud repair leakage of 1770 gpm evaluated in Safety Evaluation GE-NE-B13-01739-05, Part B.3.

Information Request 13

Provide the analysis performed to demonstrate the integrity of the core spray piping when subjected to shroud displacements of 0.904 inches and 0.61 inches in the horizontal and vertical directions, respectively.

Response 13

The core spray piping analysis is attached (Attachment 6). The core spray piping was modeled and analyzed with the SAP finite element analysis program. The analysis includes a 0.904 inch horizontal shroud movement and a 0.61 inch vertical shroud movement. The maximum strain was calculated at 1% which occurs at the elbow in the riser section of the piping. A 2.5% strain is permissible to prevent material cold-working. Although the 1% strain may result in local plastic deformation, it will not affect the performance of the core spray piping.

Information Request 14

The NMP1 Reactor Core Shroud Repair Design Summary discussed the maximum vertical displacement during a MSLB. Provide the analysis which supports the conclusion that the tie rods elastically stretch during a MSLB with postulated failed welds. Provide information on the maximum separation of each postulated failed weld during a MSLB.

Response 14

A main steam line break, coupled with a seismic event for the bounding weld failure scenario which assumes welds H1 through H6B are failed, results in a maximum weld separation of 0.61" vertically at weld H6B. The vertical displacement results in the elastic elongation of the C-spring and elastic/plastic elongation of the tie rods. The stabilizer assembly and the shroud have been analyzed for this event and found to remain within the allowable stress limit, however, limited plastic distortions are permitted for this event. The elastic springback combined with the equipment weight will close all gaps once the event is over. The allowable stress limits assure there are no structural failures, but the equipment requires inspection and possible repair before service can resume.

The upward loads produced by this event are restrained by the tie rods and keep the welds above the core plate closed and in compression. The welds below the core plate will separate at the highest weld failure. Should welds H8, H7, and H6B all remain sound, the upward load is only that of the shroud head ΔP and the upward displacement will be less. Weld separation will occur at the highest failed weld.

Information Request 15

In the design specification 25A5583, Rev. 2, the ASME Code Section IX, Welding and Brazing Qualification, 1989 Edition was referenced. Identify under what conditions will welding be applied during the fabrication and installation of the core shroud repair components. What are the controls or mitigation methods that will be implemented to minimize the magnitude of the residual stresses and material sensitization when applying welding?

Response 15

There is no welding or brazing anticipated with the core shroud repair hardware. The ASME Code section was referenced in the event of a modification to the shroud repair, which required welding, was required. If a modification were required, details would be provided to the Commission for review and approval.

Information Request 16

BWRVIP has issued the following documents to provide guidelines for visual examination (VT) and ultrasonic examination (UT) of core shrouds: (1) BWRVIP, "Standards for Visual Inspection of Core Shrouds," September 8, 1994; and (2) BWRVIP Core Shroud NDE

Uncertainty & Procedure Standard, November 21, 1994. Provide a confirmation that the guidelines in these documents will be followed in the examination of the core shroud and repair assemblies. The subject BWRVIP documents should also be referenced in the appropriate examination specification.

<u>Response 16</u>

NMPC will perform examination of core shroud welds in accordance with the BWRVIP documents "BWRVIP Standards for Visual Inspection of Core Shrouds" and "BWRVIP Core Shroud Uncertainty and Procedure Standard." NMPC has reviewed the appropriate specifications for shroud weld examinations to verify these documents are referenced. Where references were omitted, the specifications are being revised.

<u>Information Request 17</u>

The NRC staff noted that in Section 4.0 of Repair Examination in the field disposition instruction (FDI) specification (0245-90800), the required resolution for the television camera is defined as capable of resolving a 0.001-inch wire on a neutral gray background. This requirement should be changed to be consistent with the required resolution of a 0.0005-inch wire as recommended in the BWRVIP documents referenced in Item 16 above for visual examination of core shrouds.

<u>Response 17</u>

Field Disposition Instruction (FDI) specification (0245-90800) will be revised prior to performance of the inspections to require use of a camera capable of resolving a 0.0005 in. wire against a neutral gray background. The FDI as originally submitted was incorrect. NMPC review identified this error and NMPC has directed that the FDI be revised accordingly.

<u>Information Request 18</u>

In the safety evaluation of GE core shroud repair design (GE-NE-B13-01739-05, Rev. 1) Part A.2 Materials, GE referenced a statement from the shroud repair fabrication specification (GE-NE Specification: 25A5584, Rev. 1), which stated that the successful completion of the sensitization testing (ASTM A262, Practice A or E) shall be accepted as evidence of the correct solution heat treatment and water quenching if time and temperature charts and water quenching records are not available. The NRC staff considers that in a good quality assurance program, accurate records of time, temperature, and cooling rate are necessary to be maintained as evidences that a proper heat treatment has been performed. Therefore, the use of sensitization test results as a substitute for the proper heat treatment documentation is not acceptable.

Response 18

The purpose of the GE requirements on heat treatment of stainless steel core shroud repair materials is to provide material that is not sensitized. Accordingly, sensitization testing of the material after heat treatment is an accurate indicator that the heat treatment was proper. In other words, the attributes of the final material condition is considered as evidence that the engineering requirements of material performance have been met. Therefore, sensitization testing is considered an adequate alternative to detailed heat treatment records in assuring that proper heat treatment has been performed. Also, complete reliance on heat treatment records can be misleading. GE has indicated that isolated cases have occurred in which heat treatment details were recorded, even to the extent of using embedded thermocouples, but that the subject material failed to pass a sensitization test.

ASME NCA-3800 was followed to procure core shroud repair material. With respect to material test reports, NCA-3862 does not require that detailed time/temperature records for heat treatment be recorded but that specific time and/or temperature parameters be recorded if such values are specified in the underlying Section II material specification. For the austenitic stainless steels used for the shroud restraints, the only stated requirement is a minimum temperature of 1900°F (followed by rapid cooling). NMPC has reviewed and is in possession of the mill reports/certified material test reports documenting the required heat treatment.

With respect to the requirement stated in GE specification 25A5584, GE routinely adds additional requirements for heat treating stainless steel that is over and above the ASME Code. When material is ordered in heat lots from a primary melter, it is possible to get such detailed records of heat treatment. However, in the current environment of performing internals repairs, materials are procured in small quantities, often from a third party supplier out of a warehouse inventory. In these cases, detailed records showing complete conformance to the additional GE requirements are not always retrievable. It is in these cases where, as an alternative to detailed heat treatment records, the attributes of the final material condition is considered as evidence that the engineering requirements for materials performance have been met. NMPC has reviewed and is in possession of all of the test reports required by the GE Specification 25A5584 including the sensitization test results.

In summary, it is NMPC's position that the material ordering requirements are appropriate for the intended use and are in conformance with applicable codes, regulations, and quality standards.

Information Request 19

General Electric (GE) stated in their safety evaluation (GE-NE-B13-01739-05, Rev. 1) of core shroud repair design that the tie rod threads (low carbon type 316 or type 316 stainless steel) would be induction annealed after machining to remove a possible cold worked layer.

a) *Provide details regarding how the induction heating process was qualified and the results of your metallurgical evaluation of the tie rod threads after induction annealing such as its effect on the material hardness, surface oxidation and the state of sensitization.*

b) *GE stated that a minimum of 0.030 inches of Alloy X-750 materials will be removed after high temperature annealing as a control of intergranular attack (IGA). Will this process or any other process be applied to Type 316 tie rod threads after induction annealing to ensure there is no IGA? Provide the test data to support that the removal of 0.030 inches surface material would effectively eliminate the IGA effect resulting from the high temperature annealing.*

c) *Will induction annealing or any other process be applied to machined threads made of Alloy X-750 such as toggle bolts to minimize the effect of cold work?*

Response 19

a) The induction annealing of the tie rod threads is as follows:

Induction heat at about 8 kHz; hold at 1900°F to 2000°F for 1 minute ±3 seconds. Force air cool to below 200°F, and air cool to ambient. The hardness of the machined surface is reduced from Rc 30-35 to Rb 80-81. The thin, amorphous, layer of smeared metal was recrystallized back into normal grains and the grain growth resulting from this operation was minimal (9 to 6-7.5). The data for this work is in GE Laboratory Investigation 08068.

b) There were no material removal processes used on the 316 stainless steel tie rod threads after the induction annealing.

c) There were no experiments conducted on the effects of resolution heat treatment of X-750 machined surfaces because metallurgical investigations (GE Laboratory Numbers 08085 and 08127) show the surfaces are unaffected by machining and do not require such heat treatment.

Information Request 20

Identify all the threaded areas and locations of crevices and stress concentration in each component of the core shroud repair assemblies. In the planning of inservice inspection (ISI) those areas should be emphasized for inspection because these areas are most susceptible to stress corrosion cracking. Please provide these information in tables and supplement it with sketches.

Response 20

NMPC will review the potential degradation mechanisms for these areas (threaded, creviced, areas of stress concentration). NMPC will factor these considerations into the ongoing BWRVIP work on repair assembly reinspection requirements for

applicability to NMP1. Once the BWRVIP work on repair reinspection frequency and scope is established as it applies to NMP1, NMPC will submit plans for reinspection.

Information Request 21

Provide details of your controls in the practices of machining, grinding and threading to minimize the effect of cold work, such as amount of materials to be removed in each pass, application of coolant and sharpness of the tool.

Response 21

The amount of material removed in a single pass depends on the part and the particular machine doing the work. Parts are generally rough machined to within 0.10" of final size and skim passes are used to achieve the final dimensions. Features where cold work may be detrimental, such as the stainless steel tie rod threads, are heat treated after machining as specified on the drawing. Tests show that Inconel X-750 surfaces are unaffected by machining and do not require heat treatment. The application of coolant and sharpness of the tool is considered adequate provided a surface finish of 125 root mean square or better is obtained.

Information Request 22

In Part B.12.1 of GE-NE-B13-01739-05, Rev. 1, GE stated that all parts have been designed so that they can be removed and replaced. This design feature should be taken advantage of when planning ISI of the components of the core shroud repair assemblies. The NRC staff realizes that the repair assemblies may be inspected by a combination of visual and ultrasonic examinations. However, the NRC staff has some concerns regarding the reliability of such inspection to identify the potential degradation in the threaded joints and areas of crevices and stress concentration, which have limited access for inspection. Please provide a discussion and/or propose an alternative inspection such as disassembling the threaded joints for inspection to ensure that the areas mentioned above in the repair assemblies will be adequately inspected for early detection of potential degradation.

Response 22

NMPC will review the potential degradation mechanisms for these areas (threaded, creviced, areas of stress concentration). NMPC will factor these considerations into the ongoing BWRVIP work on repair assembly reinspection requirements for applicability to NMP1. Once the BWRVIP work on repair reinspection frequency and scope is established as it applies to NMP1, NMPC will submit plans for reinspection.

Information Request 23

Provide details of your planned baseline ISI (location, extent, frequency, methodology and justification) of the core shroud before and after repair.

Response 23

Other than a VT-3 Visual Inspection of the completed repair assembly, NMPC does not plan an ISI Baseline Inspection.

NMPC plans to inspect, in support of the repair installation, the following welds:

1. Required portions (four locations, adjacent to the repair tie-rods, each of 26 inches minimum in length) of the H9 weld by enhanced visual examination of the top surface.

2. All areas, made accessible by the inspection tooling, of H8 using volumetric techniques.

3. The NMP1 shroud has two (2) vertical welds per each shell section and two (2) ring segment welds per each ring section. Required portions (each six inches in length) of vertical welds V9, V10, V11 and V12, as they intersect the H5 circumferential weld, will be inspected from the inside surface by enhanced visual examination. For weld identification and location refer to shroud drawings previously enclosed as part of NMPC's August 23, 1994 submittal.

4. Accessible portions of the ring segment welds V5 and V6 from the top/inside surface with enhanced visual examination if possible.

GE FDI 0245-90800 will be revised, as appropriate, prior to performance of the inspections. The core shroud reinspection scope and schedule will be addressed through the ongoing work within the BWRVIP as outlined in the response to Information Request 20 and 22.

Information Request 24

Provide details of your planned ISI (location, extent, frequency, methodology and justification) of the installed core shroud repair components. Your planned inspection should consider the staff recommendation in Item 22.

If complete information for Items 23 and 24 cannot be provided at this time, identify the date when such information will be provided.

Response 24

NMPC will conduct a VT-3 inspection by camera of the installed repair assembly during the current outage. Re-inspection scope and schedule for the repair assembly and/or components will be addressed through the ongoing work within the BWRVIP as outlined in the response to Information Requests 20 and 22.

Information Request 25

Identify the lubricants that would be used on the machined threads during installation. What are the controls of the content of chlorides, sulfides, halogens and other elements that are known to promote stress corrosion cracking in stainless steel and high nickel alloy?

Response 25

The lubricant used for machined threads during installation is thread lubricant D50YP5B. The lubricant is referenced on GE Modification Drawing 107E5679 Rev. 4., Note 2, GE Modification Drawing Parts List PL107E5679 Rev. 3, Item 14 and the Stabilizer Installation Specification 25A5585, Rev. 2, Section 5.1.6.

GE Specification D50YP12 provides the controls of the content of halogens, nitrate, sulfur and other elements in lubricant D50YP5B. The specification defines the requirements for limiting impurities in lubricants that are in crevices exposed to BWR primary reactor coolant at temperatures above 165°F. Impurity limits are as follows:

1. The maximum allowable level of halogens, when both sulfur and nitrates are less than 1 ppm, is 450 ppm.

2. The maximum allowable level of sulfur, when both halogens and nitrates are less than 1 ppm, is 630 ppm.

3. The maximum allowable level of nitrate, when both total halogens and total sulfur are less than 1 ppm, is 820 ppm.

4. Allowable simultaneous levels of halogens, sulfur and nitrates in combination can be determined from the nomograph in Addendum A in the following manner when the values of any two of the impurities are known, or are assumed. Place a straightedge through the two known or assumed values on their scales and read the third value from its scale. The third value must not be exceeded.

5. Allowable combined levels of halogens, sulfur and nitrates can also be determined using the formula below.

$$\frac{\text{ppm Halogens}}{35.453} + \frac{\text{ppm Sulfur}}{48.096} + \frac{\text{ppm Nitrates}}{62.004} < 13.2$$

6. The maximum allowable level of any single low melting point metal is 200 ppm.

7. The maximum allowable level of all low melting point metals in combination is 500 ppm.

Information Request 26

Provide a discussion on how the magnitude of the spring preload will be monitored to ensure there is no substantial relaxation of the preload. Please also discuss the safety consequences if the spring preload is completely relaxed.

Response 26

The preload applied by the tie rod nut assures all connections are initially tight. The designed repair uses mechanical locking methods (such as crimped jam nuts on top of the tie rod nuts) for threaded connections. The differential thermal expansion between the shroud and stabilizer provide the load to assure any failed weld remains tight during reactor operation. That is, the preload calculations conservatively neglect the small 3000 pound mechanical preload applied to each tie rod by tightening the tie rod nuts. Therefore there are no plans to monitor the magnitude of the spring preload, but the stabilizer will be visually examined to assure the assembly remains tight. Complete loss of the mechanical preload has no affect on the thermal preload and there are no safety consequences.

Information Request 27

Recently, IGSCC was observed in the welds (heat-affected zones) of the top guide and core support plate in an overseas BWR. Therefore, the staff recommends that the welds in the top guide and core support plate at Nine Mile Point Unit 1 should be inspected during the upcoming refueling outage to ensure there is no unacceptable degradation.

Response 27

NMPC has reviewed the details provided by GE RICSIL 071 which discusses the IGSCC observed in the top guide and core plate of the overseas BWR. The review for applicability to NMPC was conducted as documented in NMPC Deviation Event Report DER 1-94-2395. The disposition of the DER referenced the BWRVIP analysis provided to the NRC by the BWRVIP on December 23, 1994, "Request for Additional Information Regarding the Impact of BWR Core Plate and Top Guide Ring Cracking," which was found to be bounding for the NMP1 BWR-2 design. NMPC has concluded that 360° separation of either or both of these welds, if it existed, would present no safety or operability concerns for NMP1.

Recently, NMPC received GE SIL 588 which supersedes RICSIL 071. SIL 588 provides an evaluation of cracking significance similar to the above referenced BWRVIP analysis and additionally, provides recommendations for inspection of restraining devices where they are taken credit for in the analysis. The BWRVIP does not yet have a position or recommendation for these restraining devices.

NMP1's top guide configuration utilizes alignment pins but does not have wedges. As discussed in the BWRVIP RAI, credit is taken for eight (8) lateral support brackets. The brackets are attached by intermittent fillet welds to the shroud and are

sized to be within 1/16 inch of the top guide ring. Insufficient clearance exists to perform a visual examination of the attachment welds.

NMPC intends to attempt the following inspections during the current refueling outage as recommended in SIL 588. Clearances in the annulus between the top guide and upper shroud around the core spray header do not permit an inspection to be performed from the top side of the top guide.

1. The top guide alignment pins will be verified as in place by VT-3 inspection if visible.

2. The lateral support brackets will be verified as in place by VT-3 inspection from below the top guide if visible.

3. The core plate bolts will be verified as in place by VT-3 inspection.

ATTACHMENT 3

BWR Technology
GE Nuclear Energy
San José, CA

January 10, 1995

To: J. D. Lazarus — S. Ranganath
G. M. Gordon — G. L. Stevens
H. A. Levin — B. A. McAllister

From: B. M. Gordon

Subject: Suitability of Nine Mile Point 1 H8 Shroud Weld Repair Bracket

References:

1. B. M. Gordon, "The Definition of a Crevice and its Application to the BWR," PMT Transmittal GENE-178-006-0792 Rev. 1, March 30, 1993.

2. J. W. Oldfield and W. H. Sutton, "Crevice Corrosion of Stainless Steels," British Corrosion Journal, Vol. 13, No. 1, 1978.

3. L. Liu, "Duane Arnold Energy Center Recirculation Inlet Safe End Repair Program," Iowa Electric Light and Power, December 8, 1978.

4. J. D. Lazarus and G. M. Gordon, private communication, January 6, 1995.

5. D. E. Delwiche, "Collet Retainer Tube Design Modification Development Report," NEDE-24004 Class III, June 1977.

The proposed Nine Mile Point 1 (NMP 1) H8 shroud weld repair bracket configuration is presented in Figure 1. Concern has been raised over the presence of crevices in the bracket design since it is well documented that crevices can result in premature intergranular stress corrosion cracking (IGSCC) initiation in the BWR structural materials. [1] This NMP 1 H8 weld repair bracket is characterized by two (2) areas of line contact between the faceted Alloy X-750 upper bracket and the cylindrical Type 304 stainless steel shroud, four (4) areas of line contact between the curved Alloy X-750 lower bracket and the conical Alloy 600 support shroud support cone (different radii of curvatures) and the one (1) oval crevice area between the curved Alloy X-750 lower bracket and the cylindrical stainless steel clad reactor pressure vessel (RPV) wall. Line contacts are geometrically considered crevices. [1] However, geometry alone is not sufficient to classify a particular geometrical configuration as an electrochemical crevice from a IGSCC perspective since many other factors affect the nature of a crevice. [2]

For example, a strict geometrical crevice definition provides no credit for solution flow. Experimental data and analysis has revealed that if flow induces one crevice volume change per day, the local crevice environment as identified by a decrease in pH is not established. [3] Therefore, daily flow between the repair bracket and the shroud, support cone and RPV wall would eliminate this concern. A second critical IGSCC factor is tensile stress, i.e., if the tensile stress is below the IGSCC threshold limit for creviced geometries, no IGSCC will occur. This is the specific IGSCC mitigating case for the NMP 1 H8 weld repair bracket as evaluated by a the stress rule index (SRI) evaluation. [4]

The SRI approach for IGSCC evaluation was primarily developed from laboratory test data on uncreviced specimen tests that indicated that cracking only occurred above the material's yield strength. Thus, the sum of the normalized primary stress ratio (using the yield strength) and the normalized secondary stress ratio (using the pseudo-elastic stress corresponding to the yield strain) was defined as the SRI. The limit on the SRI was set at 1.0 for uncreviced geometrical configurations. However, early tests on creviced tensile specimens revealed that IGSCC can occur under aggressive water chemistry conditions, (e.g., high dissolved oxygen contents, high conductivity) at tensile stress levels as low as 50% of the at temperature yield stress ($0.5S_y$). Since the crevice condition threshold limit is $0.5S_y$, the corresponding SRI limit for creviced conditions was set at 0.5. If the crevice is located at a weld, the weld residual stress must also be included in the SRI calculation. In instances where the crevice acts as a discontinuity, the stress concentration resulting from the "crack" like crevice should also be included in the calculation. In general, this limit is difficult to meet for welded crevices since weld residual stress and stress concentrations can lead to SRI values exceeding 0.5.

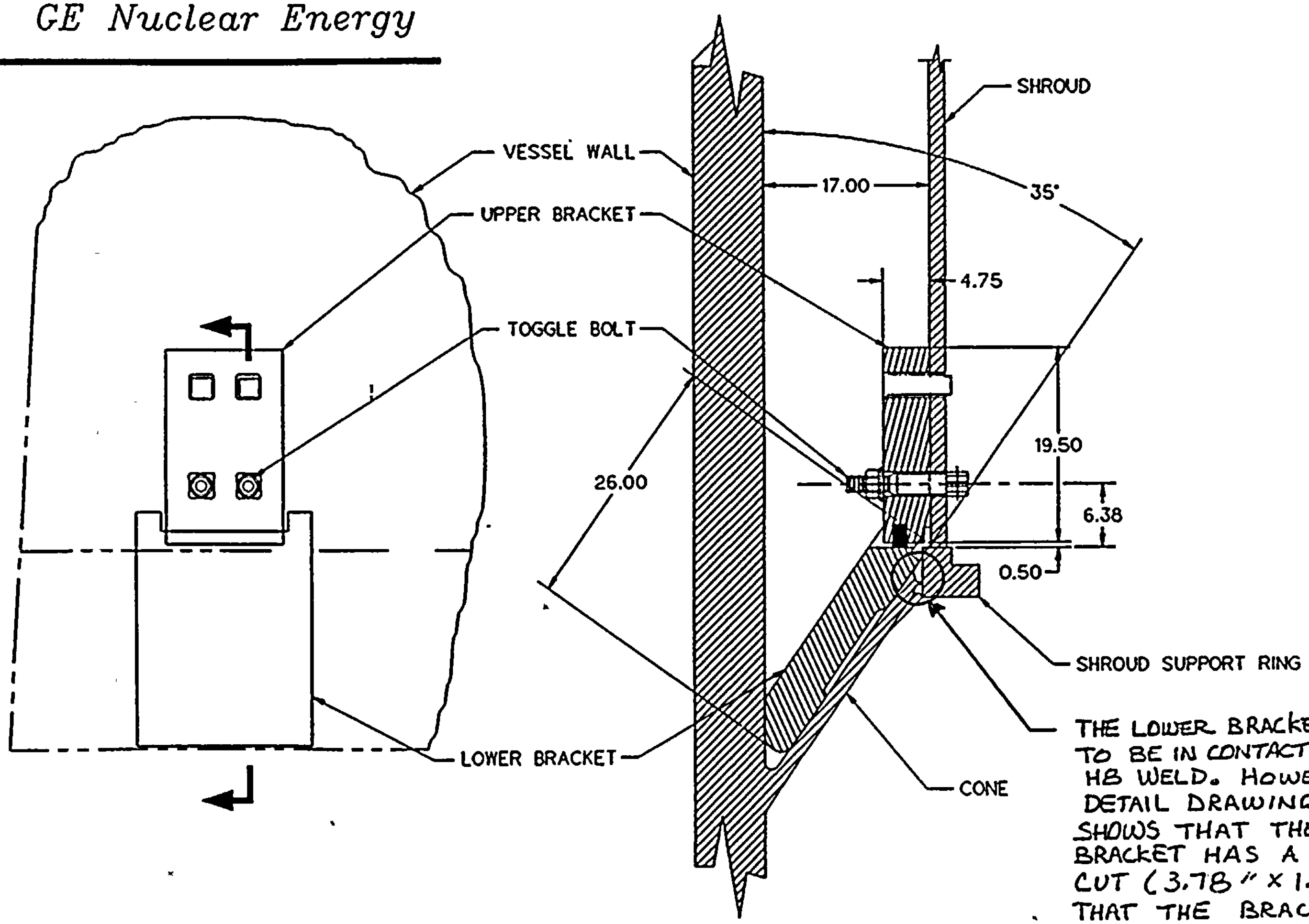
Since the SRI crevice limit of 0.5 is a major design concern for the "creviced" NMP 1 H8 weld bracket, it was considered prudent to present some of the supporting data for the establishment of this limit. For example, constant load IGSCC tests were performed on 2.54 cm (1 inch) wide Type 304 stainless steel collet retainer tube-to outer tube weld section specimens cut from collet retainer tubes (CRTs) in high oxygen (8 ppm) 288°C environments. [5] The weld sections contained the creviced portion of the weld joint with the weld heat affect zone (HAZ). The table below presents the results of this study:

Specimen	% YS	Hours	Time to IGSCC
7-304SS	39	1409	No IGSCC
8-304SS	30	5612	No IGSCC
9-304SS	50	2833	IGSCC

Since IGSCC was only obtained under accelerated conditions, i.e., 8 ppm oxygen, at an applied stress of 50% of the 288°C yield stress, a SRI value of 0.5 was designated as the crevice limit for BWR conditions. It should be noted that the stress value on these specimens does not include any weld residual stress since the weld was rather small and the IGSCC occurred far removed from the weld area, i.e., cracking was strictly a crevice effect. Further support for this limit is provided by the analysis of the IGSCC of creviced Alloy 600 shroud head bolts from various BWRs where IGSCC occurred at SRI values of 0.62 and 0.74.

With this stress limit in mind, the NMP 1 H8 weld repair bracket was specifically designed to be characterized by tensile stresses below the threshold value for IGSCC. In fact, the stresses in the regions with the geometrical line contact crevice are compressive and not tensile. Finally, in accordance with the IGSCC guidelines, the H8 repair bracket attachments points are carefully located on the Type 304 stainless steel shroud and Alloy 600 support cone away from any welded joints, i.e., no welds are creviced. Thus, no crevice related IGSCC is anticipated for the bracket or its contacting components.

GE Nuclear Energy



THE LOWER BRACKET APPEARS TO BE IN CONTACT WITH THE H8 WELD. HOWEVER, GE DETAIL DRAWING 178B3732 SHOWS THAT THE LOWER BRACKET HAS A SECTION CUT (3.78" x 1.25") SUCH THAT THE BRACKET WILL NOT CONTACT THE H8 WELD.

RC 2/1/95

NINE MILE SHROUD REPAIR
H8 BRACKET CONFIGURATION

FIGURE 1

ATTACHMENT 4

GE Nuclear Energy

GE Nuclear Energy
MC 747
175 Curtner Avenue
San Jose, CA 95125
(408) 925-2806

September 26, 1994

To: Roy Corrieri,
George Inch
Nine Mile Point - Unit 1

From: J.A. Villalta/A. Mahadevan
GE Nuclear Energy

Re: Displacement of Nine Mile Point - Unit 1 Shroud due to DBE and Recirculation Line Break

Technical specifications typically require 90% control rod insertion within three and one-half seconds after the initiating signal. In actuality, most rods scram sooner, approximately within two and one-half seconds. Assuming a finite delay time of one second from the onset of the DBE to scram initiation, control rod insertion should be complete in about four and one-half seconds.

Attachment 1 shows the lateral displacement is approximately two inches at 7.5 seconds after the onset of the DBE. For this lateral displacement, tests have shown the ability to fully insert control rods is assured. Since the control rod will insert fully well within the 7.5 second period, there are no safety implications relative to scram ability from a DBE.

Attachment 2 shows the shear forces and the moments following a recirculation line break. Under these loads, the cracked shroud could potentially tilt on one edge. However, the restoring moment due to the weight of the shroud is well above the moments due to the recirculation line break loads. Thus, the shroud will not tip or experience lateral motion as a result of a recirculation line break.

CC: S. Ranganath (GE)
A. Mahadevan (GE)

NINE MILE POINT - UNIT 1

ATTACHMENT 1: SEISMIC ASSESSMENT OF THE EXISTING SHROUD IN ITS UNMODIFIED CONFIGURATION

Given/Assumption:

1. LOCA (Steam and/or Recirc. Break) does not occur concurrent with DBE.

2. Within 3 seconds after the onset of a seismic event, the scram is complete.

3. The applicable seismic load is the licensing basis DBE.

Assessment:

1. A detailed computer model is utilized, which includes the reactor building, shield wall and pedestal, RPV, and the reactor internals. The 360 degree through-wall cracks at the weld locations are modeled and the licensing basis DBE time history load is applied. The resultant response time history of the displacements at the top guide and the core plate are generated. Review of the response time history shows that the displacement of the top guide relative to the core plate reaches a value of 2 inches after a time lapse of 7.5 seconds upon the onset of the DBE. Since scram is to be completed within 4.5 seconds from the onset of the seismic event, including a finite delay time of one second for the scram signal, the displacement will not impair the insertion of control rods during a scram. Vertical seismic effects are deemed insignificant and moreover vertical displacement does not impair control rod insertion during scram.

2. A simple static calculation is performed to determine the restoring effects of the vertical inertia of the shroud against the tipping effects of the seismic load, for a hinged condition at H8. This indicates that the lateral load due to the seismic event will not overcome the restoring moment due to vertical inertia. As a result, tipping is deemed not possible and thus impairment of rod insertion during scram, due to excessive displacement is deemed unlikely.

3. The physical attributes in the annulus between the RPV and the shroud near the shroud head region are investigated:

   - At the shroud head flange, in the 0-180 deg. diametrically opposite orientation, two 6.5" x 11" x 4" thk. lugs exist. The clearance between the lug and the RPV wall inner surface is 1". This would limit the displacement of the shroud at the top guide elevation to less than 1" in the 0-180 deg. direction.

     In the 60-240 deg. diametrically opposite orientation, core spray piping penetrates the RPV and runs vertically down the annulus for 18" and is welded to the sleeve

at the shroud head flange. This 6" dia. sch.40 piping provides substantial stiffness/restraint in the lateral direction in the 60-240 deg. orientation.

In any other diametrical orientation, due to stiffness component of the core spray line penetrations and the clearance limitations at the lugs described above, there is considerable restraint against displacement, limited to about 2" at the top guide location.

ATTACHMENT 2: ASSESSMENT OF NINE MILE POINT - UNIT 1 SHROUD FOR BLOWDOWN LOAD

The blowdown load is analyzed by examining the expected flow pattern near the recirculation suction nozzle with a potential flow model. The analysis is conducted for subcooled blowdown conditions, when the largest load is placed on the shroud.

Assumptions:

1. Viscosity effects are ignored. This is a small effect.
2. The annulus gap is assumed to be constant, the same as at H7. Although the actual gap changes, the constant gap assumption provides an accurate result for the blowdown load for the shroud assembly above H7. The annulus gap change at H3 does not affect the result much because H3 is far enough away from the suction nozzle.
3. All flow properties are constant across the annulus gap.

Assessment:

The calculated lateral force above the H7 weld is shown in Figure 1. The lateral force on the shroud assembly above H7 is 86 kips. The corresponding blowdown moment, taken about H7, is 4700 kip-inches (see Figure 2). This moment is much smaller than the restoring moment calculated in the seismic assessment of the shroud (3905 kip-feet). Thus, tipping due to a lateral blowdown load is not deemed possible and impairment of control rod insertion during scram is unlikely.

Figure 1. Lateral Shear Force vs. Elevation for Nine Mile Point Unit 1 Blowdown Analysis

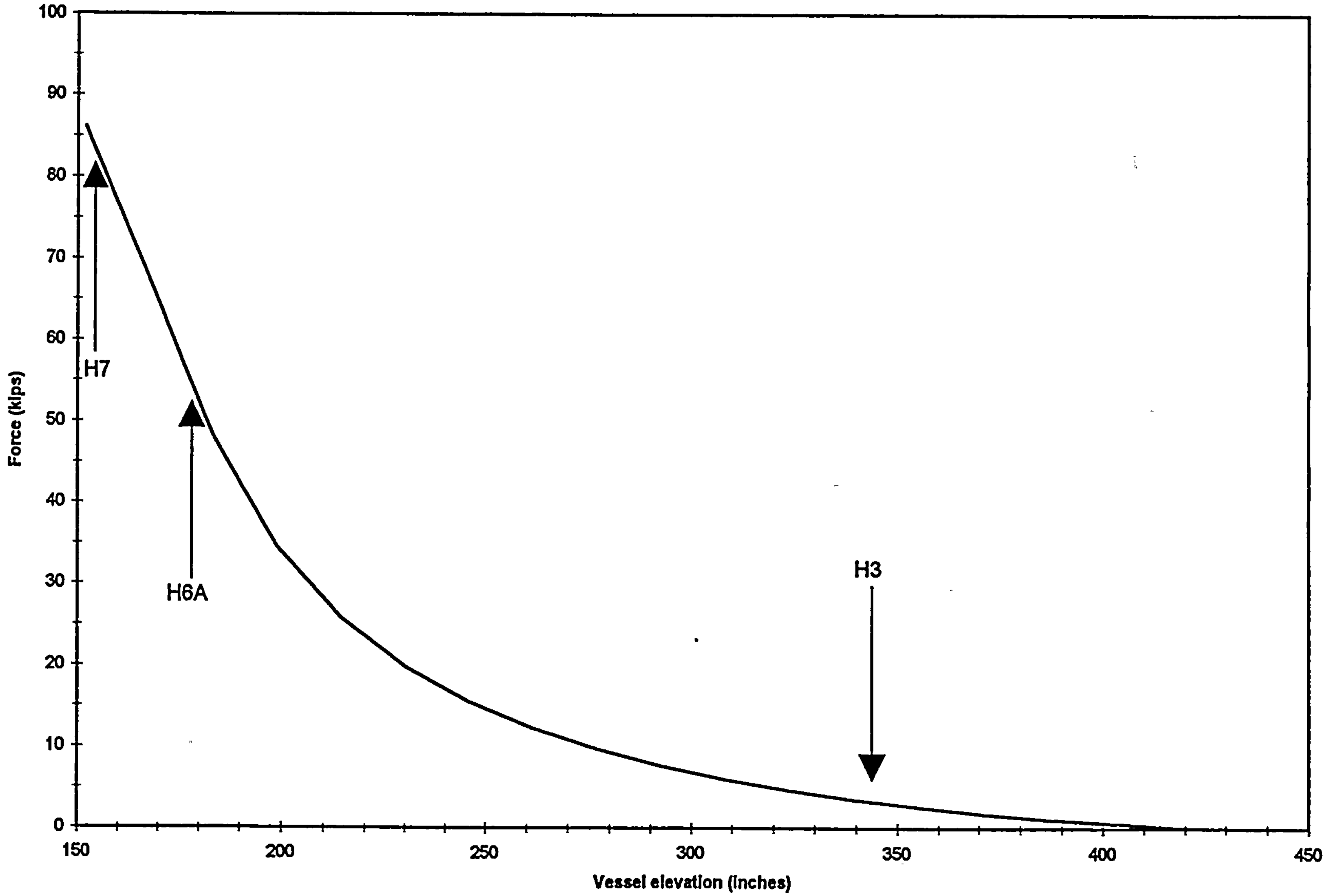
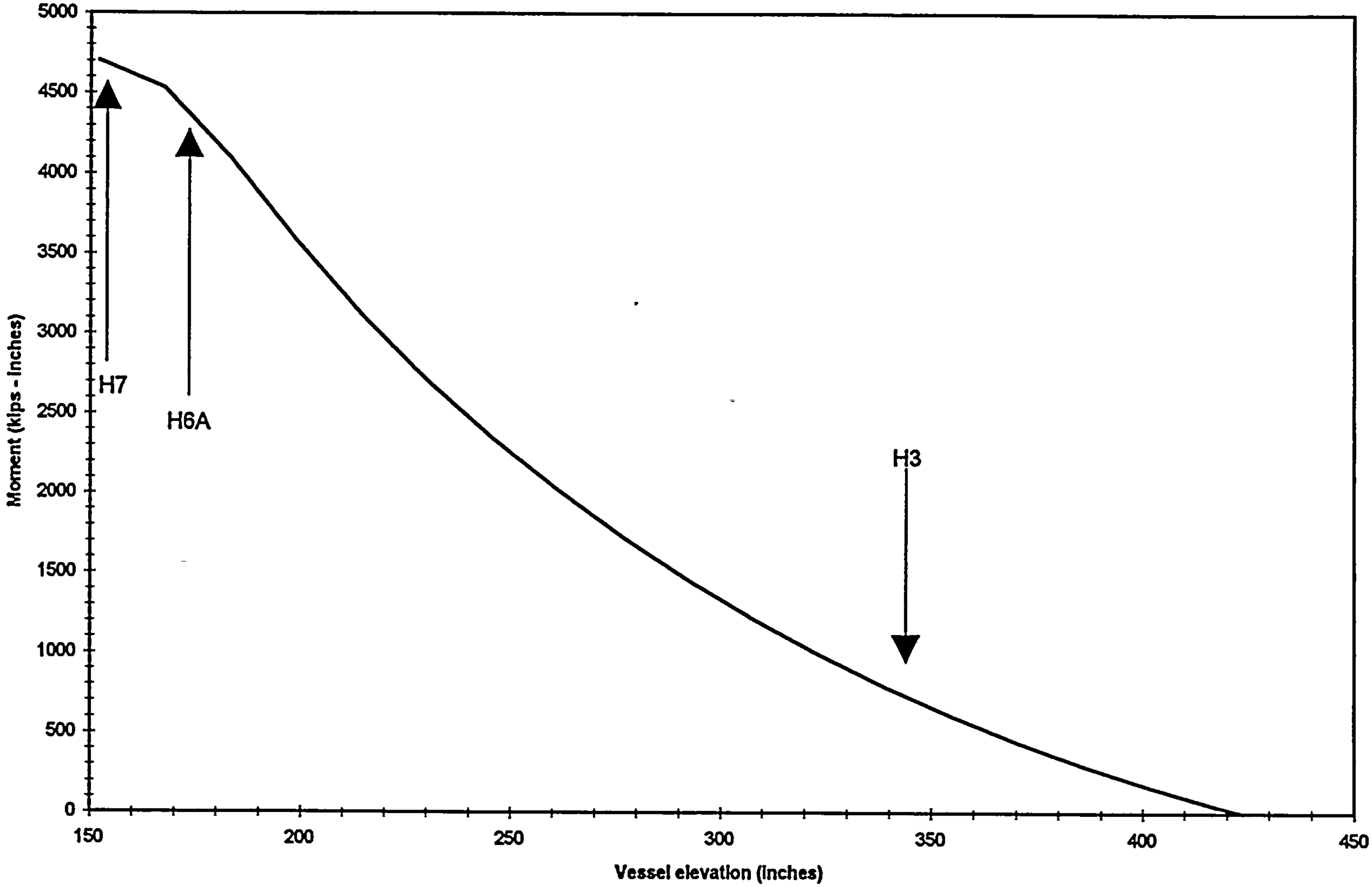


Figure 2. Moment vs. Elevation for Nine Mile Point Unit 1 Blowdown Analysis



ATTACHMENT 5

GE Nuclear Energy

Engineering Services Verification Cover Sheet
(Reference EOP 42-6.00 and EOP 25-6.00) Sheet 1 of ____

DRF NO./SECTION B13-01739

Subject: ANNULUS FLOW

1 **VERIFICATION REQUIREMENT** Designated Verifier ____

1A APPLICATION. (System/Project/Program)
NMP-1 SHROUD REPAIR

1B METHOD OF VERIFICATION. Checking, Alternate Calc, Indiv. Design Review, Team Design Review, Test (circle as needed); Other (describe)

1C SCOPE. (Identify what is to be verified (e.g., level of detail).
Verify dimensions and calculations for Annulus Flow

1D INPUTS. Identify any GE and external interfaces and requirements, assumptions, input documents, test analyses, reasons for changes. Dwg. 237E433, 706E231, 22A2903

1E OUTPUTS. Identify output document(s) or analysis results to be verified.

1F Responsible Engineer T. Gleason [signature] Date 2/20/95 Comp. 771
(Print name and Sign)

2 **INDEPENDENT VERIFICATION**
Comments: ☒ No ☐ Yes (See Attached)

2A VERIFICATION STATEMENT: The method and scope of verification as stated in 1B and 1C are appropriate. The inputs as identified in 1D. All comments and technical issues are resolved. The verification establishes that the output identified in 1E is correct and is adequate for its intended application as identified in 1A.

2B Independent Verifier W.F. Farrell [signature] Date 2/20/95 Comp. 771
(Print name and Sign)

3 **APPROVAL OF VERIFICATION**

3A MANAGER'S APPROVAL: All design requirements have been identified and all technical issues are adequately resolved. The verification described in the above sections 1 and 2 is sufficient to issue/apply the results.

3B Resp. Manager or Delegate: ____ Date ____ Comp. ____
(Print name and Sign)

4A Are there attached sheets for 1A,1B,1C,1D,1E,2A,3A? (circle as applicable)

SD-0568 (1/92)

DRF B13-01739

TEGleason

NMP-1 ANNULUS FLOW

The nominal dimensions forming the flow annulus are:

Vessel ID = 213 inches	(drawing 237E433)
Upper Shroud OD = 189 inches	(drawing 706E231)
Lower Shroud OD = 179 inches	(drawing 706E231)

The axial flow area in the annulus between the shroud and the vessel is calculated below.

Upper Annulus Area = $\pi/4 \times (213^2 - 189^2) = 7577.5$ in.^{2}
Area of Core Spray Pipes = $(\pi/4 \times 5.56^2) \times 4 + (\pi/4 \times 6.625^2) \times 2 = 166$ in.^{2}
Net Upper Annulus Flow Area = 7577.5 - 166 = 7411.5 in.^{2}

Lower Annulus Area = $\pi/4 \times (213^2 - 179^2) = 10{,}467.8$ in.^{2}

The reactor flow is 67,470,000 lb/hr as shown on the GE Reactor System Heat Balance, 22A2903. The specific volume of the fluid at 1000 psi and 545 F is 0.0216 ft^3/lb.

Flow Velocity = Q/A = 67,470,000 lb/hr x 0.0216 ft^3/lb x 144 / 3600 / Area
Flow Velocity = (58,294 ft-in^2/sec) / Area

The stabilizer restriction to axial flow in the upper annulus region is calculated on the attached sketch.

Area of the four assemblies = 4 x 98.3 = 393.2 in.^{2}

Per cent of flow area = 393.2 x 100 / 7411.5 = 5.3%

Flow velocity without stabilizers = 58,294 / 7411.5 = 7.9 ft/sec

Flow velocity with stabilizers = 58,294 / (7411.5 - 393.2) = 8.3 ft/sec

The flow in the major part of the lower shroud is restricted only by the four tie rods.

Tie rod area = $4 \times (\pi/4 \times 3.5^2) = 38.5$ in.^{2}

Per cent of flow area = 38.5 x 100 / 10,467.8 = 0.37%

Flow velocity without stabilizers = 58,294 / 10,467.8 = 5.6 ft/sec

Flow velocity with stabilizers = 58,294 / (10,467.8 - 38.5) = 5.6 ft/sec

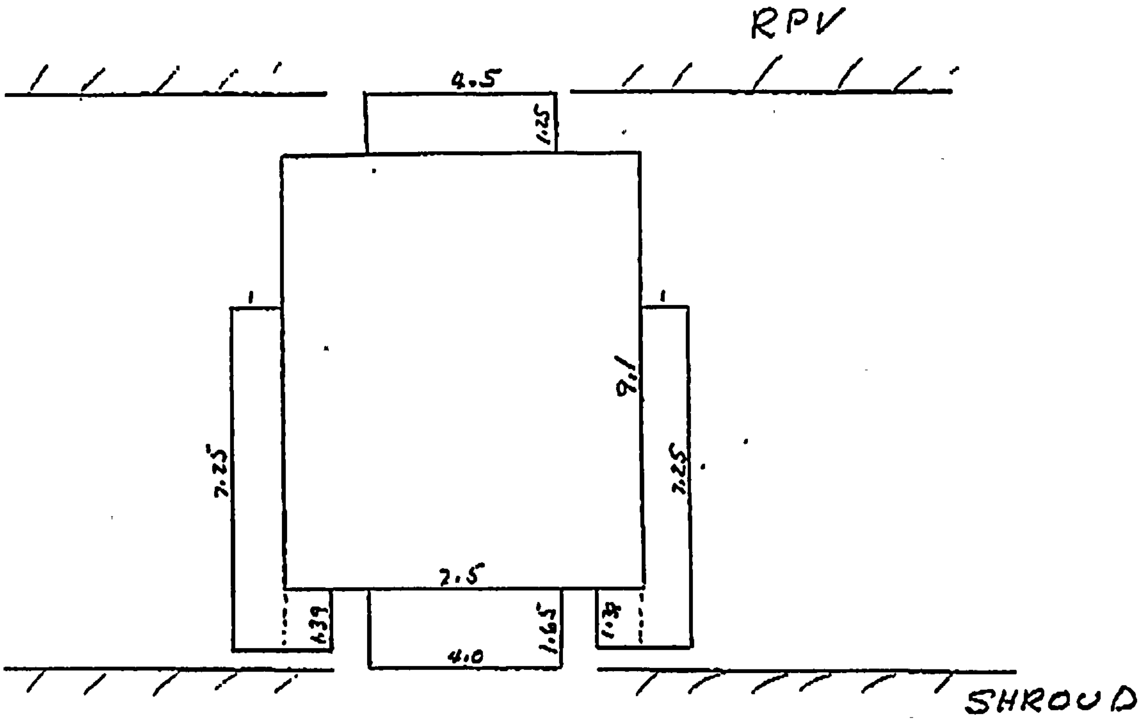
In the area of the lower spring, the flow is restricted by both the lower spring and the C-spring. The lower spring is 4.0 inches wide and extends through the entire annulus. The C-spring is a rectangle 4.2 in. by 4.45 in.

Spring area = 4 x [(4 x 17) + (4.2 x 4.45)] = 346.8 $in.^2$

Per cent of flow area = 346.8 x 100 / 10,467.8 = 3.3%

Flow velocity with stabilizers = 58,294 / (10,467.8 - 346.8) = 5.8 ft/sec

PLAN VIEW OF UPPER SUPPORT + SPRING



AREA:

(4.5)(1.25) =	5.63	IN²
(7.5)(9.1) =	68.25	IN²
(4)(1.65) =	6.60	IN²
2(7.5)(1) =	15.00	IN²
2(1.39)(1) =	2.78	IN²
	98.26	IN²

REF. DWGS: 112D6561 – UPPER SUPPORT
112D6562 – SUPPORT
112D6563 – UPPER SPRING
112D6588 – TOP SUPPORT

ATTACHMENT 6

GE Nuclear Energy

Engineering Services Verification Cover Sheet
(Reference EOP 42-6.00 and EOP 25-6.00) Sheet 1 of ___

DRF NO./SECTION B13 - 01739 - 06

Subject: Nine Mile Point 1, Shroud Repair Program - Core Spray Piping Analysis

1 **VERIFICATION REQUIREMENT** Designated Verifier Tom Tsai

1A APPLICATION. (System/Project/Program)

1B METHOD OF VERIFICATION. Checking, Alternate Calc. Indiv. Design Review, Team Design Review, Test (circle as needed); Other (describe)

1C SCOPE. (Identify what is to be verified (e.g., level of detail).
To verify the mathematical modeling for the Core Spray piping system and the accuracy of seismic movement inputs for computation.

1D INPUTS. Identify any GE and external interfaces and requirements, assumptions, input documents, test, analyses, reasons for changes.
GE Design Drawing, 706E231 and 706E234

1E OUTPUTS. Identify output document(s) or analysis results to be verified.
SAP4G07 computer analyses, snumb 2689U and 2680U

1F Responsible Engineer Chung J. Lung [signature] Date 1/26/95 Comp. 771
(Print name and Sign)

2 **INDEPENDENT VERIFICATION**

Comments: X NO ___ YES (See Attached)

2A VERIFICATION STATEMENT: The method and scope of verification as stated in 1B and 1C are appropriate. The inputs as identified in 1D. All comments and technical issues are resolved. The verification establishes that the output identified in 1E is correct and is adequate for its intended application as identified in 1A.

2B Independent Verifier Tom Tsai [signature] Date 1/26/95 Comp. 771
(Print name and Sign)

3 **APPROVAL OF VERIFICATION**

3A MANAGER'S APPROVAL: All design requirements have been identified and all technical issues are adequately resolved. The verification described in the above sections 1 and 2 is sufficient to issue/apply the results.

3B Resp. Manager or Delegate: Thomas Gleason Date ___ Comp. ___
(Print name and Sign)

4A Are there attached sheets for 1A, 1B, 1C, 1D, 1E 2A, 3A? (circle as applicable)

SD-056B (1/92)

GE NUCLEAR ENERGY
175 Curtner Avenue
San Jose, Ca. 95125

cc: Peter Walier

To: Thomas Gleason

From: Chung J. Lung C.L.

Date: Jan. 26, 1995

Subject: Nine Mile Point 1 - Shroud Repair Program, Core Spray Piping Analysis for the Seismic movement.

References:

1. GE-NE-B13-01739-03, "Nine Point 1 Nuclear Power Station, Seismic Analysis" Rev.0, Dec.16, 1994
2. GE Design Drawing, Shroud Core Structure, 706E231, Rev.3, 1969 and 706E234, Rev.5, 1969
3. SAP4G07 Users Manual, NEDO-10909, Rev.7, 1979

Attachment:

1. Nine Mile Point 1, Core Spray Piping Analysis Model
2. Piping Stress Analysis, Load Case #1 - combined radial and vertical seismic movement, computer run 2689U and Micro Fiche #0278
3. Piping Stress Analysis, Load Case #2 - combined tangential and vertical seismic movement, computer run 2690U and Micro Fiche #0280

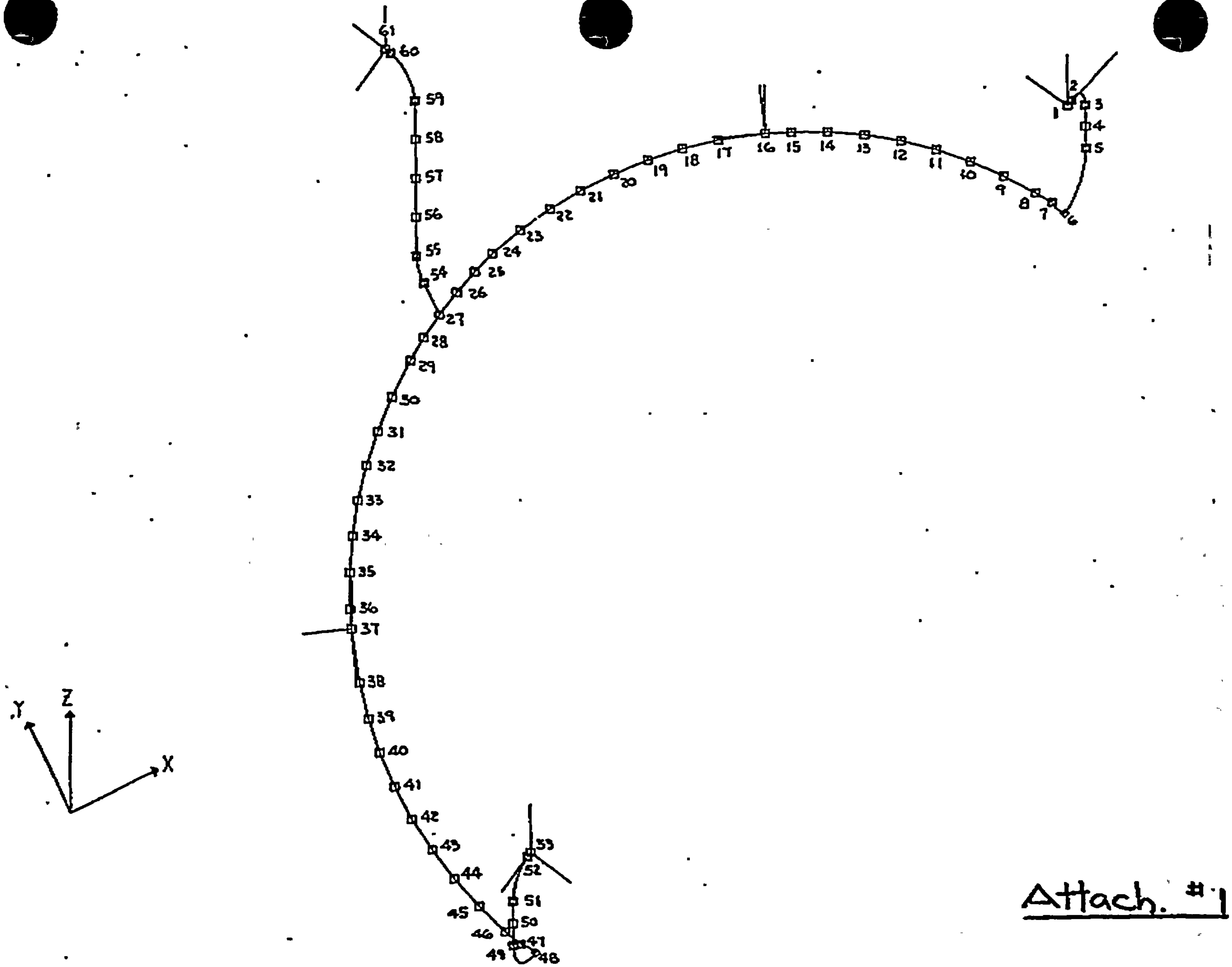
The Core Spray piping has been analyzed for the shroud movement in a postulated OBE event. Maximum shroud displacements in the horizontal and vertical directions were obtained from a separate seismic analysis; 0.904 in. for the horizontal and 0.610 in. for the vertical (Ref. 1). With the piping fixed at the RPV nozzle, the horizontal displacement was applied at the shroud nozzles either in the radial or in the tangential direction with the combined movement of the vertical displacement. For the bracket support on the shroud wall, it was also subjected to the shroud movement. Since the bracket support was designed as a guide, the piping could move freely in the tangential direction when the piping expanded, i.e. the bracket support would react to the shroud movement only in the radial and vertical directions.

There are two Core Spray pipings, only the one routed from 152 degree to 330 degree was modeled (Ref. 2). Since this Core Spray piping has a lower shroud nozzle or a shorter riser length than the other, this piping model analysis would govern. The analysis was performed using the computer code SAP4G07 (Ref. 3), and two load cases were analyzed;

(1) Combined radial and vertical movement,
(2) Combined tangential and vertical movement.

The table below is a summary of the results showing the maximum_strain in the piping and nozzles and the bracket loads.

Component	Strain, %	Location (node)	Load Case
Piping	0.80 0.45 0.42 0.26	elbow (59) tee (27) elbow (59) tee (27)	(1) (1) (2) (2)
RPV Nozzle	0.15	@ 242° (60)	(1)
Shroud Nozzle	0.08	@ 330° (1)	(1)
Bracket Load	6005 lb (radial) 8464 lb (vertical)	@ 290° (16)	(1)



NINE MILE POINT, CORE SPRAY PIPING
ANGLES 25. 115. 90.

14

ATTACHMENT 2

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Licensing Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary document 383HA718, Revision 0, *Thermal Cycles, Reactor Vessel and Nozzle - Description, Bases and Assumptions*, (GE Company Proprietary Information), issued February 23, 1973. The proprietary information is delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed bases, assumptions, methods and processes which GE has developed and applied to the initial BWR reactor designs and thousands of subsequent evaluations of the thermal cycles of the reactor vessel and nozzles for many plants and applications. The application for which this document is presently being requested is for a hardware design modification (stabilizers for the shroud

horizontal welds) intended to be installed in a reactor to resolve the reactor pressure vessel core shroud weld cracking concern. The development and approval of this design modification utilized systems, components, and models and computer codes that were developed at a significant cost to GE, on the order of several hundred thousand dollars.

The development of the design methods and evaluation process contained in the paragraph (2) document along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset, on the order of several million dollars.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 22nd day of February 1995.

George B. Stramback
General Electric Company

Subscribed and sworn before me this 22nd day of February 1995.

PAULA F. HUSSEY
COMM. #1046120
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires DEC 1,1998

Paula F. Hussey
Notary Public, State of California