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Document Control Branch (Document Control Desk)

See Rpt.

SUBJECT: Forwards final version of proprietary GE-NE-B13-01739-04, "NMP Unit 1 Shroud Repair Hardware Stress Analysis" & GE-NE-B13-01739-05,rev 1, "...Safety Evaluation GE Core Shroud..." in response to GL 94-03.Repts withheld.

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NIAGARA MOHAWK

NINE MILE POINT NUCLEAR STATION/P.O. BOX 63, LYCOMING, NEW YORK 13093/TELEPHONE (315) 343-2110

January 23, 1995
NMP1L 0894

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: *Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors" (TAC No. M90102)*

Gentlemen:

Niagara Mohawk's letter dated January 6, 1995, provided the Commission the Nine Mile Point Unit 1 Reactor Core Shroud Repair Design Summary (Enclosure 1) and supporting documentation (Enclosures 2 and 3). Enclosure 2 of our letter included a preliminary version of GE-NE-B13-01739-04, Nine Mile Point Unit 1 Shroud Repair Hardware Stress Analysis, and indicated that a final version of the analysis would be provided to the Commission by January 21, 1995. The purpose of this letter is, in part, to provide to you the final version of GE-NE-B13-01739-04.

Also, final verification of the shroud repair supporting analyses resulted in minor numerical changes to the calculated shroud displacements and calculated leakage values through the machined shroud holes. These changes have resulted in revisions to our Core Shroud Repair Design Summary and GE-NE-B13-01739-05, which were submitted in our January 6, 1995 letter. These documents are being re-submitted with revisions indicated by "bars" in the left hand margin. In addition, revisions have been made to documents 25A5583 and FDI 0245-90800 and to several drawings included in Enclosure 2 and Enclosure 3, respectively, of our January 6, 1995 submittal. These revised documents are also being re-submitted for your review.

Certain supporting documentation 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 these documents from public disclosure in accordance with 10CFR2.790(b)(1). Affidavits executed by General Electric detailing the reasons for the request to withhold the proprietary

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information have been included. Non-proprietary versions of the subject documents will be submitted to the Commission by January 31, 1995.

Very truly yours,

C. D. Terry
Vice President - Nuclear Engineering

CDT/JMT/kab
Enclosure

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

General Electric Company

AFFIDAVIT

I, **David J. Robare,** being duly sworn, depose and state as follows:

(1) I am Manager, ALMR Project Management, 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 report GENE-B13-01739-04, *NMP1 Shroud Repair Hardware Stress Analysis,* Revision 0, Class III (GE Company Proprietary Information), dated December, 1994. 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;

The development and approval of the BWR Shroud Repair Program was achieved at a significant cost, on the order of one million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(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)

David J. Robare, 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 18 TH day of JANUARY 1995.

David J. Robare
General Electric Company

Subscribed and sworn before me this 18th day of January 1995.

MARY L. KENDALL
COMM. # 987864
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires MAR 26, 1997

Mary L. Kendall
Notary Public, State of California

General Electric Company

AFFIDAVIT

I, David J. Robare, being duly sworn, depose and state as follows:

(1) I am Manager, ALMR Project Management, 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 report GENE-B13-01739-05, *Safety Evaluation for Installation of Stabilizers on the NMP1 Core Shroud,* Revision 1, Class III (GE Company Proprietary Information), dated January, 1995. 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).

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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 results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of the core shroud repair for the BWR.

The development and approval of the BWR Shroud Repair Program was achieved at a significant cost, on the order of one million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(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.

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David J. Robare, 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 18 TH day of JANUARY 1995.

David J. Robare
General Electric Company

Subscribed and sworn before me this 18th day of January 1995.

MARY L. KENDALL
COMM. # 987864
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires MAR 26, 1997

Mary L. Kendall
Notary Public, State of California

ENCLOSURE 1

Nine Mile Point Nuclear Station - Unit 1
Reactor Core Shroud Repair Design Summary

1.0 PURPOSE

This enclosure provides a summary of Niagara Mohawk Power Corporation's (NMPC) design details for the permanent repair of the 304 stainless steel circumferential welds for the Nine Mile Point Unit 1 (NMP1) reactor core shroud. This design is being submitted for review and approval by the NRC staff.

1.1 Background and Scope

Cracks have been observed in the core shrouds of several BWRs. The NRC issued Generic Letter 94-03 which requires inspection and/or repair. The NMP1 shroud welds have not been examined using the currently required non-destructive examination (NDE) techniques. NMPC will inspect the shroud during the upcoming February 1995 refuel outage. Should the weld examination results show that shroud cracking is not acceptable for continued plant operation, a shroud repair will be implemented. NMPC may implement a preemptive repair of the shroud welds in lieu of NDE; however, this issue is still being evaluated by NMPC.

The reactor core shroud repair is designed to structurally replace shroud welds H1 through H8. Figure 1-1 depicts the NMP1 shroud welds. Welds H1 through H6B are all of the circumferential shroud welds. Weld H7 attaches the shroud to the forged stainless steel shroud support ring. Weld H8 is a bimetallic weld that attaches the stainless steel support ring to the Inconel core support cone.

The NMP1 shroud repair consists of two separate design features. Tie-rod assemblies combined with core plate wedges replace welds H1 through H7 and the upward vertical load carrying capability of weld H8. Separate H8 weld brackets replace the downward vertical load carrying capability of weld H8.

As previously mentioned, NMPC is currently evaluating options for repair and examination. The primary options currently under evaluation are:

1. Examine shroud welds H1 through H8 in accordance with the BWR VIP Inspection Criteria and install the tie-rod assemblies and/or the H8 weld brackets only if cracking is found to be unacceptable for continued plant operation.

2. Implement a preemptive repair of welds H1 through H7 in lieu of examination. Examine weld H8 in accordance with the BWR VIP Inspection Criteria and install the H8 weld brackets only if cracking is found to be unacceptable for continued operation.

1.2 Core Shroud Physical Description

The core shroud, as shown in Figure 1-2, is a Type 304 stainless steel cylinder which surrounds the core and provides a barrier to separate the upward flow of coolant through the core from the downcomer recirculation flow. The recirculation inlet and outlet plenums are separated by the shroud and the Inconel 600 shroud support cone. The shroud support cone is designed to sustain the differential expansion of the ferritic reactor vessel and the austenitic stainless steel shroud without high stresses. The shroud support cone sustains essentially all of the vertical weight of the core structure and the steam separator assembly, except for the interior fuel assembly weights which are transmitted to the guide tubes.

The principal design stresses produced in the shroud and shroud support cone are due to the differential upward/downward pressure loading on the core under normal/upset operating and accident conditions; deadweight loadings, thermal expansion and the vertical and horizontal thrusts developed on the core and core structure during an earthquake.

The core shroud supports the upper core grid (top guide) which provides lateral support and alignment at the top of the fuel assemblies contained in each grid opening. The shroud also supports the lower core grid (core plate) which provides lateral guidance for the bottom of the fuel assemblies.

1.3 Shroud Safety Design Basis

The reactor internals, of which the core shroud and shroud support cone are a part, have the following basic functions to assure the safety design basis is satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired:

- To limit deflections and deformation to assure that the Emergency Core Cooling Systems (ECCS) can perform their safety functions during anticipated operational occurrences and accidents.

- To maintain partitions between regions within the reactor vessel to provide correct coolant distribution for all normal plant operating modes.

- To provide positioning and support for the fuel assemblies, control rods, incore flux monitors, and other vessel internals and to ensure that normal control rod movement is not impaired.

The shroud repair stabilizer assemblies and H8 bracket supports are designed to maintain the above shroud functions in the event welds H1 through H8 are cracked 360° circumferentially through-wall.

1.4 Core Shroud Fabrication Details

Specific NMP1 shroud fabrication details were previously provided to the NRC in NMPC's initial response to Generic Letter 94-03 dated August 23, 1994.

2.0 EVALUATION

2.1 Scope of Modification Design

The NMP1 shroud modification is designed to provide an alternative load path for the Type 304 stainless steel circumferential welds (welds H1 through H7), and for the shroud support ring to shroud support plate weld (weld H8). The NMP1 shroud modification design therefore provides structural integrity for, and takes the place of, each of the circumferential welds H1 through H8 in the NMP1 core shroud, without taking credit for weld integrity.

If the H8 weld inspection results confirm weld structural integrity, NMPC may elect not to install the H8 support brackets. If the brackets are not installed, credit would be taken for the weld integrity of H8; thus the designed shroud repair would only provide an alternative load path for welds H1 through H7. The shroud repair design analyses envelope either of these two noted scenarios.

2.2 Shroud Stabilizer Design Description

The repair is designed in accordance with the criteria set forth in BWROG VIP Core Shroud Repair Design Criteria, Revision 1, September 12, 1994. The NMP1 stabilizer repair was designed by General Electric Nuclear Energy and is similar to the shroud stabilizer design installed in the Edwin I. Hatch Nuclear Plant Unit 1. The design of the NMP1 core shroud repair is illustrated in the attached drawings (Figures 2-1 through 2-5).

The design of the NMP1 core shroud stabilizers consists of four tie rod assemblies, six H8 bracket supports and four core plate wedges.

Tie Rod Assembly Description

Each tie rod assembly is axis-symmetrically located in the RPV annulus. Each tie rod assembly consists of a tie rod, upper support, upper spring, top support, middle support, lower lateral and axial springs, lower support with two toggle bolts, and other minor components. The ends of the tie rod assemblies are attached at the top to the upper shroud head flange and at the bottom to the Inconel shroud conical support. The shroud head is notched at four azimuth locations (eight notches) using electric discharge machining (EDM) to accommodate the installation of the upper stabilizer support. At the bottom, two holes are machined through the angled conical shroud support for attaching each tie rod assembly.

The tie rod assemblies are designed to prevent unacceptable lateral or vertical motion of the shroud shell sections, assuming complete failure (360° through-wall) of one or more of the circumferential shroud welds. Each cylindrical shell and ring section of

the shroud is prevented from unacceptable motion by the stabilizers. The functions of each tie rod assembly component are as follows:

- The tie rods serve to provide an alternative vertical load path from the upper support of the tie rod assembly through the shroud support cone. These tie rod assemblies maintain the alignment of the core shroud to the reactor vessel. The tie rods also prevent upward vertical displacement of the shroud if weld H8 were completely failed.

- The upper support bracket combined with the upper lateral spring is designed to restrain lateral movement of the shell between welds H1 and H2, the ring between H2 and H3 and the shell between H3 and H4.

- The lateral rigid support (limit stop) located at the midpoint of the tie rods is designed to restrain lateral movement of the shell between welds H4 and H5. The rigid support is also provided for the tie rod so that the tie rod's natural frequency will be higher than that of the forcing frequency due to flow induced vibration.

- The lower lateral spring contacts the shroud and the RPV and is designed to restrain lateral movement of the shell between welds H5 and H6, the ring between welds H6A and H6B and the shell between H6B and H7.

- The lower axial spring is designed to provide axial flexibility of the tie rods to accommodate postulated temperature transients.

- The lower support with toggle bolts is designed to provide an attachment of the tie rod assemblies to the shroud conical support and to minimize leakage between the RPV lower plenum inlet flow and the RPV annulus flow.

Core Plate Wedge Description

The shroud repair also consists of four core plate wedges (spacers) located in the annulus between the core support plate and the inside of the shroud. In the event that welds H6A and H6B failed, the wedges would provide a direct load path from the core plate to the shroud to help distribute the lateral loads occurring during a seismic event. The shroud cylinder at this location is restrained in the lateral direction by the lower tie rod lateral spring. The wedges are held in place by clamping against the existing angle brackets that position the existing shield blocks.

Weld H8 Support Bracket Description

The tie rod assemblies combined with the core plate wedges are designed to carry all of the design loads of welds H1 through H7 and the upward vertical loads of weld H8. However, additional support is required at the bottom of the shroud in the event of an H8 weld failure to prevent the downward displacement of the shroud. The shroud repair therefore includes six bracket supports located at azimuth locations between tie rod locations. Each of the six bracket supports consists of an upper and lower bracket. Four holes are machined through the shroud above weld H7. The upper bracket is attached to the shroud by the four holes using two toggle bolts and two shear keys. The lower bracket rests on the Inconel conical support and bears against the vessel wall, and is held in place by the ears of the upper bracket.

2.3 Design and Code Considerations

The design and code requirements for the shroud repair are specified in two separate documents. One document, "Shroud Repair Hardware Design Specification (25A5583)," defines the design and performance requirements for the stabilizers. The second document, "Shroud Repair Code Design Specification (25A5586)," defines the American Society of Mechanical Engineers (ASME) Code design requirements for the shroud repair.

The shroud stabilizer construction shall be performed in accordance with an ASME Section XI Replacement Program per the requirements of Article IWA-7000. The NMP1 core shroud was not designed to ASME Section III design criteria and thus it is not considered an ASME component. However, Section XI requires inservice inspection (ISI) of the core support structures. The required replacement program is different from most replacement programs, because the stabilizers are not a direct replacement. Instead, the structural functions of the shroud horizontal welds are replaced by new components. Any defects found in welds H1 through H8 are structurally acceptable after the installation of the shroud stabilizers.

Because the core shroud was not designed to ASME Section III, the core shroud stabilizers are not required to be designed to ASME Section III criteria. However, material properties for the stabilizers will be in accordance with ASME Section III, Appendices, 1989 Edition and the nomenclature for stress intensity used in the design will be the same as that used in ASME III, Subsection NB, 1986 Edition and ASME III, Subsection NG, 1983 Edition with Addenda through Summer 1984.

The shroud stabilizers shall meet or exceed the original construction requirements for the shroud. Design of the core shroud stabilizers meets the structural criteria as specified in the NMP1 Updated Final Safety Analysis Report (UFSAR). Loads and load combinations in the UFSAR that are applicable to the core shroud have been included in the design specifications for the modification. The stabilizers change the points of application of the forces applied to the RPV by the core shroud. These new force application points and force distributions were analyzed per the RPV original Code of Construction.

The shroud repair is designed for a design life of 25 years (the remaining design life of the plant, plus possible life extension beyond the current operating license), to include 20 Effective Full Power Years. All shroud repair hardware shall be designed so that it can be removed and replaced. This is to provide full access to the annulus area for possible future inspections and/or maintenance/repair activities that may prove necessary.

2.4 Materials and Fabrication Considerations

The NMP1 shroud stabilizers used materials and fabrication methods for manufacturing of the NMP1 modification assembly subcomponents as specified in the document entitled "Fabrication of Shroud Stabilizer (25A5584)." The materials and fabrication methods used for the NMP1 shroud repair are consistent with those materials used for the Hatch Unit 1 repair and are as follows:

- upper and lower springs, upper nuts, upper and lower brackets, lower bracket nuts and toggle bolts and the H8 brackets, toggle bolts and shear keys are to be fabricated from nickel-based (Ni-Cr-Fe) alloy X-750 which has been heat treated at 1975 ± 25°F, followed by air cooling at 1300°F and age hardening

- tie rods, core plate wedges and other remaining components in the assemblies are to be fabricated from either Type 316 or 316L austenitic stainless steels, heat treated at 1900 - 2100°F, followed by quenching in circulating water to a temperature below 400°F

Alloy X-750 was selected for the springs, nuts, and upper brackets due to its inherent high strength, its low coefficient of thermal expansion in comparison to that of Type 304 stainless steel (which was used for construction of the NMP1 core shroud), and its resistance to IGSCC when placed in service in typical BWR operating environments. The X-750 material is certified to American Society for Testing and Materials (ASTM) Standard B637, Grade UNS NO7750 material requirements. Alloy X-750 is a precipitation-hardened Inconel material which has been accepted for use in nuclear environments by ASME Code Section III. ASTM Standard B637 lists the chemistry and material property requirements for precipitation-hardened nickel-based alloy bars and forgings. ASTM Standard B637 is equivalent to ASME Specification SB637, and is acceptable as a basis for certifying alloy X-750. The modification components fabricated from Alloy X-750 have 0.030 inches of the material removed from the surface after final annealing or pickling treatment, in order to minimize surface conditions that increase the susceptibility of the material to intergranular attack (IGA).

Type 316L stainless steel selected for the tie rods and remaining components was procured with a carbon content of less than 0.020%. Types 316 and 316L are acceptable ASME Code Section III materials for use in nuclear environments. The low carbon content and solution heat treatment of the 316 materials lowers the degree of sensitization of the steels. However, all procured 316 or 316L materials will be required to be tested for sensitization in accordance with methods delineated in ASTM Standard A262, Procedures A or E. The procurement and test practices will provide high resistance to IGSCC. The tie rod threads were induction annealed after machining the threads to remove a possible cold work layer. It

should be noted that all pieces of the assemblies will be mechanically locked in place; no welding will be done to assemble the stabilizer hardware. This will lower the residual stresses in the stabilizer assembly components, and thereby produce higher resistance to IGSCC.

The selection of material and methods of fabricating these materials should minimize the susceptibility of these materials to IGSCC. The absence of welding in the modification design should also reduce the susceptibility of the tie rod assemblies to IGSCC.

2.5 Systems Evaluation

Niagara Mohawk evaluated the response of plant systems with the shroud stabilizers installed for the following loading conditions:

	EVENT	LOAD COMBINATIONS
1.	NORMAL OPERATION	Normal Pressure, Dead Weight, Thermal
2.	UPSET 1	Upset Pressure, Dead Weight, Upset Thermal
3.	UPSET 2	Upset Pressure, Dead Weight, OBE (=DBE)
4.	EMERGENCY 1	Normal Pressure, Dead Weight, DBE
5.	EMERGENCY 2	Steam Line LOCA, Dead Weight
6.	EMERGENCY 3	Exit Recirc. Line LOCA (includes asymmetric load), Dead Weight
7.	FAULTED 1	Steam Line LOCA, Dead Weight, DBE
8.	FAULTED 2	Inlet Recirc. Line LOCA, Dead Weight, DBE
9.	FAULTED 3	Exit Recirc. Line LOCA (includes asymmetric load), Dead Weight, DBE

The above load combinations are consistent with the plant licensing basis except that the DBE was conservatively combined with LOCA loads, which was not required by the licensing basis. Consistent with the plant licensing basis, the DBE, in lieu of an Operating Basis Earthquake (OBE), is combined with the upset pressure loads since the plant licensing basis does not define an OBE. The upset transients described in the NMP1 UFSAR were reviewed and the bounding upset thermal event was determined to be a transient wherein the annulus water temperature decreases to 300°F while the reactor inlet plenum fluid temperature remains at 545°F. This situation could occur with the loss of feedwater followed by restoring the feedwater flow, but without feedwater heating. This event results in the largest temperature difference between the shroud wall and the tie rod assemblies.

The pressure differences across the shroud support cone, core plate and shroud head for the above events are provided in the design specification. The accident pressure differences for the above events are consistent with the values listed in the UFSAR. General Electric is

currently performing NMP1 specific TRACG analyses. Preliminary results indicate that the LOCA loads due to a recirculation line break are significantly less than the UFSAR values.

Also evaluated were the expected core plate and top guide displacements during all of the above loading conditions with postulated through-wall cracking at different weld locations including cracking at all welds concurrently. The predicted deflections of the core plate and top guide during all transient and accident conditions listed above have been calculated by Niagara Mohawk to be within the allowables defined in the design specifications, and therefore, would have no impact on control rod insertion. These allowables are bounded by the allowables discussed in a GE report, GENE-771-44-0894 Rev. 2, "Justification for Allowable Displacements of the Core Plate and Top Guide Shroud Repair," dated November 16, 1994. Revision 2 of this report was recently revised to include the final results of CRD Performance Evaluation Testing and Driveline Misalignment (GE Report NEDC-32406, September 1994).

The NMP1 design specifies the maximum allowable permanent horizontal deflection of any point on the shroud adjacent to either the H2 or the H3 weld (i.e., the top guide support) shall be less than 2.1 inches divided by a minimum safety factor (SFmin), during all of the above load combinations. The maximum permanent horizontal deflection of any point on the shroud adjacent to either the H6A or H6B weld (i.e., core plate support) shall be less than 0.75 inches divided by SFmin for the above load combinations. The maximum transient elastic horizontal deflection during a seismic event adjacent to either the H6A or H6B weld shall be less than 1.68 inches divided by SFmin. The values of SFmin are 2.25 for normal and upset, 1.5 for emergency and 1.125 for faulted conditions, consistent with the above noted GE document.

The bounding load combinations for comparison of the top guide and core plate horizontal displacements to allowables were the (Upset 2) and (Faulted 1) cases. The horizontal displacements summarized below are within the allowable values, therefore, insertion of the control rods is assured.

		Displacement	Allowable
Upset 2:	Top Guide	0.25"	0.93"
	Core Plate	0.03"	0.75"
Faulted 1:	Top Guide	0.64"	1.87"
	Core Plate	0.36"	1.49"

The maximum horizontal permanent deflection of any part of the shroud other than the top guide support ring and the core plate support ring that is not directly supported by either the upper or lower radial springs is limited to approximately 0.75 inches by mechanical limit stops. These stops do not perform any function unless a section of the shroud, for example between H4 and H5, becomes disconnected and a combined LOCA plus seismic event occurs. If this unlikely scenario occurs, the stops will limit the horizontal displacement to approximately 0.75 inches, which is equal to one-half the shroud thickness. A displacement

equal to one-half of the shroud wall thickness results in minimal leakage from the core to the downcomer region because the shroud sections still overlap each other. The limit stops do not invalidate the linear seismic analysis discussed in Section 2.6 because very little mass is associated with any potentially disconnected and unsupported section of the shroud.

The allowable vertical displacement of the shroud was determined based on the attendant leakage through a crack during normal operation. This is discussed in detail later in this section. The allowable vertical displacement of the shroud was also determined by the allowable vertical displacement of the top guide during an accident to ensure the fuel overlaps the top guide and by ensuring that the core spray function is not impacted. The vertical displacement of the core support plate is limited by the control rod guide tubes to an acceptable value of approximately one-half inch.

The maximum vertical displacement occurs during the MSLB accident scenario. Based on peak differential pressures listed in the UFSAR, Niagara Mohawk determined that for approximately six seconds during which the LOCA loads exceed normal operating pressures, the tie rods will elastically stretch a maximum of 0.61 inches, assuming postulated through-wall shroud cracking. This vertical displacement is momentary, and the top of the shroud will return to rest on the lower portion. After the six-second lift, no significant shroud bypass will occur. Further, minor shroud bypass leakage is not considered safety significant during the MSLB accident because there is no loss of coolant from the lower vessel area, and the small vertical lift of 0.61 inches will not adversely impact the safety function of the core spray system or cause the top guide to exceed the top of the fuel.

Niagara Mohawk also performed a leakage flow evaluation for normal and upset pressure conditions. The hardware designed to repair the shroud with identified cracks for NMP1 requires the machining of several holes through the shroud head flange for the installation of the upper support. There are a total of eight holes. Each of these holes will have some clearance, which will allow a small amount of leakage flow to bypass the steam separation system. As part of the stabilizer design, the shroud support cone will have eight holes, which also allows a small amount of core flow leakage through the clearance between the holes and the mating bolts. As part of the H8 weld bracket design, the lower shroud will have 24 holes, which also allow a small amount of core flow leakage through the clearance between the holes and the mating bolts and shear keys. In addition, there are nine welds in the shroud that may develop cracks, either above or below the core plate elevation. These cracks present another leakage flow path for the core flow. During normal operation, thermal tightening of the tie rods prevents upward motion of the shroud and crack separation for all crack scenarios.

The shroud head leakage flow includes steam flow, which effectively increases the total carryunder in the downcomer by a maximum of about 0.02% at 100% rated power and 85 to 100% rated core flow. The carryunder from the separators is based on the applicable separator test data at the lower limit of the operating water level range. The combined effective carryunder from the separators and the shroud head leakage at 85 to 100% rated core flow is about 0.17%, and is bounded by the design value of 0.25%. The impact of the flow leakage along with the associated carryunder increase is considered below.

The impact of the leakage results in an overprediction of core flow by about 0.6% of core flow. This overprediction is small compared to the core flow measurement uncertainty of 5% for non-jet pump plants used in the Maximum Critical Power Ratio (MCPR) Safety Limit evaluations. Additionally, the decrease in core flow resulting from the overprediction results in only about 0.2% decrease in calculated MCPR. Therefore, it is concluded that the impact is not significant.

The computer code used to evaluate performance under plant anticipated abnormal transients and calculate fuel thermal margin includes carryunder as one of the inputs. The effect of the increased carryunder due to shroud repair leakage results in greater compressibility of the downcomer region and, hence, a reduced maximum vessel pressure. Since this is a favorable effect, the thermal limits are not impacted.

The limiting condition is the recirculation discharge line break. The severity of the limiting event is primarily determined by the core spray flow to the upper plenum region. The leakage through the shroud repair holes does not have an impact on the core spray flow or the cooling to the fuel rods or fuel channel. Therefore, the ECCS results are unchanged by the shroud leakage.

The increased carryunder due to shroud bracket-hole leakage results in an increase in the core inlet enthalpy by about 0.1 BTU/lb, compared with the no leakage condition. The combined impact of the reduced core inlet subcooling and the reduced core flow due to the leakage results in a minor effect (~1.2 days) on fuel cycle length and is considered negligible.

Niagara Mohawk has concluded that there is an insignificant impact of the leakage flows through the shroud repair holes and postulated shroud cracks on the steam separation system performance, core monitoring, fuel thermal margin, ECCS performance and fuel cycle length. The results show that at rated power and 85 to 100% rated core flow the leakage flow from the repair holes is equal to a maximum combined leakage of about 0.7% of core flow. This leakage flow is sufficiently small so that the steam separation system performance, core monitoring, fuel thermal margin and fuel cycle length remain adequate. Also, the impact on ECCS performance is insignificant, and hence, the licensing ECCS evaluation for the normal condition with no shroud leakage is applicable.

2.6 Structural Evaluation

The NMP1 shroud repair has been designed to both vertically and horizontally support the top guide, core support plate, and shroud head and to prevent core flow bypass to the annulus region. The shroud repair will support the fuel assemblies and maintain the correct fuel channel spacing to permit control rod insertion by limiting the displacement of the shroud under postulated accident scenarios.

Extensive stress analysis was performed for all of the shroud repair parts and affected reactor components. This analysis was divided into four separate parts for convenience: (i) the conical support (including the adjacent reactor vessel and weld H9), (ii) the shroud and tie rod assemblies, (iii) the repair hardware components, and (iv) the reactor vessel. Separate three-dimensional (3-D) finite element models were developed to evaluate items (i) and (ii),

and hand calculations using basic strength of materials formulas were used to evaluate item (iii). Each of these evaluations and the associated results are described briefly in the paragraphs which follow. The complete details of the evaluation can be found in GE report B13-01739-04, Revision B. It should be noted that some of the results given here are preliminary in the sense that design inputs are still being finalized; however, the conclusions resulting from the use of final design inputs are not expected to change from those shown here. The fourth part of the stress analysis addressed the new loads applied to the reactor vessel as a result of the installation of the shroud stabilizers. This analysis is contained in GE report 24A6426.

Conical Support Evaluation

A detailed finite element model of the shroud conical support was used to perform stress analysis of the conical support. The model consisted of a 90° vessel/conical support segment. A 90° segment (i.e., ¼ model) was utilized since the shroud repair consists of four nearly equally spaced tie rods. A portion of the reactor vessel was included in the model so that the appropriate interaction at the conical support junction could be accounted for. The vessel ends were modeled far enough away from the junction so that end effects were insignificant. The repair hardware connection was placed in the center of the model (i.e., at 45°) so that edge effects from the 0° and 90° planes were insignificant in the region of interest. The shroud support ring and the shroud were not included in the finite element model (i.e., the H8 weld was assumed to be completely failed). This configuration is conservative in that the conical support receives no additional support from the shroud support ring and bounds all other weld configurations.

This model was also used to evaluate the six H8 weld repair brackets. Although the model effectively considered four of these brackets (since the model was a ¼ model), the stress results demonstrated that the effects from one bracket do not influence the stresses at other bracket locations; therefore, the 90° model was deemed adequate for fully evaluating all loads imposed on the conical support by all of the shroud repair hardware and all vessel loads.

Stresses due to reactor pressure, lower shroud pressure drop (ΔP), thermal events, and all reactions from the shroud repair hardware were considered in the analysis. Deflection of the conical support was also evaluated from a safety standpoint and found to be acceptable.

Shroud and Tie Rods Evaluation

Several other detailed finite element models were developed to evaluate stresses in the shroud and repair hardware components. The first model consisted of a 180° shroud segment composed of shell, gap (representing cracks), 3-D truss, beam and spring elements. Repair spring and vertical tie rod assemblies were also included in this model as 3-D truss elements and lower brackets as 3-D beam elements representing the repair hardware global mechanical characteristics. A 180° segment was necessitated by the need to evaluate the non-symmetric loads. This model allowed for complete stress evaluation of the shroud.

Several other finite element models were constructed using 3-D solid elements for the spring and tie rod components to separately evaluate stresses in each of these components.

As with the conical support, stresses due to reactor pressure, shroud pressure drop (ΔP), thermal events, and all reactions from the shroud repair hardware were considered in the analysis and found to be acceptable. Horizontal deflection of the shroud at the top guide and core plate locations was also evaluated from a safety standpoint and found to be acceptable.

Repair Hardware Components Evaluation

Detailed hand calculations of the repair hardware components and the H8 weld repair brackets at locations not specifically covered by the above finite element models were performed for structural analysis purposes. The bounding loading conditions for the evaluation of stresses in the H8 brackets are the (Emergency #3) and (Faulted #2) loadcases. Stresses in the H8 brackets and the shroud were below allowable stresses for these bounding loadcases. The bounding loading conditions for the tie rod assemblies are the (Emergency #2) and (Faulted #1) loadcases. Stresses in the tie rod assemblies were below allowable stresses for these bounding loadcases.

Reactor Vessel Evaluation

The upper and lower lateral springs and the H8 brackets put new design mechanical loads into the reactor vessel. A finite element (FE) model of the reactor vessel shell was developed and the radial loads from the springs and H8 bracket supports were applied. All of the stress intensities due to the new design mechanical loads satisfy the allowable stress intensities of the original code of construction.

Seismic Analysis

The dynamic seismic analysis for the NMP1 shroud repair modification is documented in report GENE-B13-01739-03. The mathematical, beam element structural model used for the analysis includes the reactor building, shield wall/pedestal, RPV, reactor internals, and the repair modification hardware, all coupled. The model was analyzed using the SAP4G07 computer program.

The current licensing basis Design Basis Earthquake (DBE) was used in the analysis. A synthetic time history was generated based on the horizontal DBE spectra in accordance with the guidelines contained in the U.S. NRC Standard Review Plan, NUREG-0800. DBE results were combined with upset as well as emergency and faulted conditions. To add conservatism to the shroud repair design, the DBE loads were combined with the LOCA loads, although the plant licensing basis does not require this conservative load combination.

No detailed RPV and internals dynamic analysis documentation existed for NMP1 against which the results of the new detailed dynamic model could be benchmarked. The new coupled model was thus generated utilizing the available licensing basis data and analyzed and verified using the methodologies employed in modern plant designs. The licensing basis condition was, however, simulated by additionally analyzing the model without the shroud stabilizers and without any cracks, to form a new benchmark run. The resultant component loads based on the new shroud repair seismic analysis compared favorably with the component loads in the benchmark run.

The structural stiffness properties were calculated for the tie rods and the top and bottom springs, taking into consideration the local stiffness at the shroud head and the shroud conical support. The model being axisymmetric, the equivalent rotational stiffness offered by the tie rod system was incorporated into the model. The displacements of the shroud at the top guide and core plate elevations were calculated based on the total horizontal force and spring constant of each stabilizer spring. Only one spring was assumed to be effective at a time. The spring constant values of 24,000 lb./in at the top guide, and 336,000 lb./in at the core plate elevation were determined by finite element analysis.

A bounding combination of cracked/uncracked cases were analyzed. The cases analyzed bound the various hypothetical cracked scenarios, and yield maximum loads for the modification hardware design. The stabilizer design is based on the worst case scenario to ensure control rod insertion and safe shutdown of the reactor.

The NMP1 shroud repair includes six H8 weld brackets at the interface between the shroud and the conical shroud support skirt to support the shroud in the vertical direction against free fall, should the H8 weld fail completely. The seismic analysis evaluated the scenario where the H8 weld failed and the corresponding downward load on the H8 support bracket, due to the moment caused by the horizontal seismic motion, was taken into account.

For the Coupled model, an axisymmetric lumped mass model was developed for the seismic analysis. The model was constructed as an assemblage of lumped masses connected by massless beam elements and spring elements. In the "horizontal" model, only the horizontal translation and the corresponding rotational degrees-of-freedom were included. The structural properties of the various elements including the proposed structural modifications were incorporated in the model. Hydrodynamic masses were calculated and modeled in order to account for the dynamic coupling of the fluid mass with the solid mass.

For the Weld Crack model, analysis iterations were performed to reflect the scenarios wherein 360 degree through-wall, circumferential cracks were assumed at the various crack locations on the shroud.

For the analysis, the cracks were represented as hinges or rollers, depending upon the assumed crack condition and the loading event. In a given circumferentially cracked plane, the crack is assumed to resist only lateral shear, if no lifting or separation of the crack plane occurs. In such a case, the crack plane is modeled as a hinge. If vertical separation of the crack plane is assumed or anticipated, the resistance to both shear and moment are lost. In this case, the crack plane is modeled as a roller. The tie rods are preloaded with a pre-determined thermal load. This preload maintains a compressive clamping force on the shroud which would keep the cracked welds from separating during an upset condition event. Consequently, a hinged assumption is applicable for the upset condition. For the emergency or faulted condition event involving a LOCA, the possibility of the shroud lifting momentarily exists, which would cause a separation at a postulated crack. To represent such a scenario, the postulated crack was assumed as a roller.

For the case with all welds cracked, weld H1 was modeled as a roller; all other welds are modeled as hinges. H1 being the uppermost weld, the roller condition at H1 represents maximum crack separation due to LOCA upward pressure, and minimum downward

compressive (crack closure) load due to deadweight. All other welds were modeled as hinges since there is sufficient deadweight to maintain contact (crack closure) and offer shear resistance.

Vertical seismic inertia load was not analyzed using the computer model. Any potential vertical amplification during the short period of time when a portion of the shroud may tend to lift is judged to be small, since that portion of the shroud is only connected in the vertical direction to the remainder of the shroud with the tie rods. The tie rods cannot apply a vertical upward force on the lifted portion of the shroud. Thus, vertical excitation cannot be transferred from the unlifted shroud to the lifted portion of the shroud.

In the repaired condition (with the modification hardware in place), the uncracked case yielded the most governing response spectra. These spectra was compared with the spectra generated using the benchmark model (without modification hardware and cracks) which showed that the spectra were almost identical, demonstrating the insignificant impact of the repair modification on the piping/RPV interface seismic loads.

The repair hardware was designed for the potential for vibration, and to keep the vibration to a minimum. The natural frequency of the repaired shroud, including the repair hardware, has been determined. The usage factor due to cyclic stresses caused by vibration will be less than 1.0 for the design life of the repair hardware.

Flow Induced Vibration Analysis

The potential for flow induced vibration has been evaluated by calculating the lowest natural frequency of the tie rods and the highest vortex shedding frequency due to the water in the downcomer. The tie rods are 3.5 inches in diameter and 136.6 inches long. The tie rods are threaded on both ends. One end is connected with a nut to a support assembly and the other end is threaded to an axial spring member. The spring member is anchored to the reactor vessel support cone by a pin and clevis arrangement. The assembly is thermally preloaded to 79,670 lb. A mid-span support is included which reduces the effective length of the tie rod. The calculated lowest natural frequency of the assembly is 28 Hz. The potential excitation forces come from the water flow and from the shroud which has a natural frequency much lower than the stabilizer assembly.

The stabilizer assemblies are located in the annulus between the shroud and vessel at approximately 90°, 170°, 270°, and 350° degree locations. The flow in this region is primarily parallel to the tie rods. The maximum axial flow in the annulus at 105% rated core flow is calculated to be 5.8ft/sec. The maximum cross radial flow occurs at the inlet to the recirculation nozzles which flair out in the vessel ID to approximately 40 inches in diameter. The flow velocity at this diameter is 9.3 ft/sec. Although there is no stabilizer assembly at this location, the vortex shedding frequency for this flow velocity is only 7 Hz. This is well below the 28 Hz lowest natural frequency of the stabilizer assembly. This combination satisfies the standard GE design goal of a factor of three between excitation frequency and lowest natural frequency.

2.7 Pre-modification and Post-modification Inspection Requirements

The requirements for pre-modification and post-modification inspection of the tie rod assemblies are given in General Electric Field Disposition Instruction (FDI) No. 0245-90800. FDI No. 0245-90800 Section IV., Step 1.0 will require that field examiners perform a VT-1 examination of the accessible areas of the RPV wall, shroud support cone and weld H9, adjacent to the attachment point for the shroud stabilizer lower support and H8 weld brackets.

FDI No. 0245-90800 Section IV., Step 4.0 requires that field examiners perform a VT-1 examination of the completed modification. The post-modification inspections for the tie rod assemblies will include VT-1 examinations of all the clevis pins used in the modifications, each core plate wedge assembly; each stabilizer assembly in contact between the RPV wall and the upper contact, mid-support and lower contacts, each stabilizer assembly in contact between the shroud and the upper support and lower spring and each jam nut on each of the eight toggle bolt assemblies to verify crimping. The post-modification inspections for the H8 brackets will include VT-1 examinations of the foot of the upper bracket of the H8 weld assembly to confirm that it is resting or contacting the lower bracket and each jam nut on each of the six upper bracket assemblies to verify crimping has occurred. All VT-1 examinations will be accomplished using a television camera which is capable of resolving a 0.001 inch diameter wire on a neutral gray background, and from a distance and with light that has been demonstrated capable of detecting IGSCC.

Niagara Mohawk will augment their Inservice Inspection programs to include examination of the repair/modification designs (i.e., as stated in Section 2.2.7 of staff SER "Safety Evaluation on Boiling Water Reactor (BWR) Core Shroud Repair Design Criteria," which was issued to the BWR VIP Repair Technical Subcommittee on September 29, 1994). Niagara Mohawk will submit its plans for augmented inspections within 90 days following completion of NMP1's 1995 refueling outage.

3.0 Conclusion

Based on Niagara Mohawk's review of the shroud modification hardware from design, code reconciliation, materials, fabrication, structural, systems, installation and inspection considerations, as discussed above, Niagara Mohawk concludes that the proposed modification is in accordance with the BWR VIP Core Shroud Repair Design Criteria dated September 12, 1994 and the NRC Safety Evaluation on BWR VIP Core Shroud Repair Design Criteria. Niagara Mohawk has also concluded that the NMP1 shroud repair is consistent, where applicable, to the previous NRC accepted Hatch Unit 1 installed shroud repair. Therefore, the proposed modification provides an acceptable level of quality and safety as required by 10CFR50.55a(a)3 and is acceptable for installation in the NMP1 Reactor Pressure Vessel.