CATEGORY 2

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9704100242 DOC.DATE: 97/04/08 NOTARIZED: YES DOCKET #
FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
AUTH.NAME AUTHOR AFFILIATION
MCCORMICK,M.J. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

See Reports

SUBJECT: Forwards proprietary & non-proprietary repts from GE re GL 94-03, "Intergranular Stress Corrosion Cracking in BWRs." List of repts,encl.Encls withheld,per 10CFR2.790(b)(i).

DISTRIBUTION CODE: AP01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 41+62
TITLE: Proprietary Review Distribution - Pre Operating License & Operating R

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-1 LA	1	1	PD1-1 PD	1	1
	HOOD,D	1	1			
INTERNAL:	ACRS	1	1	FILE CENTER 01	1	1
	OGC/HDS3	1	0			
EXTERNAL:	NRC PDR	1	~~0~~ 1 w/o prop			

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

TOTAL NUMBER OF COPIES REQUIRED: LTTR 7 ENCL ~~5~~ 6

C A T E G O R Y 2 D O C U M E N T

NIAGARA MOHAWK

GENERATION BUSINESS GROUP

NINE MILE POINT NUCLEAR STATION/LAKE ROAD, P.O. BOX 63, LYCOMING, NEW YORK 13093/TELEPHONE (315) 349-2660
FAX (315) 349-2605

MARTIN J. McCORMICK JR. P.E.
Vice President
Nuclear Engineering

April 8, 1997
NMP1L 1200

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

RE: Nine Mile Point Unit 1
Docket 50-220
DPR-63

Subject: *Generic Letter 94-03 "Intergranular Stress Corrosion Cracking (IGSCC) in Boiling Water Reactors"*

Gentlemen:

By letters dated January 6, 1995 and January 23, 1995, Niagara Mohawk Power Corporation (NMPC) submitted an application for repairs to the Nine Mile Point Unit 1 (NMP1) core shroud. The shroud repairs and use of stabilizer assemblies (tie rods) were submitted as an alternate to the requirements of the ASME Code, Section XI, as allowed by 10CFR50.55a (a)(3)(i). The staff provided approval of the proposed alternate repair by letter dated March 31, 1995. The approval letter and attached safety evaluation required NMPC to submit re-inspection plans for the shroud and repair assemblies prior to the next refueling outage planned for 1997. By letter dated February 7, 1997, NMPC submitted plans for re-inspection of the core shroud vertical welds and repair assemblies in accordance with the criteria provided by the "BWR Vessel and Internals Program" (BWRVIP) document BWRVIP-07.

During the 1997 refueling outage, NMPC conducted core shroud vertical weld inspections per the approved documents and observed vertical weld cracking which exceeded the screening criteria. Additionally, inspections of the four tie rod assemblies found the tie rod nuts to have lost some preload and identified damage to the lower wedge retainer clips on three tie rods. Further details of the as found conditions are provided in Enclosures 1 and 2.

A001

By phone calls on March 20, 1997 and April 2, 1997, NMPC informed the staff of the inspection findings and indicated that analysis of the vertical weld cracking and restoration plan of the shroud tie rod assemblies would be submitted to the NRC prior to restart of the unit. This letter and the attached enclosures provide root cause, corrective actions and the final design documentation which establishes the acceptability of the as found vertical weld

cracking for a minimum of 10,600 operating hours (above 200°F), determines an appropriate weld re-inspection schedule, provides details of the actions taken to restore the tie rods to the as designed condition and describes a modification of the lower wedge retainer clip design.

The modified lower wedge retainer clips are part of the tie rod assemblies which, as noted above, are not included under the ASME Code Section XI definition for repair or replacement. As such, the design details of the modified retainer clips are being submitted to the staff for review and approval as an alternative repair pursuant to 10CFR50.55a (a)(3)(i). The enclosed analyses provide justification for continued operation of NMP1 during the upcoming cycle utilizing the updated 10CFR50.55a approval as proposed herein.

Enclosures 1, 2 and 5 are considered by their preparer, General Electric (GE), to contain proprietary information exempt from disclosure pursuant to 10CFR2.790. Therefore, on behalf of GE, NMPC hereby makes application to withhold these documents from public disclosure in accordance with 10CFR2.790 (b)(1). An affidavit executed by GE detailing the reasons for the request to withhold the proprietary information has been included in Enclosure 7. A non-proprietary version of these documents has been included with this letter as Enclosure 8.

SUMMARY OF INSPECTION FINDINGS AND ANALYSES

I. Core Shroud

A. *Core Shroud Inspections Performed*

The NMP1 core shroud has four GE core shroud stabilizer assemblies installed. These assemblies were installed during the RFO-13 (1995) refueling outage. The installation was done as a pre-emptive repair of the core shroud horizontal welds H1 through H7 in lieu of baseline shroud inspection of these horizontal welds. The GE shroud stabilizer design requires vertical weld integrity in order for the shroud stabilizers to satisfy the design basis assumption of horizontal welds H1 through H7 being through wall cracked 360°. The pre- and post-shroud repair installation inspection scope during RFO-13, included a sample inspection of the vertical welds at the intersection of a selected high fluence weld (the H5 weld). The inspection included 6 inches above and below the H5 location along the V9, V10, V11 and V12 welds. The inspection was an enhanced visual examination performed from the inside diameter (ID). This visual examination was intended as a sample inspection. This inspection scope was approved by the NRC as part of the safety evaluation report (SER) issued for the NMP1 core shroud stabilizer design.

The inspection of the NMP1 vertical welds in the current refueling outage (RFO-14) was performed consistent with the BWRVIP-07 guidelines for the reinspection of BWR core shrouds. These guidelines also utilized a sampling

approach for the vertical core shroud welds. The option selected by NMPC was to complete a visual inspection of 25% of the equivalent total vertical weld length from either the outside diameter (OD) or ID. As part of the inspection plan, GE defined screening criterion for minimum required uncracked vertical welds on a per weld basis. The ring segment welds were excluded from the vertical welds requiring inspection based on GE analysis of the ring segment welds submitted to the staff for review by letter dated February 7, 1997. As a result of inspection findings, the inspection scope was expanded using an enhanced visual inspection method supplemented by ultrasonic inspection (UT).

B. *Core Shroud Inspection Results*

The initial RFO-14 inspection of the vertical welds identified cracking over the entire OD length of the V10 weld using enhanced visual inspection techniques. The inspection plans were then expanded to establish minimum required uncracked ligament on the vertical welds which are required to meet the shroud stabilizer repair design basis assumptions. The vertical weld cracking evident on the OD of both the V9 and V10 welds was extensive. The extent of cracking identified on the OD had not previously been identified at other BWRs. As a result, a complete baseline inspection of the NMP1 accessible portions of certain core shroud horizontal and vertical welds was performed in order to establish an overall material condition assessment of the NMP1 core shroud. Detailed descriptions of both vertical and horizontal welds cracking is provided in Enclosure 1. The individual inspection results have received N.D.E. Level III review by GE and NMPC personnel. The documentation of inspection results is being compiled for final quality assurance review. This review will be completed by April 20, 1997.

C. *Root Cause of Vertical Weld Cracking*

This shroud baseline inspection has enabled NMPC to establish that the cracking at the vertical welds V9 and V10 is consistent with the expected IGSCC cracking of BWR core shrouds. Both the horizontal weld cracking in the beltline H4 weld and the vertical weld cracking in the beltline V9 and V10 welds is occurring in the heat affected zone (HAZ) of the welds. The assessment of the IGSCC cracking is included in enclosed analyses and reports. Several independent evaluations were also performed for NMPC to obtain an accurate assessment of the cause and acceptability of vertical weld cracking.

These evaluations have concluded that the cracking noted on the vertical welds V9 and V10 is IGSCC. The stresses that cause cracking in the vertical welds are weld residual and fabrication stresses and to a lesser extent the stress resulting from internal pressure (hoop stress). The NMP1 shroud horizontal and vertical welds are clearly susceptible to IGSCC. The high carbon Type 304

stainless steel material was initially sensitized by the welding process. The material's susceptibility was further enhanced by surface cold work and surface strains from the fabrication process. Irradiation would also add to the susceptibility over the operating time. Finally, the tensile surface residual stresses and surface fabrication stresses led to the IGSCC initiation.

The inspection data from UT of these welds has established the cracking depth. The pattern of crack depth is consistent with the calculated fluence axial and radial profiles. The estimated fluence for these welds is in the 2 to 4.5 x 10^{20} n/cm^2 (>1 MEV). This fluence places these welds in a range for which the radiation enhanced IGSCC conditions exist. The evaluations performed have concluded that the observed cracking is associated either with weld HAZ or sites where fabrication related welding or grinding was apparent. The overall conclusion is that this cracking is not unique and can be attributed to welding residual stresses and fabrication fit up induced stresses.

D. *Relationship of Vertical to Horizontal Weld Cracking*

The baseline inspection has identified one location at the intersection of H5 and V9 where a horizontal crack in the HAZ of H5 has linked with a vertical crack in the HAZ of V9. This case is isolated and has not been identified in other locations. In fact, the majority of the cracking appears to start approximately 6 to 10 inches down from the horizontal H4 weld HAZ. The shroud horizontal and vertical weld baseline inspection of the NMP1 core shroud which has been performed provides a point of reference for future sample inspection of the core shroud. This baseline and future sample inspections will allow NMPC to monitor the actual IGSCC crack growth rate which will be used to maintain the required design basis margins.

GE has completed analyses regarding the potential impact the core shroud stabilizer assemblies could have on vertical weld cracking. The results have shown that any hoop stress induced at the vertical welds due to shroud stabilizer thermal preload is negligible. The overall conclusion is that the shroud stabilizers had no effect on the shroud vertical weld cracking identified at V9 and V10.

The vertical weld 9 and V10 cracking was reviewed by independent experts in IGSCC cracking of BWR core shrouds. Enclosure 3 contains the results of a qualitative assessment of the visually observed cracking on the H4, V9, V10 and H5 welds. This evaluation has concluded that the IGSCC cracking is similar in nature to the cracks seen in other BWRs and that the specific conditions for the particular cracking patterns can be explained by normal fabrication practices used in manufacturing the core shroud. In an effort to better define how these fabrication processes can explain the cracking, detailed finite element modeling have been performed. Overall the results show that the

welding and fabrication process can explain the cracking pattern observed on the vertical welds. These analyses calculated through-thickness stress intensity solutions and crack growth studies. The results clearly support the bounding analysis approach being used to define the proposed operating interval between inspections.

E. *Core Shroud Vertical Weld Analytical Approach for Acceptance*

An analysis of the vertical welds used to define the proposed shroud vertical weld reinspection interval has been performed consistent with approved BWRVIP shroud analysis methods. The criteria applied are those set forth in the BWRVIP core shroud inspection and evaluation document. The approach being applied for the vertical welds analysis assumed that all horizontal welds are cracked 360° through wall consistent with the core shroud stabilizer design basis. The assumption of horizontal weld 360° cracking requires sufficient vertical weld integrity to ensure that the design basis assumption of stacked right cylinders is maintained. The analysis approach relies upon sizing of the through wall vertical weld cracking with UT. These through thickness cracks have been analyzed consistent with the BWRVIP core shroud inspection and evaluation guidelines accounting for ASME Code Section XI safety factors, design basis loads, inspection uncertainty consistent with the BWRVIP-03 guidelines, and the currently bounding NRC core shroud crack growth assumption of 5×10^{-5} inches/hr. Based on these assumptions, the required core shroud re-inspection interval has been determined to be at least 10,600 operating hours as described in Enclosure 1.

The attached analysis of the vertical welds includes an assessment of the potential leakage from postulated through wall vertical cracking. The overall thermal hydraulics assessment has concluded that the leakage would be negligible. The overall conclusion is that this leakage has no impact on the design basis for normal upset or accident conditions. The attached Enclosure 1 provides the required detailed discussion on this subject.

In conclusion, the vertical weld cracking condition has been reviewed and been determined to not represent an unreviewed safety question based on applying the NRC approved core shroud inspection and evaluation guidelines. These guidelines provide the analysis basis to define an acceptable inspection interval based on as found IGSCC cracking of core shrouds. The required interval established by the attached analyses is 10,600 hours of operation.

II. Core Shroud Stabilizer Assemblies (Tie Rods)

A. *Tie Rod Inspections Performed*

During the current refueling outage, post-operational inspections were conducted on the core shroud stabilizer (tie rod) assemblies. Tie rod deficiencies were found, including improper as found torque on the tie rod nuts, and damage to the retainer clips on the lower spring wedges. These findings resulted in root cause evaluations and additional inspections and testing of the tie rods.

B. *Tie Rod Inspection Findings*

Enclosure 2 contains the detailed data on the as-found condition, root cause of those deficiencies, validation of the root cause and corrective actions taken. Gaps were identified on the clevis pin to lower support hook contact and under the tie rod nut to top support contact. It was determined that preload of the tie rods had been lost, to some degree, on each tie rod. Also, the lower spring wedge retainer clip was broken at the 90° tie rod location and visibly damaged at the 270° and 350° tie rod locations. The 90° tie rod lower spring wedge was found bottomed on its guide rod, not in contact with the vessel as originally installed. The remaining contact points, springs and retainer clips were found in their proper positions.

C. *Root Cause of Tie Rod Degradation*

The root cause for the tie rod degradation is attributed to recognition that the tie rod design did not consider the effect of installation tolerances for the lower support bolt holes. Because of this, the installation procedures did not contain specific criteria for the location of the toggle bolts during installation of the lower support. The lower support toggle bolts are nominally 4.000" in diameter. The measured electric discharge machining (EDM) holes in the shroud cone ranged from 4.090" to 4.110". Since the position of the lower support bolts within the machined holes was not procedurally controlled during installation, the relative position of the bolts within the holes was variable.

During heatup, the expansion of the shroud and tie rods generates a force sufficient enough to overcome the installed friction forces and move the lower support up the shroud cone. This translates into a vertical movement of the tie rod. This movement was sufficient to apply a load on the lower spring wedge retainer clip such that it failed within one cycle of operation. Additionally, the lower spring wedge retainer clip was not designed to accommodate differential movement given the frictional loads between the vessel wall and the lower spring wedge during normal and transient conditions.

D. *Corrective Actions*

Subsequent to these findings and root cause evaluation, an installation procedure was developed to restore the tie rods to their original design basis condition. Each tie rod was jacked at three locations during tie rod nut torquing to remove any gaps associated with installation tolerances. Jacks were placed under the lower support, on the vessel side of the lower support to push it up the shroud cone to remove the clearances between the toggle bolts and the shroud side of the cone holes.

Following performance of the revised installation procedure inspections were completed on each tie rod to verify the absence of gaps, proper contact and position. As a result of these inspections, it was discovered that the middle support was no longer in contact with the vessel on the 90° and 166° tie rod. This was caused as a result of the lower support assembly being moved up the cone towards the shroud. The middle support dimensions are being retaken and new middle supports will be installed prior to reload. Other locations on the tie rod assemblies with the potential for gaps and non-conforming conditions were inspected. No additional deficiencies were noted. A summary of NMPC's 10CFR50.59 safety evaluation concerning modification to the core shroud repair tie rod assemblies is provided in Enclosure 4.

E. *Analytical Acceptance Criteria*

Calculations were performed to evaluate the maximum potential displacements of the tie rod relative to the lower spring wedge. This resulted in a redesign of the lower wedge retainer clip. The modified design is described below and accommodates expected movements. The new retainer clips will be installed during the current refueling outage. The clips have been fabricated from X-750, analyzed in accordance with the ASME Code, and meet original design criteria for the tie rods.

F. *Design Change of Lower Spring Wedge Retainer Clip*

The function of the lower wedge retainer clip is to retain the lower wedge in the proper position during installation. It was not designed to experience operational loads. Lower wedge to vessel contact was assumed to move and accommodate differential thermal expansion between the tie rod assembly and the vessel. As explained in Enclosure 2, the friction force between the wedge and the vessel was sufficient to prevent movement of the wedge during thermal growth of the tie rod assembly. The latch portion of the retainer clip became loaded resulting in the overstressed condition of the retainer clip and its subsequent failure.

The retainer clip has been redesigned to accommodate movement during normal and transient conditions. The redesigned retainer clips will be installed prior to reload. Enclosure 5, "Design Report for Improved Shroud Repair Lower Support Latches," provides the results of an evaluation performed for the redesigned latch and demonstrates acceptability of the redesigned latch and its use in the original tie rod assembly.

III. Further Actions

NMPC has analyzed the as found condition of the shroud vertical welds and has established that the plant can be operated safely. A conservative interval for re-inspection of the welds has been established as described in Enclosure 1. Re-inspection, including tightness checks of the tie rod nuts, will be performed after approximately 10,600 hours of operation and NMPC will have plans for a contingency repair should one be needed at that time. NMPC plans additional analyses, during the upcoming cycle, which may justify extension of the re-inspection interval for the shroud vertical welds. The results of these analyses will be submitted to the NRC, if appropriate. A boat sample of cracked material will be mechanically removed from a shroud weld HAZ at an appropriate location prior to restart from RFO-14. As a longer term action, NMPC plans to perform analysis on the sample to establish the presence of IGSCC, the age of the cracking, whether crack growth has arrested and to investigate any other potential contributing mechanisms. This metallurgical sample is to be used to help NMPC and the industry better understand the IGSCC cracking of the BWR core shroud vertical welds.

IV. Inspection of Other Internals

NMPC has performed inspections over the operating life of the plant to meet several ASME Code, industry, BWRVIP and Augmented Regulatory requirements. These inspections provide the basis for an overall condition assessment of the RPV internals. Specifically, the inspections performed during the current refuel outage on the internal core spray annulus piping and core spray spargers, showed no crack growth of previously identified indications on the spargers. The annulus piping was found to be without flaws, including the critical welds at creviced locations. A summary of inspections performed to date of other internals is provided in Enclosure 6.

SUMMARY

NMPC has performed an evaluation of the tie rod restoration activities and the as found condition of the vertical welds and found them acceptable for continued service.

NMPC requests approval of the final design documentation for the proposed modification of the tie rod retainer clips by a revision to the existing NRC shroud repair safety evaluation

submitted as an alternate repair under 10CFR50.55 (a)(2)(i). Receipt of NRC approval is requested by April 20, 1997.

Very truly yours,

Martin J. McCormick Jr.
Vice President - Nuclear Engineering

MJM/MSL/lmc
Enclosures

xc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. S. S. Bajwa, Acting Director, Project Directorate I-1, NRR
Mr. B. S. Norris, Senior Resident Inspector
Mr. D. S. Hood, Senior Project Manager, NRR
Records Management

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Niagara Mohawk Power Corporation	)	Docket No. 50-220
	)	
Nine Mile Point Unit 1	)	

Martin J. McCormick Jr., being duly sworn, states that he is Vice President - Nuclear Engineering of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the document attached hereto; and that the document is true and correct to the best of his knowledge, information and belief.

M. J. McCormick Jr.

Martin J. McCormick Jr.
Vice President - Nuclear Engineering

Subscribed and sworn before me,
in and for the State of New York
and the County of Oswego,
this 8th day of April, 1997.

John C. Josh

NOTARY PUBLIC

JOHN C. JOSH
Notary Public, State of New York
No. 4837303
Qualified in Oswego County
Commission Expires Feb. 28, 1998

INDEX OF ENCLOSURES

ENCLOSURE 1	Assessment of the Vertical Weld Cracking on the NMP1 Shroud
ENCLOSURE 2	Shroud Repair Anomalies, Nine Mile Point Unit 1, RFO14
ENCLOSURE 3	Nine Mile Point Unit 1 Core Shroud Cracking Evaluation
ENCLOSURE 4	10CFR50.59 Safety Evaluation 96-018, Revision 1
ENCLOSURE 5	Design Report for Improved Shroud Repair Lower Support Latches
ENCLOSURE 6	Inspection History
ENCLOSURE 7	Affidavit (GE)
ENCLOSURE 8	Non-Proprietary Version of Reports

ENCLOSURE 2

SHROUD REPAIR ANOMALIES
NINE MILE POINT UNIT 1
RFO14

ENCLOSURE 4

10CFR50.59 SAFETY EVALUATION
96-018, REVISION 1

10 CFR 50.59 SAFETY EVALUATION SUMMARY
MODIFICATION TO THE CORE SHROUD REPAIR STABILIZER ASSEMBLIES

A. DESCRIPTION:

A shroud repair modification was installed in Nine Mile Point 1 Nuclear Power Plant to provide an alternate load path for the Type 304 stainless steel circumferential welds, H1 through H7. The modification ensures the structural integrity of the core shroud by replacing the function of welds HI through H7 with 4 stabilizer assemblies and four core plate wedges.

In the course of the post-installation inspection of the shroud repair, three deviations were identified, evaluated and were found acceptable for continued plant operation through the next cycle. After additional review and evaluation, additional modifications are proposed to provide the long term corrective actions.

During the spring 1997 refueling outage, two additional deficiencies were found on the shroud repair hardware. Each of the four shroud repair stabilizer assemblies were found to have less than the original installation preload and one of the lower wedge latches had failed inservice. Two other lower wedge latches also appeared to be degraded. The latch is a wishbone shaped piece, that is intended to prevent relative motion between the lower wedge and the lower spring with the assumption that sliding would occur between the lower wedge and the RPV wall. The deviations were found during required augmented In-service Inspections (ISI) and during the planned replacement of the shroud stabilizer assembly at 270°.

The root cause of the stabilizer vertical loss of preload was due to clearances between the lower support toggle bolts and the holes in the shroud support cone. The importance of the clearance between the toggle bolts and the hole was not recognized and not incorporated into the installation engineering documentation. This allowed the lower support to move up the shroud support cone toward the shroud when the plant reached normal operating conditions. The root cause of the latch failure is an incorrect design assumption regarding sliding at the vessel to lower wedge interface. A detailed discussion of the as-found condition of the stabilizer assemblies and the root cause of the deviations is included in Reference 27.

This evaluation considers the addition of the three modifications described below and how these modifications affect the Safety Evaluation for the Core Shroud Repair Design, Reference 23, 31 and 32. The references in Part E retain the same numbers with additional references applicable to the modifications.

Modification 1 The lower spring of one stabilizer assembly bears on the blend radius of the 270° recirculation nozzle. The proposed modifications is to replace the tie rod and spring assembly with one having the spring on the opposite side of the tie rod. This proposed modification relocates the spring to bear on the RPV as intended.

Modification 2 The lower spring contact with the shroud do not extend beyond weld H6A at any of the four locations. As result, the barrel section between welds H5 and H6A is not laterally restrained during a steam line LOCA combined with a DBE as was intended. The proposed modification adds an extension piece to extend the spring contact beyond weld H6A and restore this feature to its intended function. The extended contact and the core plate wedges also provide an redundant load path between the core plate and the lower spring as was intended in the in the original design.

The above two noted modifications have been reviewed and approved by the NRC in Reference 32.

Modification 3 There is clearance between the toggle bolts and the shroud support cone that can effect the axial tightness of the stabilizer assemblies. The lower wedge latches may become loaded due to differential vertical displacement greater than intended by the original design of the latches. There are two corrective actions. The first is to remove the clearance between the toggle bolts and the shroud support cone. This has been accomplished with the Reference 28 procedure. The removal of the clearances restores the stabilizer assemblies to their originally intended design and does not represent a modification. The second corrective action was to install new modified latches which are more tolerant of differential vertical displacement.

A.1 270° STABILIZER ASSEMBLY MODIFICATION:

Following the installation of the core shroud repair a visual inspection of the as-installed assembly hardware showed the lower spring wedge on the 270° stabilizer assembly bearing on the blend radius of the recirculation nozzle. The wedge was intended to bear on the RPV wall.

The proposed modification is to replace the tie rod and spring assembly with one having the spring on the opposite side. The modification moves the spring sufficiently such that it will bear on the RPV originally as intended. The modification utilizes existing hardware which was built as a spare along with the other stabilizer assemblies. Only minor rework is required to relocate the lower spring and the rework has no affect on the hardware function. The modification does not require additional penetrations through the shroud support cone or any additional EDM work. The modification uses the same lower support and upper spring assemblies and there is no change to the actual tie rod location.

Additional analysis has been done to address the design where the lower springs are no longer located 90° apart. The non-uniform lower spring spacing affects the net spring characteristic when the horizontal seismic load is directed between two springs. The analysis show the loads and displacements remain acceptable for all conditions.

A.2 H6A CONTACT EXTENSION MODIFICATION:

The lower spring contacts with the shroud do not extend above the H6A weld as was intended. The design function can be restored by adding a U shaped extension piece to extend beyond weld H6A. The extension piece fits over the existing lower contact with the legs of the U extending around the sides of the existing lower contact. The steps at the ends of the legs fit under the lower contact to prevent axial movement. A tang at the top extension fits in the gap between the lower contact and the lower spring to restrict the horizontal movement. The added extension piece is captured in all directions on the existing lower contact. The legs of the extension are spring loaded to provide a positive clamping force against the sides of the lower contact. The spring force is not required to capture the part but is sufficient to prevent any free movement or vibrations. With this arrangement, the added extension piece is captured in all directions and is held secure by the spring loaded clamping force.

The hardware for both modifications is designed and fabricated to the same design basis (Ref. 1) as the original shroud repair hardware. The design life of all repair hardware will be for twenty-five years (the remaining life of the plant, plus life extension beyond the current operating license), to include 20 Effective Full Power Years.

The modified stabilizer assembly includes the same design features as the original hardware. All parts are locked in place or captured by mechanical devices. The stresses in the stabilizer do not change and

remain less than the allowable stresses. The repair hardware is fabricated from intergranular stress corrosion resistant material. There is no welding in the construction or installation of the shroud repair hardware. The fast flux levels at the stabilizers are well below the damage threshold which could result in the degradation of material properties. After 25 years of operation, the maximum fast fluence at the shroud repair components will be well below the value to cause damage. Therefore, it is very unlikely that a component will fail.

A.3 LATCH MODIFICATION AND STABILIZER INSTALLATION CORRECTIVE ACTION:

The design of the new improved shroud repair lower support latches have been analyzed in detail in Reference 30. The design of the new latches maintains the original design function. The function of the original latch was to secure the wedge to the lower spring. This is primarily needed when the wedge looses contact with the reactor vessel wall. This is an important function since the wedge will otherwise slide down and create excessive gaps. The new latch design maintains the wedge support capability and can readily support the dead weight and flow forces which could act to push the wedge down. The new latch design incorporates another spring which can tolerate vertical displacements. Therefore, the original functional requirement is accomplished while adding more flexibility in the vertical direction to accommodate vertical displacements. Under the most probable operating and sliding conditions the new latch design is expected to perform satisfactorily for the remaining life of the plant. Even for worst case postulated conditions, the latch is capable of operating without failure throughout the next operating cycle.

The new latches can tolerate a differential vertical displacement for the worst case thermal transient event (loss of feedwater event) without experiencing an overstress condition. Also for normal plant operation, the maximum vertical differential displacement under probable wedge interaction conditions (assuming no slippage between the RPV and the wedge) is 0.10 inches. Under this deflection the stresses in the new latches will be less than the stress limit established to prevent stress corrosion in X-750 material for a 40 year lifetime. A comparison of the original latch design to the new design has been performed using common finite element modeling methods. The results show that the new latch is 8 to 12 times more capable of tolerating vertical displacements than the original design. This order of magnitude improvement in the design provides assurance that the new latch will perform satisfactorily in the next operating cycle.

The removal of the clearance between the toggle bolts and the shroud support cone will assure that the tie rod vertical forces will be as intended in the original design. The vertical clearances in the stabilizer assemblies were eliminated using the procedure included in Reference 28. Each of the four stabilizer assemblies were then torqued to the original required installation value. With the tie rod in a tight condition at startup, the proper vertical thermal expansion loads can be accomplished during the heatup of the reactor, and maintain the hold down forces on the shroud through subsequent heatups and cool downs.

A.4 MATERIALS:

The installed stabilizers tie rods are fabricated entirely from the type 316, 316L stainless steel (both with a carbon content less than 0.02%) or alloy X-750. The added contact extension and modified latches are fabricated from alloy X-750. The replacement components for the 270° tie rod modification will be fabricated using the same materials as the currently installed stabilizers. The fabrication requirements for the two proposed tie rod modifications will be in accordance with the previously approved fabrication requirements for the NMP-1 core shroud stabilizers. There is no welding required during fabrication or installation.

B. ANALYSIS:

The applicable criteria and conformance for this analysis is as follows. The criteria is the same criteria that was used for the original Shroud Repair Design Safety Evaluation, Reference 23. The conformance sections specifically address the three proposed modifications.

B.1 Design Life (Criteria):

The design life of all repair hardware will be for twenty-five years (the remaining life of the plant, plus life extension beyond the current operating license), to include 20 Effective Full Power Years.

B.1.1 Repair Design Life (Conformance):

The hardware for the three modifications is fabricated to the same design basis, including material requirements, as the original shroud repair hardware. All repair hardware has been designed for a design life of twenty-five years (the remaining life of the plant, plus life extension beyond the current operating license), to include 20 Effective Full Power Years. This requirement is documented in reference 1.

Assuring an adequate design life is mainly a material selection and process control effort, for this equipment. The selection of low carbon stainless steels and high nickel alloys assures the best available materials for the nuclear reactor environment. Solution annealing and sensitization testing are imposed to guard against inter granular stress corrosion cracking (IGSCC). Process chemical controls are imposed to assure that contamination by heavy metal and chlorine or sulfur compounds will not occur. This is the same design selections and controls imposed for a standard forty year plant life. There is nothing in the equipment or installation that puts a specific limit on how long it can be used, such as creep or radiation degradation.

The stresses in the latch are within ASME code limits and the latch is analyzed to be resistant to stress corrosion for a minimum of 2 years assuming conservative worst case displacements in the retainer. It is fully expected that the retainer will last for a significantly longer time based on the factor of improvement which has been demonstrated from the original design. For the expected sliding case where the movement is always along the wedge/spring interface, the retainer will last for a least the remaining life of the plant. The retainers will be inspected at the next outage to determine which type of sliding is occurring in order to validate the service lifetime of the retainers.

B.2 Safety Design Basis (Criteria):

To assure the safety design basis is satisfied and that the safe shutdown of the plant and removal of decay heat are not impaired, the repair hardware shall assure that the core shroud will maintain the following basic safety functions:

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

B.2.1 Safety Design Basis(Conformance):

The changes in the lower spring spacing affects the system spring characteristics for loads acting between two contacts. Additional seismic analysis (Reference 24) calculated core support displacements for the bounding conditions. The section below is revised to include the maximum displacements based on modified lower spring spacing and includes the gap between the shroud and the contact extension. All displacements remain acceptable. The new modified latch design on the lower spring wedge does not effect the maximum displacements below.

- The core spray piping analysis performed to support the shroud repair included a shroud displacement of 0.904 in. horizontally and 0.65 in. vertically, caused by a fault condition. This displacement will not create an unacceptable loading condition in the ECCS piping and therefore will perform its intended safety function. The proposed modifications do not change the maximum displacements calculated for the original shroud repair at the upper shroud. Therefore there is no change in loading of the core spray piping.

- The proper decay heat removal requires that the shroud to remain as a flow boundary to force water through the fuel and not allow a large leakage into the downcomer region. The maximum permanent horizontal offset of adjacent shell sections, that are not directly supported by either the upper or lower springs, is limited by structural stops to 0.75 in. Since the wall of the shroud is 1.5 in. thick, the shroud will still function properly as a flow boundary within the reactor.

- The safe shutdown of the plant is a function of the SCRAM capability. The core support plate and the top guide must be kept aligned within test limits so that friction between the control rods and fuel bundles will not impair proper motion. The worst case condition exists when the top guide moves one direction and the core support moves the opposite. This creates the maximum angle between the fuel bundles and the guide tubes. The maximum temporary calculated horizontal displacement of the top guide is 0.904 in. and the maximum for the core support is 0.85 in. The corresponding allowable displacement are 1.87 in. and 1.49 in. There is no calculated permanent horizontal displacement of the top guide and the maximum permanent displacement for the core support is 0.48 inches. The corresponding allowable core support permanent displacements is 0.67 inches.

B.3 Flow Partition (Criteria):

Repairs to the core shroud are not required to totally prevent leakage from the core region into the downcomer annulus. However, the design shall ensure that cracked welds do not separate under normal operations as a minimum. Design will account for leakage from the region inside the shroud into the annulus region during normal operation. The leakage should not exceed the minimum subcooling required for proper recirculation pump operation and the core bypass flow leakage requirements assumed in the reload safety analysis shall be maintained. The design will also verify acceptable leakage through the flow partition resulting from weld separation during accident and transient events.

B.3.1 Flow Partition (Conformance):

The original shroud repair design ensured that cracked welds will not separate under normal operations. The original shroud repair design accounted for leakage from the region inside the shroud into the annulus region during normal operation. The leakage does not exceed the minimum subcooling required for proper recirculation pump operation and the core bypass flow leakage requirements assumed in reload safety analyses is maintained.

There are no requirements for allowable leakage during the accident (LOCA and/or seismic). After the accident, the leakage is limited by the allowable deflections such that the shroud section does not displace sufficiently to open any vertical flow areas. The maximum permanent horizontal displacement of a shroud cylindrical section that is not directly supported by either the upper or lower springs is less than 0.75 inch, which is equal to one half of the thickness of the shroud. Thus, leakage after an accident will be limited to the leakage through a crack. Since the pressure difference across the shroud is small, the leakage will be small.

The three proposed modifications have no affect on the potential weld crack separation or any potential leakage path. The three modifications do not require any new holes or penetrations through the shroud/shroud support. Therefore the leakage calculations and performance predictions in References 23 and 29 remain valid. The added contact extension provides assurance the maximum permanent displacement of the shroud cylinder between weld H5 and H6A remains less than 0.75 inch.

B.4 Flow Induced Vibration(Criteria):

The repair shall be designed to address the potential for vibration, and to keep vibration to an acceptable level. The natural frequency of the repaired shroud, including the repair hardware, shall be determined. The vibratory stresses shall be less than the allowable stresses of the repair materials. Forcing functions to be considered include the coolant flow and the vibratory forces transmitted via the end point attachments for the repair. Testing may be used as an alternative or to supplement the vibration analysis.

B.4.1 Flow Induced Vibration (FIV) (Conformance):

The original shroud repair was designed to address the potential for vibration, and to keep 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. Forcing functions considered included the coolant flow and the vibratory forces transmitted via the end point attachments for the repair. Details of the original vibration analysis are provided in Reference 23. The three repair modifications have no affect on the natural frequency of the stabilizer assembly or on the vortex shedding frequency. Therefore the original vibration evaluation in Reference 23 remains valid for the stabilizer assemblies.

The potential for vibration of the new extension pieces has been considered. Forcing functions considered, included the vibratory forces transmitted from the stabilizer assemblies and coolant flow. The stabilizer vibratory forces are low, as demonstrated in the original vibration analysis, therefore vibratory forces imposed on the extension pieces are low. The coolant flow will not vibrate the lower contact extensions because the extensions are captured in all directions on the existing lower spring assembly. The lower contact extension is a "U" shaped part which fits around the existing lower contact. Steps at the ends of its legs extend under the lower contact to prevent axial movement. A tang towards the top fits in the gap between the lower contact and the lower spring to prevent horizontal movement. A positive spring force from the legs keep the part tight and prevent random vibrations.

The only time that FIV is of interest is when the lower wedge loses contact with the vessel wall. This can occur during hydrotest, maximum seismic conditions, and during the limiting upset thermal feedwater event. These events have short duration with the longest potential duration being 8 hours for the hydrotest event. The loss of contact at the lower spring support is not a concern in either the tie rod assembly or the subassembly of the latch and lower wedge for the following reasons:

- The time when contact is lost is a relative short duration and the associated number of cycles is limited.

- An independent calculation of the new latch and lower wedge assembly shows that the natural frequency is sufficiently high to avoid flow induced vibration.

- The clearance which is created between the wedge and the vessel wall is less than 0.050" which will limit the motion of the lower wedge in the lateral direction. This prevents any significant contact forces from being produced, and contact would dampen out any excitation of the lower wedge. The relative radial movements between the vessel and the shroud are such that surface contact is likely to remain at one of the two surfaces during the postulated events.

- Even postulating that no support is present at the lower spring, analysis has been performed for the tie rod assembly which demonstrates that flow induced vibration will not occur.

In conclusion, none of the shroud repair components are susceptible to flow induced vibration when contact is lost at the lower spring contact.

B.5 Loading on Existing Internal Components(Criteria):

Increased stress on existing internal components, used in the repair, is acceptable as long as the current plant licensing basis are met. Increases in applied load shall be demonstrated to be acceptable.

- The repair shall be designed so as to produce acceptable loading on the original structure of the shroud, consistent with the criteria provided herein.

- The repair should minimize stresses introduced into the shroud consistent with the criteria provided so as to not aggravate further shroud cracking.

- The repair should minimize the loading on the supporting structures of the shroud, such as the shroud support cone and the RPV wall, to stay within the original design allowable stresses of these structures.

- Supplemental seismic analysis for the proposed modifications shall conform to the same methodology and criteria used in the original shroud repair seismic analysis as documented in the FSAR.

B.5.1 Loading on Existing Internal Components (Conformance):

- Stresses on the original structure of the shroud, which are directly impacted by the shroud repair hardware, have been demonstrated to be acceptable. The results of this evaluation are documented in references 4, 5 and 11 for all of the postulated accidents.

- The original shroud repair was designed to minimize stresses introduced into the shroud consistent with the criteria provided so as to not aggravate further shroud cracking. The addition of the contact extensions , the modification to the 270° tie rod and the addition of modified lower wedge latches has an insignificant affect on the component loads and stresses. In addition analyses included in Reference 29 have been completed regarding the potential impact the shroud stabilizer assemblies could have on vertical weld cracking. The results have shown that any hoop stress induced at the vertical welds due to shroud stabilizer thermal preload is negligible. The overall

conclusion is that the shroud stabilizers had no affect on the shroud vertical weld cracking identified at V9 and V10. Therefore the evaluation in Reference 23 remains valid.

- The original shroud repair design minimized the loading on the supporting structures of the shroud, such as the shroud support cone and the RPV wall, to stay within the original design allowable stresses of these structures. The results of this evaluation are documented in references 4, 5 and 11 for all of the postulated accidents. Relocating the 270° lower spring assembly changes the spacing between the adjacent lower spring assemblies. The change in spacing affects the net spring characteristics and load distribution when two springs share the horizontal seismic load. Analysis show the load on any one spring does not exceed the loads used in the original stress evaluation, Reference 24. The stress evaluation remains valid for the modified 270° stabilizer modification.

B.5.1.1 Seismic Analysis (Conformance):

The modifications adding the contact extensions and modified lower wedge latches have no affect on the seismic analysis.

Relocating the lower spring affects the original seismic analysis. Supplemental seismic analysis was made using the same methodology and criteria as was used in the original seismic analysis. The changes in the spacing between lower springs and affects the effective spring characteristics when two springs share the horizontal seismic loads. Springs less than 90° apart increase the effective spring constant and springs greater than 90° tend to lower the spring constant. Equivalent spring constants were determined for the bounding conditions and additional seismic calculations were made to determine loads and displacements (Reference 24). The individual spring loads do not exceed the loads used in the original stress evaluation (Reference 25) and the calculated displacements remain acceptable (Part B.2.1).

B.6 Annulus Flow Distribution (Criteria):

The design shall not adversely affect the normal flow of water in the annulus region, or the normal balance of flow in this region. The design shall not adversely restrict the flow of water into the recirculation suction inlet.

B.6.1 Annulus Flow Distribution (Conformance):

None of the three modifications adversely affect the normal flow of water in the annulus region, or restrict the flow in any way that would adversely affect normal balance of flow in this region. The design does not adversely restrict the flow of water into the recirculation suction inlet.

B.7 Emergency Operating Procedure (EOP) Calculations (Criteria):

Inputs to the EOP calculations, such as bulk steel residual heat capacity and reduction of reactor water inventory shall be addressed based on repair hardware mass and water displacement.

B.7.1 Emergency Operating Procedure (EOP) Calculations (Conformance):

The addition of the spring contact extensions and new latches have an insignificant affect on the EOP calculations, such as bulk steel residual heat capacity and reduction of reactor water inventory since the quantity of steel added is negligible as compared to the mass and volume of the existing shroud repair hardware and reactor internals.

B.8 Radiation Effects on Repair Design (Criteria):

The design of the repair shall account for the affects of irradiation relaxation utilizing end-of-life fluence on the materials.

B.8.1 Radiation Effects on Repair Design (Conformance):

The original design of the repair accounts for the affects of irradiation relaxation utilizing end-of-life fluence on the materials. In accordance with Reference 1, the design considers an End-of-Life preload relaxation for the upper and lower springs. The radiation level is less than the limit contained in the UFSAR . The basis for this is documented in reference 11 (design basis for reference 1).

The contact extension has a positive spring loaded clamping force around the lower contact. The initial installation clamping force is not required to keep the part captured or for the part to remain functional. Radiation relaxation may reduce, but will not eliminate the positive clamping load. A postulated reduction in the initial clamping load due to radiation relaxation is not a concern because the extension pieces are captured in all directions as discussed in Part B.4.1 and any amount of positive clamping load will prevent free movement or random vibrations of the extension pieces. A positive spring force in the latch is achieved by compressing the latch prior to insertion into the hole within the lower wedge. A postulated reduction in the initial compression load due to radiation relaxation is also not a concern for the latches as they are captured by recessed areas in the wedge and the lower spring.

B.9 Thermal Cycles (Criteria):

The repair hardware shall consider the effects of thermal cycles for the remaining life of the plant. Analysis shall use original plant RPV thermal cycle diagrams. The design shall assume a number of thermal cycles equal to or greater than the number assumed in the original RPV design. Alternatively, thermal cycles defined by actual plant operating data may be employed if technically justified. Using this thermal cycle information repair components and the repaired shroud shall be evaluated for fatigue loading along with any other design vibratory loads.

B.9.1 Thermal Cycles (Conformance):

The original shroud repair hardware analysis considered the effects of thermal cycles for the remaining life of the plant as documented in Reference 5. The analysis considered thermal expansion for the varying temperatures and material combinations of the shroud, shroud support cone, reactor vessel and the shroud repair stabilizers for normal and upset thermal conditions. The stresses resulting from the thermal cycles have been evaluated by a fatigue analysis. The results show that its effect on fatigue life of the plant is negligible. The three modifications have an insignificant effect on previous fatigue analysis.

The analysis provided in Reference 30 has evaluated the modified lower wedge latches for their capability to withstand loading conditions due to thermal differential vertical displacements between the RPV and the stabilizer lower spring. The analysis concluded that for normal plant thermal cycles as well as transient thermal cycles (loss of feedwater event), the new latches when considering the most probable loading conditions will handle these thermal cycles satisfactorily for at least the remaining plant life. The removal of the clearance between the toggle bolts and the shroud support cone will assure that the differential vertical displacements are limited to the design values used in the Reference 30 analysis.

B.10 Chemistry/Flux (Criteria):

The design shall recognize the use of existing and anticipated water chemistry control measures for BWRs and shall consider the affects of neutron flux on any materials used in the repair.

B.10.1 Chemistry/Flux (Conformance):

Since the materials for the three modifications are the same as was used for the installed shroud repair hardware, existing and anticipated water chemistry control measures and the affects of neutron flux on the materials have been addressed and will have no effect on the repair hardware.

B.11 Loose Parts Consideration (Criteria):

Repair hardware mechanical components shall be designed to minimize the potential for loose parts inside the vessel. The design repair shall use mechanical locking methods for threaded connections. All parts shall be captured and held in place by a method that will last for the design life of the repair.

B.11.1 Loose Parts Consideration (Conformance):

The modified stabilizer assembly has been designed to minimize the potential for loose parts inside the vessel. The design repair uses mechanical locking methods (such as crimped jam nuts) for threaded connections. All parts are captured and held in place by a method such as pinning, staking, spring retainers, interference fits, and crimping that will last for the design life of the repair.

The lower contact extension is captured in all directions on the existing lower spring assembly. The lower contact extension is a "U" shaped part which fits around the existing lower contact. Steps at the ends of its legs extend under the lower contact to prevent axial movement. A tang towards the top fits in the gap between the lower contact and the lower spring to prevent horizontal movement. A positive spring force from the legs keep the part tight and prevent random vibrations. The spring force is not required to ensure the extension is secured to the existing lower contact. A positive spring force in the latch is achieved by compressing the latch prior to insertion into the hole within the lower wedge. The latches are captured by recessed areas in the wedge and the lower spring so they can not become a loose part.

Loose Parts Generated by the Repair Process:

Special tooling/equipment is being provided that will be tested and personnel will be trained on full scale mockups to assure adequate controls exist to minimize the potential for vessel internals damage or loose parts. Protective shields have been designed that can be installed as needed to protect the Feedwater Sparger, Core Spray Line and the Recirculation nozzles. NMPC and GE installation procedures/travelers will be used to establish Foreign Material Exclusion (FME) controls. All tools and equipment used in the Vessel and Spent Fuel Pool will be properly secured.

B.12 Inspection Access (Criteria):

The repair design shall be such that inspection of reactor internals, reactor vessel, ECCS components and repair hardware is facilitated. The installed repair hardware shall not interfere with refueling operations and shall permit servicing of internal components. All parts shall be designed so that they can be removed and replaced. This is to provide full access to the annulus area for other possible future inspections and/or maintenance/repair activities that may prove necessary in the future.

B.12.1 Inspection Access (Conformance):

None of the three modifications affect the access for inspections. All parts have been designed so that they can be removed and replaced.

B.13 Crevices (Criteria):

The repair design shall be reviewed for crevices to assure that criteria for crevices immune to stress corrosion cracking acceleration are satisfied.

B.13.1 Crevices (Conformance):

The selection of the materials for the modification hardware is the same as the original hardware and assures that criteria for crevices shown to be immune to stress corrosion cracking acceleration are satisfied.

B.14 Materials (Criteria):

All materials used shall be in conformance with the BWR VIP requirements.

B.14.1 Materials (Conformance):

Materials for the three modifications have the same requirements as the original shroud repair hardware and are in conformance with the BWR VIP requirements.

B.15 Maintenance/Inspection (Criteria):

The designed repair shall minimize the need for future inspections and maintenance of the repair components. The designed repair shall minimize the requirement for future inspections of the affected shroud joints.

B.15.1 Maintenance/Inspection (Conformance):

The stabilizer assemblies including the three modifications are currently inspected under the NMP1 Augmented Inservice Inspection Program (LDCR No. 1-94-ISI-009, Rev. 3).

B.16 Installation Issues (Criteria):

Tooling/equipment used for installation of repair components shall be evaluated in accordance with Reference 9 and shall consider the following:

- Heavy loads
- Shutdown System Status (N+1)
- Rigging
- Hole Cutting Method

B.16.1 Installation Issues (Conformance):

The modified stabilizer assemblies have the same installation requirements as the original stabilizer assembly with the exception that a special procedure (Reference 28) was developed and performed to

ensure the clearances were removed between the toggle bolts and the holes on the shroud side of the support cone. This procedure ensures that the tie rods remain tight and are restored to their original design mechanical preload. No hole cutting is required for either modification. The installation activities associated with the proposed modifications were evaluated in a separate safety evaluation (Ref. 26).

B.17 Existing Reactor Internals (Criteria):

The design shall not rely on existing reactor internals or components to carry loads that have experienced cracking in the industry (e.g. shroud head bolt lugs, stub tubes).

B.17.1 Existing Reactor Internals (Conformance):

None of the three modification rely on existing reactor internals or components to carry loads that have experienced cracking in the industry (e.g. shroud head bolt lugs, stub tubes).

C. UNREVIEWED SAFETY QUESTION DETERMINATION:

1. **Could the proposed change or activity increase the probability of occurrence of an accident previously evaluated in the SAR? No.** The affected plant systems and components will be capable of performing their intended functions with the three core shroud stabilizer modifications installed. These modifications restore the shroud repair stabilizers to their intended design condition. As the modifications are being provided to the plant's safety-related design requirements, the probability of a component failure is not increased. The three modifications impose a negligible change to the plant operating conditions. Neither modification will interact with any component assumed to initiate an accident in the UFSAR. Nor will the failure or presence of the modifications initiate an accident evaluated in the UFSAR.

2. **Could the proposed change or activity increase the consequences of an accident evaluated previously in the SAR? No.** The calculated Peak Clad Temperature (PCT) will remain below 2200°F, and all structures, systems and components (SSC) used to mitigate the (radiological) consequences of the accidents in the SAR are independent of the three proposed modifications, and thus, the consequences of the accidents will not be affected. The abnormal events in the UFSAR that potentially could be affected by the installation of the stabilizers were evaluated, and they remain unchanged.

   The three proposed modifications impose no change to the plant operating conditions, and thus, there is no affect on any LOCA and transient analyses.

   LOCA-Radiological analysis is based on the plant's engineered safety features (ESF) functioning within design parameters, and the radioactive material source terms. The three modifications will not adversely affect any ESF, and thus, the ESF functions will not be affected. The radioactive material source terms are based on the regulatory limit PCT of 2200°F. As the PCT for Nine Mile Point 1 will remain below this regulatory limit, the source terms will not be affected. Therefore, the consequences of the LOCA-Radiological analysis will not change.

   The MSLB analysis release is limited by the capacity of the MSL Flow Restrictors, and uses UFSAR allowables for source terms. As the three modifications will not affect either of these, the consequences of the MSLB analysis will not change.

   Seismic analyses (Ref. 6) show that the stabilizers will remain functional following an earthquake.

3. **Could the proposed change or activity increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR? No.** The three modifications are designed and constructed as safety related components. No adverse equipment interactions will be created by installing the three modifications. The Installation Processes and Tooling will not adversely effect any internal components important to safety discussed in the SAR. Therefore, the probability of equipment malfunctions is not increased.

4. **Could the proposed activity increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR? No.** The installation of the three modifications ensures that the shroud stabilizer assemblies will perform their intended functions. Thus, consequences of a malfunction of equipment important to safety is not increased. The three modifications and the shroud stabilizers perform a passive function that does not interface with any

equipment that is used to mitigate the radiological consequences of a malfunction in the UFSAR. The effects of the shroud repair stabilizer assemblies on the consequences of potentially affected transients are negligible. As the stabilizer assemblies, including the three modifications, do not adversely affect equipment "Important to Safety," the consequences of all transients will not change. The Installation Processes and Tooling will not adversely effect any equipment important to safety, as discussed previously. Therefore, there is no increase to the consequences of component malfunctions.

5. **Could the proposed activity create the possibility of an accident of a different type than any evaluated previously in the SAR? No.** The stabilizers, including the three modifications, are designed to the structural criteria specified in the Nine Mile Point 1 UFSAR. All of the loads and load combinations specified in the UFSAR, that are relevant to the core shroud, have been evaluated, and are within design allowables. The stabilizers, including The three modifications, do not add any new operational/failure mode or create any new challenge to safety-related equipment or other equipment whose failure could cause a new type of accident. In addition, the stabilizers or the three modifications do not create any new component/system interactions or sequence of events that lead to a new type of accident.

   It has been postulated that if a core shroud had a 360° crack and a MSLB accident occurred, the upper shroud and the top fuel support could lift. If the top fuel support lifted sufficiently, the tops of the fuel bundles could move (shift), which might prevent the control blades from completely inserting (partial scram). This event would be an accident of a different type. However, the core shroud stabilizers would limit the shroud from moving, and thus, prevent the top fuel support from lifting. The proposed changes to the lower spring, the addition of the lower extensions and new modified latches have no affect on the ability of the stabilizer to perform this function. The three modifications also ensure that the barrel section of the shroud between welds H5 and H6A and the core support displacements are limited during a MSLB or recirculation LOCA when combined with an earthquake.

   Therefore, the modifications do not increase the probability of occurrence of an accident of a different type than any evaluated previously in the SAR.

6. **Could the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR? No.** The stabilizers, including the three modifications, structurally replace the shroud horizontal welds. The three modifications include the same design features as the as-installed stabilizers. All equipment assumed to operate in the transient analyses, and the safety-related structures, systems and components will not be adversely affected by the stabilizers, including the three modifications. All components interacting with the stabilizers will perform their intended functions. The stabilizers, including the three modifications, do not increase challenges to or create any new challenge to equipment. The stabilizers, including the three modifications, do not create any new sequence of events that lead to a new type of malfunction. Therefore, the possibility of a different type of component malfunction than evaluated in the SAR is not created.

7. **Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No.** The Technical Specifications Bases, the UFSAR (including the shroud repair design basis documents listed in the UFSAR) and the NRC safety evaluation (SE) of the NMP1 shroud repair were reviewed. The USFAR and the NRC SE define the acceptance limits for calculated displacements / stresses as the "design allowable" displacement / stresses. That is, neither the USFAR nor the NRC SE define the safety margin as the difference between the

previously calculated displacements / stresses and the design allowables. Therefore, increases in displacements / stresses as a result of the proposed modifications will not reduce the margin of safety as defined by the USFAR and the NRC SE, provided the calculated displacements/stresses remain less than the original design allowables.

The analysis completed for the 270° tie rod modification, the lower spring contact modification and the lower wedge latch modification demonstrated that the original shroud repair calculated reactor internals and repair hardware stresses are bounding, therefore the margin of safety is not reduced. The analysis for the proposed modifications also indicate that the calculated maximum core support temporary (0.85") and permanent (0.48") horizontal displacements increased. These increases do not reduce the margin of safety as defined above, because the displacements remain below the design allowable temporary (1.49") and permanent (0.67") displacements.

D. CONCLUSION:

This evaluation has investigated modifications to the shroud repair stabilizers at Nine Mile Point 1 which will restore them to their intended design function. The modifications include relocating a lower spring assembly to properly bear against the RPV, adding extensions to assure the spring contacts on the shroud extend beyond weld H6A and installing new latches which are more tolerant of differential vertical displacement. Additionally new installation requirements were implemented to ensure the tightness of the stabilizer assemblies. The plant licensing bases have been reviewed. This review demonstrates that these modifications can be installed (1) without an increase in the probability or consequences of an accident or malfunction previously evaluated, (2) without creating the possibility of an accident or malfunction of a new or different kind from any previously evaluated, (3) and without reducing the margin of safety in the bases of a Technical Specification. Therefore, installation of these three modifications do not involve an unreviewed safety question.

E. REFERENCES:

1. GE-NE Specification: 25A5583, Rev. 2, "Shroud Repair Hardware, Design Specification"

2. GE-NE Specification: 25A5586, Rev. 1, "Shroud Repair Code, Design Specification"

3. UFSAR, Rev. 12, Nine Mile Point 1

4. GE-NE Document: 24A6426, Rev. 1, "Reactor Pressure Vessel Stress Report"

5. GE-NE-B13-01739-04, Rev. 0, "Shroud Repair Hardware Stress Analysis"

6. GE-NE-B13-01739-03, Rev. 0, "Seismic Design Report of Shroud Repair for Nine Mile Point 1 Nuclear Power Plant"

7. NRC Generic Letter 94-03, July 25, 1994, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors"

8. Niagara Mohawk Procedure: N1-MMP-GEN-914, "Lifting of Miscellaneous Heavy Loads"

9. GE-NE Specification: 386HA852, "Reactor Servicing Tools"

10. GE-NE Document: NEDO-10909, Rev. 7, "SAPG07, Static and Dynamic Analysis of Mechanical and Piping Components by Finite Element Method"

11. GE-NE Document: DRF B13-01739, "Nine Mile Point 1 Shroud Stabilization"

12. GE-NE Procedure: NM-SM-TP&P-04, "EDM Actuators"

13. Niagara Mohawk Procedure: N1-ODG-11, "Outage Safety Assessment"

14. Niagara Mohawk Procedure: NIP-OUT-01, "Shutdown Safety"

15. GE-NE "Post Inspection Plan"

16. GE-NE Specification: 21A1104, Rev. 0, "Specification for Reactor Pressure Vessel"

17. BWROG VIP Core Shroud Repair Design Criteria, Rev. 1, September 12, 1994

18. GE-NE Specification: 25A5584, Rev. 1, "Fabrication of Shroud Repair Components"

19. GE-NE Drawing: 237E434, Rev. 5, "Reactor Vessel Loadings" GE Drawing

20. GE-NE Specification: 383HA718, Thermal Cycles, Reactor Vessel and Nozzle, Description Basis and Assumptions

21. GE-NE-A0003981-1-13, Rev. 1, "Performance Impact of Shroud Repair Leakage for NMP1", 12/15/94

22. Niagara Mohawk Document: SO-EOP-M018,

23. GE-NE--B13-01739-05, Rev. 1, SAFETY EVALUATION, GE Core Shroud Repair Design

24. Supplement 1, GENE-B13-01739-03, Rev. 0, Nine Mile Point 1, Seismic Analysis, Core Shroud Repair Modification

25. Supplement 4, GENE-B13-01739-04, Nine Mile Point 1, Shroud Repair Hardware Stress Analysis

26. NMPC Safety Evaluation No. 95-007 Rev.1, Nine Mile Point 1, Core Shroud Repair Installation.

27. GENE-B13-01739-40, Shroud Repair Anomalies, Nine Mile Point Unit 1, RFO14.

28. NMP-SHD-003, Lower Wedge Latch Replacement and Tie Rod Torque Checks.

29. GENE-523-B13-01869-043, Assessment of the Vertical Weld Cracking on the NMP1 Shroud, April 1997.

30. GENE B13-01739-22, Design Report for Improved Shroud Repair Lower Support Latches.

31. NRC Safety Evaluation of the NMP1 Core Shroud Repair Dated 3/31/95.

32. NRC Safety Evaluation Related to Modifications to Correct Core Shroud Repair Deviations, Dated 3/3/97.

ENCLOSURE 5

DESIGN REPORT FOR IMPROVED SHROUD REPAIR LOWER SUPPORT LATCHES

ENCLOSURE 6

INSPECTION HISTORY

Nine Mile Point Unit 1
Invessel Visual Inspection
Summary of Inspections Performed
Refueling Outage '97

Workscope

The following identifies the invessel visual inspections during the 1997 refueling outage:

Core Spray Piping

"A" core spray piping, welds, and brackets (attachment welds)
"B" core spray piping, welds, and brackets (attachment welds)
There were no relevant indications noted:

Core Spray Spargers

Upper spargers "A" and "C" looking at the spargers, sparger welds, including the tee box welds, nozzles, nozzle welds and brackets (attachment) welds.

Lower spargers "B" and "D" looking at the spargers, sparger welds, including the tee welds, nozzles, nozzle welds and brackets (attachment) welds.

Two indications were recorded (1) crack at nozzle 23A and one on nozzle 26A both indications were observed on previous data. There is no apparent difference in the crack length from 1995 until 1997.

Steam Dryer

All of the steam dryer, banks and skirts, lifting lugs. Close attention to clips, lower stiffener, and areas with previous indications as noted below:

Bank 2, Clip 5	The previously identified indication was noted with no growth or change.
Bank 2, Clip 2	The previously identified indication was noted with no growth or change.
Locking Channel at 225°	The previously identified indication was noted with no growth or change.
Bank 2, Lower Stiffener, 1" Hole	The previously identified indication was noted with no growth or change.
Bank 4, Clip 5	The previously identified indication was noted with no growth or change.

Moisture Separator

Examination of the moisture separator showed no new indications and no growth or change in indications located on the 102 standpipe bracket.

Upper Core Grid

Examined bolting, wedges and verified general cleanliness.

Incore Dry Tubes

SIL409-IDTC 1245 no recordable indications noted.

SIL409-IDTC 3645 one indication was noted and recorded on the dry tube shaft just below the collar.

VES Cladding

Evaluated various areas during examination of all components within the vessel this outage.

Feedwater Sparger

All feedwater spargers, end brackets, pins, wedge blocks and flow holes were examined with no indications noted. In addition, the blend radius of all four feedwater nozzles were examined and found acceptable.

Clad Patch

Located at 180 degrees, 77" down the vessel wall.

Fuel Support Casting

Several accessible core locations were inspected for debris, erosion corrosion and seating surfaces.

(w:\ivirfo97.wpd)

ENCLOSURE 7

AFFIDAVIT (GE)

General Electric Company

AFFIDAVIT

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

(1) I am Project Manager, Regulatory 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 reports GE-NE 523-B13-01869-043, *Assessment of the Vertical Weld Cracking on the NMP1 Shroud*, Revision 0, Class III (GE Nuclear Energy Proprietary Information), dated April 1997, GENE B13-01739-40, *Shroud Repair Anomalies Nine Mile Point Unit 1, RF014*, Revision 0, Class III (GE Nuclear Energy Proprietary Information), dated April 1997, and GENE B13-01739-22, *Design Report for Improved Shroud Repair Lower Support Retainers*, Revision 0, Class III (GE Nuclear Energy Proprietary Information), dated April 1997. 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 results of analytical models, methods and processes,

including computer codes, which GE has developed and applied to perform evaluations of indications in the core shroud 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 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.

(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 7th day of April 1997.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 7th day of April 1997.

Notary Public, State of California

JULIE A. CURTIS
Commission # 1113834
Notary Public — California
Santa Clara County
My Comm. Expires Oct 20, 2000

General Electric Company

AFFIDAVIT

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

(1) I am Project Manager, Regulatory 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 drawings *Reactor Modification/Installation Drawing*, 107E5679, Revision 7, and those drawings listed in the attachment. These documents, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by General Electric Company. The independently proprietary elements that are drawings are marked as proprietary information.

(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., (4)b. and (4)e., 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 constitutes a confidential compilation of information, including detailed design drawing results of a hardware design modification (stabilizer 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 detailed results of the analytical models, methods, and processes, including computer codes, and conclusions from these applications, represent, as a whole, an integrated process or approach which GE has developed, and applied to this design modification. The development of the supporting processes was at a significant additional cost to GE, in excess of a million dollars, over and above the large cost of developing the underlying individual proprietary report and drawings information.

(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 5th day of April 1997.

George B. Stramback
George B. Stramback
General Electric Company

Subscribed and sworn before me this 5th day of April 1997.

Julie A. Curtis
Notary Public, State of California

JULIE A. CURTIS
Commission # 1113834
Notary Public — California
Santa Clara County
My Comm. Expires Oct 20, 2000

ATTACHMENT

Drawings

112D6546, Rev. 3, Tie Rod, Spring Assembly
112D6573, Rev. 3, Upper Support Assembly

CATEGORY 1

REGULAT INFORMATION DISTRIBUTION STEM (RIDS)

ACCESSION NBR:9704100242 DOC.DATE: 97/04/08 NOTARIZED: YES DOCKET #
FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
AUTH.NAME AUTHOR AFFILIATION
MCCORMICK,M.J. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

See Reports Withholding Granted 5/5/97

SUBJECT: Forwards proprietary & non-proprietary repts from GE re GL 94-03, "Intergranular Stress Corrosion Cracking in BWRs." List of repts,encl.Encls withheld,per 10CFR2.790(b)(i).

DISTRIBUTION CODE: AP01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 22+327
TITLE: Proprietary Review Distribution - Pre Operating License & Operating R

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-1 LA	1	1	PD1-1 PD	1	1
	HOOD,D	1	1			
INTERNAL:	ACRS	1	1	FILE CENTER 01	1	1
	OGC/HDS3	1	0			
EXTERNAL:	NRC PDR	1	1 w/out prop			

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

TOTAL NUMBER OF COPIES REQUIRED: LTTR 7 ENCL ~~8~~ 6

[illegible]

CATEGORY 2

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9704100242 DOC.DATE: 97/04/08 NOTARIZED: YES DOCKET #
FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
AUTH.NAME AUTHOR AFFILIATION
MCCORMICK,M.J. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

See Reports

SUBJECT: Forwards proprietary & non-proprietary repts from GE re GL 94-03, "Intergranular Stress Corrosion Cracking in BWRs." List of repts,encl.Encls withheld,per 10CFR2.790(b)(i).

DISTRIBUTION CODE: AP01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 11+338
TITLE: Proprietary Review Distribution - Pre Operating License & Operating R

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-1 LA	1	1	PD1-1 PD	1	1
	HOOD,D	1	1			
INTERNAL:	ACRS	1	1	FILE CENTER 01	1	1
	OGC/HDS3	1	0			
EXTERNAL:	NRC PDR	1	0			

C A T E G O R Y 2 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

TOTAL NUMBER OF COPIES REQUIRED: LTTR 7 ENCL 5

DH
An Extra Copy
Single

NIAGARA MOHAWK
GENERATION
BUSINESS GROUP

NINE MILE POINT NUCLEAR STATION/LAKE ROAD, P.O. BOX 63, LYCOMING, NEW YORK 13093/ TELEPHONE (315) 349-2660
FAX (315) 349-2605

MARTIN J. McCORMICK JR. P.E.
Vice President
Nuclear Engineering

April 8, 1997
NMP1L 1200

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

RE: Nine Mile Point Unit 1
Docket 50-220
DPR-63

Subject: *Generic Letter 94-03 "Intergranular Stress Corrosion Cracking (IGSCC) in Boiling Water Reactors"*

Gentlemen:

By letters dated January 6, 1995 and January 23, 1995, Niagara Mohawk Power Corporation (NMPC) submitted an application for repairs to the Nine Mile Point Unit 1 (NMP1) core shroud. The shroud repairs and use of stabilizer assemblies (tie rods) were submitted as an alternate to the requirements of the ASME Code, Section XI, as allowed by 10CFR50.55a (a)(3)(i). The staff provided approval of the proposed alternate repair by letter dated March 31, 1995. The approval letter and attached safety evaluation required NMPC to submit re-inspection plans for the shroud and repair assemblies prior to the next refueling outage planned for 1997. By letter dated February 7, 1997, NMPC submitted plans for re-inspection of the core shroud vertical welds and repair assemblies in accordance with the criteria provided by the "BWR Vessel and Internals Program" (BWRVIP) document BWRVIP-07.

During the 1997 refueling outage, NMPC conducted core shroud vertical weld inspections per the approved documents and observed vertical weld cracking which exceeded the screening criteria. Additionally, inspections of the four tie rod assemblies found the tie rod nuts to have lost some preload and identified damage to the lower wedge retainer clips on three tie rods. Further details of the as found conditions are provided in Enclosures 1 and 2.

By phone calls on March 20, 1997 and April 2, 1997, NMPC informed the staff of the inspection findings and indicated that analysis of the vertical weld cracking and restoration plan of the shroud tie rod assemblies would be submitted to the NRC prior to restart of the unit. This letter and the attached enclosures provide root cause, corrective actions and the final design documentation which establishes the acceptability of the as found vertical weld

A001 2nd Distribution
1/1