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TAC No. M90270
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Response to Request for Additional Information
Core Shroud Stabilizer Design Submittal

Gentlemen:

By letter dated September 15, 1994, the NRC staff requested that Georgia Power Company (GPC) provide additional information on the Hatch Unit 1 Core Shroud Stabilizer design which was submitted for NRC review on September 2, 1994.

Enclosed is GPC's response to the Request for Additional Information. Also enclosed are the results of the final stress analysis and a summary of the dynamic testing previously performed for General Electric by Hitachi.

Sincerely,

J. T. Beckham, Jr.

Enclosure: GPC's response to NRC request for additional information, Core Shroud Stabilizer Design

Attachments:
1. 383HA617 Rev. 0
2. GENE 771-44-0894 Rev. 1
3. Final Test Report CRD Performance Evaluation Testing with Driveline Misalignment
4. Hitachi Testing Report Summary
5. Final Stress Intensities and Displacements

TWL/eb

Ltr. Encl.
Change: NRC PDR 1 1NP
NOAC 1 1NP
A001

Georgia Power

cc: <u>*Georgia Power Company*</u>
Mr. H. L. Sumner, Nuclear Plant General Manager
NORMS

<u>*U.S. Nuclear Regulatory Commission, Washington, D.C.*</u>
Mr. K. Jabbour, Licensing Project Manager - Hatch

<u>*U.S. Nuclear Regulatory Commission, Region II*</u>
Mr. S. D. Ebneter, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

Enclosure

Edwin I. Hatch Nuclear Plant
Response to NRC Request for Additional Information
Core Shroud Stabilizer Design

NRC Requested Information

1. The overall three dimensional model of the shroud was developed using a COSMOS/M FEA software, 1.70 version. Provide information regarding validation of this code by the licensee or other users. Also, indicate whether or not it has been benchmarked or approved by the NRC.

GPC Response

COSMOS/M is a PC based structural finite element analysis (FEA) computer code developed by Structural Research and Analysis Corporation (SRAC) of Los Angeles California. It has been verified for use in the nuclear power industry per the requirements of 10CFR50 Appendix B and the applicable sections of ANSI/ASME QA-1 and related supplements. Per contractual obligations, SRAC transmits "bug-reports" to GE whenever there are deficiencies discovered for resolution. To our knowledge, COSMOS/M has not been formally reviewed by the NRC staff.

COSMOS/M user guide shows a close comparison between FEA results and closed form solutions of over 100 problems of different types of elements and loading conditions. For validating the COSMOS/M Code for the Hatch Unit 1 shroud analysis application, the verification problems for the elements used in the shroud analysis (3D brick, 3D beam, Shell 4T, rigid R bar, spring and gap) were reanalyzed using the 1.70 version of the Code. The results showed good comparison with the closed form solutions.

NRC Requested Information

2. The maximum compressive load in the tie rod is computed to be 60,700 lbs. during a DBE. It is the staff's understanding that the compressive loads would be transferred to the shroud. However, a concern remains that during the load transfer from the tie rod to the shroud, the upper stabilizer assembly may become detached. Provide clarification on how the hardware assembly remains stable when subjected to compressive loading.

GPC Response

The hardware assembly remains stable when the shroud is subjected to compressive loads. The stabilizer components are assembled such that compressive loads will not be accepted in the tie rods. The tie rod penetrates through the bottom of the upper bracket and is held in place with a nut on the top side only of the upper bracket. This provides a shear connection, which will prevent separation of the tie rod and the upper bracket

under tensional load but will be relieved under compressive loads. The top of the upper bracket is installed over the top, and down inside slotted grooves of the shroud greater than 4 inches, therefore the upper bracket cannot come off of the top of the shroud during compressive loading of the shroud.

NRC Requested Information

3. The rotation of the top guide ring due to failure of H-2 and H-3 welds could result in a loss of preload in the tie rods. This may result in unacceptable displacements of the cracked shroud during faulted events. Provide additional information regarding tie rod preloading to preclude such consequences.

GPC Response

Postulated cracking of H2 and H3 will not affect the design preload requirement. If both H2 and H3 are completely failed, the preload in the tie rods will prevent the lift off of the shroud head during normal operation.

The preload on the tie rods is not intended to prevent a small amount of upward motion during the time when the peak forces from the main steam line Loss of Coolant Accident (LOCA) is applied across the shroud head or during the Design Basis Earthquake (DBE). Per the existing design basis calculations, the main steam line LOCA peak shroud head differential pressure is 30 psi. The tie rods will elastically stretch a maximum of 0.4 inches for approximately 6 seconds during which the LOCA pressures exceed the normal operating pressures. In this event, the upper stabilizer assemblies prevent unacceptable top guide ring displacement. For all other events, the preload in the tie rods will prevent upward motion.

NRC Requested Information

4. It is the staff's understanding that GE is currently revising its estimate of the blowdown loads during a recirculation line break event and the revised calculations would be completed by early October 1994. What impact would the revised loads have on the repair hardware?

GPC Response

General Electric (GE) has completed a TRACG analysis of a domestic 251 inch BWR/3. The results of this analysis have been extrapolated to encompass Plant Hatch (218 inch, BWR/4) and included in the design requirements of the Hatch Unit 1 shroud stabilizers. The specifics of this analysis are discussed in GPC's response to Question 9.

NRC Requested Information

5. The lower spring has a device at its bottom which is attached to the shroud support gusset with a pin. The pin passes through a three-inch hole which is machined in the shroud support gusset. The calculated bearing stress on the inner surface of the hole in the gusset is close to allowable values for the faulted condition loads. The gussets need to be inspected for degradation or damage prior to the installation of the repair hardware to ensure their structural integrity during postulated accident conditions. Describe your plans for inspection.

GPC Response

Accessible areas of attachment welds for the four gussets to be used will be examined with an enhanced VT-1.

NRC Requested Information

6. It is stated in the licensee's evaluation that if one stabilizer is postulated to fail during normal plant operation, there would be no consequence to the shroud (even it is cracked) or to the other three stabilizers. Has this scenario been evaluated analytically? If so, provide the supporting calculations and analysis.

GPC Response

This postulated condition has not been specifically analyzed but is enveloped by other conditions that have been analyzed. During normal operation each stabilizer has a vertical force of approximately 57 Kips (due to thermal preload) and a horizontal force of less than 10 Kips (due to preload). If one stabilizer is postulated to fail, the other three stabilizers will carry the additional vertical load which will result in a total vertical load for each stabilizer of approximately 76 Kips. During the upset thermal loading condition, the vertical load in each stabilizer is approximately 150 Kips. Since the allowable stress intensities are the same for normal and upset conditions, the three remaining stabilizers will be acceptable for normal operation. The insignificant increase in leakage through postulated cracks will not affect normal operation or Emergency Core Cooling System (ECCS) performance after the postulated failure of one stabilizer.

NRC Requested Information

7. It is the staff's understanding that any rotational displacement of the cracked shroud during postulated accident conditions would be limited by the intermediate stops and other physical constraints within the vessel. Based on geometrical considerations, provide an estimate of such limiting displacements and indicate what impact these displacements may have on the ability to insert the control rods.

GPC Response

If it is assumed that the shroud can rotate, then the movement due to rotation of the shroud at the locations which are important for control rod insertion is limited by the upper and lower springs to acceptable values. Horizontal movement due to rotation of the other sections of the shroud is limited to a maximum horizontal movement of 1.5 inches. This worst case condition could potentially occur if the shroud rotated about a remaining ligament at 45°. The radial motion at 135° and 315° would be limited to 0.75 inches and the tangential movement at 225° would be limited to 1.5 inches. Such postulated movement would not have any significant consequence to control rod insertion capability.

Rotational displacement of the shroud is restrained in the same manner as any other horizontal movement. No rotation can occur unless the shroud has separated in the vertical direction at a postulated crack. Without vertical separation, the intergranular crack surfaces are held in intimate contact by the tie rods and therefore, can react the postulated torsional loads. At the top guide and core plate elevations, radial outward motion is restrained by linear springs. At other locations, radial outward motion is limited by mechanical stops.

NRC Requested Information

8. Provide documentation of the results of testing related to the evaluation of the effects of displacements of the core plate and top guide on control rod insertion. Also, provide Reference 2.2 of the GE Report No. GENE-771-44-0394, Rev. 1, "Justification of Allowable Displacements of the Core Plate and Top Guide Shroud Repair," dated September 2, 1994. The referenced document relates to scram testing of a full-scale control rod drive line with simulated core plate, fuel and top guide on a shaker table. These tests were performed by Hitachi in 1978.

GPC Response

Copies of 383HA617 Rev 0 (Attachment 1), GENE-771-44-0894, Rev 1 (Attachment 2), and the Final Test Report "CRD Performance Evaluation Testing with Driveline Misalignment" (Attachment 3) are attached. A summary of the requested Hitachi report is included as Attachment 4.

NRC Requested Information

9. Provide information about the model (i.e., Quad Cities TRACG model, potential flow) used to evaluate shroud response to structural loadings resulting from design-basis events for Hatch 1. Explain why Quad Cities TRACG model is applicable to Hatch 1. The Hatch 1 core shroud stabilizer design submittal, dated September 2, 1994, provided displacements of the top guide, core shroud, and core plate. How were the displacements calculated? Explain why this method of evaluation is acceptable.

GPC Response

A detailed TRACG model of a domestic 251 inch BWR/3 was completed. This model was used to analyze a recirculation line LOCA. This model is described in GENE-L12-00819-05, which has been submitted to the NRC by Commonwealth Edison Company. The predicted transient pressure distributions in the annulus between the shroud and the Reactor Pressure Vessel (RPV) were integrated to obtain the maximum force and moment applied to the shroud during a postulated recirculation line LOCA. This analysis was used to obtain the maximum values of force and moment applied to the Hatch Unit 1 shroud. A similarity analysis was performed with the assumptions that (1) the initial pressure distributions in the analysis is similar between the two BWRs, (2) the degree of subcooling and the recirculation suction line size are approximately the same between the two BWRs and (3) the effects of RPV Internal Diameter (ID), annulus thickness and shroud Outside Diameter (OD) on the blowdown load can be calculated based on the change in velocity in the annulus and the projected area of the shroud. The effects of jet pump blockage area are minor and were not included in the similarity analysis. The overall accuracy of the similarity analysis combined with additional calculational margins used in the original BWRs analysis is judged to be adequate for the Hatch Unit 1 application.

The similarity factor used for converting TRACG results from one BWR to another under the above assumptions is:

$$\frac{D_1 H_1 / (D_1 + T_1)^2 T_1^2}{D_2 H_2 / (D_2 + T_2)^2 T_2^2}$$

Where:

D = Shroud Outside Diameter
H = Shroud Height
T = Thickness of annulus between RPV and Shroud
1 and 2 = Plant 1 and Plant 2

The results for Hatch Unit 1 is that the 251 inch BWR/3 analysis results must be multiplied by a factor of 1.235.

The displacements of the shroud at the top guide and core plate elevations were conservatively calculated by dividing the total horizontal force by the spring constant of each stabilizer spring. Only one spring was assumed to be effective at a time. At the top guide elevation, the horizontal force was divided by 20,000 lb/inch and at the core plate elevation by 150,000 lb/inch. These spring constant values were determined by finite element analysis.

NRC Requested Information

10. Explain why the Faulted 2 case is less limiting than the Faulted 1 case, as stated in the Hatch 1 core shroud design submittal.

GPC Response

The main steam line LOCA (Faulted condition 1) pressure forces are sufficient to elastically stretch the tie rods. Thus, the appropriate seismic model boundary condition is a roller for the uppermost crack. The recirculation line LOCA (Faulted condition number 2) does not cause an increase in the vertical loads on the shroud above the normal operation values. Since the tie rods hold postulated shroud cracks in intimate contact during normal operation, the appropriate seismic model boundary condition is a hinge.

The assumption of a roller boundary condition in the seismic analysis significantly increases the horizontal seismic displacements and forces. Faulted condition number 1 results is higher horizontal forces than Faulted condition number 2. Since the horizontal and vertical forces are higher during Faulted condition number 1 than

during Faulted condition number 2, Faulted condition 2 is less limiting than Faulted condition 1.

The maximum value of moment on the shroud during the recirculation line LOCA is less than the available moment of the dead weight of the shroud. Thus the shroud will not tip. The weight of the shroud is not included in the seismic analysis, therefore it can be used to react the moment due to the recirculation line LOCA.

NRC Requested Information

11. A model superposition method was adopted in the design of the repair hardware, which is a linear analysis. The shroud is considered rigid in the vertical direction, even when fully cracked, because the seismic uplift forces are small as compared to dead weight. However, during a steamline break event, combined with a DBE, the uplift forces would be large enough to separate portions of the cracked shroud. Provide justification for using a linear analysis in this scenario.

GPC Response

The vertical uplift force due to the main steam line LOCA exceeds the normal operating uplift forces only for approximately 6 seconds, per the existing design basis calculations.

A vertical seismic model was not included in the original seismic analysis of Hatch Unit 1. Any potential vertical amplification during the short period of time when a portion of the shroud may have lifted is judged to be small, because the lifted portion of the shroud is only connected in the vertical direction to the remainder of the shroud with the tie rods. The tie rods can not apply a vertical upward force on the lifted portion of the shroud. Thus vertical excitation can not be transferred from the unlifted shroud to the lifted portion of the shroud.

NRC Requested Information

12. The maximum permanent and transient deflections at various weld locations are provided in the design specification; however, it is not clarified how these limits would ensure control rod insertability and acceptable leakage during postulated accident conditions. Additional information which clarifies the relationship of the specified deflection limits to system leakage and control rod insertion needs to be provided.

GPC Response

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 sections do not displace sufficiently to open vertical flow areas. The maximum permanent horizontal displacement of any shroud cylindrical section is less than 0.75 inch, which is equal to one half of the thickness of the shroud. Thus leakage after an accident will only be that which can leak through a crack. Since the pressure difference across the shroud after an accident is small, the leakage will be small.

The allowable displacement of the shroud at the top guide and core plate elevations are determined based on control rod insertion testing. The logic for the development of the allowable displacements during and after a seismic event is given in GENE-771-44-0894, Rev. 1 (Attachment 2). The allowable core plate displacements have been increased based on the recent testing performed at GE. GENE-771-44-0894 is based on a maximum tested core plate permanent displacement of 0.25 inch which showed no effect on control rod insertion. The recent tests have shown that permanent displacements of 0.88 inch will not prevent control rod insertion and permanent displacements of 0.75 inch will permit scram within Technical Specifications limits. Thus the final allowable core plate displacements are based on a maximum loss of function displacement from testing of 0.75 inch. The final test report, which documents all testing, will be completed by September 26, 1994. The attached test report "CRD Performance Evaluation Testing with Driveline Misalignment" (Attachment 3) includes testing up to a permanent displacement of 0.625 inch.

NRC Requested Information

13. Some further information is needed in terms of the contents of GE documents P50YP102, Arc Welding of Austenitic Stainless Steel, and P10JYP2, Age Hardening of Ni-Cr-Fe Alloy X-750. For P50YP102 include a discussion of the acceptable methods of welding, and welding qualifications and requirements which are appropriate for the repairs proposed.

    For P10JYP2, include a discussion of the fabrication methods and heat treatments of any X750 Alloys proposed for the repairs. Are the Ni-Cr-Fe Alloy X750 (precipitation hardened Nickel based alloy) being fabricated in accordance with the guidelines of EPRI NP-7032, Rev. 1, Material Specification for Alloy X750 for Use in LWR Internal Components? If so, which method(s) of heat treatment(s) is (or are) acceptable for X750 materials used in the repairs?

In addition, is fabrication of austenitic stainless steels being done in accordance with the guidelines of EPRI Document 84-MG-18, Volumes I and II? These documents were referenced in the VIP Task Group Submittal on Repairs.

GPC Response

a. Welding has been specified as a contingency. No welding is planned on the Hatch Unit 1 stabilizers during the fabrication or installation.

   Specification P50YP102 defines the requirements for arc welding of austenitic stainless steel. The document includes requirements for:

   1) Welding materials with reference to ASME or GE specifications for weld metal.

   2) Control of delta ferrite.

   3) Preparation of base material prior to welding.

   4) Welding processes.

      - Control of grinding.
      - Heat input controls.

   5) Qualification of welders to Section IX of the Code.

   6) General acceptance criteria (required attributes) of the completed welds.

b. The processing requirements for the Alloy X-750 parts are specified in the Fabrication Specification 25A5573, Rev. 0. The material shall be in accordance with ASTM B-637 (ASME SB-637), Type 3 condition, with additional requirements specified in paragraphs 3.2.1.1 (maximum cobalt), 3.2.1.2 (hot forming), 3.2.1.3 (heat treatment), 3.2.1.4 (IGA testing), and 3.2.1.5 (age hardening).

   Fabrication Specification 25A5573, Rev. 0 requires that, after hot forming, the Alloy X-750 parts shall be solution annealed at 1975°F±25°F, held at temperature for 60-70 minutes for the thickest section and air cooled. The parts are to be age hardened at 1300°F±25°F for 20 hours minimum and air cooled. There are some minor differences between the requirements of 25A5573 and

three other specification type documents: P10JYP2, ASME/ASTM SB-637/B-637 and EPRI NP-7032, Rev. 1. The requirements of the four documents are summarized in Table 1.

c. The solution heat treatments specified in 25A5573, 1975°F±25°F, is consistent with EPRI NP-7032 for the CIB condition. EPRI NP-7032 states that this temperature range will optimize IGSCC resistance while not causing an increase in grain size, that may be unacceptable for some properties. The air cool after solution heat treatment is specified in 25A5573, P10JYP2 and SB-637/B-637 Type 3. NP-7032 specifies a water or oil quench. The cooling rate after the solution anneal is a critical factor in setting up the material for the subsequent precipitation heat treatment. Too low of a cooling rate at this stage and the required tensile properties in SB-637/B-637/NP-7032 will not be achieved after the precipitation heat treatment. On a semi-finished part, controlled air cool may be preferable to a quench to reduce the potential for distortion problems. NP-7032 specifies a nominal precipitation heat treatment temperature of 1320F. The other three referenced specifications use a nominal temperature of 1300F. As discussed above, the precipitation heat treatment is required to develop the minimum property requirements. Experience has shown that both processes will achieve the required properties. Based upon the above discussion it is concluded that the process defined in 25A5573 is in accordance with the guidelines of EPRI NP-7032, Rev. 1, for the CIB condition and is acceptable for use.

d. The 316 stainless steel with low carbon is primarily purchased as bar stock for the tie rods. The EPRI Document 84-MG-18 does not address bar stock. The tie rod fabrication is per the standard GE practice with the additional step of an induction heating stress relief of the machined threads. This process was qualified to result in a hardness of < RB 80. The following summarizes the level of implementation of the guidelines of 84-MG-18:

1) The Type 316 bars have carbon content less than 0.020%.

2) The bars are in the solution annealed condition (Guideline I, Par. 4.3).

3) All heating and mechanical treatments, including control of sensitization, have been controlled and approved by GPC (Guideline I, Par. 4.4).

4) Surface contamination with low melting elements per the Fabrication Specification 25A5573 and Specification P50YP211 "Cleaning and Cleanliness Control of Reactor System components," (Guideline I, Par. 4.6).

5) The hardness through the cross-section of the threads, from surface to surface, was controlled to a maximum hardness of Rockwell B 92 (Guideline I, Par. 6.2).

6) ASTM A-262 Practice E tests were performed to verify IGSCC resistance. 84-MG-18, Guideline I, Par. 10.5 specifies a Practice A test after a sensitization heat treatment. This test is similar to the Specification E50YP20

NRC Requested Information

14. Clarify whether the shroud stabilizers are being designed in accordance with ASME Section III, 1965 Edition, ASME Section III, 1989 Edition, or some other construction code or design record. If another construction code or design record is being used, please designate what it is, and provide it for the staff's review.

GPC Response

The shroud stabilizers are being designed to ASME Code Section XI 1980 Edition and Addenda through Winter 1981, which permits the use of the original Code of Construction or Owner's Requirements. The original shroud was not designed to the ASME Code. Instead it was designed to GE Specification 21A3319. The Design requirements for the shroud stabilizers are contained in GE Specifications 25A5125 and 25A5572, which were previously submitted September 2, 1994 for NRC review.

The interface loads between the shroud and the Unit 1 RPV were analyzed to the original RPV code of construction, ASME III, 1965 Edition with Addenda through Winter 1966. These loads were shown to meet all of the requirements of the Code.

NRC Requested Information

15. Are the examinations for intergranular surface attack (GE E50YP11) being performed in accordance with codes, standards or other documents accepted by the staff? Please discuss and comment on the contents of E50YP11. Are liquid penetrant examinations (GE E50YP11) being performed in accordance with codes, standards or other documents accepted by the staff? Please discuss and comment on the contents of E50YP11.

GPC Response

a. E50YP11 is intended to be used on the process material to verify that pickling (chemical treatment to remove oxides and/or base metal) or heat treatment procedures are not introducing intergranular attack. The examinations included in this specification are not required if a minimum 0.030-inch is removed from all surfaces by a mechanical technique after the pickling and/or heat treatment processes. This specification is not a replacement for the standard tests which were specified in Fabrication Specification 25A5573 to determine the carbide precipitation or intergranular attack susceptibility; i.e., Specification E50YP20 (Determination of Carbide Precipitation in Wrought Austenitic Stainless Steels) and ASTM A-262 (Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel) Practice E.

b. Fabrication Specification 25A5573 specifies that all final part surfaces shall be inspected to the requirements of the E50YP22A class of this specification with the additional requirements defined in paragraph 5.3.1 of the Fabrication Specification:

   1. No indication interpreted as a crack is allowed.
   2. No linear indications longer than 0.063-inch are allowed.

   With the supplemental requirements of 25A5573, the acceptance criteria of E50YP22A are consistent with the ASME Code, Section III, Subparagraph NG-5352.

NRC Requested Information

16. Type XM-19 steel portions of the repair should be heat treated such that the yield Strength of the XM-19 parts will remain below 90 ksi. Since SM-19 stainless steels with yield strengths above 90 ksi have been shown to be susceptible to IGSCC, indicate how you intend to limit the yield strength of XM-19 to within acceptable limits.

GPC Response

Type XM-19 material is not used in the Hatch Unit 1 stabilizers.

TABLE 1

SUMMARY OF ALLOY X-750 HEAT TREATMENT PROCEDURES

Specification	(FS)25A5573	P10JYP2D	SB-637 Type 3	EPRI NP-7032 CIB Condition
Solution Anneal				
Temperature	1975±25°F	2000±25°F	1975-2050°F	1975±25°F
Solution Time	60-70 minutes	Not Specified	1 to 2 hours	1 to 2 hours
Cooling process	Air Cooled	"Fast Cool" Min 20°F/min to below 800°F	Air cool	Cool by water or oil quench
Precipitation Heat Treatment				
Temperature	1300±15°F	1300±25°F	1300±25°F	1320±25°F
Time	20 hours min	20-21 hours	20-24 hours	20-22 hours
Cooling process	Air cool	Cooled at min rate 25°F/min below 1150°F	Air cool	Air cool

Equalizing heat treatment is not allowed.