UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
PROPOSED MODIFICATION FOR THE HATCH UNIT 2 CORE SHROUD
GEORGIA POWER COMPANY, ET AL.
EDWIN I. HATCH NUCLEAR PLANT, UNIT 2
DOCKET NO. 50-366

1.0 BACKGROUND

In boiling water reactors (BWRs) the core shroud is a stainless steel cylinder within the reactor pressure vessel (RPV) that provides lateral support to the fuel assembly. The core shroud also serves to partition feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core. The RPV, core shroud and other RPV internals are designed to accomplish three basic safety functions:

- provide a refloodable coolant volume for the reactor core to assure adequate core cooling in the event of a nuclear process barrier breach;
- limit deflections and deformation of internal safety-related RPV components to assure that control rods and Emergency Core Cooling Systems can perform their safety functions during anticipated operational transients and/or design basis accidents;
- assure that the safety functions of the items above are satisfied with respect to shutdown of the reactor and that proper removal of decay heat is accomplished.

In 1991, cracking of the core shroud was visually observed in a foreign BWR. The crack in this BWR was located in the heat-affected zone of a circumferential weld in the mid-core shroud shell. The General Electric Company (GE) reported the cracking found in the foreign reactor in Rapid Information Communication Services Information Letter (RICSIL) 054. GE identified the cracking mechanism as intergranular stress corrosion cracking (IGSCC).

A number of domestic BWR licensees have recently performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic experience. The combined industry experience from plants which have performed inspections to date indicates that both axial and circumferential cracking can occur in the core shrouds of GE designed BWRs, and that extensive cracking can occur in circumferential welds located both in the upper and lower portions of BWR core shrouds. The cracking reported in the Brunswick Unit 1 core shroud was particularly significant since it was the first time that extensive 360° core shroud cracking had been reported by a licensee in a domestic BWR. The 360° core shroud crack at Brunswick Unit 1 was located at weld H3 which joins the top guide support ring to the mid-core shroud shell. Information Notice 93-79 was issued by the NRC on September 30, 1993, in response to the observed cracking at Brunswick Unit 1.

The cracks reported by the Commonwealth Edison Company (ComEd) in the Dresden Unit 3 and Quad Cities Unit 1 core shrouds were of major importance, since they signified the first reports of 360° cracking located in lower portions of BWR core shrouds. These 360° cracks are located at core shroud weld H5, which joins the core plate support ring to the middle core shroud shell in both the Dresden and Quad Cities Units. Information Notice 94-42 and its Supplement were issued by the NRC on June 7 and July 19, 1994, to alert other licensees of the core shroud cracking discovered at Dresden Unit 3 and Quad Cities Unit 1.

On July 25, 1994, the Nuclear Regulatory Commission issued Generic Letter (GL) 94-03 (Reference 1) to address the potential for cracking in core shrouds and to request licensees to take certain actions. By letter dated August 24, 1994, Georgia Power Company (GPC) responded to GL 94-03. GPC indicated support of, and participation in, the BWR Vessel Internals Project (BWRVIP) and plans to install a permanent preemptive repair of the shroud in both Hatch units. A permanent repair was subsequently installed on Unit 1 during the fall 1994 refueling outage. The repair encompassed the entire set of circumferential welds in the core shroud and involved the installation of four tie-rod assemblies in the annulus region around the core shroud. GPC submitted the details of the planned repair for the Unit 2 core shroud on July 3, 1995 (Reference 2). Supplemental information in response to the staff's request for additional information dated August 17, 1995 (Reference 3) was provided by GPC on August 25, 1995 (Reference 4).

2.0 EVALUATION

2.1 Scope of the Modification

The function of the Unit 2 core shroud repair is to structurally replace all circumferential welds from the H1 weld at the top of the core shroud to the H8 weld at the bottom of the core shroud. The Unit 2 core shroud contains a total of nine circumferential girth welds. These welds are labeled H1 through H5, H6A, H6B, H7, and H8. (See Figure 1 for the identification of the core shroud welds.) The only significant cracking of BWR core shrouds has been associated with these welds.

The core shroud repair is designed to restrain the core shroud head, the top guide support ring, and the core support plate, and to limit upward displacement of the core shroud to acceptable levels during normal, upset and postulated accident conditions. The modification has been designed as an alternative to the requirements of the ASME Boiler and Pressure Vessel (BP&V) Code pursuant to 10 CFR 50.55a(a)(3)(i). The repair design provides structural integrity for, and takes the place of, all circumferential welds subject to cracking in the core shroud. The repair is designed for the remaining life of the plant and any possible life extension beyond the current operating license. The repair is also designed to accommodate uprated power conditions corresponding to 105% rated power (2558 MWt).

Details of the modification are contained in a number of GE proprietary reports which were reviewed by the staff. These are contained in References 4 through 13 and References 15 through 17.

2.2 Shroud Modification Design Description

The core shroud repair design consists of four tie-rod stabilizer assemblies installed 90° apart in the core shroud/reactor vessel annulus. Each assembly consists of a tie-rod, an upper bracket, upper stabilizers, a lower spring, a middle support assembly, and a collet mount connected by a solid rod. The assemblies which are designed and fabricated as safety-related components, are used to maintain the alignment of the core shroud assuming all circumferential welds are cracked 360° through-wall.

At the top of the shroud, each stabilizer assembly fits into a slot that is machined partially into the top shroud flange just below the shroud head. The stabilizer upper bracket is inserted into this slot and extends downward to below weld H3 providing support for the upper stabilizer. The tie-rod passes through a hole in the upper bracket and is held against the upper bracket with a nut. The tie-rod extends downward approximately 151 in. to the lower spring. At the middle of the tie-rod, a support is installed between the tie-rod and the reactor pressure vessel (RPV) to minimize the potential for vibration, and provide a limit to the potential motion of the shroud between welds H4 and H5. The bottom of the tie-rod threads into the lower spring which has a clevis at its bottom that is attached to a collet connecter with a pin. The collet connects to the shroud support through a hole that is machined in the shroud support.

Each cylindrical section of the shroud is prevented from unacceptable motion by the stabilizers even if it is assumed that its respective welds contain 360° through-wall cracks. The motion of the sections above H1, between H1 and H2, and between H2 and H3, is restrained by the upper bracket which contacts the shroud and is radially restrained by the upper stabilizers that contact the inside wall of the RPV. The upper bracket prevents unacceptable motion by a limit stop which is part of the upper tie-rod support. The lower spring contacts the shroud such that it prevents unacceptable motion of the section between H5 and H6A. An extension of the lower spring prevents unacceptable motion of the section between H6A and H6B, as well as the section between H6B and H7. The section between H7 and H8 is prevented from unacceptable motion by a limit stop which is part of the collet assembly. Together, the stabilizer assemblies and the lateral restraints resist both vertical and lateral loads resulting from normal operation and design accident loads, including seismic loads and postulated pipe ruptures.

The stabilizer assemblies are installed with a small vertical preload which assures that all components are tight after installation and during cold shutdown, and provides approximately 3500 lbs. of axial load on the 3.75 in. diameter tie-rods. The upper bracket, upper stabilizers, lower spring, and collet are fabricated from alloy X-750. The tie-rod is fabricated from Type XM-719 stainless steel. The spring bracket, and tie-rod materials have a smaller coefficient of thermal expansion than does the Type 304L shroud. Thus, the stabilizer assemblies are thermally preloaded when the reactor is at operating conditions. The spring constant of the stabilizers in the vertical direction is designed to provide a total vertical preload at operating conditions which is greater than the net upward applied loads on the shroud. Thus, if a combination, or all, of welds H1 through H8 were completely cracked, the stabilizers would vertically restrain the shroud such that no vertical displacement would occur during normal operation, which minimizes potential leakage through the cracks. Vertical separation for any and all welds is precluded except for the postulated design events addressed in Section 2.4.6 of this SE. Similarly, the upper, middle and lower spring assemblies are installed with a small preload during cold shutdown. During normal operation, the lateral expansion of the core shroud and the spring assemblies due to thermal growth is greater than that of the RPV, providing additional preload and support for the core shroud. This preload will restrict the lateral core shroud displacements during postulated accident conditions within acceptable limits, and assure prompt rod insertion during these conditions.

2.3 Structural Evaluation

2.3.1 Core Shroud and Tie-Rod Stabilizer Assemblies

The tie-rod stabilizer assemblies were designed using the ASME Code Section III, 1989 Edition, subsections NB and NG as a guide. The original ASME Code Section III (1968 Edition and addenda through Summer 1970) for the design and construction of the RPV did not contain design requirements for core support structures. The additional loads placed on the RPV by the stabilizer assemblies were evaluated to the original design code.

The load combinations required by the Hatch Unit 2 UFSAR for normal, upset, emergency, and faulted conditions included consideration of three seismic events. They are the Operating Basis Earthquake (OBE), Design-Basis Earthquake (DBE) and the 1/2 Seismic Margin Earthquake (SME). The characteristics of these earthquakes are discussed in Section 2.3.3 of this SE. The SME was recommended for inclusion as an additional DBE case as a result of the Hatch seismic margin assessment (Reference 14).

A three-dimensional finite element analysis using the ANSYS code was used to calculate the shroud structural response. Design loads were obtained by appropriate combination of the mechanical preload; loads from differential thermal expansions of the shroud and repair hardware; gravity, including loads from buoyancy effect; pressure differences; and seismic loads. The use of the ANSYS code is acceptable to the staff.

The Main Steam Line Break (MSLB) LOCA loads were applied as uniform static upward pressures on the core support plate and the shroud head. The LOCA pressures, together with the gravity loads, were used in the analyses for a MSLB LOCA event. Pressures and gravity loads were combined with either the peak DBE or 1/2 SME seismic loads as appropriate in the analyses for the simultaneous MSLB LOCA and DBE event.

The Recirculation Line Break (RLB) LOCA produces a spatial and time varying lateral pressure in the shroud/reactor vessel annulus. The initial acoustic phase of this transient is very abrupt relative to the shroud inertia and frequencies, and does not have a significant effect on the shroud or stabilizer performance. The remainder of the transient extends over a relatively long period of time and as such, is considered a static pressure.

This static pressure produces a 25,000 lb. lateral load in the shroud section between welds H4 and H5. This load, combined with gravity and normal operating pressure differences, was used in the analyses for an RLB LOCA event. The LOCA pressures, operating pressure, and gravity loads were combined with the peak DBE seismic loads in the analyses for the simultaneous RLB LOCA and DBE. The evaluations show that the RLB LOCA loads were bounded by the MSLB LOCA loads for the design of the stabilizer assembly. The staff has reviewed the details provided in the licensee's submittal (Reference 9) relative to the RLB LOCA and finds the use of MSLB LOCA loads for the design of the stabilizer assembly acceptable.

The limiting upset loading condition event which GPC evaluated is the cold feedwater transient. During this transient, due to injection of cold feedwater into the core shroud annulus, a maximum temperature difference of 133°F between the hot core shroud and the cooler tie-rod stabilizer assembly components could exist. This would cause an increase in the tensile load on the stabilizer and an increase in the compressive load on the core shroud. GPC evaluated this condition and determined that the stresses in the stabilizer and in the core shroud for this condition would both be less than the ASME Code upset allowable stress and less than the material yield stress, thus preventing permanent deformation, which is acceptable. GPC also determined that this event is the only case which produces any fatigue in need of consideration. For this event, the maximum calculated fatigue usage was found to be insignificant compared to the allowable usage and is, therefore, acceptable.

GPC has also investigated the effects of radiation on the repair design. Specifically, GPC determined that the fast flux levels on the stabilizer are low compared to levels which could degrade material properties. Further, the service temperature for this application has no significant effect on the degradation of the repair materials.

The NRC staff has reviewed the methodology and results of the stress analysis of the core shroud and tie-rod stabilizer assembly and has determined it meets the appropriate criteria to assure core shroud structural integrity and, therefore, is acceptable.

2.3.2 Evaluation of Postulated Critical Weld Failures

GPC evaluated an enveloping combination of postulated cracked/uncracked welds to define the worst case for the core plate and top guide displacements in order to ensure control rod insertion and safe shutdown during the assumed normal, upset, emergency and faulted conditions required by the UFSAR. Each postulated through-wall cracked weld was modelled as a hinge or roller to determine the limiting displacement. GPC provided the maximum allowable transient and permanent displacements of the core plate and top guide. The staff agrees that these maximum displacements are reasonable and therefore acceptable. The predicted worst case lateral transient deflection of the core plate support ring is 0.24 inch for a load combination of a DBE assuming all welds cracked. This is less than the allowable limit of 1.12 inches for this emergency event. The worst lateral transient displacement of the top guide support ring is 2.23 inches during an DBE plus MSLB LOCA assuming weld H8 is cracked and acting as a roller. The allowable transient deflection of the top guide for this faulted event is 4.0 inches. The core shroud top guide and core plate have been determined to undergo no permanent lateral displacements during the limiting faulted conditions. The staff has reviewed the maximum transient deflections provided in Table 6 of Reference 8. These are less than the allowable transient deflections provided in Section 4.2.3 of the Hatch Unit 2 core shroud repair design specifications (Reference 5). The allowable core plate displacements are provided in the GE proprietary report "Justification of Allowable Displacements of the Core Plate and Top-Guide Shroud Repair" (Reference 16). This report has been previously reviewed by the staff and was found to be acceptable. The staff therefore finds the lateral displacements of the top guide and core plate acceptable.

The limiting loads in the tie-rods and the limiting loads in the upper and lower springs occur for different postulated core shroud crack combinations. The limiting loads in the tie-rods occur under the DBE plus operating pressure combination, assuming a through-wall crack in weld H4 when it behaves as a hinge. The limiting loads in the radial direction on the upper springs occur under the DBE plus operating pressure combination where it is assumed that all horizontal welds in the core shroud are cracked and represented as hinges. The limiting loads in the radial direction on the lower springs occur under the DBE plus MSLB LOCA combination where it is assumed that weld H8 in the core shroud is cracked and represented as a roller. The mid span support is designed to prevent radial deflections of the core shroud from exceeding acceptable limits. The upper and lower springs are similarly designed to prevent the radial deflection of the top guide support ring and the core plate support ring from exceeding acceptable limits. The maximum deflection of any part of the shroud that is not directly supported by the upper or lower springs is limited by mechanical limit stops to approximately 0.75 inch which is equal to one half of the shroud thickness. This results in overlapping of the shroud sections by at least 0.75 inch. The staff has reviewed the licensee's evaluation and finds it acceptable.

The tie-rod stabilizer assembly preload prevents the vertical separation of the core shroud at all potential crack locations during normal operation. The critical cracked weld locations are H2 and H3 since the failure of these welds has a significant effect on the vertical stiffness of the core shroud. This is due to the relatively large deflections in the top guide support ring when vertical loads are applied. GPC evaluated the effects of a postulated failure of the H5 and H6 welds in combination with postulated failures of welds H2 and H3, on the vertical core shroud stiffness and separation at the lower weld locations. The most severe consequences are determined to occur if these welds are postulated to be initially intact but fail subsequently during plant operation. For this scenario, GPC's calculations indicate that there is sufficient preload to prevent weld separation due to the change in rigidity of the core shroud structure. GPC determined that the tie-rod stabilizer assembly cold preload could be reduced a substantial amount due to the application of the core shroud head weight when it is installed if the core shroud stiffness is reduced the maximum amount. However, since the mechanical cold preload is only a small part of the total hot operating preload, there will be no separation at any welds during normal operation. The staff has reviewed GPC's evaluation and finds it reasonable and acceptable.

GPC reported that the maximum expected vertical separation would be 0.588 inch for the postulated DBE plus dead weight plus MSLB LOCA load combination. This displacement is momentary since the tie-rod stabilizer assemblies and the weight of the core shroud and the internals will close the gap once the event is over. GPC also determined that this motion would not place any loading on the control rod guide tubes because they are designed to accommodate a transient vertical motion of this magnitude. The staff finds these results reasonable and acceptable.

2.3.3 Seismic Analysis

A seismic model that contained the reactor building, RPV internals, the core shroud and repair stabilizers was used in determining of the natural frequencies and mode shapes in the East-West and North-South directions. The model was also utilized in the computations of the dynamic response to the DBE and 1/2 SME time histories including shears, moments, and displacements. The model was a two-dimensional linear elastic dynamic model consistent with the original design model in the UFSAR, and the synthetic OBE and 1/2 SME time histories used are based on the UFSAR response spectra. The peak ground accelerations for the OBE and 1/2 SME are 0.08g and 0.15g respectively. With the exception of the nuclear core and the core shroud (including the repair hardware), these models were identical to the original seismic models. The nuclear core was updated to the projected Cycle 13 configuration. The seismic models incorporated the tie-rod stabilizer assemblies and the core shroud with postulated 360° through-wall cracks. The tie-rod stabilizer assemblies were modelled as an equivalent rotational spring and incorporated into the stick model, and these were assumed to resist the horizontal seismic loading acting on the core shroud. However, due to the postulated cracked welds, the structural behavior of the core shroud is non-linear, with different mass and stiffness characteristics causing the dynamic properties of the core support shroud and the tie-rod stabilizer assemblies to vary, depending on the particular load combination and the postulated cracked weld configuration. To permit the application of linear elastic analysis, the core shroud was represented by a number of stick models, in which the critical cracked welds were represented by hinges or rollers. The seismic analyses were performed considering these loading conditions and core shroud models as bounding cases. These analyses were performed using the GE proprietary computer program SAP4G07 that has been accepted for this application. The material damping ratios used in the seismic analyses were the same as those used in the original design analysis and are consistent with the UFSAR.

In order to account for uncertainties in the seismic input and modelling of the core shroud repair, GPC evaluated key input parameters such as the rotational stiffness of the tie-rod stabilizer assemblies along with the various crack locations and joint roller or hinge configurations by performing analyses of bounding conditions. Also, the response spectra from both the DBE and 1/2 SME time histories envelope the smoothed UFSAR spectra used as a target.

Forces and moments due to vertical seismic loading were calculated as a multiplier of the dead weight in accordance with the UFSAR. The peak horizontal and vertical seismic loads were combined by absolute summation with other loads in the core shroud and the repair hardware analyses.

The staff has reviewed the methodology and results of the seismic analysis of the core shroud and the repair hardware, and has found them to be consistent with the UFSAR criteria, and therefore acceptable.

2.3.4 Evaluation of Shroud Shell and Shroud Support Plate

Evaluations of the core shroud shell and core shroud support plate stresses were performed by GPC using finite element models and the ANSYS computer code. The effect of additional loads from the core shroud repair were evaluated for the combined loadings resulting from the specified normal operating, upset, emergency and faulted conditions. The shroud seismic loads were obtained in terms of spring loads, tie-rod moments, and shroud shear/moment diagrams. The maximum values of these parameters required for repair hardware analyses were obtained by scanning the system seismic time-history analysis results. These maximum values occurred at different times and thus did not form a consistent set of parameters required for shroud cylinder analyses. Therefore, the effects of combined seismic, operational and faulted condition loads were analyzed separately from the effects of the seismic loads and the governing seismic loads were considered in the evaluation of the appropriate load combination. As a result of these evaluations, the stresses were shown to be within the ASME Code allowable stresses. The staff has reviewed these results and finds them reasonable and acceptable.

2.3.5 Evaluation of Core Spray Piping

GPC analyzed the core spray lines (CSL) inside the RPV for the seismic loads and anchor movements which result from the assumption of a worst case cracked shroud with the installation of the stabilizer assemblies. The CSL is a run of 5-inch diameter pipe which conducts core spray flow from the RPV nozzle thermal sleeve to the shroud. The CSL does not provide significant restraint to the shroud and was designed with flexibility to accommodate the thermal expansion of the shroud relative to the RPV. Postulated shroud cracking has been shown to induce end-to-end seismic displacement (anchor movements) for the CSL, which is larger than that considered for the original CSL design. The larger end-to-end OBE displacement for an assumed all welds cracked case was analyzed for a representative CSL piping configuration. This analysis, using ASME Code, Section III, Subsection NB piping rules as a guide, demonstrated compliance with fatigue requirements for normal and upset events including ten cycles of OBE. Since the primary plus secondary stress range exceeded 3Sm, the simplified elastic-plastic method of ASME Code Section, Subsection NB-3653.6 was applied. The stress resulting from end-to-end CSL displacement for one cycle of steam line break LOCA plus DBE or 1/2 SME is classified as secondary and is therefore not required by ASME Code Section III to be evaluated. However, as a functional check it was shown that the maximum strain in the CSL during this faulted event is less than one percent which is well below the minimum 25 percent ultimate strain for the 304 stainless steel piping material specification. An added margin of safety exists in the CSL analysis because the stabilizer stiffness used in the analysis was lower than the actual stiffness of the Hatch Unit 2 stabilizers. The actual displacements in the Hatch Unit 2 CSL is therefore likely to be 25 percent less than that predicted by the analysis thus providing a large margin to failure. The staff finds these results reasonable and acceptable.

2.3.6 Evaluation of Flow-Induced Vibration

The potential for flow-induced vibration was evaluated by GPC by calculating the lowest natural frequency of the tie-rods and the highest vortex shedding frequency due to the water in the downcomer. The lowest natural frequency is 69 Hz and the maximum vortex shedding frequency is 4.4 Hz. The results show ample margin between excitation frequency and lowest natural frequency compared to the standard design goal of a factor of 3. The staff therefore concludes that no flow-induced vibration fatigue of the rod stabilizer assembly components is likely to occur.

2.3.7 Loose Parts Considerations

All components of the stabilizer assemblies are locked in place with mechanical devices. Loose pieces cannot occur without the failure of a locking device. The stress in the stabilizer components during normal plant operation are less than one-third of the normal event allowable stresses. If a tie-rod stabilizer assembly were to fail during normal operation, the leakage through any through-wall cracks would increase but would not be detectable. If the failed tie-rod stabilizer assembly part came completely loose, it could fall onto the core shroud support plate or be swept into the recirculation pump suction line. The consequences of such a loose part would be consistent with other postulated loose parts. GPC is requested to submit an inspection plan of the Hatch Unit 2 core shroud hardware approximately 6 months following restart of Hatch Unit 2. (See Section 2.5.2). If the licensee's tie-rod stabilizer assembly inspection results, following the first fuel cycle of operation, indicate that further measures are necessary to assure that the tie-rod stabilizer assemblies (or parts thereof) will not become loose or detached during plant operation, GPC will be required to augment the inservice inspection plan to address these additional measures.

Installation of the stabilizer assemblies require the machining of eight slots in the core shroud head flange just below the shroud head, and drilling/machining four holes in the shroud support plate. These machining activities could generate small objects or debris that may remain in the reactor after the repair is installed. Electric Discharge Machining (EDM) generates swarf, which is very fine particles comprised of carbon, nickel, iron, chromium, etc. (the elements contained in the EDM electrode, and the shroud and shroud support material). These particles are very small (approximately 1-50 microns). Greater than 95 percent of the swarf generated is collected by the EDM electrode flushing system. However, when the EDM electrode breaks the shroud support, the flushing system cannot collect the swarf. This swarf remains in the reactor. The amount of swarf is very small, representing less than a tenth of one percent of the total generated. Consequently, it is considered insignificant.

The minute sand-like particles resulting from the EDM process are too fine and small to be caught at one of the fuel spacers. Most likely, these particles will be carried by the cooling flow up through the top of the upper tie plate. They will eventually be removed from the reactor coolant by the reactor water cleanup (RWCU) system. Therefore, there is no potential for fuel fretting due to the EDM process.

The potential for the particles generated by the repair process could cause control rod drive (CRD) seal wear and was therefore also evaluated. Because the particles generated are so small, they will most likely be carried by the cooling flow through the length of fuel bundles and then discharged from the reactor core through the top of the upper tie plate, or by the core bypass flow through the core region and then discharged through the top guide. The upward flow direction makes it highly unlikely these particles will be deposited on the top of the core plate so that they can migrate to the bottom of the control rod guide tubes where they can be sucked into the CRD. Therefore, it is very unlikely these particles will have any significant effect on CRD seal wear or adverse effects on CRD operation.

In addition to the CRD seals, the potential for the particles generated by the repair processes adversely affecting the reactor recirculation pump seal performance or life was evaluated. The objective of preventing damage to the recirculation pump seals is achieved by venting the seals after maintenance and purging the seals during operation. GPC evaluated the impact which the remaining metal particles/fillings would have on reactor operation, and determined that the suspended particles will be carried away to the reactor water cleanup (RWCU) system where they will be removed and will not increase any short or (long-term) degradation of the CRD or recirculation pump wear.

The staff reviewed the licensee's evaluation for loose parts consideration and potential for debris generated by the repair and finds it acceptable, subject to inspection pursuant to an inspection plan.

2.3.8 GPC 10 CFR 50.59 Safety Evaluation of Core Shroud Repair

In Reference 11, GPC provided its 10 CFR 50.59 Safety Evaluation of the core shroud repair. In accordance with 10 CFR 50.59, GPC determined that no unreviewed safety question will result and no Technical Specification revision will be involved as a result of the implementation of the core shroud repair. The staff agrees with this determination, and concludes that no license amendment, pursuant to 10 CFR 50.90, is necessary.

2.3.9 Conclusion

GPC has demonstrated that the maximum stresses in the core shroud and the tie-rod stabilizer assemblies resulting from operating, upset thermal and emergency and faulted accident conditions meet the corresponding ASME Code-allowable stresses. The staff has reviewed the referenced documents, and has determined that the results are reasonable and, in general, agree with design and analysis practices employed in support of other core shroud repairs reviewed by the staff. Based on the foregoing discussion, the staff therefore concludes that the proposed core shroud repair modification is acceptable from a structural standpoint.

2.4 Systems Evaluation

2.4.1 Introduction

GPC provided information to demonstrate that fuel geometry and core cooling would be maintained given the unlikely occurrence of a through-wall failure of any horizontal weld during normal operations and design basis events with the core shroud repair installed. Fuel geometry must be maintained to ensure control rod insertion while core cooling is ensured by proper emergency core cooling system (ECCS) performance. The GPC submittals provided analyses of the principal effects and issues of operating the plant with postulated circumferential core shroud welds cracked and tie-rod stabilizer assemblies installed. Some of the conditions analyzed by GPC included tie-rod stabilizer assembly induced leakage, core shroud weld crack leakage, downcomer flow characteristics, lateral displacement of the core shroud, and vertical separation of the core shroud. The staff has reviewed these portions of the GPC's submittals and is providing the following evaluation of GPC's findings.

2.4.2 Tie-Rod System Induced Leakage

The installation of the tie-rod assemblies requires the machining of four holes through the shroud support plate and eight slots in the shroud head flange using the EDM process and a Trepan style drill. The licensee estimates that a small amount of core flow leakage will occur based on the clearances of the collet assembly parts passing through the shroud support plate. The total calculated leakage from the installation of the tie-rod assemblies was estimated to be 0.085% of core flow (185 gpm) at 100% uprated power (105% rated power) and 105% rated core flow (Reference 4). The staff does not consider this leakage rate to be significant with regards to total core flow and therefore, it is acceptable.

The installation of the tie-rod assemblies also requires the machining of eight slots into the shroud head flange in order to install the long upper supports. The shroud head flange is located above the H1 weld which is the uppermost weld on the shroud and is above the top guide. At this location, core flow is considered to be two-phase flow. The slots in the shroud head flange will be sufficiently shallow to prevent any leakage from the upper core plenum to the vessel downcomer region. The staff notes that, if leakage does occur, the leakage will not bypass the core and, therefore, it is acceptable.

The Hatch Unit 2 ECCS consists of the single-train high pressure coolant injection (HPCI) system, the automatic depressurization system (ADS), the two-train core spray (CS) system, and the two-train low pressure coolant injection (LPCI) system. The staff notes that the leakage from the shroud support plate and the shroud flange to the downcomer annulus does not affect the performance of the above systems. Therefore, the ECCS performance is not affected by the physical installation of the tie-rod system.

2.4.3 Core Shroud Weld Crack Leakage

The tie-rods are installed with a cold preload to ensure that no vertical separation of any or all cracked horizontal welds will occur during normal operations. Vertical separation, if sufficiently large, could compromise fuel geometry and control rod insertion. For most plants, a maximum vertical separation of 13 to 15 inches is required for the top guide to clear the top of the fuel channels (Reference 15). The staff notes that, with the repair, the estimated vertical separation during normal operations will not affect the fuel geometry, and therefore, control rod insertion is not precluded.

However, a small leakage path could exist due to existing through-wall shroud weld cracks. The licensee conservatively modeled the cracks to provide a 0.001 inch leakage path per weld, H1 through H8 (Reference 11). The licensee estimated that the total leakage from all potentially cracked welds, H1 through H8, having postulated 360° through-wall cracks was approximately 130 gpm (0.045% of core flow) at 100% uprated power (105% rated power) and 105% rated core flow (Reference 11). Although shroud crack leakage is unlikely due to the preload on the tie-rod, the licensee concluded that there are no consequences associated with the repair installed based on these small leakages during normal operations. The staff acknowledges that the total leakage is insignificant and will not affect the performance of the ECCS.

2.4.4 Downcomer Flow Characteristics

The licensee analyzed the available flow area in the downcomer with the four tie-rod assemblies installed. The size of the tie-rod assemblies are small compared to the size of the jet pump assemblies and thus, the tie-rod assemblies are not expected to significantly affect the flow characteristics in the downcomer. However, since the downcomer annulus is smaller at the top of the shroud with other existing obstructions such as the core spray lines, the licensee evaluated the flow blockage area due to the shroud repair upper supports. GPC's analysis demonstrated that the installation of the tie-rod stabilizer assemblies will decrease the as-built downcomer flow area by approximately 6 percent (Reference 4). The staff reviewed the downcomer flow calculation which accounted for the core spray piping, miscellaneous bolts, lugs, and brackets, and the upper supports of the tie-rod stabilizer assemblies. Based on the licensee's analyses, the staff concluded that the installation of the tie-rod stabilizer assemblies will not have a significant impact on the downcomer flow characteristics.

Additionally, the licensee evaluated the corresponding pressure drop due to the decrease in downcomer flow area. The licensee estimated that the loop pressure drop due to the installation of tie-rod assemblies is negligible. Based on this information and information from other reviews of similar core shroud repairs, the staff concluded that the impact on the loop pressure drop is insignificant. Therefore, the staff agrees with the licensee that the installation of the tie-rod assemblies should not affect the recirculation flow of the reactor.

2.4.5 Potential Lateral Displacement Of The Shroud

The licensee also evaluated the maximum lateral displacement of the shroud at the core support plate and top guide under normal operations and load combinations such as design basis earthquake (DBE), main steam line break (MSLB), and recirculation line break (RLB). Lateral displacement of the shroud could damage core spray lines and could produce an opening in the shroud, inducing shroud bypass leakage and complicating recovery. Maximum permanent displacements of the shroud are limited to approximately 0.75 inch by mechanical limit stops. This lateral displacement is equal to half the thickness of the shroud, and accordingly, the separated portions of the shroud would remain overlapped during worst case conditions. Additionally, a permanent lateral displacement of the top guide or core plate to the actual magnitude shown in the submittal will not significantly increase the scram time as demonstrated in Reference 16. Therefore, the staff has concluded that the maximum lateral displacement of the core shroud would not result in significant leakage from the core to the downcomer region following an accident scenario and the ability to reflood the core to 2/3 core height would not be precluded.

2.4.6 Potential Vertical Separation Of The Shroud

The licensee evaluated the maximum vertical displacement of the shroud assuming 360° through-wall cracks at any weld above or below the core support plate during an MSLB and an MSLB plus DBE. These postulated events would result in a large upward load on the shroud which could impact the ability of the control rods to insert and the ability of the core spray system to perform its safety function. As stated above, a maximum vertical separation of 13 to 15 inches is required for the top guide to clear the top of the fuel channels. With the repair installed, the maximum vertical separation during an MSLB is limited to 0.214 inch at the H6B weld location, assuming 360° through-wall failure of any of the respective welds (Reference 8). This separation is limited by the tie-rods and should not impact the core spray system. Furthermore, the licensee analyzed the effect of 360° through-wall cracks in horizontal welds during an MSLB plus a DBE. The licensee stated that this combination event would result in a maximum momentary separation of 0.588 inch at the H6B weld (Reference 2). The staff acknowledges that the ECCS performance and control rod insertion should not be impacted by either case of momentary separation. Therefore, based on this assessment, the staff concluded that postulated separation during an MSLB or an MSLB plus DBE event would not preclude any of the ECC systems from performing their safety functions.

2.4.7 Conclusion

The staff has evaluated the licensee's safety evaluation of the consequences of the proposed core shroud repair. The staff has found that the proposed repair should not impact the ability to insert control rods, the performance of the ECCS, particularly the core spray system, or the ability to reflood and cool the core. The staff concluded that the proposed repair does not pose adverse consequences to plant safety, and therefore, plant operation is acceptable with the proposed core repair installed.

2.5 Materials, Fabrication and Inspection Considerations

2.5.1 Materials and Fabrication

GPC stated in Reference 1 that Type 316 or 316L austenitic stainless steel, Type XM-19 stainless steel and nickel-based (Ni-Cr-Fe) alloy X-750 materials were selected for the fabrication of core shroud tie-rod stabilizer components. These materials have been used for a number of other components in the BWR environment and have demonstrated good resistance to stress corrosion cracking by laboratory testing and long-term service experience. Welding is not used in the fabrication and the installation of the core shroud tie-rod stabilizer, thereby, minimizing its susceptibility to intergranular stress corrosion cracking (IGSCC). The upper stabilizers, lower springs and some connecting components were made from alloy X-750. The alloy X-750 material was selected for these components because of the requirements of higher material strength and lower coefficient of thermal expansion than that of the core shroud material (Type 304 stainless steel). The tie-rods in the stabilizer assemblies were made of Type XM-19 stainless steel in a solution annealed condition with a carbon content less than 0.04%. The remaining connecting components in the tie-rod stabilizer assemblies were made from either Type 316 or 316L austenitic stainless steel with a carbon content not more than 0.02%.

GPC selected Type XM-19 instead of Type 304 or 316 stainless steel for the fabrication of tie-rods in the stabilizer assemblies because Type XM-19 material has higher resistance to sensitization, higher allowable stress and a slightly lower coefficient of thermal expansion which would increase the thermal pre-load. GPC stated that Type XM-19 was extensively tested in the mid-1970's, with the results published in Reference 17. The test results showed that Type XM-19 material has good resistance to sensitization and IGSCC. The solution annealed Type XM-19 material has been used in BWR environments with successful experience for over 20 years. The material was used for piston or index tubes in the control rod drive mechanisms and in a number of other applications.

Type 316/316L austenitic stainless steel and solution annealed alloy Type XM-19 are acceptable ASME Code Section III materials. The alloy X-750 was procured to American Society for Testing and Materials (ASTM) Standard B637, Grade UNS N07750 material (bars and forging) requirements. The heat treatment of alloy X-750 includes solution annealing at 1975°F ±25°F for 60 to 70 minutes, followed by forced air cooling, and age hardening at 1300°F ± 15°F for a minimum of 20 hours, followed by air cooling. The equalization heat treatment at 1500°F to 1800°F was prohibited because this heat treatment will produce a microstructure that would make the alloy X-750 material susceptible to IGSCC.

Type 316/316L austenitic stainless steel was procured to ASTM A-479, A-182 or A-240 with a maximum carbon content of 0.020%. The procured materials were water quenched from solution annealing at 2000°F ±100°F.

The Type XM-19 stainless steel materials were procured to ASTM specification A182, A240, A412 or A479. The materials were solution annealed at 1950°F to

1975°F, followed by forced air cooling to a temperature below 800°F in 20 minutes or less. The staff finds that the process of aircooling from the solution annealing temperature is not consistent with the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines as provided in Reference 18, where water quenching from the solution annealing temperature is specified. GPC stated that due to the straightness requirement in the fabrication of the tie-rods, it is necessary to air cool the XM-19 materials from the solution annealing temperature, because water quenching will cause excessive distortion in the materials. To support the use of air-cooled XM-19 material, GPC submitted in Reference 4 a GE report of evaluating the stress corrosion cracking of XM-19 in the BWR environment. GE's evaluation report presented several sensitization and stress corrosion studies on XM-19 and several 300 series stainless steels with various carbon contents. The results of the studies had shown that, due to its sluggish kinetics of sensitization, XM-19 exhibited good resistance to sensitization and ranked very high in stress corrosion resistance in comparison to other 300 series stainless steels. Based on the test data presented in Reference 2, the staff has determined that the air cooling rate specified in the fabrication of the tie-rods is not expected to cause any sensitization in the XM-19 material. Therefore, the staff concludes that the subject air-cooled XM-19 material is acceptable for use in the BWR environment.

All procured XM-19 and Type 316/316L stainless steel materials were tested for sensitization in accordance with ASTM Standard A262, Procedures A or E, to ensure the materials were not sensitized. The maximum hardness of the procured materials and completed parts were specified in the GE Fabrication Specification (Reference 12).

To preclude intergranular attack (IGA) as a result of high temperature annealing, GPC required IGA testing per GE E50YP11 specification to be performed for each heat and heat treat lot of materials after annealing or pickling. In lieu of IGA testing, a minimum of 0.03 inch may be removed from all surfaces after the last exposure to high temperature annealing as a control of IGA.

GPC indicated that tie-rod stabilizer assembly components are generally rough machined to within 0.10 inch of final size and skim passes are used to achieve the final dimensions. Coolant and sharp tools are used in the machining. The final machined surface finish is generally specified to be 125 root mean square or better. GPC also indicated that a Nickel-Graphite antiseize thread lubricant (D50YP5B) will be used in the installation of tie-rod stabilizer assemblies. Controls of lubricant impurities were provided in the GE Specification (D50YP12), where impurities limits were specified for halogens, sulfur and nitrates. The machining processes used for each type of material were qualified by metallographic and microhardness evaluation on test samples to minimize the cold work effect. The cold work resulting from machining is known to promote stress corrosion cracking in stainless steel. The current industry practice of re-solution-anneal of machined components is intended to remove the cold work effect. GPC does not plan to re-solution anneal any components after final machining. The staff requests that the licensee identify in their post-modification inspection plan what inspections will be performed of cold worked components that have not been re-solution-annealed.

The staff has reviewed GPC's submittal regarding the proposed core shroud repair and concludes that the selected materials and fabrication methods for the tie-rod stabilizer assemblies are acceptable.

2.5.2 Pre-modification and Post-modification Inspection

GPC's pre-modification inspection plan (Reference 1) for Hatch Unit 2 to support the repair installation consists of inspection of core shroud vertical welds and H-9 welds, and was reviewed by the staff. The selection of the welds and the scope and limitation of the inspection are briefly summarized below.

(1) Enhanced visual examination will be performed on the vertical welds intersecting H4 weld from the outside diameter (OD) surface and the inside diameter (ID) surface. About four (4) inches of each vertical weld will be examined.

(2) Enhanced visual examination will be performed on the H-9 weld from the jet pump annulus region at the four repair assembly locations. The H-9 weld connects the core shroud support plate to the reactor vessel. Approximately four (4) inches of H-9 weld at each side of the repair location will be inspected.

There are no segment welds in the core shroud support rings at Hatch Unit 2 because each support ring was made of one piece Type 304L forging. The core shroud support plate is made of 8 inches thick low-alloy steel plate and was clad with Inconel Alloy 82 and stainless steel at the top surface and bottom surface, respectively. The low-alloy steel plate is not susceptible to IGSCC and, therefore, its segment welds need not be inspected.

GPC has not yet finalized its reinspection plan for the core shroud and the tie-rod stabilizer assembly components. The staff recommends that GPC's reinspection plan should consider the following: (1) the plant-specific repair design requirements, (2) the extent and the results of the baseline inspection performed during pre-modification inspection, (3) the threaded areas and the locations of crevices and stress concentration in the tie-rod stabilizer assemblies, and (4) BWRVIP reinspection guidelines when they are established. GPC is requested to submit the Hatch Unit 2 reinspection plan for the core shroud and repair assemblies approximately 6 months after restart of Hatch Unit 2. The NRC staff will review GPC's reinspection plans when submitted. Since the core shroud and the tie-rod stabilizer assemblies are generally classified as ASME Code Class B-N-2 components (core structural support), the reinspection plan will be required to be incorporated into the plant inservice inspection (ISI) program after NRC approval.

The staff has reviewed GPC's pre-modification inspection plan. The staff concludes that the proposed inspection plan is acceptable to support the planned core shroud repair at this time. However, as discussed in Section 2.5.1, the staff requests that the licensee describe in their post-modification inspection plan, what inspections will be performed of cold worked components that have not been re-solution-annealed.

3.0 CONCLUSION

The proposed core shroud repair has been designed as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, pursuant to Title 10, Code of Federal Regulations, Part 50.55a(a)(3)(1). Based on a review of the core shroud modification hardware from structural, systems, materials, and fabrication considerations, as discussed above, the staff concludes that the proposed modifications of the core shroud is acceptable and, subject to the submittal of the post modification inspection program, will not result in any increased risk to the public health and safety.

Attachment:
Figure 1, Shroud Horizontal Welds

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

2. Letter of July 3, 1995, from J. T. Beckham, Jr., Georgia Power Company (GPC), to the USNRC Document Control Desk, relating to the Edwin I. Hatch Nuclear Plant, Unit 2 "Core Shroud Stabilizer Design Submittal" with proprietary attachments and enclosures.

3. Letter of August 17, 1995, from K. Jabbour, NRC, to J. T. Beckham, Jr., GPC, requesting GPC to provide additional information on the proposed core shroud repair design for Hatch Unit 2.

4. Letter of August 25, 1995, from J. T. Beckham, Jr., GPC, to the USNRC Document Control Desk, with enclosures and proprietary attachments, providing responses to request for information from the NRC on the Hatch Unit 2 core shroud modification.

5. GENE-B11-00637-006, Revision 0, "GE Responses to NRC Questions, Hatch Unit 2 Shroud Repair," June 1995.

6. GENE Specification 25A5718, Revision 0, "Shroud Repair Hardware, Design Specification," Hatch Unit 2, May 1995.

7. GENE Specification 25A5717, Revision 1, "Shroud Stabilizers, Code Design Specification," Hatch Unit 2, June 1995.

8. GENE-B11-00637-003, Revision 0, "Edwin I. Hatch Nuclear Plant Unit 2 Shroud Repair Seismic Analysis Report for OBE, DBE & 1/2 SME," May 1995.

9. GENE-B11-00637-002, Revision 0, "Shroud Mechanical Repair Program, Hatch Unit 2, Shroud and Shroud Repair Hardware Stress Analysis," June 1995.

10. GENE Specification 25A5721, Revision 0, "Shroud Stabilizers, Stress Report, Hatch Unit 2," June 1995.

11. GENE-B11-00637-005, Revision 0, "Safety Evaluation for the Installation of Stabilizers on the Edwin I. Hatch Nuclear Plant Unit 2 Core Shroud."

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13. GENE FDI No. NT2-0121-12900, Revision 0, "Field Disposition Instruction, Hatch Unit 2, Shroud Repair Program," June 1995.

14. Seismic Margin Assessment of the Edwin I. Hatch Plant" EPRI Report NP-7217-SL, June 1991.

15. GE Document GENE-523-A107P-0794, Revision 1, "BWR Shroud Cracking Generic Safety Assessment," August 1994.

16. GE Document GENE-771-44-0894, Revision 2, "Justification of Allowable Displacements of the Core Plate and Top Guide Core Shroud Repair," November 16, 1994 (Proprietary).

17. GE Document NEDE-21653-P, "XM-19 Materials Qualification Report," July 1997, (Proprietary).

18. Boiling Water Reactor Owners Group, Vessels and Internals Project (BWROG-VIP) Document BWROG-VIP-9410, "BWR Core Shroud Repair Design Criteria," Section 5.10.7, August 18, 1994.

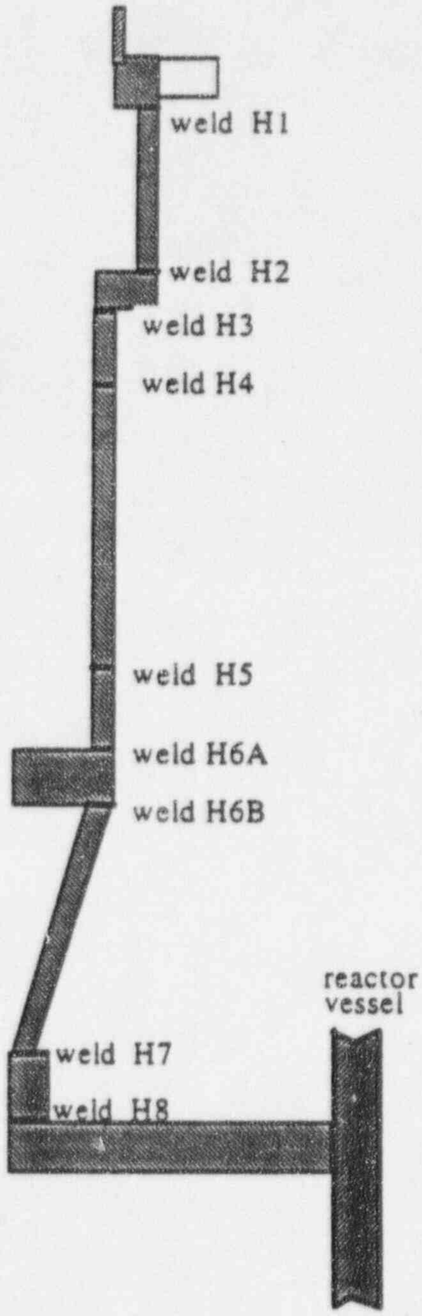

FIGURE 1 SHROUD HORIZONTAL WELDS

Attachment