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Docket No. 50-321

Mr. J. T. Beckham, Jr.
Vice President, Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

During the current maintenance/refueling outage for the Edwin I. Hatch Nuclear Plant, Unit No. 1, inspection of Recirculation (RECIRC) and Reactor Heat Removal (RHR) system piping revealed a number of unacceptable ultrasonic indications. You took corrective action to repair this piping and reported the results of the inspection, analysis and repairs in a letter to the NRC dated January 27, 1983. Based on our review of your report and related discussions with your staff, we prepared the enclosed Safety Evaluation of the inspection, analysis and repair of the Hatch RECIRC and RHR piping. We found that the repairs were sufficient but that additional leak detection requirements should be implemented prior to startup of the plant and that you should be required to submit a plan prior to the next refueling outage for inspection of the RECIRC and RHR piping during that outage.

We discussed these findings with you and you agreed to implement them. In a subsequent letter, dated February 10, 1983, you: (1) submitted a proposed modification to the leak detection Technical Specifications (TSs) to implement this agreement and (2) committed to provide an augmented inservice inspection program to the NRC three months prior to the next scheduled maintenance/refueling outage.

We have reviewed the proposed TS change and the commitment concerning submittal of the inspection program and found that they are consistent with the requirements specified in the Safety Evaluation and, on this basis, conclude that they are acceptable.

The Commission has therefore issued the enclosed Amendment No. 93 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The amendment consists of the addition of a condition to the license and changes to the TSs.

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The amendment revises the license for Hatch Unit No. 1 to: (1) add a condition concerning submittal of plans for inspection of the RECIRC and RHR piping systems during the next refueling outage for Commission review and approval, and (2) change the TSs to augment the leak detection requirements.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

*ORIGINAL SIGNED BY
JOHN F. STOLZ*

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 93
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

As to form of Amdt + F.R. Notice

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	GRIngram	GRivenbark/cb	JStolz	GLainas			
DATE	2/11/83	2/11/83	2/11/83	2/11/83	2/11/83		

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated February 10, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-57 is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and by revising paragraph 2.C.(2) and adding paragraph 2.C.(5) to read as follows:

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 93, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

2.C.(5) The licensee shall submit, for the Commission's review and approval, plans for inspection and/or modification during the next refueling outage (following Cycle 7 operation and prior to startup for Cycle 8 operation) of the Recirculation and Reactor Heat Removal Systems piping. These plans shall be submitted to the Commission at least three months prior to the start of the next refueling outage.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: FEBRUARY 11 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

Remove	Insert
3.6-7	3.6-7
3.6-8	3.6-8
--	3.6-8a (new page)

3.6.F.2.c. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.6.F.2.c.3. Primary coolant pH shall be measured at least once every 8 hours whenever reactor coolant conductivity is >2.0µmho/cm at 25°C.

d. Whenever the reactor is not pressurized, a sample of the reactor coolant shall be analyzed at least every 96 hours for chloride ion content and pH.

G. Reactor Coolant Leakage

1. Unidentified and Total

Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F:

a. reactor coolant system leakage into the primary containment from unidentified sources shall not exceed 5 gpm when averaged over a 24 hour period;

b. reactor coolant system leakage into the primary containment from unidentified sources shall not increase more than 2 gpm when averaged over a 24 hour period; and

c. the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm when averaged over a 24 hour period;

when checked in accordance with 4.6.G.

2. Leakage Detection Systems

a. At least one of the leakage measurement instruments associated with each sump shall be operable and two of the other three leakage detection systems identified in Table 3.2-10, note c shall be operable when irradiated fuel is

G. Reactor Coolant Leakage

Unidentified sources of reactor coolant system leakage shall be checked by the drywell floor drain sump system and recorded at least once per 4 hours. Identified sources of reactor coolant system leakage shall be checked by the equipment drain sump system and recorded at least once per 4 hours. The readings provided by the primary containment atmosphere particulate radioactivity monitoring system, the primary containment radioiodine monitoring system, and the primary containment gaseous radioactivity monitoring system shall also be recorded at least once per 4 hours.

G. Reactor Coolant Leakage

2. Leakage Detection Systems (Cont'd)

a. (Continued)

in the reactor vessel and reactor coolant temperature is above 212°F. Further, the primary containment atmosphere particulate radioactivity monitoring system shall be among these two operable systems, or samples shall be obtained and analyzed at least once each 4 hours.

b. From and after the date that any two of the four systems identified in Table 3.2-10, note c are made or found to be inoperable, but with the primary containment atmosphere particulate radioactivity monitoring system operable, reactor power operation may continue for the succeeding 30 days provided the primary containment atmosphere particulate radioactivity monitoring system reading is checked and recorded at least once each 4 hours.

c. From and after the date that any two of the four systems, including the primary containment atmosphere particulate radioactivity monitoring system, identified in Table 3.2-10, note c are made or found to be inoperable, reactor power operation may continue for the succeeding 30 days provided samples of the containment atmosphere are obtained and analyzed at least once each 4 hours.

G. Reactor Coolant Leakage

3. Shutdown Requirements

a. If the conditions of 3.6.G.1.a or 3.6.G.1.c cannot be met, reactor coolant system leakage will be reduced to within the specified limits within 4 hours or an orderly shutdown shall be initiated. If the condition of 3.6.G.1.b cannot be met, the source of reactor coolant leakage shall be identified or reduced within 4 hours or an orderly shutdown shall be initiated. The reactor shall be in the Hot Shutdown condition within the next 12 hours and in the Cold Shutdown condition within the following 24 hours.

b. If the conditions of 3.6.G.2 cannot be met, Specification 3.6.G.3.a shall apply unless an inoperable system is sooner made operable.

c. If three of the four leak detection systems are made or found to be inoperable, Specification 3.6.G.3.a shall apply.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSPECTION, ANALYSES AND REPAIR OF RECIRCULATION AND BWR SYSTEMS PIPING

AT EDWIN.I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

Introduction

During the current 1982 maintenance/refuel outage, augmented inservice inspection was performed on 28 austenitic stainless steel welds in the 4", 12", 22" and 28" recirculation (RECIRC) piping system and 11 austenitic stainless steel welds in the RHR piping system. The results of ultrasonic tests (UT) indicated that three RECIRC welds (two 22" end cap to pipe welds and one 22" sweepolet to manifold weld) and two RHR welds (one 20" elbow to pipe weld and one 24" pipe to pipe weld) showed reportable linear indications. Because of the linear indications found on two 22" RECIRC end cap to manifold welds, the other two end cap to manifold welds were also ultrasonically examined and were found to show linear indications. Subsequently, an additional 19 stainless steel welds in the 12", 22" and 28" RECIRC piping systems were selected for ultrasonic inspection and no reportable linear indications were found.

Overall, a total of 7 welds (5 welds in the RECIRC system and 2 welds in the RHR system) were found to show linear indications. All indications were reported to be located in the general area of the base material heat affected zone (HAZ).

During this outage, a total of 57 welds in the RECIRC and RHR piping systems were ultrasonically inspected. The selection of the welds for inspection was based on ASME Section XI requirements or the NUREG 0313 Revision 1 guidelines including the consideration of stress rule index, carbon content and the IGSCC experienced by other BWR plants. Southern Company Services (SCS) and Southwest Research Institute (SWRI) performed the ultrasonic tests for Georgia Power Company. Their UT procedures and calibration standards were satisfactorily evaluated on IGSCC cracked pipe samples at Battelle-Columbus in accordance with I&E Bulletin 82-03.

NuTech has evaluated the flaws found in the seven welds mentioned above. Code stress analysis and various fracture mechanics analysis were performed. The results of the evaluation indicate that six of the seven flawed welds required weld overlay repair to restore the original design safety margins. The 22" RECIRC sweepolet to manifold weld was shown by analysis to continue to have the original design safety margin for at least one fuel cycle.

The six welds being reinforced by a weld overlay of IGSCC resistant materials are four 22" RECIRC end cap to manifold welds, one 20" RHR elbow to pipe weld, and one 24" RHR pipe to pipe weld. The design life of each overlay repair was calculated by NuTech to be at least 5 years. Acoustic emission (AE) devices with automatic strip chart recording will be installed on the unrepaired sweepolet to manifold weld to enhance the capability of the leak detection system to detecting small leaks.

After all weld overlay repairs, the code required hydrostatic test and nondestructive examination will be performed.

Description of Cracks

The reportable linear indications found on 7 stainless steel welds in RECIRC and RHR piping systems are described in detail in the Attachment to the Georgia Power Company submittal of January 27, 1983. These indications are located generally in the base material of the heat affected zone (HAZ) and are characterized predominantly as short axial or transverse cracks. Only two circumferential cracks were found; these were in the RHR system, at an elbow to pipe weld. These were determined to be 1 1/2 inches long, not over 33% of the wall thickness in depth. Numerous short, but deep axial cracks, were found in all 4 end cap to manifold welds with the lengths varying from 1/4 inch to 1/2 inch and with depths up to 72% of wall thickness. Seven short transverse cracks (1/4 to 1/2 inch) were found in the sweepolet to manifold weld. The depth of these cracks is reported to be very shallow, not exceeding 12% of the wall thickness. The axial cracks found in the RHR pipe to pipe weld are also short

(1/4 to 1/2 inch long) with the depth not exceeding 47% of wall thickness. The deepest axial crack reported was 94% of the wall thickness and about 3/8 inches long; it was found at the elbow to pipe weld.

Short, deep axial cracks have been noted previously, and leaks emanating from them were noted and reported at Quad Cities 1 in 1980, and Monticello in 1982. They probably occur in locations with high residual welding stresses in the circumferential direction. They are typically short because the sensitized heat affected zone (HAZ) extends less than 1/2 inches on either side of the weld, and integranular stress corrosion cracking requires sensitization to be present. As noted at Quad Cities 1, however, such cracks can propagate into and through the weld ,if it has high carbon and low ferrite.

Axial cracks are of much less concern from a safety standpoint than circumferential cracks, which can grow through the wall and around the circumference of the pipe, for two reasons. First, the service stress on the axial crack is almost all caused by pressure, and typically the pressure stress is low compared to the total stress acting on a circumferential crack, where bending stresses can be significant. Second, because IGSCC is confined to the sensitized material of the narrow HAZ, axial cracks cannot grow to significant lengths.

Axial or radial cracks, if short, are very difficult to detect and size by UT because they form under the crown of the weld, and it is usually difficult to direct the sound beam at the proper angle. They often can only be detected at very limited transducer locations. It should also be mentioned that because they can only be short in relation to the wall thickness, and the stresses tending to open them are low, they will cause very little actual leakage, perhaps not enough to be detected with normal procedures.

In summary, although axial or radial IGSCC cracks are hard to find and size by UT, they will cause only small leaks and will not grow long enough to initiate a pipe burst unless the piping itself is completely sensitized.

Description of the Overlay Reinforcement

The weld overlay was installed by depositing IGSCC resistant 308 L weld metal 360 degrees circumferentially around the pipe. The weld deposited band over the cracks reinforces the pipe and introduces beneficial reduced stresses. The minimum overlay thickness was selected to restore the original design safety margins. The minimum overlay length used was equal to $\sqrt{rt}$, where r is the mean radius of the pipe and t is the wall thickness. This is the minimum length required to ensure that there would be no adverse end effects.

The ends of the overlay are tapered at a nominal angle of 45 degrees. The overlays on the end cap to manifold welds (RECIRC), elbow to pipe weld (RHR) and the pipe to pipe weld (RHR) have a nominal thickness of 0.25 inch, 0.4 inch and 0.3 inch respectively; and their respective lengths are 6.5 inches, 7 inches and 10 inches. During overlay welding of the RHR pipe to pipe weld, toe cracks were observed adjacent to one end of the overlay. The toe cracks with a depth approximately 1/32 to 1/16 inch were formed due to portion of the stainless steel overlay being deposited on a neighboring Inconel weld. The overlay was subsequently extended 2 inches to a total length of 10 inches to cover completely the area where the toe cracks were removed by grinding. Inconel weld material was used for the last portion of overlay which overlapped the neighboring Inconel weld.

Effect of Overlay Repair on the Recirculation or RHR Systems

The weld overlay repair causes both an axial and radial shrinkage underneath the overlay. The shrinkage induces beneficial residual compressive stresses in the cracked pipe, but may adversely affect the weld joints in other locations of the systems if the shrinkage is of significant magnitude. These effects have been evaluated by NuTech for GPC and are judged to have no deleterious effects on the recirculation or RHR systems. The results are summarized below.

The effects of radial shrinkage are limited to the regions adjacent to and underneath the overlay. NuTech indicated that based on their work performed for Monticello, the radial shrinkage stresses are less than yield stress at distances greater than 4 inches from the end of the overlay.

NuTech has evaluated the effect of the weld overlay axial shrinkage on the Recirculation and RHR systems by using their computer program PISAR. The 4 end cap weld overlays are adjacent to the recirculation manifold free ends and will not induce stress in any other section of the piping. In the RHR system, the axial shrinkage of the elbow weld overlay and pipe to pipe weld overlay was measured to be 0.25 inch and 0.19 inch, respectively. The measured axial shrinkage is imposed in the model as boundary condition during evaluation. The maximum secondary stress calculated from the model is less than 9 ksi and NuTech considers this to be acceptable.

We have noted that the flawed sweepolet to manifold weld is only 2 inches away from a nearby end cap to manifold weld overlay. There is a potential concern that the radial shrinkage of the end cap overlay may affect the flaw in the sweepolet weld. We have concluded that this will not present a serious safety concern for the following reasons:

(1) Although the unrepaired sweepolet weld is only about 2 inches away from the end cap to manifold repair overlay, the identified cracks in the sweepolet weld are at least 8 inches away. The magnitude of the radial shrinkage stresses at this distance are not expected to be of major significance.

(2) The cracks found in the sweepolet to manifold weld are all short cracks transverse to the weld. As previously discussed, transverse cracks will not grow to any significant lenth as their length is limited to the width of the sensitized heat affected zone.

(3) The bending stress induced by the weld overlay is displacement controlled (self equilibrating) and would tend to be relieved by initiation of cracking.

(4) As will be discussed later, GPC is in the process of installing two acoustic emission (AE) devices on the unrepaired sweepolet to manifold welds to detect the leakage. The AE devices are very sensitive and leakage as small as 0.1 gpm can readily be detected. Therefore, any small leakage emanating from the unrepaired sweepolet will immediately be detected.

Code Stress Analysis

The repaired piping was evaluated according to Section III, and was found to meet all requirements including seismic and fatigue requirements. This was done by conservatively developing a finite element model with the use of ANSYS computer program. In the model, the material was removed to represent the cracks. Although the geometrical configuration is not typical of Code design, the stress analysis was performed using the Code rules. The fatigue analysis used the standard set of transient conditions which consist of 38 startups, 25 small temperature change cycles and one emergency cycle every five years, and included a strength reduction factor of 5 in the calculation. The calculations show that the weld overlay repaired end cap to manifold welds, elbow to pipe weld and pipe to pipe weld will meet all Code requirements for at least 5 years.

Fracture Analysis

Background

NuTech performed the following three types of fracture analyses to show that the safety margins against failure are at least equivalent to the margins inherent in the ASME Code.

Allowable Crack Depth Evaluation

The calculation of allowable crack depth is based on a new proposed flaw evalaution methodology for Section XI of the Code. This includes IWB 3640, "Acceptance Criteria for Flaws in Austenitic Stainless Steel Piping," and the associated Appendix C, "Evaluation of Flaws in Austenitic Stainless Steel Piping." Although these new sections have not yet been approved through the Main Committee, they have been approved through the first levels, and full approval is expected shortly.

The basis for this criterion is the well known and accepted limit load for plastic collapse method of analysis. Specific development of this method for the evaluation of flaws in stainless steel piping has been done under EPRI contracts, and has been described in several reports, including References 1 and 2. For Code use, this calculational method has been used to develop simple tables, from which acceptable flaw sizes and shapes as a function of applied stresses can be read directly. These are Tables IWB 3642-1 and -2 for axial cracks, and Tables 3641-1 and -2 for circumferential cracks. There are separate tables for Normal Conditions and Emergency and Faulted Conditions, with different safety margins. The tables provide a safety margin of between 2.5 and 3 for Normal Conditions, and about 1.5 for Emergency and Faulted Conditions. These are consistent with the overall basis of the Code.

It is noted that the presence of more than one crack does not change the calculations. Multiple axial cracks do not interact, and are treated separately.

Crack Growth Evaluation

The crack growth due to fatigue and IGSCC is calculated based on rules provided in Appendix C to the proposed IWB-3640 for Section XI of the Code. The methodology for evaluating fatigue propagation appears acceptable, but we still have some reservations about the IGSCC crack growth rate given in the Code. This is of no concern for the repaired cracks and unrepaired crack at Hatch Unit 1 as will be described later.

Ultimate Failure Load

The ultimate failure load is calculated with a tearing modulus analysis. This type of elastic-plastic fracture mechanics was initially developed on an NRC contract, and has been widely accepted and used during the past 5 years. It is recognized that the limit load approach is conservative, and that much larger margins are actually present in many cases.

Tearing Modulus calculations were performed for both the repaired welds (end caps, elbow and pipe to pipe) and the unrepaired sweepolet weld. As expected, the calculations show that very large margins against failure are present.

End Cap to Manifold Welds Repair Evaluation

The flaws found in end cap to manifold welds are all short axial cracks. The largest axial crack has a length of 1/2 inch and a depth of 72% wall thickness. NuTech performed an Appendix C evaluation of the most limiting flaw in the repaired end cap to manifold welds. The thickness of the overlay pipe wall is 1.24 inch with a minimum overlay thickness of 0.25 inch. The allowable crack depth for the end cap to manifold welds is determined to be 75 percent of the wall thickness from Table IWB-3642-1. This corresponds to a crack depth of 0.93 inch in the weld repaired by overlay. NuTech calculated the crack growth due to fatigue for 5 years of operation to be less than 0.05 inch.

They also calculated the crack growth due to IGSCC for two cases. The first case was calculated by conservatively assuming an infinitely long crack and by considering beneficial residual stess due to the weld overlay. In this case, the crack will grow to a depth of approximately 0.85 inch in five years, which is below the allowable crack depth of 0.93 inch. The second case considers the worst case for the end caps by assuming an axial crack completely through the original pipe wall. The crack will not propagate into the overlay weld material due to its high resistance to IGSCC but will grow approximately 0.05 inch due to fatigue in five years of operation. Thus, the total crack depth is about 1.05 inch which is about 82% of the overlay pipe wall thickness. This exceeds the allowable crack depth by 7 percent. However, the calculations for the allowable depth of the crack are overly conservative in this case, because the Code arbitrarily cuts off the allowable depths given in the tables for axial cracks to 75% of the wall thickness. Extrapolation of the values in the table would show the allowable crack depth in excess of 82%. Therefore, the overlay design is judged to be acceptable for 5 years.

We have reviewed NuTech's Appendix C calculation and agree with their conclusion regarding the acceptability of the overlay design based on the net section limit load analysis.

NuTech also performed a tearing modulus evaluation of the repaired end cap to manifold welds based on the postulation that the largest size the existing crack could reasonably be expected to grow to be a one inch radius flaw. The predicted burst pressure (ultimate failure load) based on this flaw configuration is in excess of 5500 psig which is more than 4 times of the design pressure. This indicates that the design pressure, even in the presence of this worst case assumed crack, has a safety factor well in excess of that inherent in the ASME Code.

RHR System 20 mil Elbow to Pipe Weld Repair Evaluation

The largest cracks found in the elbow to pipe weld are an axial crack of depth 94% of wall thickness and length of 3/8 inch, and a circumferential crack of length 1-1/2 inches and depth of 33% of wall thickness. NuTech performed similar calculations to show the acceptability of the elbow to pipe weld repair.

The overlay has a minimum thickness of 0.4 inch and the repaired pipe section has a minimum thickness of 0.76 inch. The allowable crack depth determined from Tables IWB-3642-1 and IWB-3641-1 is 75% of wall thickness for both the most limiting axial and circumferential cracks found in elbow to pipe weld. The largest axial crack is essentially through the pipe wall and will propagate only by fatigue to a distance less than 0.05 mils. The axial crack depth after five years of operation would be 0.81 inch (70% of overlaid wall) which is less than the allowable crack depth of 75%. The circumferential crack depth after 5 year's crack growth due to fatigue and IGSCC is about 26% of the overlaid wall thickness which is substantially less than the allowable of 75%. NuTech also performed a Tearing Modulus analysis based on a postulated radius flaw of 0.8 inch. The predicted ultimate failure load is in excess of 3 times the normal operating loads which provides a safety factor on normal operating loads larger than that inherent in the ASME Code, even in the presence of this worst case assumed crack.

We have reviewed NuTech's Appendix C calculation and agree with their conclusion that the overlay design is acceptable based on the net section limit load analysis. We also note that there are 2 circumferential cracks, each with a length of 1.5 inches in the elbow to pipe weld. These two circumferential cracks are located on different sides of the weld. It is very unlikely that the two circumferential cracks are linked to each other. We made a calculation based on Table IWB-3641-1 to show that with a crack depth of 40% wall thickness, the allowable length of circumferential crack can be half of the pipe circumference (~ 33 inches). Therefore, even if the two circumferential cracks are linked together, it will still meet the Appendix C requirement.

Pipe to Pipe Weld Repair Evaluation

The flaws found in pipe to pipe weld are all short axial cracks. The largest axial crack has a length of 3/8 inch and a depth of 47% wall. The overlay applied has a minimum thickness of 0.3 inch and the repaired pipe section has a minimum wall thickness of 1.14 inches. NuTech performed similar calculations to show the acceptability of the pipe to pipe weld repair. The allowable crack depth determined from Table IWB-3642-1 is 75% of wall thickness for the most limiting crack found in the pipe to pipe weld. NuTech calculated the crack growth due to fatigue and IGSCC in the next five years. The calculated crack growth is small and will not exceed the allowable crack depth. NuTech also performed tearing modulus analysis based on a postulated 1.14 inch radius flaw. The predicted ultimate failure load is in excess of 4 times the normal operating loads which provides a safety factor on normal operating load larger than that inherent in the ASME Code, even in the presence of this worst case assumed crack.

We have reviewed NuTech's Appendix C calculation and agree with their conclusion that the overlay design is acceptable based on the net section limit load analysis.

Sweepolet to Manifold Weld Evaluation

Seven short transverse cracks (perpendicular to the weld) were found in the sweepolet to manifold weld. The largest transverse crack is approximately 1/2 inch in length with a depth approximately 12% of the pipe wall. Due to the difficulty in applying overlay in this pipe branch area, the flawed sweepolet to manifold weld was not repaired. NuTech performed similar Appendix C calculation to show that the subject sweepolet to manifold weld is acceptable for operation at least for one fuel cycle in the present unrepaired condition. The calculation showed that the allowable crack depth is 75% of pipe wall and the maximum crack depth after 5 years of operation is predicated to be 0.38 inch (38% of wall thickness), which is well below the allowable crack depth. NuTech also performed a tearing modulus analysis with a postulated 0.50 inch radius flaw in the weld. This flaw is larger than any existing crack will grow to become in one fuel cycle based on the crack growth rate used for Appendix C. The predicted ultimate failure load is in excess of 3.3 times the normal operating loads which provides a safety factor on normal operating load larger than that inherent in the ASME Code even in the presence of this worst case assumed crack.

We have received NuTech's evaluation and have concluded that the continuous operation of Hatch Unit 1 for at least one fuel cycle with the subject sweepolet to manifold weld in the as flawed condition does not represent a safety concern, and is acceptable provided that augmented leak detection requirements which will be discussed later, are implemented. The bases for our conclusion are:

(1) The cracks found in the sweepolet to manifold weld are all short transverse cracks. Any growth in the crack length by IGSCC is limited by the width of the sensitized HAZ, which is generally not over 1/2 inch.

(2) Assuming the unlikely event that the cracks eventually grow through the wall, the leakage emanating from such short transverse cracks will be relatively small and will not cause any significant loss of reactor coolant.

(3) As will be discussed later, the licensee is in the process of installing 2 Acoustic Emission (AE) devices adjacent to the subject sweepolet to manifold weld. AE is a very sensitive device capable of detecting very tiny steam leaks. In the unlikely event of a through wall crack, this device will provide an early warning to the operator to initiate appropriate corrective action.

(4) As will be discussed later, the licensee will implement augmented reactor coolant leakage detection requirements, which include more frequent monitoring and more restrictive leakage limits.

Conclusion of the Fracture Analysis Review

The safety margins provided by the overlay repair to the cracked end cap to manifold welds, elbow to pipe weld and pipe to pipe weld and the safe margins of the unrepaired sweepolet to manifold weld are shown by the proposed Code rules to be acceptable. Crack propagation to the extent of leakage is considered very unlikely.

Tearing modulus analyses of cracked welds show that even larger safety margins exist than are inherent in the Code approach.

Augmented Leak Detection

The Hatch Unit 1 plant Technical Specifications require the operator to initiate corrective action when 5 gpm unidentified leakage is detected. NuTech performed an evalaution based on the leak rate calculated in reference (3) and concluded that there is considerable margin between the crack length to produce 5.0 gpm leakage and the critical crack length. We have reservations regarding these calculations; therefore, we consider that tighter leak rate limits should be imposed. Specifically, we are concerned about the following:

(1) IGSCC cracks are known to be very tight and branched. Preliminary experimental data provided by Argonne National Laboratory have shown that the leakage rate from IGSCC can be less than is usually assumed or calculated.

(2) All reactor water leaked from the pipe during normal operating conditions will not be collected by the sump monitoring system. A large portion of the leakage will be flashed into steam or evaporated before reaching the sump. Therefore, for the sump monitoring system to register 5 gpm leakage, more than 5 gpm has to be leaked out from the pipe.

In view of the above considerations, we have determined that the licensee should augment his leak detection procedures in accordance with the recommendations in NUREG-0313, Rev. 1, by implementing the following items prior to the start-up of Hatch Unit 1:

a. An additional operational limit on reactor coolant system leakage of an increase in unidentified leakage of two gallons/minute or more within any 24-hour period. On exceeding this limit, or the existing limits of 5 gallons/minute unidentified leakage or 25 gallons/minute total leakage (averaged over a 24-hour period), the reactor will be placed in a cold shutdown condition within 24 hours for inspection.

b. Drywell leakage will be measured and recorded every four hours.

c. At least one of the leakage measurement instruments associated with each sump will be operable.

d. The drywell atmospheric particulate radioactivity monitoring system will be operable or a sample shall be taken and analyzed every four hours.

We conclude that implementation of the above measures will provide additional assurance that possible cracks in pipes will be detected before growing to a size that will compromise the safety of the plant.

In addition, during the licensee's presentation to NRC on February 1, 1983, the licensee has decided to install two local acoustic emission devices on the flawed sweepolet to manifold weld to monitor the potential leakage. This system could provide even more assurance that small leaks will not go undetected.

Inservice Inspection Plan

The licensee proposed the following future inspection plans for the austenitic stainless steel reactor coolant pressure boundary piping in their submittal of January 27, 1983:

(1) The six overlay repaired welds and 50% of the recirculation sweepolet to manifold welds including the one found to have cracks will be examined during the next three successive scheduled refueling outages.

(2) All other stainless steel welds will be examined in accordance with the licensee's June 29, 1981 response to NRC Generic Letter 81-04 dated February 26, 1981, regarding the implementation of NUREG 0313, Rev. 1.

We have reviewed the licensee's proposal and have determined that their proposed inspection plan is not adequate especially regarding the welds in the recirculation piping system. As NRC is in the process of revising NUREG 0313, Rev. 1, the licensee's June 29, 1981 response to NRC Generic Letter 81-04 will be addressed separately at a later date.

Due to the recent occurrences of IGSCC in recirculation lines in several BWR plants including Hatch Unit 1, such lines must be considered "service-sensitive," and augmented ISI must be consistent with recommendations of NUREG 0313, Rev. 1. Therefore, to increase the assurance of the integrity of recirculation piping in Hatch Unit 1, we have determined that an augmented ISI program similar to that delineated below should be carried out during the next refueling outage.

(1) The six overlay repaired welds and the one unrepaired sweepolet to manifold weld should be ultrasonically examined.

(2) A minimum of ten welds in recirculation piping of 20 inches diameter, or larger should be ultrasonically examined. Those circumferential welds not ultrasonically inspected during 1982 refueling outage should be selected for inspection first.

(3) A minimum of ten welds of the jet pumps inlet riser piping and associated safe-ends should be ultrasonically examined. Those circumferential welds not ultrasonically inspected during 1982 outage should be selected for inspection first.

(4) Stainless steel piping welds in other systems should be examined in accordance with the guidelines provided in the NUREG 0313, Rev. 1 or its subsequent revision as appropriate.

Summary and Conclusion

We have reviewed Georgia Power Company's submittal dated January 27, 1983 regarding the actions taken during this refueling outage on the analyses and repairs of recirculation and RHR piping system welds in the Hatch Unit 1 plant. This includes description of the defects found, description of repairs performed, stress and fracture analysis and future inspection plan.

We conclude that the Hatch Unit 1 plant can safely return to power and operate in its present configuration at least until the next refueling outage, provided the following items are satisfactorily completed prior to startup:

(1) The Code-required hydrostatic test and nondestructive examination on overlay repaired welds should be satisfactorily completed.

(2) The additional leak detection requirements as listed in the section on Augmented Leak Detection should be properly implemented.

Nevertheless, we still have concern regarding the long term growth of small IGSCC cracks that may be present but not detected during this inspection.

Therefore, we require that plans for inspection in accordance with the requirements provided herein and/or modification of the recirculation and RHR piping systems during the next refueling outage be submitted for our review and comment before the start of the outage.

References

Reference 1. EPRI NP-2472-SY "The Growth and Stability of Stress Corrosion Cracks in Large-Diameter BWR Piping", July, 1982.

Reference 2. EPRI NP-2705-SR "Structural Mechanics Program: Progress in 1981, October, 1982.

Reference 3. EPRI NP-2472, "The Growth and Stability of Stress Corrosion Cracks in Large-Diameter BWR Piping," July, 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 93 to Facility Operating License No. DPR-57, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, which added a license condition and revised Technical Specifications (TSs) for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility) located in Appling County, Georgia. The amendment is effective as of the date of issuance.

The amendment revises the license for Hatch Unit No. 1 to: (1) add a condition concerning submittal of plans for inspection of the Recirculation and Reactor Heat Removal Piping Systems during the next refueling outage for Commission review and approval, and (2) change the TSs to augment the leak detection requirements.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 10, 1983, (2) Amendment No. 93 to License No. DPR-57, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 11th day of February 1983.

FEBRUARY 1 1 1983

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing