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Georgia Power
the southern electric system

L. T. Gucwa
Manager Nuclear Engineering
and Chief Nuclear Engineer

NED-84-509

September 25, 1984

U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 2900
101 Marietta Street, NW
Atlanta, Georgia 30323

REFERENCE:
RII: RCL
50-321
Inspection Report
84-30

ATTENTION: Mr. James P. O'Reilly

GENTLEMEN:

The following information is submitted in response to Inspection Report 84-30, which concerns the inspection conducted by Messrs: R. V. Crlenjak and P. Holmes-Ray of your office from July 21 to August 20, 1984. One apparent violation was identified.

VIOLATION:

"Technical Specification 6.8.1.a requires that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

Contrary to the above, procedures were not properly implemented in that on August 1, 1984, two snubbers were removed from Unit 1 Residual Heat Removal (RHR) system in violation of work instructions (MR 1-84-3763 work process sheet) which required the worker to "obtain necessary clearances and notify responsible engineer". No clearances were obtained prior to removal of snubbers E11-RHRH-193 or E11-RHRH-199.

This is a Severity Level IV violation (Supplement I)".

RESPONSE:

Admission or denial of alleged violation: The violation occurred.

Reason for the violation: The violation resulted from a failure of contractor personnel to follow written and verbal instructions. The contractor, Reactor Control Incorporated (RCI), removed the two snubbers from the operable RHR system without obtaining the proper clearances from Operations personnel. The controlling document for the snubber work was Maintenance Request (MR) 1-84-3763, issued on July 13, 1984.

Georgia Power



RESPONSE: (Continued)

The work process sheet for the snubber work required that the necessary clearances be obtained prior to modifying the snubbers. In addition, the contractor was verbally cautioned against performing work which could make the snubbers inoperable. Both the written and verbal instructions were violated when the two snubbers were removed without the necessary clearances.

Corrective steps which have been taken and the results achieved: Snubbers E11-RHRH-193 and E11-RHRH-199 are listed in Table 3.6.L of Unit 1 Technical Specifications and are therefore required to be operable during power operation. Removal of these two snubbers placed the Unit in an action statement requiring cold shutdown within 36 hours. Immediate action was taken to re-install the snubbers. Snubber E11-RHRH-193 was re-installed within 32 hours and Snubber E11-RHRH-199 within 37 hours. Limiting conditions for operation were not exceeded because the absence of a single snubber places the unit in an action statement requiring replacement of the snubber within 72 hours.

Corrective steps which will be taken to avoid further violations: A meeting was conducted with RCI personnel to emphasize equipment clearance and tagging procedures, welding procedures, and quality assurance concepts. RCI was directed to contact the designated Georgia Power Company representative prior to performing any installations or removals. A letter was sent to RCI management stressing the importance of plant procedures.

Date when full compliance will be achieved: Full compliance was achieved on August 2, 1984 when both snubbers were replaced and the contractor personnel informed of plant procedural requirements.

Please contact this office if you have any questions.

Very truly yours,

William E. Burns /for

L. T. Gucwa

JH/mb
xc: J. T. Beckham, Jr.
H. C. Nix, Jr.
Senior Resident Inspector

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L. T. Gucwa
Manager Nuclear Engineering
and Chief Nuclear Engineer

NED-84-506

September 26, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKET 50-321
OPERATING LICENSE DPR-57
EDWIN I. HATCH NUCLEAR PLANT UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON 1984 REFUELING OUTAGE INSPECTION PROGRAM

Gentlemen:

Georgia Power Company (GPC) hereby provides the following information in response to the September 17, 1984 telecopy from the Plant Hatch NRC Licensing Project Manager, Mr. G. Rivenbark, requesting additional information concerning our May 31, 1984 submittal on the 1984 Plant Hatch Unit 1 refueling outage inspection plans for stainless steel piping. The four topics addressed in the telecopy were sampling plan, qualification of examination personnel, leak detection and leakage limits, and plans for other inspections in selected components as a result of IGSCC observed at other utilities.

SAMPLING

The number of welds scheduled to be examined by size and category as identified in NRC Generic Letter 84-11 are:

Welds Not Examined Previously

| | | |
|---|---|---|
| 4" Recirc | - | None to be examined, 100% examined during previous outage |
| 12" Recirc | - | 6 welds to be examined |
| 22" Recirc | - | None to be examined, 100% examined during previous outage |
| 28" Recirc | - | 6 welds to be examined |
| 20" RHR | - | None to be examined, 100% examined during previous outage |
| 24" RHR | - | None to be examined, 100% examined during previous outage |
| 6" RWCU | - | 3 welds to be examined |

Subtotal - 15 welds to be examined

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Welds Examined Previously

4" Recirc - 1 weld to be examined (bypass removed during 1977 refueling outage)
12" Recirc - 5 welds to be examined
22" Recirc - 3 welds to be examined (in addition, 5 more welds are to be examined due to weld overlay repairs or being left unrepaired - see below)
28" Recirc - 2 welds to be examined
20" RHR - 1 weld to be exami ed (in addition, 1 more weld to be examined due to weld overlay repair - see below)
24" RHR - 1 weld to be examined (in addition, 1 more weld to be examined due to weld overlay repairs - see below)
6" RWCU - 2 welds to be examined

Subtotal - 15 welds to be examined

Overlay Repaired Welds (Note: affected piping size only shown)

22" Recirc - 4 welds to be examined
20" RHR - 1 weld to be examined
24" RHR - 1 weld to be examined

Subtotal - 6 welds to be examined

Cracked, Unrepaired Welds (Note: affected piping size only shown

22" Recirc - 1 weld to be examined

Subtotal - 1 weld to be examined

Total - 37 welds to be examined from above four categories

GPC has reviewed the requirements of NRC Generic Letter 84-11 and has determined that the scope of examination must be expanded to meet minimum requirements for 6" RWCU not examined previously and 4" Recirc examined previously. Consequently, four welds (vice three noted above) will be examined for 6" RWCU not examined previously and two welds (vice one noted above) will be examined for 4" Recirc examined previously in lieu of that shown above. Therefore, the grand total of examinations has increased to thirty-nine (39) welds requiring examination for the four categories specified in NRC Generic Letter 84-11.

PLANS FOR OTHER INSPECTIONS

In addition to the above proposed sample, the following examinations will be performed as a result of recent cracking experienced at other BWR utilities:

a) At least one recirculation outlet nozzle-to-safe end weld and two recirculation inlet nozzle-to-safe end welds will be examined during the upcoming Plant Hatch 1 outage. These inspections were committed previously by GPC to NRC through the GE BWR Owners Group response to NRC regarding cracking in Inconel-clad safe ends and nozzles;

b) Fifty percent (50%) of the recirculation inlet nozzle thermal sleeve attachment welds will be examined during the upcoming outage. Should unacceptable indications be observed, the remaining 50% would then be examined, radiation levels permitting. The configuration of these welds at the Plant Hatch units differ significantly from those units observing cracking in this type weld in that the thermal sleeve does not weld to the nozzle safe end at either Plant Hatch unit. This was discussed in considerable detail with NRC Region II personnel during a telephone conversation on August 16, 1984; and

c) Both "A" and "B" recirculation loop jet pump instrumentation nozzle safe end-to-penetration seal welds will be examined during the upcoming outage. The safe end-to-nozzle welds for these particular nozzles will not be examined since they were examined during the previous outage.

QUALIFICATION OF UT PERSONNEL

The information provided in our May 31, 1984 submittal was specific regarding qualification of UT personnel in that:

a) It did indicate that procedures similar to those previously qualified at Battelle-Columbus (BCL) for IGSCC detection would be used. The latest approved revision of the procedure technically meets or exceeds the originally BCL-qualified procedure, e.g., calibration requirements, recording requirements, etc. Further, similar procedures, techniques, etc. have been reviewed and found acceptable for use through NRC Region II I&E inspections at Plant Hatch during inservice inspection activities; and

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b) It did indicate that various Levels II and III inspection personnel who will perform data evaluation have been qualified in the detection of IGSCC through the process currently in effect at the EPRI NDE Center. NDE personnel under contract to our primary inservice inspection group, Southern Company Services, who have qualified at the EPRI NDE Center in the detection and interpretation of IGSCC may also perform examinations and evaluations, as appropriate. As is the case of the procedures, inspector qualification is subject to audit by NRC regional personnel and have met the necessary requirements to date.

With regard to qualification of personnel in sizing of IGSCC, NRC has not formally notified GPC through implementation letter, bulletin, etc. per the NRC's review and approval process that this is a requirement. In anticipation of any such future sizing qualification requirement, the primary inservice inspection group to be used at Plant Hatch has several Levels II and III personnel on its staff qualified through the EPRI NDE Center for the sizing of cracks. Those personnel and any subcontractor personnel similarly qualified can be used for sizing of IGSCC indication depth should reportable indications be observed.

LEAK DETECTION AND LEAKAGE LIMITS

Our May 31, 1984 submittal indicated that proposed Technical Specification changes to augment these existing reactor coolant leakage detection requirements were submitted to you by letters dated February 10 and 11, 1983. The proposed changes were subsequently reviewed and approved as discussed in the NRC's Plant Hatch Unit 1 Safety Evaluation Report dated February 11, 1983. The proposed changes meet the intent of the leak detection and leakage limits discussed in Attachment 1 of NRC Generic Letter 84-11. No changes other than those discussed in Section 2.5 of Attachment 1 to our May 31 submittal are planned.

Should you have any questions in his regard, please contact this office.

Sincerely yours,

William E. Burns /for

L. T. Gucwa

JAE/mb

xc: J. T. Beckham, Jr.
H. C. Nix, Jr.
J. P. O'Reilly (NRC- Region II)
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J. T. Beckham, Jr.
Vice President and General Manager
Nuclear Generation

NED-84-510

September 26, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
OPERABILITY CONCERN REGARDING RESIDUAL
HEAT REMOVAL SERVICE WATER SYSTEM PUMPS

Gentlemen:

On September 25, 1984, a discussion regarding the operability of the Residual Heat Removal Service Water System (RHRSW) Pumps of Plant Hatch Unit 2 was held between representatives of Georgia Power Company (GPC) and members of the NRC staffs of Nuclear Reactor Regulation and Region II Office of Inspection and Enforcement. Pursuant to that discussion, GPC herein submits for your review and concurrence a description of the circumstances surrounding our concern and our course of action to resolve that concern.

Recently, as part of the consideration of a possible future modification to up-grade the service life of the Plant Hatch RHRSW Pumps, the pump vendor, Johnston Pump Company, was asked to provide design input. While reviewing the latest revision of the seismic analysis, Johnston Pump Company found that an apparent inconsistency existed between the bolt materials assumed in the seismic analysis and those shown on available documentation of the pump installation. Johnston Pump Company informed our architect-engineer of their findings and questioned what bolt material was installed in the pump columns. Their question raised a concern on the part of our architect-engineer regarding a potential impact on pump operability. If this discrepancy were determined in fact to exist, the bolted pump columns might not be of sufficient strength to remain operable following a postulated seismic event.

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The pump assemblies, as purchased for Plant Hatch Unit 2, were assembled using bolts manufactured of SA-193 Grade B8 material. Subsequently, a design modification was implemented to relocate the pump column seismic support from a location below the water level to one above the high water level to facilitate repairs and normal service removal. At the time of this support relocation, a seismic reevaluation by Johnston Pump Company was requested. This reevaluation was performed with the assumption that SA-354 grade DB bolts would be used in the modified pump column. This change in bolt material was required due to an increase in the calculated bolt stresses predicted for an Operating Basis Earthquake (OBE) seismic event to stresses greater than those allowed by ASME Code Section III, 1971 for SA-193 Grade B8 bolts. The bolt stresses during normal pump operation (maximum normal operating stress is approximately 11,500 psi) are well below the code allowable stress of 15,000 psi as specified by ASME Section III, 1971, for the SA-193 Grade B8 bolts. It should be noted that the ASME codes by which the acceptability of the loading of the bolts is determined have an inherent margin of safety. Documentation found to date of the as-modified pumps does not reflect that the bolting material assumed in the seismic reevaluation was used - and still may not after the review is complete. Hence, GPC is concerned that a deficiency potentially exists which could adversely affect the operability of the RHRSW pumps.

Since a final determination of the actual strength properties of the installed bolts has not been ascertained by either the available documentation or materials examination, GPC has undertaken what is believed to be the most conservative approach in resolving the uncertainty about the bolts. To this end, three efforts are being pursued concurrently. First, in order to determine with certainty what bolt material was installed in the flanges, a sample of the installed bolts has been obtained and will be analyzed by an independent laboratory in Atlanta to determine the material of manufacture and the associated strength characteristics. Until this analysis is made, a final determination of the status of the RHRSW pumps cannot be made with certainty. The results of this material analysis are expected to be available by September 28, 1984.

Second, a reanalysis of the seismic loadings on the bolts in question is underway using improved seismic analysis techniques and input assumptions to determine if the original bolts of SA-193 Grade B8, should they still be installed, might be acceptable for pump support after the postulated OBE.

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The reanalysis of the seismic loading, if it is still required in light of the third aspect of the program, will be available by September 28, 1984.

A third effort, which we feel to be very conservative, is that GPC is replacing the questionable bolts in the RHRSW Pumps without waiting for the results of the above two efforts. Sufficient bolts of the SA-354 grade DB material assumed in the Johnston Pump Company seismic analysis are available at the plant site and are presently being installed. First priority in the bolt replacement effort is the replacement of the questionable bolts in one pump in each of the RHRSW subsystem loops to remove any doubt concerning the short term operability of the RHRSW system in the minimum possible time. This work is being performed on an around-the-clock basis with as many personnel assigned as can reasonably be expected to work within the area of the pumps. Bolt replacement is expected to be accomplished on one pump in each loop by 2400 Eastern Daylight Savings Time (EDST) on September 27, 1984. Because the work is of a sequential nature, it is anticipated that one pump will be completed approximately 24 hours prior to the stated time on September 27th. The remaining two RHRSW pumps will have the questionable bolts replaced as soon as possible, but no later than 1800 EDST on October 2, 1984.

It is GPC's position that because of the period of time before these three efforts can be accomplished, additional actions are necessary even though the original bolts may prove to be acceptable. The Plant Deputy General Manager declared the RHRSW pumps inoperable upon the recommendation of the Plant Hatch Plant Review Board as a conservative approach to plant operations. The action statement of Technical Specification 3.7.1.1(4) has not been implemented based upon the subject telephone conversation with the NRC staff and a concern for the optimization of plant safety discussed below.

Because of the nature of this particular situation, the safest and most conservative action is to maintain the unit in its present operational condition. By so maintaining the unit, the RHRSW pumps are not required to operate. If the plant were to undertake a shutdown, the risks of a possible abnormal plant transient would be increased. In addition, going to a shutdown condition requires the RHRSW pumps to operate. The analysis done by our engineering support indicates that, in the event of an OBE, the stresses on the pump column will be greater for an operating pump than for a non-operating pump. This is due to the fact that the stresses on the pump column are additive and, should the unlikely OBE occur (an annual risk of

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exceeding an 0.08g OBE acceleration is estimated to be $4.5 \times 10^{-4}$ - reference S.T. Algemissen and D. M. Perkins Probablistic Estimate of Maximum Acceleration in Rock in the Contiguous U.S., USGS 76-416, 1976), the total loading on the bolts is decreased by an amount corresponding to that portion of the ASME code loading which is derived from the pressure component due to pump operation. Thus a lower stress will be realized in the event of an OBE with the plant in operation.

In order to further minimize the possibility of any plant abnormal transients which would require the use of the pumps, action has been taken to minimize power changes and testing during the period of time until the bolts on at least two RHRSW pumps have been replaced and the pumps placed back in service. Further, the replacement of the bolts can be accomplished more efficiently in an environment where all pumps are stopped. Thus, while the program to resolve the bolt material question is underway, it is GPC's intention to continue operation on Plant Hatch Unit 2 and, thereby, maintain the optimum condition of safety under the existing circumstances.

The Plant Hatch Plant Review Board and the corporate Safety Review Board have reviewed the circumstances reported herein and concur with the conclusions and actions described. If you have any questions regarding this letter, please call my office.

Yours very truly,

J. T. Beckham, Jr.

xc: H. C. Nix, Jr.
J. P. O'Reilly (NRC- Region II)
Senior Resident Inspector

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Manager Nuclear Engineering
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NED-84-508

September 26, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
UPDATE OF EQUIPMENT QUALIFICATION PROGRAM
JUSTIFICATIONS FOR CONTINUED OPERATION

Gentlemen:

On July 24, 1984, Georgia Power Company (GPC) submitted as letter NED-84-395 the 10 CFR 50.49 equipment qualification program Justifications for Continued Operation (JCOs) which were still effective for Plant Hatch. Since that submittal GPC has determined that revisions to seven of the Hatch Unit 1 JCOs and addition of one new JCO are required due to changes which have occurred in the qualification status of certain equipment.

Enclosed are the eight new or revised Unit 1 JCOs. These pages should be used to replace the pages with identical attachment and page numbers which were transmitted by our July 24, 1984 letter. It should be noted that pages 2 and 3 of Attachments 1 and 2 of the July 24, 1984 submittal are deleted since the equipment covered by those JCOs is now fully qualified. In addition, page 46 of Attachment 2, enclosed with this letter, is a new JCO to cover a limit switch which was recently added to the scope of the Hatch equipment qualification program.

Attachment 3 to our July 24, 1984 letter has not been revised since there have been no changes to the Unit 2 JCOs since that date. The three attachments to our July 24, 1984 letter, along with the revisions now being submitted, justify continued operation of Plant Hatch equipment for which complete environmental qualification per 10 CFR 50.49 has not yet been demonstrated.

Very truly yours,

William E. Burns /for

L. T. Gucwa

CBS

Enclosures

xc: H. C. Nix, Jr.
J. P. O'Reilly
Senior Resident Inspector

8410020213 PDR

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J. T. Beckham, Jr.
Vice President and General Manager
Nuclear Generation

NED-84-512

September 27, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
SUPPLEMENTAL INFORMATION REGARDING
RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM PUMPS

Gentlemen:

Georgia Power Company (GPC), pursuant to the request of Mr. J. A. Olshinski, Director, Division of Reactor Safety, Region II Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, herein submits the following information to supplement our letter NED-84-510 dated September 26, 1984 regarding the operability of the Residual Heat Removal Service Water (RHRSW) pumps:

1. The effort to replace questionable bolts as of 8:00 a.m., September 27, 1984, is on a schedule ahead of the commitments made in our September 26, 1984 letter. RHRSW pump 2E11-C001B has had the bolts in question replaced and is reinstalled. 2E11-C001A is expected to be returned to service after bolt replacement by noon today. RHRSW pump 2E11-C001D has been removed from service and is in the process of bolt replacement. The last of the four RHRSW pumps, 2E11-C001C will undergo bolt replacement when pump A has been restored to service. There are no known obstacles which will prevent replacement of the remaining questionable bolts by the dates committed to in our earlier correspondence. If any situation arises which will preclude our meeting our stated commitment, appropriate NRC Region II personnel will be notified at the first opportunity.

Georgia Power



2. Subsequent to the submittal of letter NED-84-510, GPC management learned that sufficient bolts to modify all four RHRSW pumps were not, in fact, on the plant site as was stated in that letter. The misunderstanding on this point originated in the invoicing and transmittal of the packaged bolts. However, an adequate supply of suitable bolts has been located and has been procured on an expedited basis by use of dedicated aircraft. Delivery is expected to support the committed schedule. Additionally, a clarification is required regarding the material of the replacement bolts currently being installed. SA-354 Grade BD bolts were not available in a time frame to support the early effort on 2E11-C001 A and B. Johnston Pump Company supplied a substitute material, A490 Grade BD, for the SA 354 Grade BD material. This substitute material is acceptable under the provisions of ASME Section II, Part A, "Requirements for SA 354 Bolting Material", 1983 edition for use in this application. The use of this bolt material has been analyzed by the pump vendor, Johnston Pump Company, and verified to be fully acceptable. RHRSW pumps 2E11-C001C and D will be rebolted with acceptable bolts manufactured from either A490 Grade BD or SA 354 Grade BD.

3. A review has been conducted of the Plant Hatch Unit 2 Technical Specifications to determine if any secondary impacts on plant operations would result from the inoperability of the RHRSW pumps. Two such Limiting Conditions for Operations (LCOs) were found--3.6.2.2(b), Suppression Pool Cooling and 3.9.12(a), Reactor Coolant Circulation During Refueling Operations. LCO 3.6.2.2(b) calls for action which is similar to and bounded by LCO 3.7.1.1(4) discussed in our September 26, 1984 letter. LCO 3.9.12(a) only applies when the plant is in the refueling mode and therefore is not a concern for continued plant operations. These LCOs and the Applicability Statement 3.0.3 are the only known plant Technical Specifications to be of applicability, to our knowledge and belief. In each case the action required is bounded by the discussion contained in our September 26, 1984 letter.

4. In order to minimize the possibility of any abnormal plant transient which would require the use of the RHRSW pumps, the Plant Hatch Deputy General Manager (acting for the General Manager) issued a memorandum on September 25, 1984 to the Manager of Operations and the Superintendent of Operations which, in part, called for the following:

- The reactor remained in operation (the reactor will continue in operation as long as plant conditions permit);
- All load increases were suspended immediately;
- All startup testing was suspended; and
- All work which would significantly increase the risk of a plant trip, with the exception of the required surveillance tests, was suspended.

These actions were reviewed and concurred with by the Plant Review Board. A malfunction of a recirculation pump controller resulted in a load decrease from approximately 700 mwe to 342 mwe during the evening of the 26th. Load level has been maintained at the reduced level in accordance with our commitment to limit transients whenever possible. These operating restrictions will be rescinded upon restoration of one RHRSW pump per subsystem loop to a known acceptable operability condition through replacement of questionable bolts.

5. A standing order was issued on September 26, 1984, which gave guidance to plant personnel regarding preferred plant operations in the event of a reactor trip. Briefly, the Standing Order called for the plant to be maintained in a hot standby condition following a reactor scram, if plant circumstances would allow. Further, the operation of the RHRSW pumps was prohibited unless absolutely necessary to maintain the reactor in a safe condition or to protect plant equipment and the general public. It noted that the decision to operate the pumps should come from the Operations Supervisor on shift. This order should minimize the possibility, to the extent possible, of the operation of the RHRSW pumps. These operating restrictions will be rescinded upon restoration of one RHRSW pump per subsystem loop to a known acceptable operability condition through replacement of questionable bolts.

6. As noted in our September 26, 1984 letter, in the event of an Operating Basis Earthquake (OBE) the stresses on the pump column will be greater for an operating pump than for a non-operating pump. Since the submittal of that letter, we have received a more detailed quantification of the stresses seen in the OBE and Design

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Basis Earthquake (DBE) for operating and non-operating pumps. A summary of these stresses is provided in the enclosed Table 1. It can be seen that the allowable stresses for the material which is in question (SA-193) are greater than the seismic loads for non-operating pumps in the event of an OBE, as noted in our September 26, 1984 letter. Further, the table shows that the material used in the replacement bolts (SA-354/A490) has an allowable stress well in excess of the seismic loadings on the pumps in the operating or non-operating mode in the event of an OBE. If one compares the Relocation Analysis OBE loading from line 1 of the table to the listed allowable for the SA-354/A490 material, the multiplier of 1.5 should be applied for the flat face flange assumption and the allowable becomes 45,000psi.

Should you require any further clarification or amplification regarding the Plant Hatch RHRSW pumps, please contact my office.

Yours very truly,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

WEB/mb

Enclosure

xc: H. C. Nix, Jr.
J. P. O'Reilly (NRC-Region II)
Senior Resident Inspector

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Manager Nuclear Engineering
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September 25, 1984

U. S. Nuclear Regulatory Commission
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Inspection Report
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ATTENTION: Mr. James P. O'Reilly

GENTLEMEN:

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VIOLATION:

"Technical Specification 6.8.1.a requires that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

Contrary to the above, procedures were not properly implemented in that on August 1, 1984, two snubbers were removed from Unit 1 Residual Heat Removal (RHR) system in violation of work instructions (MR 1-84-3763 work process sheet) which required the worker to "obtain necessary clearances and notify responsible engineer". No clearances were obtained prior to removal of snubbers E11-RHRH-193 or E11-RHRH-199.

This is a Severity Level IV violation (Supplement I)".

RESPONSE:

Admission or denial of alleged violation: The violation occurred.

Reason for the violation: The violation resulted from a failure of contractor personnel to follow written and verbal instructions. The contractor, Reactor Control Incorporated (RCI), removed the two snubbers from the operable RHR system without obtaining the proper clearances from Operations personnel. The controlling document for the snubber work was Maintenance Request (MR) 1-84-3763, issued on July 13, 1984.



Package dupe

Georgia Power

U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II - Suite 2900
September 25, 1984
Page Two

RESPONSE: (Continued)

The work process sheet for the snubber work required that the necessary clearances be obtained prior to modifying the snubbers. In addition, the contractor was verbally cautioned against performing work which could make the snubbers inoperable. Both the written and verbal instructions were violated when the two snubbers were removed without the necessary clearances.

Corrective steps which have been taken and the results achieved: Snubbers E11-RHRH-193 and E11-RHRH-199 are listed in Table 3.6.L of Unit 1 Technical Specifications and are therefore required to be operable during power operation. Removal of these two snubbers placed the Unit in an action statement requiring cold shutdown within 36 hours. Immediate action was taken to re-install the snubbers. Snubber E11-RHRH-193 was re-installed within 32 hours and Snubber E11-RHRH-199 within 37 hours. Limiting conditions for operation were not exceeded because the absence of a single snubber places the unit in an action statement requiring replacement of the snubber within 72 hours.

Corrective steps which will be taken to avoid further violations: A meeting was conducted with RCI personnel to emphasize equipment clearance and tagging procedures, welding procedures, and quality assurance concepts. RCI was directed to contact the designated Georgia Power Company representative prior to performing any installations or removals. A letter was sent to RCI management stressing the importance of plant procedures.

Date when full compliance will be achieved: Full compliance was achieved on August 2, 1984 when both snubbers were replaced and the contractor personnel informed of plant procedural requirements.

Please contact this office if you have any questions.

Very truly yours,

William E. Burns /for

L. T. Gucwa

JH/mb
xc: J. T. Beckham, Jr.
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