Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-6526

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

Georgia Power
the southern electric system

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

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

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ATTACHMENT 1

SHEET DELETED

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ATTACHMENT 1

SHEET DELETED

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ATTACHMENT 2

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ATTACHMENT 2

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ATTACHMENT 1

T47-N003, T47-N009 and Cable N3-04

These are RTD instruments and associated cable which monitor the drywell temperature in the area of the reference legs on the RPV water level instruments. The instruments are used by the operators to density compensate the level instruments based on drywell temperatures.

The level instruments do not require the density compensation in order for all automatic system actuations to take place to assure plant safety. The temperature instruments in conjunction with the level instrument will be used by the operator to confirm proper injection system operations. The maximum expected error in the level instruments is 25 inches which corresponds to a drywell temperature of 340°F. The operator is very aware of this error and if in doubt will apply the maximum error in the safe direction. If the instrument provided an incorrect reading the operator could be misled but the automatic system actuations which occur on high and low RPV level will protect the core and will alert the operator to the fact that he is being misled by the temperature elements. Therefore, plant safety is assured.

Based on the above, continued operation is justified.

ATTACHMENT 1

CABLES N1-4, H1-16, M5-20, N3-03, and N3-19

The specific cables noted above are of the PVC type which have not been tested to the standards required by the DOR guidelines which are an attachment to I. E. Bulletin 79-01B. Justification for continued plant operation until replacement of the cables is achieved is principally based on IEEE paper entitled" Insulation and Jackets for Control and Power Cables in Thermal Reactor Nuclear Generating Stations, (IEEE Transactions on Power Apparatus and Systems Vol. PAS-88, No. 5, May 1969)."

A key consideration in the evaluation of this report relative to the acceptability of the cable installed at HNP-1 is that under no circumstances will the accident dose of radiation occur concurrent with the harsh environment created by a high energy line break since all the subject cables are located exterior to the drywell. With this in mind radiation can be evaluated independently of the high energy line break.

The IEEE paper in Table X1 concludes that PVC has an "Overall threshold of damage" of 5×10^5 Rads. Were it not for the failure of the test cable in a simulated steam environment after being irradiated to 5×10^6 Rads the overall threshold of damage would have been 5×10^6 Rads. Since we are not considering a steam environment concurrent with the maximum radiation, the 5×10^6 Rads threshold of damage compares favorably with the highest actual accident dose of 1.86×10^6 Rads in the area of the NE and SE corner rooms.

The IEEE paper in Table II indicates that the useful life of PVC cable in terms of elongation loss is greater than 115 years even after being irradiated to 5×10^6 Rads. Although further evaluations of cable life are possible, these evaluations are not necessary because the cable is being replaced.

The maximum accident temperature of 214°F presents no significant problem because PVC cable, as pointed out in the IEEE paper on page 534, has a useful life of 200 hous at 136°C (276°F). Additionally, a steam environment presents no significant problem because PVC cable, as pointed out in the IEEE paper on page 534 and Table 1X, was found suitable in a steam environment greater than 9 days.

ATTACHMENT 2

E51-N023A&B, E51-N026A&B, E51-N025A&B, E51-N027A&B

These instruments are temperature sensors which monitor temperatures in areas where the RCIC steam line is located and initiate an isolation signal in the event of a steam leak in the RCIC steam line.

The instruments could be subjected to a harsh environment due to radiation in the unlikely event of a LOCA. No credit is taken in the accident analysis for the operation of these switches for the mitigation of a LOCA.

In the event of a HELB the instruments could be subjected to a harsh environment due to temperature and pressure. A high degree of confidence exists that the instruments will operate to isolate the HPCI steam line break since test experience shows that the switches will operate up to a test temperature of 212°F. The set point of these switches is 170°F which is well below the temperature for which test experience exists.

This piece of equipment has been used during normal operation in plants similar to Hatch 1 for the past eleven (11) years. To date, no age related common mode failures have been reported. These devices have experienced relatively limited service in Hatch 1 as they have been used during normal operation for only five (5) years. With this limited service, this equipment is not expected to fail before replacement.

Based on the above, continued operation is justified.

ATTACHMENT 2

E41-C002 (LS-4)

This limit switch is attached to the HPCI Turbine stop valve and controls the signal which opens the HPCI system discharge valve as well as activates the turbine regulator ramp generator on HPCI turbine start.

The limit switch is not required for a HELB in the HPCI room which is a rupture of the HPCI steam supply line. No credit is taken for the operation of the HPCI turbine to mitigate the consequences of a HPCI steam line break.

The limit switch could be subjected to a harsh environment due to radiation after a LOCA. The limit switch is a NAMCO Model D200G-ST-2 and does not have any documented radiation exposure test data. The manufacturer has provided a list of non-metallic parts which are used in the subject switch. Each part has been reviewed to determine its radiation damage threshold value. The results of the evaluation has demonstrated that the most susceptible part has a threshold value 5 times higher than the maximum expected total integrated dose in the HPCI room; therefore, the switch is expected to operate as designed.

Based on the above, continued operation is justified.