NRC Central

August 2, 1978

Robert M. Lazo, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Walter H. Jordan
881 West Outer Drive
Oak Ridge, TN 37830

In the Matter of
Northern States Power Company
(Monticello Nuclear Generating Plant, Unit 1)
Docket No. 50-263

Gentlemen:

Enclosed, for your information, is a Staff memorandum regarding the applicability to Monticello of certain regulatory actions taken with respect to Hatch, Unit 2. As noted in the memorandum, the Staff plans to perform evaluations to establish the relevancy of items (2) and (4) to Monticello. Although certain regulatory action by the Staff may be taken with respect to these items as they apply to Monticello, we do not believe that they raise issues which require resolution by this Licensing Board.

Sincerely,

Stephen H. Lewis
Counsel for NRC Staff

Encl.: (1) Memorandum fm V. Stello, Jr. to M. Grossman, July 24, 1978
(2) "Safety Evaluations in Support of Exemptions from Certain Requirements of the Commission's Rules and Regulations" (Hatch 2), June 1978

cc: (See Page 2)

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

JULY 2 4 1978

MEMORANDUM FOR: Milton Grossman, Hearing Division Director
and Chief Counsel, ELD

FROM: Victor Stello, Jr., Director
Division of Operating Reactors

SUBJECT: BOARD NOTIFICATION-MONTICELLO

We have examined the four exemptions recently issued in connection with the operating license for Hatch, Unit 2. This was done to determine their applicability to Monticello and, if applicable, to recommend that you advise the ASLB accordingly. The four exemptions concern questions regarding:

(1) full conformance of the ISI program to the requirements of 50.55 a(g);

(2) RPS power supply to perform its intended function under postulated conditions of single failure and earthquakes;

(3) full conformance of the Mark I containment with the requirements of GDC 50 in Appendix A of Part 50; and,

(4) full conformance of the pressure vessel surveillance program with the requirements of Appendices G and H to Part 50.

It appears that items (1) and (3) are not relevant since Monticello has been granted exemptions on these issues. Items (2) and (4) may be relevant to the Monticello hearing. We plan to perform evaluations to establish the relevancy to Monticello.

We believe that the potential applicability of items (2) and (4) warrants Board notification. We suggest that the Board be informed without delay because their decision is imminent.

Victor Stello, Jr., Director
Division of Operating Reactors

Enclosure:
Safety Evaluation for
Hatch 2

ENCLOSURE 1

JUNE 1978

SAFETY EVALUATIONS IN SUPPORT OF

EXEMPTIONS FROM CERTAIN

REQUIREMENTS OF THE COMMISSION'S

RULES AND REGULATIONS

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

GEORGIA POWER COMPANY

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

AND

CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-366

SAFETY EVALUATIONS IN SUPPORT OF

EXEMPTIONS FROM CERTAIN

REQUIREMENTS OF THE COMMISSION'S

RULES AND REGULATIONS

We have determined that the Edwin I. Hatch Nuclear Plant Unit No. 2 requires exemptions from certain requirements of (1) Section 50.55a(g)(2) of 10 CFR Part 50, (2) Criterion 2 of Appendix A to 10 CFR Part 50, (3) Criterion 50 of Appendix A to 10 CFR Part 50, and (4) Appendices G and H to 10 CFR Part 50. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Our safety evaluations supporting the granting of these exemptions are contained herein.

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF SECTION 50.55a(g)(2) OF 10 CFR PART 50

I. INTRODUCTION

In FSAR Amendment No. 36, the Georgia Power Company (GPCo) requested relief or exemption from certain preservice inspection requirements. On the basis of our review of this information, we advised GPCo that we would require the additional information in Questions 121.16, 121.17, 121.19 and 121.20 to complete our evaluation of this matter. Georgia Power Company provided the additional supporting information in FSAR Amendment Nos. 41, 42, 43, 44 and 45. As a result of our review of this information, we have determined that an exemption to 10 CFR 50.55a "Codes and Standards" is required and have also determined that an exemption regarding this matter is justified. Our basis for this conclusion is discussed in the subsequent paragraphs of this report.

For nuclear power facilities whose construction permits were issued on or after January 1, 1971, but before July 1, 1974, 10 CFR 50.55a (g)(2) specifies that components shall meet the preservice examination requirements set forth in editions of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and Addenda in effect six months prior to the date of the issuance of the construction permit. The provisions of 10 CFR 50.55a (g)(2) also state that components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which become effective.

Therefore, our evaluation consisted of determining the areas where GPCo met 10 CFR 50.55a(g)(2) requirements and the areas where exemptions to the regulation were necessary and the basis for these exemptions.

II. TECHNICAL EVALUATION CONSIDERATIONS

A. The Edwin I. Hatch Nuclear Plant, Unit No. 2, received a Construction Permit in December 1972. In accordance with 10 CFR 50.55a, the preservice inspection must conform with the ASME Code, Section XI, 1971 Edition, including Addenda through Summer 1971. The ASME first published rules for inservice inspection in the 1970 Edition of Section XI. No preservice or inservice inspection requirements existed prior to that date. Since the Hatch Unit No. 2 plant system design and ordering of long lead time components were well underway by the time the Section XI rules became effective, full compliance with the access and inspectability requirements was difficult to achieve. As can be seen in Section III below, which discusses individual welds or examination categories, a large portion of the required volumetric examinations were performed.

B. Verification of as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the Hatch Unit No. 2 primary pressure boundary was fabricated, contain examination and testing requirements which by themselves provide the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions and postulated accidents reviewed in the FSAR and described in the plant design specification. As a part of these examinations the primary pressure boundary full penetration welds were volumetrically inspected (radiographed) and the sysetm was subjected to hydrostatic pressure tests.

C. The intent of a preservice examination is to establish a reference or base line prior to the initial operation of the facility. The results of subsequent inservice examination can then be compared to the original condition to determine if changes have occurred. If review of the inservice inspection results show no change from the original condition no action is required. In the case where base line data are not available, all indications must be treated as new indications and disposed of accordingly. Section XI of the ASME Code contains acceptance standards which are used as the basis for evaluating the acceptability of such indications. Therefore, conservative disposition of defects found during inservice inspection can be accomplished even though preservice information is not available.

D. Other benefits of preservice examination include providing redundant or alternate volumetric inspection of the primary pressure boundary using a test method different from that employed during the component fabrication thereby increasing the overall probability of finding all significant fabrication flaws. Successful performance of a preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using a similar test method.

In the case of Hatch Unit No. 2, a large portion of the code required preservice examinations were performed. We have concluded that failure to perform 100% preservice examination of the welds specifically identified below will not significantly affect the assurance of the initial system integrity or the ability to subsequently detect and correct service-induced defects.

E. In some instance where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, we will require that these or supplemental examinations be conducted as a part of the inservice inspection program. We have concluded that requiring these supplemental examinations to be performed at this time (before plant startup) would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. The performance of supplemental examinations, such as surface examinations, in areas where volumetric inspection is difficult will be more meaningful after a period of operation. Acceptable pre-operational integrity has already been established by similar Section III fabrication examinations and the probability of system degradation between these examinations and initial plant startup is small.

In cases where parts of the required examination areas cannot be effectively examined because of a combination of component design/current inspection technique limitations, we will continue to evaluate the development of new or improved volumetric examination techniques. As improvements in these areas are achieved, we will require that these new techniques be made a part of the inservice examination requirements of those components or welds which received a limited preservice examination.

F. The FSAR contains information on the preservice examination of ASME Code Class 1 and Class 2 components. For Hatch Unit No. 2, 10 CFR 50.55a (g)(2) requires that the preservice examination conform with Section XI, through the Summer 1971 Addenda. For Class 1 components, specific examination requirements are contained in Section XI, Summer 1971 Addenda. While not all the specific examinations have been conducted, for the reason set forth above, those examinations performed provide an adequate level of assurance of the preservice structural integrity and the ability to subsequently detect and correct service-induced defects. Specific examination requirements for Class 2 components are not contained in Section XI, Summer 1971 Addenda. Therefore, we will evaluate the preservice examination of Class 2 components as supplemental information in our subsequent evaluation of the applicant's initial inservice inspection plan.

III. EXEMPTIONS REQUIRED

Section 50.55a states that as a minimum, the system and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f),(g) and (i) of this section meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission or the Atomic Energy Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing or inspection of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

We have reviewed the information submitted by the Georgia Power Company related to the preservice examination of the Edwin I. Hatch Nuclear Plant, Unit No. 2. Based on this information and our review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Boiler and Pressure Vessel Code, Section XI, have been determined to be either impractical or would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety as provided in 10 CFR 50.55a.

Therefore, pursuant to 10 CFR Section 50.12 specific exemption for those preservice requirements is justified as follows:

Piping Pressure Boundary

1. Item B4.5 Circumferential and Longitudinal Piping Welds

   Code Requirement: .The examination areas shall include essentially 100% of the longitudinal and circumferential welds and the base metal for one wall thickness beyond the edge of the weld. Longitudinal welds shall be examined for at least one foot from the intersection with the edge of the circumferential weld selected for examiantion. In the case of pipe branch connections, the areas shall include the weld metal, the base metal for one pipe wall thickness beyond the edge of the weld on the main pipe run, and at least two inches of the base metal along the branch run.

Exemption Requested: An exemption was requested from performing 100% of the code volumetric examination requirement.

Reason for Request: The design and arrangement of the piping sytems and components limits some examinations due to geometric configuration or accessibility. Generally, these limitations exist at pipe-to-fitting welds, where examination can be fully performed only from the pipe side, the fitting geometry limiting or even precluding examination from the opposite side. Welds having such restrictions were examined to the extent practical. In instances where the location of pipe supports or hangers restricts the access available for examination of pipe welds, examinations were performed to the extent practical unless removal of the support is permissible without unduly stressing the system.

Approximately 98% of the required examinations were completed. The table on the following pages identifies the location and supporting information for the piping pressure boundary welds for which exemptions are requested.

Bases and Conclusions: We conclude, for the piping system welds listed in the table on the following pages, that (1) the approximately 98% preservice ultrasonic examination, (2) the construction code radiographic examination, and (3) the fabrication or supplemental surface examination provide an adequate level of assurance of preservice structural integrity.

2. Item B4.9 Integrally Welded Supports

Code Requirement: The examination areas shall include essentially 100% of the integrally-welded external support attachments. This includes the welds to the pressure-retaining boundary and the base metal beneath the weld zone and along the support attachment member for a distance of two support thicknesses.

Exemption Requested: An exemption was requested from performing 100% of the code volumetric examination requirement.

Reason for Request: The design and geometric configuration of the piping system integrally-welded supports, identified in the FSAR in response to Questions 121.16 and 121.20, was such that examinations could not be performed to the extent required by Article IWB-2600. The welds that were not examined completely by volumetric methods can be categorized as follows: (1) welds that were accessible but only the base metal could be examined by ultrasonic techniques, (2) welds that were accessible but weld concavity or the small size prevented acoustic coupling or (3) welds that were inaccessible due to main steam line whip restraints. Surface examinations were performed on the integrally-welded attachments during original fabrication or to supplement the limited volumetric examinations.

Bases and Conclusion: We have determined that the limited ultrasonic examination supplemented by surface examination for the accessible welds is a satisfactory alternate examination for the Section XI code requirement. For the welds which are inaccessible due to interference from protective systems, we have determined that the construction code examinations provide an adequate level of assurance of preservice structural integrity.

IV. PUBLIC INTEREST REGARDING COMPLIANCE WITH SECTION 50.55a(g)(2) OF 10 CFR PART 50

Our technical evaluation has not identified any practical method by which the Edwin I. Hatch, Unit No. 2, preservice inspection program can meet the ASME Code, Section XI, requirements of 10 CFR Part 50, Paragraph 50.55a(g)(2). Requiring specific compliance with this paragraph would include the following actions: delay the startup of the plant and remove significant portions of the primary pressure boundary piping system; redesign and fabricate, if possible, new sections for the piping system within the available space; reweld the new primary pressure boundary piping; and repeat the system hydrostatic pressure test. The as-built structural integrity of the primary pressure boundary piping is not dependent on the required Section XI preservice examination since the applicable construction codes contain examination and testing requirements which by themselves provide the necessary assurance of structural integrity. We believe the public interest is served by not imposing the certain provisions of 10 CFR Part 50, Paragraph 50.55a(g)(2), that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

TABLE 121.16
SECTION XI EXAMINATION CATEGORY B-J

Weld Identification Number and Weld Type	Required Examinations	Completed Examinations	Supplemental Examinations	Fabrication Examinations
2B31-1RC-4AA Branch Connection-to-Cap	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT
2B31-1RC-4AB Branch Connection-to-Cap	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT
2B31-1RC-4BC Branch Connection-to-Cap	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT
2B31-1RC-4BD Branch Connection-to-Cap	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT
2B31-1RC-28A-17 Tee-to-Cross	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT RT (root)
2B31-1RC-28A-18 Cross-to-Reducer	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT (ID & OD) Straight Beam UT
2B31-1RC-28B-17 Tee-to-Cross	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT RT (root)

TABLE 121.16 (Cont'd)

Weld Identification Number and Weld Type	Required Examinations	Completed Examinations	Supplemental Examinations	Fabrication Examinations
2B31-1RC-28B-18 Cross-to-Reducer	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	PT	RT, PT (ID & OD) Straight Beam UT
2E11-1RHR-24A-R-1 Valve-to-Valve	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	MT	RT, PT Post Stress RT & PT
2E11-1RHR-24B-R-1 Valve-to-Valve	0° Weld Scan Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	MT	RT, PT Post Stress RT & PT
2E11-1RHR-24A-R-1A Valve-to-Penetration	0° Weld Scan 0° Lamination Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	MT	RT, PT Post Stress RT & PT
2E11-1RHR-24B-R-1A Valve-to-Penetration	0° Weld Scan 0° Lamination Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle Beam	MT	RT, PT Post Stress RT & PT
2B21-1MS-24C-15 Pipe-to-Valve Weld	0° Weld Scan 0° Lamination Angle Beam Transverse Angle Beam	0° Weld Scan 0° Lamination Transverse Angle Beam	MT	RT, PT Root RT & PT
2B21-1MS-24B-14 Elbow-to-Pipe	0° Weld Scan 0° Lamination Angle Beam Transverse Angle Beam	0° Weld Scan Transverse Angle	None	RT, PT

Where, RT = Radiography; UT = Ultrasonic Testing; PT = Penetrant Testing; MT = Magnetic Particle Testing

V. CONCLUSIONS

Based on the foregoing we have determined that, pursuant to 10 CFR Section 50.12, a specific exemption as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption 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. We have concluded that this exemption would be 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 this action.

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF CRITERION 2 OF APPENDIX A TO 10 CFR PART 50

I. INTRODUCTION

The design of the Hatch Unit 2 reactor protection system power supply is essentially the same as that of previously-licensed BWR/4 reactors. The reactor protection system power supply consists of two high-inertia alternating current motor-generator sets and an alternate alternating current power supply.

During our review of the Hatch Unit 2 operating license application, we questioned the capability of the reactor protection system power supply to accommodate the effects of earthquakes without jeopardizing the capability of the reactor protection system to perform its intended safety function. We determined that a sequence of events initiated by the occurrence of an earthquake can be postulated which could result in damage to the reactor protection system components with the attendant potential loss of capability to scram the plant. We, therefore, conclude that the Hatch Unit 2 reactor protection system power supply design is not in conformance with the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50 and that an exemption from certain requirments of Criterion 2 of Appendix A to 10 CFR Part 50 is required and justified. The bases for our conclusions are discussed in the following sections.

II. TECHNICAL EVALUATION CONSIDERATIONS

Criterion 2 of Appendix A to 10 CFR Part 50 requires in part that systems important to safety, such as the reactor protection system, be designed to withstand the effects of earthquakes. The Hatch Unit 2 reactor protection system is a Class IE system, hence it is seismic Category I. The reactor protection system power supply, however, is not seismically qualified. A sequence of events initiated by the occurrence of an earthquake can, therefore, be postulated which could result in damage to the reactor protection system components with the attendant potential loss of capability to scram the plant. This sequence of events includes (1) the occurrence of an earthquake that would cause the undetected failure of a voltage sensor, (2) the failure of the motor-generator set resulting in abnormal output voltage, (3) persistence of this abnormal output voltage undetected by visual observation and surveillance testing for a time sufficient to damage reactor protection system components, and (4) failure of these components in such a manner that results in loss of scram capability (instead of in the fail-safe mode).

Therefore, we require that prior to startup following the first scheduled refueling outage, the applicant install a Class IE system approved by us capable of de-energizing the reactor protection system power supply when its output voltage exceeds or falls below limits within which the equipment being powered from the power supply has been designed and qualified to operate continuously and without degradation. With such a system, the reactor protection power supply design will be in conformance with the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50. The operating license will be conditioned accordingly.

III. EXEMPTION REQUIRED

As a result of our review of the Hatch Unit 2 reactor protection system power supply design, we determined that a sequence of events initiated by the occurrence of an earthquake can be postulated which could result in damage to the reactor protection system components with the attendant potential loss of capability to scram the plant. We, therefore, conclude that the reactor protection system power supply design is not in conformance with the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50.

We, therefore, require that prior to startup following the first scheduled refueling outage, the applicant install a Class IE system approved by us capable of de-energizing the reactor protection system power supply when its output voltage exceeds or falls below limits within which the equipment being powered from the power supply has been designed and qualified to operate continuously and without degradation. With such a system, the reactor protection system power supply design will be in conformance with the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50. The operating license will be conditioned accordingly. In the interim, however, we conclude that an exemption from the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50 is required and justified. The bases for our conclusion are as follows:

(1) The most likely failure mode of the reactor protection power supply motor-generator sets is complete loss of output. This is not a concern because the reactor protection system is fail-safe, i.e., a scram would result.

(2) There have been no reported failures in the Class IE loads which are connected to these motor-generator sets which can be attributed to an over-voltage or under-voltage condition in the sets.

(3) It is our judgment that the occurrence of the sequence of events necessary to result in loss of the capability to scram the plant is unlikely. This sequence of events includes (a) the occurrence of an earthquake that would cause the undetected failure of a voltage sensor, (b) the failure of the motor-generator set resulting in abnormal output voltage, (c) persistence of this abnormal output voltage undetected by visual observation and surveillance testing for a time sufficient to damage reactor protection system components, and (d) failure of these components in such a manner that results in loss of scram capability (instead of in the fail-safe mode).

(4) The technical specifications will require that the over-voltage, under-voltage, and under-frequency relays be calibrated and that the tripping logic and generator output breaker be functionally tested following an operating basis earthquake. It is our judgment that the likelihood that a seismic event of a lesser intensity than the operating basis earthquake will damage non-Class IE equipment to the extent that a safe shutdown cannot be initiated is so small as to not require consideration.

(5) It is our judgment that the likelihood that an operating basis earthquake will occur during the interim period that would (a) result in the occurrence of the sequence of events necessary to result in loss of the capability to scram the plant and (b) cause damage to non-Class IE equipment to the extent that a safe shutdown cannot be initiated in the time necessary to detect the seismic event and to initiate a safe shutdown is negligible considering the favorable operating history of this design.

IV. PUBLIC INTEREST REGARDING COMPLIANCE WITH CRITERION 2 OF APPENDIX A TO 10 CFR PART 50

To require specific conformance with the applicable requirement of Criterion 2 of Appendix A to 10 CFR Part 50 would necessitate delaying the startup of the plant until a Class IE system approved by us capable of de-energizing the reactor protection system power supply when its output voltage exceeds or falls below limits within which the equipment being powered from the power supply has been designed and qualified to operate continuously and without degradation is designed, fabricated, installed, and tested. The applicant estimates, and we agree, that such a system cannot be designed, fabricated, installed, and tested before the end of the first refueling outage.

The applicant estimates that the cost of replacement power to serve the needs of its customers during this period of time is approximately 93 million dollars. This amount is based on a replacement power cost of approximately 200,000 dollars per day and a 69 percent plant capacity factor. In addition, the applicant estimates that the capital cost of the plant will increase during this period of time by approximately 50 million dollars. This amount is based on the seven percent per annum allowance for funds used during construction. Finally, the applicant estimates that approximately 200 people would have to be maintained on the payroll during this period of time at a total cost of approximately 7.5 million dollars. We have reviewed these costs and their bases and conclude that they are reasonable.

It is our judgment, based on the favorable operating experience attained with essentially the same reactor protection system power supplies on operating BWR/4 reactors and on the sequence of events that must occur in order to result in the loss of capability to scram the plant, that the benefits of allowing the plant to operate during this period of time while a system that will enable the reactor protection system power supply to conform to the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50 is designed, fabricated, installed, and tested outweigh the cost to the public of delaying the startup of the plant.

We, therefore, conclude that the public interest is served by not imposing the applicable requirement of Criterion 2 of Appendix A to 10 CFR Part 50 until the end of the first scheduled refueling outage since such an imposition would be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

V. CONCLUSIONS

Based on the foregoing, we have determined that, pursuant to Section 50.12 of 10 CFR Part 50, a specific exemption as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption 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. We have concluded that this exemption would be insignificant from the standpoint of environmental impact and pursuant to Paragraph (d)(4) of Section 51.5 of 10 CFR Part 51 that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

Mark I

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF CRITERION 50 OF APPENDIX A TO 10 CFR PART 50

As discussed in Sections 3.8.1 and 6.2.1 of our Safety Evaluation Report for the Edwin I. Hatch Nuclear Plant Unit No. 2 (NUREG-0411) dated June 1978, we have completed our review of the generic Mark I Containment Short-Term Program conducted by the Mark I Owners Group, of which the applicant is a member, and the applicant's plant-unique analysis for the Hatch Unit 2 containment. The results of our review are documented in our "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, dated December 1977.

Based upon our review, we have concluded that Hatch Unit 2 can be operated safely, without undue risk to the health and safety of the public, during an inteirm period of approximately two years while a methodical, comprehensive Long-Term Program is conducted. This conclusion has been made based on our determination: (1) that the magnitude and character of each of the hydrodynamic loads resulting from a postulated design basis loss-of-coolant accident have been adequately defined for use in the Short-Term Program structural assessment of the Mark I containment system; and (2) that, for the most probable loads induced by a postulated design basis loss-of-coolant accident, a safety factor to failure of at least two exists for the weakest structural or mechanical component in the containment system for Hatch Unit 2.

As described in Section IV of NUREG-0408, our evaluation of the capability of each facility's Mark I containment system to withstand the recently identified loss-of-coolant accident related hydrodynamic suppression pool loads indicates that, although each of the structural and mechanical components of these containment systems meet the Short-Term Program structural acceptance criteria (i.e., a safety factor to failure of at least two), the demonstrated safety margins of certain components under these loading conditions are less than that which is necessary to satisfy the requirements of Section III of the ASME Code. Consequently, we conclude that the demonstrated safety margin of the Hatch Unit 2 containment system with respect to such loading conditions does not comply with our current interpretation of "sufficient margin" as prescribed by Criterion 50, "Containment Design Basis," of Appendix A to 10 CFR Part 50. For long-term operation of Hatch Unit 2, we require that the structural and mechanical components of the containment system meet the acceptance criteria of the ASME Code to the maximum extent practicable for the loads and loading combinations identified in the Mark I Containment Long-Term Program and approved by us.

However, we have found that: (1) the Hatch Unit 2 containment system design still retains sufficient margin under present conditions to preclude failure and thus provides reasonable assurance of no undue risk to the health and safety of the public, (2) the objective of the Mark I Containment Long-Term Program (i.e., to restore the originally intended design safety margins) is acceptable, and (3) the Mark I Owners' Program Action Plan for the Long-Term Program is reasonably designed to satisfy the Long-Term Program objectives. Therefore, we have found that operation of Hatch Unit 2, in conformance with the conditions specified in NUREG-0408, will not endanger life or property or the common defense and security.

In the absence of any safety problem associated with the operation of Hatch Unit 2 until the Long-Term Program is completed, there appears to be no public interest consideration favoring restriction of the operation of Hatch Unit 2. Accordingly, pursuant to Section 50.12 of 10 CFR Part 50, we have granted the applicant an exemption from Criterion 50 of Appendix A to 10 CFR Part 50, with respect to loss-of-coolant accident related hydrodynamic suppression pool loads, for an interim period until completion of the Long-Term Program (approximately two years), provided that the conditions specified in NUREG-0408 and any resulting technical specification requirements are maintained. To this extent, this exemption encompasses any related requirements of Section 50.55(a) of 10 CFR Part 50 and Criterion 1 of Appendix A to 10 CFR Part 50.

Furthermore, we have determined that the granting of this examption 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. We have concluded that this exemption would be insignificant from the standpoint of environmental impact and pursuant to Paragraph (d)(4) of Section 51.5 of 10 CFR Part 51 that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF APPENDICES G AND H TO 10 CFR PART 50

I. INTRODUCTION

In FSAR Section 5.2.4.2, the Georgia Power Company (GPCo) requested that the NRC staff evaluate their method of compliance with 10 CFR Part 50, Appendices G and H. On the basis of our review of this information, we advised GPCo that we would require the additional information in Questions 121.4, 121.5, 121.6, 121.10, 121.11, 121.12 and 121.14 to complete our evaluation of this matter. Georgia Power Company provided the additional supporting information in FSAR Amendment Nos. 18, 22, 35 and 41. As a result of our review of this information, we have recently determined that an exemption to 10 CFR Part 50, Appendices G and H is required and have also determined that an exemption regarding this matter is justified. Our basis for this conclusion is discussed in the subsequent paragraphs of this report.

II. TECHNICAL EVALUATION CONSIDERATIONS

A. The objective of Appendix G is to specify minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of water cooled power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. Specimens of the material of fabrication are required to be tested and the data used to develop safe operating condition limits for the reactor pressure vessel.

The objective of Appendix H is to monitor the change in fracture toughness properties of ferritic materials in the reactor vessel beltline region of water cooled power reactors resulting from exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens placed in the vessel before operation and withdrawn periodically during operation and tested to obtain fracture toughness data. These data permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The requirements of Appendices G and H to 10 CFR Part 50 are intertwined. Appendix G requires that the properties of reactor vessel beltline region materials, including welds, be monitored by a material surveillance program conforming with Appendix H. Appendix H in turn requires that the surveillance specimens be taken from locations alongside the fracture toughness test specimens required in Appendix G and that the specimen types comply with Appendix G except that drop weight specimens are not required.

The bulk of the detailed procedures and practices to be followed are given by way of reference to the ASME Code and ASTM Standards.

Determination of compliance with Appendices G and H requires therefore consideration of a cascade of requirements.

B. The requirements of 10 CFR Part 50, Appendices G and H became effective on August 16, 1973, after the construction permit for Hatch Unit No. 2 was issued. When Appendices G and H were published in the FEDERAL REGISTER on July 17, 1973, the Statement of Consideration stated the following: "...the Commission recognizes that there may be an interim period when, for plants now under construction, the method of compliance with certain provisions may be determined on a case-by-case basis. For example, if the test data needed to establish certain fracture control requirements are not available because they were not required at the time material sampling was done, estimated values that are appropriately conservative may be acceptable."

This statement was in recognition of the fact that compliance with Appendices G and H to 10 CFR Part 50 requires in turn compliance with Appendix G of Section III of the ASME Boiler and Pressure Vessel Code (the Code). Appendix G of Section III was first published in the Summer 1972 Addenda to the Code while the construction code for the reactor vessel of Hatch Unit No. 2 was the 1968 Edition including Addenda through Summer 1970. It is this disparity in time between the actual fabrication of the vessel and the effective date of Appendices G and H to 10 CFR Part 50 that brings about the need for consideration of an exemption. The practices employed in the 1968 edition of the

Code to assure adequate fracture toughness,although representing good technical practice, are not precisely those required to completely satisfy Appendices G and H to 10 CFR Part 50.

As discussed below the number and type of specimens taken during fabrication as well as some of the procedural, administrative and documentation requirements vary from full compliance with Appendices G and H to 10 CFR Part 50. In the following evaluation the staff has considered each type of variance and assessed the importance of those variances on the fulfillment of the safety objective of the regulation as well as the feasibility of requiring absolute compliance with the regulation.

III. EXEMPTIONS REQUIRED

We have reviewed the information submitted by the Georgia Power Company related to their method of compliance with 10 CFR Part 50, Appendices G and H. Based on this information and our review of the design, geometry, and materials of construction of the components, the requirement to comply with certain provisions of 10 CFR Part 50, Appendices G and H, have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Therefore, pursuant to 10 CFR Section 50.12 specific exemption for those requirements is justified as follows:

A. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

<u>Exemption Requested</u>: An exemption was requested by the Georgia Power Company to substitute an alternative Material Surveillance Program for the requirements of 10 CFR Part 50, Appendix H.

<u>Reason for Request</u>: Georgia Power Company stated in the FSAR that the reactor vessel surveillance program specimens meet the requirements of 10 CFR Part 50, Appendix H, and ASTM E 185-73 except for the following:

1. The base metal specimens are of the longitudinal rather than the transverse orientation.

2. Two of the three groups of impact specimens are in sets of eight rather than sets of 12 specimens.

3. The materials are from the beltline material but were chosen at random from the three beltline plates rather than in accordance with E 185-73.

<u>Bases and Conclusions</u>: The Charpy base metal impact specimens are of the longitudinal orientation consistent with the requirements of ASTM E 185-70, the rules in effect prior to the publications of 10 CFR Part 50 Appendix H. ASTM E 185-73 requires that the base metal specimens be of the transverse orientation; i.e., the major axis of the specimens be machined normal to the principal rolling direction for plates and normal

to the major working direction for forgings. Longitudinally oriented specimens are machined parallel to the principal rolling direction of the plates. The evaluation of the effects of irradiation can be performed with either transverse or longitudinal base metal impact specimens. Transversely-oriented Charpy V-notch specimens generally produce more conservative fracture toughness curves during laboratory tests of surveillance specimens. However, the equivalent conservatism will be obtained by applying established standard correlating factors to the test data obtained from available longitudinal specimens. In addition, as the material surveillance specimens are irradiated during reactor operation, the use of either transverse or longitudinal base metal specimens will become less significant because the available information from actual reactor vessel materials show that the beltline weldments will become the controlling factor in the evaluation of irradiation effects.

ASTM E 185-73 requires that each exposure set contain a minimum of 12 base metal Charpy specimens, 12 weld metal Charpy specimens, and 12 HAZ Charpy specimens. In the capsules installed in the reactor, two of the three groups of impact specimens are in sets of eight rather than sets of 12 specimens. ASTM E 185-70 required that each exposure set contain a minimum of eight base metal and weld metal impact specimens and eight impact specimens from the heat-affected zone. Based on industrial practice, existing material surveillance programs for operating reactors are currently being evaluated with eight specimens. Our technical evaluation determined that eight specimens are sufficient to establish the effect of radiation on the fracture toughness properties of the beltline material.

ASTM E 185-73 requires that material specimens be selected from locations in the beltline region with fracture toughness properties that will limit plant operation. When the irradiated capsules are withdrawn and tested at designated intervals, our detailed evaluation of the results will ensure that the fracture toughness properties from randomly selected base metal specimens represent the limiting conditions. In addition, as the material surveillance specimens receive irradiation, the random selection of base metal specimens will be less significant since the residual elements in the beltline weldments will become the controlling factor. However, the random selection will not be significant during the first 10 years of operation because the regions near geometric discontinuities, remote from the beltline, are the controlling factor for initial operation. After 10 years of operation, the results from the first surveillance capsule will determine the limiting conditions and this process will continue throughout the life of the plant.

B. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"

Exemption Requested: An exemption was requested by the Georgia Power Company to substitute an alternative method of compliance for the requirements of 10 CFR Part 50, Appendix G.

Reasons for Request: Georgia Power Company stated in the FSAR that it is not possible to comply with 10 CFR Part 50, Appendix G, with components which were purchased to earlier ASME Code requirements without the replacement of large amounts of materials, reworking of fabricated components, and the revision of most of the design analysis for these components.

GPCo proposes to provide operating limitations on pressure and temperature for the reactor pressure vessel based on fracture toughness properties as the basic method of compliance with Appendix G of the ASME Code, Section III. The operating limitations were established for normal heatup and cooldown and during hydrostatic testing using as a guide, Appendix G, of the ASME Code Section III.

Bases and Conclusions: We have determined that the essential requirements for the Edwin I. Hatch, Unit No. 2, reactor vessel to comply with 10 CFR Part 50, Appendix G, are the following:

1. A material surveillance program in accordance with 10 CFR Part 50, Appendix H. This requirement may seem redundant after the discussion in III. A. above but this is a consequence of the intertwining of the requirements of Appendices G and H discussed in Section II.

2. Material surveillance specimens fabricated and tested in accordance with ASME Code, Section III, Article NB-2300, Summer 1972 Addenda.

3. The development of operating condition limitations based on the ASME Code, Section III, Appendix G, Summer 1972 Addenda.

With regard to 1 and 2 above the material surveillance program and the specific requirement for transversely oriented impact specimens in Paragraph NB-2322 have previously been discussed associated with compliance with Appendix H.

In addition to the structure and content of the material surveillance program as discussed in conjunction with our review of compliance with Appendix H above and the development of operating limits to be discussed as the last item of consideration, there are a number of procedural, administrative and documentation requirements contained in Appendix G. The purpose of these requirements is to ensure that data of the quality and quantity necessary to fulfill the goals of this Appendix are developed; such as,

> Paragraph III.B.3 related to the calibration of test instruments,
>
> Paragraph III.B.4 related to the qualification of test personnel,
>
> Paragraph III.B.5 related to records and certifications.

Given that the actual ordering of materials, the fabrication of the Hatch Unit No. 2 vessel and the development of the testing program for the first material samples occurred well before Appendix G was effective, we have found little meaning in a step by step comparison of the procedures of Appendix G cited above and the actual procedures employed. We have instead reviewed the procedures actually employed on their merits.

These procedures are contained in the 1968 edition of the Code, paragraphs N-331 and N-332, and by reference in ASTM E 208 and ASTM A 370. These are long standing procedures that have been utilized successfully over many years for nuclear and non-nuclear components.

Based on our review we have concluded that the practices followed by the applicant in relation to each of the matters cited above would provide data of a quantity and quality sufficient to accurately characterize the fracture toughness properties of the materials being tested. The goals of Appendix G to 10 CFR Part 50 would therefore be fulfilled with respect to these matters.

Item 3, the development of operating limits, is the final item necessary to be considered in the review of the requested exemption.

The requirements for development of operating limits are given in Appendix G to 10 CFR Part 50 by reference to Section III, Appendix G, and Article NB-2300, Summer 1972 Addenda of the Code. Appendix G of Section III is a non-mandatory appendix for ASME Code applications.

and is written in terms of general recommendations, guidelines, opinions, and proposed alternatives. Recognizing this format, 10 CFR Part 50, Appendix G, paragraph IV.A.2.a states "The calcuation procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture making appropriate allowance for all uncertainties in the data and analyses."

To implement the intent expressed by theStatement of Consideration and Paragraph IV.A.2.a and to establish an orderly and consistent licensing review process, Standard Review Plan (SRP) Sections 5.2.3, 5.3.1 and 5.3.2 were published. A specific objective of SRP Section 5.3.2 was to establish a method of compliance with 10 CFR Part 50, Appendix G during the interim period cited in the Statement of Consideration. This was accomplished in SRP Section 5.3.2 by defining acceptable and conservative guidelines to assure that (1) the fracture toughness of the materials for plants under construction on August 16, 1973 are assessed by using the available test data to estimate the fracture toughness in the same terms as the requirements of Appendix G of the Code and (2) the operating limitations imposed to provide the same safety margins as the requirements of Appendix G. Further, SRP Section 5.3.2 incorporates the principles and objectives of Appendix G of the ASME Code in terms of specific requirements. In addition to reproducing the essential equations and figures from Section III, Appendix G, SRP Section 5.3.2

provides conservative estimated values for fracture toughness properties that may not be available from test data and provides sample calculations for a consistent licensing review. Although a plant does not comply with the specific provisions of the ASME Code, Section III, Appendix G, an equivalent margin of safety is obtained by establishing the pressure-temperature limits defined in the Technical Specifications based on SRP Section 5.3.2. The operating limits proposed by the applicant were reviewed in accordance with SRP Section 5.3.2 and found to be acceptable thus fulfilling the goal of Appendix G.

Our technical evaluation has not identified any practical method by which the existing Hatch Unit No. 2 reactor vessel can meet the specific requirements of 10 CFR Part 50, Appendix G. However, based on our review we conclude that Georgia Power Company has provided, with the available material test data, a satisfactory alternative program to the requirements of 10 CFR Part 50, Appendix G.

IV. PUBLIC INTEREST REGARDING COMPLIANCE WITH 10 CFR PART 50, APPENDICES G AND H

Our technical evaluation has not identifed any practical method by which the existing Edwin I. Hatch, Unit No. 2, reactor vessel can meet the specific requirements of 10 CFR Part 50, Appendices G and H. Requiring specific compliance with these Appendices would include the following actions: delay the startup of the plant and remove the installed material surveillance capsules, design and fabricate new capsules containing three sets of 12 Charpy impact specimens, obtain, if possible, sufficient material from the actual Hatch Unit No. 2 beltline plates to fabricate transverse specimens, reassemble and install the new surveillance capsules. We believe the public interest is served by not imposing the certain provisions of 10 CFR Part 50, Appendices G and H, that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

V. CONCLUSIONS

Based on the foregoing, we have determined that, pursuant to 10 CFR Section 50.12, a specific exemption as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption 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. We have concluded that this exemption would be 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 this action.