SEP 2 7 1974

Docket No. 50-321

Georgia Power Company
ATTN: Mr. I. S. Mitchell, III
Vice President & Secretary
P. O. Box 4545
Atlanta, Georgia 30302

Gentlemen:

The Atomic Energy Commission has issued the enclosed Amendment No. 1 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1 in response to your application of August 16, 1974, for a change to the Technical Specifications for this license. Change No. 1 to Appendix A of the Technical Specifications, attached to the amendment, incorporates the requested change into the license. Change No. 1 also removes Temporary Restriction No. 3 from the Technical Specifications. The enclosed Safety Evaluation identifies and evaluates the changes which are covered by this license amendment.

Also enclosed is a notice of issuance which will be forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/ Karl Kniel

for Voss A. Moore, Assistant Director
for Light Water Reactors, Group 2
Directorate of Licensing

Enclosures:
1. Amendment No. 1 to DPR-57
   w/change No. 1 to Appendix
   A, Technical Specifications
2. Safety Evaluation
3. Federal Register Notice

cc: (See next page)

OFFICE▶						
SURNAME▶						
DATE▶						

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ACRS (16)

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No legal objection

OFFICE▶	L:GCR	LWR 2-1	LWR-2	OGC		
SURNAME▶	H Gearin HGearin:aw	JFStolz	for R Knie VAMoore	SHL S H Lewis		
DATE▶	9/23/74	9/27/74	9/27/74	9/27/74		

Docket 50-321

September 27, 1974

Note to Voss A. Moore

I have noted herein that OGC has "no legal objection" to this package. We are concerned, however, about the manner in which this Technical Specification limitation was removed. The note at the bottom of the "Temporary Restrictions" states that "Regulatory Operations shall advise the staff in writing upon satisfactory completion of these work items". In our opinion it is poor regulatory posture to rely in these circumstances simply upon the "advice" of the Applicant that the items have been completed.

Stephen H. Lewis

Stephen H. Lewis

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

GEORGIA POWER COMPANY

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 1
License No. DPR-57

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for amendment by the Georgia Power Company (the licensee) dated August 16, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-57 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 1."

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

FOR THE ATOMIC ENERGY COMMISSION

Karl Kniel FOR

Voss A. Moore, Assistant Director
for Light Water Reactors, Group 2
Directorate of Licensing

Attachment:
Change No. 1 to Appendix A
Technical Specifications

Date of Issuance: SEP 27 1974

GEORGIA POWER COMPANY

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 1
License No. DPR-57

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   A. The application for amendment by the Georgia Power Company (the licensee) dated August 16, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-57 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 1."

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

FOR THE ATOMIC ENERGY COMMISSION

/s/ Karl Kniel

for Voss A. Moore, Assistant Director
for Light Water Reactors, Group 2
Directorate of Licensing

Attachment:
Change No. 1 to Appendix A
Technical Specifications

Date of Issuance: SEP 2 7 1974

No legal objection

OFFICE➤	L:GCR	LWR 2-1	LWR-2-1	OGC	LWR 2 K Kniel	
SURNAME➤	HGearin	SBurwell	JStolz	S.H. Lewis	for VAMoore	
DATE➤	9/23/74	9/26/74	9/27/74	9/27/74	9/27/74	

TEMPORARY RESTRICTIONS

Prior to achieving power levels which exceed one percent of rated power, the following work items are to be completed.

1. Install seismic restraints on piping.

2. Complete pre-operational testing of the following systems:

   (a) Process liquid radiation monitors
   (b) Main stack radiation minitors
   (c) Leak detection system
   (d) Air ejector off gas system
   (e) Off gas monitor
   (f) Radwaste ventilation system
   (g) Turbine building ventilation system
   (h) Radwaste system
   (i) Main steam radiation monitors
   (j) Condensate system
   (k) Feedwater system

Regulatory Operations shall advise the staff in writing upon satisfactory completion of these work items. The temporary restriction will then be removed by the staff.

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

The Safety Limit, established to preserve the reactor coolant system integrity, applies to the limit on the reactor vessel steam dome pressure.

Objective

The objective of the Safety Limit (associated with preserving the reactor coolant system integrity) is to establish a pressure limit below which the integrity of the reactor coolant system is not threatened due to any overpressure condition.

Specifications

A. Reactor Vessel Steam Dome Pressure

1. When Irradiated Fuel is in the Reactor

The reactor vessel steam dome pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the reactor vessel steam dome pressure Safety Limit from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the reactor vessel steam dome pressure Safety Limit from being exceeded.

Specifications

A. Nuclear System Pressure

1. When Irradiated Fuel is in the Reactor

When irradiated fuel is present in the reactor vessel, and the head is bolted to the vessel, the limiting safety system settings shall be as specified below:

Protective Action	Limiting Safety System Settings (psig)
a. Scram on high reactor pressure (reactor vessel steam dome pressure)	≤ 1045
b. Nuclear system relief valves open on nuclear system pressure	4 valves @ 1080 4 valves @ 1090 3 valves @ 1100

The allowable setpoint error for each valve shall be ± 1%.

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1.2. REACTOR COOLANT SYSTEM INTEGRITY

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

A. Reactor Vessel Steam Dome Pressure

1. When Irradiated Fuel is in the Reactor

The pressure Safety Limit of 1325 psig as measured by the reactor vessel steam dome pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressure of the reactor pressure vessel (1250 psig) and coolant system piping (suction piping: 1150 psig; discharge piping 1350 psig). The pressure Safety Limit was chosen as the lower pressure resulting from the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1150 = 1380 psig; 120% X 1350 = 1620 psig).

The pressure relief system (relief/safety valves) has been sized to meet the overpressure protection criteria of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The details of the overpressure protection analysis showing compliance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels is provided in the FSAR, Appendix M, Summary Tecnical Report of Reactor Vessel Overpressure Protection. To determine the required steamflow capacity, a parametric study was performed assuming the plant was operating at the turbine-generator design condition of 105 percent rated steam flow (10.6 X 10^6 pounds per hour) with a vessel dome pressure of 1020 psig, at a reactor thermal power of 2537 Mw, and the reactor experiences the worst pressurization transient. The analysis of the worst overpressure transient, a 3 second closure of all main steam line isolation valves neglecting the direct scram (valve position scram) results in a maximum vessel pressure (bottom) of 1319 psig if a pressure scram is assumed, or 1281 psig if a neutron flux scram is assumed. In addition, the same event was analyzed to determine the number of installed valves which would limit pressure to below the code limit. The results of this analysis show that eight of the eleven installed relief/safety valves are adequate even if assuming the backup neutron flux scram, and provide 25 psi margin.

Turbine trip from high power without bypass is the most severe transient resulting directly in a nuclear system pressure increase, assuming the turbine trip scram. This event is presented in FSAR Section 14.3.1.2.2. The analysis shows that even with only nine relief valves opening the peak pressure in the bottom of the vessel is limited to 1227 psig. Peak steam

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1.2.A.1. When Irradiated Fuel is in the Reactor (Continued)

line pressure is 1192 psig, showing adequate protection for this abnormal operational transient.

2. When Operating the RHR System in the Shutdown Cooling Mode

An interlock exists in the logic for the RHR shutdown cooling valves, which are normally closed during power operation, to prevent opening of the valves above a preset pressure setpoint of 135 psig. This setpoint is selected to assure that pressure integrity of the RHR system is maintained. Administrative operating procedures require the operator to close these shutdown cooling valves prior to pressure operation. However, as a backup, the interlock will automatically close these valves when the pressure setpoint is reached. Double indicating lights will be provided in the control room for valve position indication.

2.2 REACTOR COOLANT SYSTEM INTEGRITY

A. Nuclear System Pressure

1. When Irradiated Fuel is in the Reactor

The 11 relief/safety valves are sized and set point pressures are established in accordance with the following requirements of Section III of the ASME Code:

a. The lowest relief/safety valve must be set to open at or below vessel design pressure and the highest relief/safety valve be set to open at or below 105% of design pressure.

b. The valves must limit the reactor pressure to no more than 110% of design pressure.

The primary system relief/safety valves are sized to limit the primary system pressure, including transients, to the limits expressed in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. No credit is taken from a scram initiated directly from the isolation event, or for power operated relief/safety valves, sprays, or other power operated pressure relieving devices. Thus, the probability of failure of the turbine-generator trip SCRAM or main steam isolation valve closure SCRAM is conservatively assumed to be unity. Credit is taken for subsequent indirect protection system action such as neutron flux SCRAM and reactor high pressure SCRAM, as allowed by the ASME Code. Credit is also taken for the dual relief/safety valves in their ASME Code qualified mode of safety operation. Sizing on this basis is applied to the most severe pressurization transient, which is the main steam isolation valves closure, starting from operation at 105 percent of the reactor warranted steamflow condition.

Reference 2, Figure 4 shows peak vessel bottom pressures attained when the main steam isolation valve closure transients are terminated by various modes of reactor scram, other than that which would be initiated directly from the isolation event (trip scram). Relief/safety valve capacities for this analysis are 84.0 percent, representative of the 11 relief/safety valves.

The relief/safety valve settings satisfy the Code requirements for relief/safety valves that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief/safety valve actuation is required are given in Section 14.3 of the FSAR.

2. When Operating the RHR System in the Shutdown Cooling Mode

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome pressure less than 135 psig.

2.2.B. References

1. FSAR Section 14.3, Analysis of Abnormal Operational Transients

2. FSAR Apperdix M, Summary Technical Report of Reactor Vessel Overpressure Protection.

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3.6.H. Relief/Safety Valves

When more than one relief/safety valve is known to be failed an orderly shutdown shall be initiated and the reactor depressurized to less than 113 psig within 24 hours. Prior to reactor startup from a cold condition all relief/safety valves shall be operable

I. Jet Pumps

Whenever the reactor is in the Start & Hot Standby or Run Mode with both recirculating pumps operating, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

4.6.H. Relief/Safety Valves

1. End of Operating Cycle

Approximately one-half of all relief/safety valves shall be benchchecked or replaced with a benchchecked valve each refueling outage. All 11 valves will have been checked or replaced upon the completion of every second operating cycle.

2. Each Operating Cycle

Once during each operating cycle, at a reactor pressure >100 psig each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

3. Integrity of Relief Valve Bellows

The integrity of the relief valve bellows shall be continuously monitored and the pressure switch calibrated once per operating cycle and the accumulators and air piping shall be inspected for leakage once per operating cycle.

4. Relief Valve Maintenance

At least one relief valve shall be disassembled and inspected each operating cycle.

I. Jet Pumps

Whenever both recirculating pumps are operating with the reactor in the Start & Hot Standby or Run Mode, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

3.6.G. Reactor Coolant Leakage (Continued)

would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in Specification 3.6.G on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in Specification 3.6.G, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation (Reference FSAR, Question 10.4.2). Leakage less than the magnitude specified can be detected reasonably in a manner of a few hours utilizing the available leakage detection scheme, and if the origin cannot be determined in a reasonably short time the plant shall be shut down to allow further investigation and corrective action. The total leakage rate consists of all leakage, identified and unidentified which flows to the drywell floor drain and equipment drain sump. The capacity of the drywell floor sump pumps is 100 gpm and the capacity of the drywell equipment sump pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

H. Relief/Safety Valves

The pressure relief system (relief/safety valves) has been sized to meet the overpressure protection criteria of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The details of the overpressure protection analysis showing compliance with ASME, Section III is provided in the FSAR, Appendix M, Summary Technical Report of Reactor Vessel Overpressure Protection. To determine the required steamflow capacity, a parametric study was performed assuming the plant was operating at the turbine-generator design condition of 105 percent rated steam flow (10.6×10^6 pounds per hour) with a vessel dome pressure of 1020 psig, at a reactor thermal power of 2537 Mw, and the reactor experiences the worst pressurization transient. The analysis of the worst overpressure transient, a 3 second closure of all main steam line isolation valves neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1319 psig if a pressure scram is assumed, or 1281 psig if a neutron flux scram is assumed. In addition, the same event was analyzed to determine the number of installed valves which would limit pressure to below the code limit. The results of this analysis show that eight of the eleven installed relief/ safety valves are adequate, even if assuming the backup neutron flux scram and provide 25 psi margin.

Turbine trip from high power without bypass is the most severe transient resulting directly in a nuclear system pressure increase, assuming the turbine trip scram. This event is presented in FSAR Section 14.3.1.2.2. The analysis shows that even with only nine relief valves opening the peak pressure in the bottom of the vessel is limited to 1227 psig. Peak steam line pressure is 1192 psig, showing adequate protection for this worst abnormal operational transient.

3.6.H. Relief/Safety Valves (Continued)

Experience in relief/safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failure or deteriorations. The relief/safety valves are benchtested every second operating cycle to ensure that their set points are within the ± 1 percent tolerance. The relief/safety valves are tested in place at low reactor pressure once per operating cycle to establish that they will open and pass steam.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

I. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5%, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

CHANGE NO. 1
September 27, 1974

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 1 TO LICENSE NO. DPR-57

CHANGE NO. 1 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-321

1.0 INTRODUCTION

The Atomic Energy Commission issued Facility Operating License No. DPR-57 to the Georgia Power Company on August 6, 1974. That license authorized the operation of the Edwin I. Hatch Nuclear Plant Unit 1 at reactor core power levels not in excess of 24 megawatts thermal in accordance with the Technical Specifications appended thereto. The activities authorized by that license are limited to fuel loading, low power testing, and operation at 1% of the facility's power level of 2436 megawatts thermal until the remaining items listed in Technical Specifications, Appendix A, have been satisfactorily completed. The outstanding work items which are to be completed prior to the staff authorizing operations at higher power levels are listed under Temporary Restrictions in Appendix A.

By letters dated August 16, 1974, Georgia Power Company amended their application (Amendment 47) to describe a design change to the overpressure protection system in which the two safety valves on the reactor primary coolant boundary are replaced by two relief/safety valves, and requested a change in the Technical Specifications, Appendix A, to provide for station operation with the modified design.

The purposes of this Safety Evaluation are:

(a) to describe the Regulatory staff review of the design change from 2 safety valves and 9 relief/safety valves to 11 relief/safety valves;

(b) to describe the evaluation of the changes to Technical Specifications, Appendix A, that are necessary to provide for the modified design;

(c) to document that the following work item has been completed and is therefore removed from the list of work items described in the Temporary Restrictions:

"3. Install main steam safety valves. (The primary coolant system shall be vented to the atmosphere prior to installation of the main steam safety valves.)"

2.0 DISCUSSION

2.1 OVERPRESSURE PROTECTION SYSTEM DESIGN CHANGE (Section 5.2.2 of the SER)

By Amendment 47, submitted by letter dated August 16, 1974, the Georgia Power Company revised portions of FSAR Section 4.4 Pressure Relief System and Appendix M, Summary Technical Report of Reactor Overpressure Protection in support of its proposed design change in the overpressure protection system for the Edwin I. Hatch Nuclear Plant Unit 1. The licensee proposed to replace the two Dresser Type safety valves with two Target Rock relief/safety valves. The overpressure protection system will consist of 11 relief/safety valves all of which are of similar design. The new relief/safety valves are to be installed on the same flanges provided for the two safety valves. However, the steam released from all relief/safety valves will be discharged through an exhaust pipe and submerged rams head into the torus pool. The design change eliminates the potential safety valve discharge directly into the drywell atmosphere. The design modification provides an increased overpressure relief capacity, i.e., the relief/safety valves are rated at about 800,000 lbs/hr each, and the replaced safety valves are rated at about 640,000 lbs/hr. The design modification also provides a lower pressure (earlier) actuation of the two valves should the reactor vessel experience a pressure transient; i.e., the two relief/safety valves are set to open on a steam line pressure of 1080 psig and 1090 psig, but the replaced safety valves were to be set to open at 1240 psig.

The new relief/safety valves and their discharge piping are designed, constructed and tested to the same requirements as the nine other, previously approved relief/safety valves.

The transients from both the turbine trip without bypass and main steam line isolation valve (MSLIV) closure were analyzed and the MSLIV closure transient was determined to be the more severe, assuming a failure of the valve position scram. The transient analyses presented in Amendment 47 are based on delay and stroke times for the relief/safety valves of 0.4 and 0.1 seconds respectively consistent with current analyses for other BWR nuclear steam supply systems. This modification in the analysis results in a small increase in the peak pressures calculated for the above transients.

The proposed overpressure protection system will meet the requirements of Article 9, Section III of the ASME Pressure Vessel Code, 1968.

2.2 Changes to Technical Specification Appendix A to Provide for Modified Design

By letter dated August 16, 1974, Georgia Power Company requested revisions to the Technical Specification, Appendix A to provide for station operation with the overpressure protection system modified as described in Amendment 47 to the application. The requested revisions remove reference to the 2 safety valves and their set point pressure, increase the relief/safety valves (by 2) and specify their respective set point pressures, and update the description of the overpressure protection analysis to present the results of the new analysis reported in Amendment 47.

2.3 Completion of Work Item 3 of the Temporary Restrictions

Regulatory Operations, Region II, has been advised that all eleven of the relief/safety valves comprising the overpressure protection system as described in Amendment 47 have been tested, installed and declared operational in accordance with station procedures.

3.0 EVALUATION

3.1 Overpressure Protection System Design Change

The licensee's evaluation of the modified overpressure protection system was performed using the same methods, transients, and requirements which were previously reviewed and found acceptable by the Regulatory staff. The results of these transient analyses confirm that the proposed overpressure protection system design will satisfy the requirement that the maximum pressure experienced by the reactor pressure vessel will be less than the Code limit of 1375 psig. Further, the licensee reported that the peak pressure experienced by the reactor vessel for main steam line isolation valve closure was 1281 psig, indicating a margin of 94 psi lower than the Code limit. This peak pressure is 5 psi lower than previously reported in FSAR Section 4.4.6 for the same transient. In view of the change in valve closure time used in the analysis, we conclude that the real overpressure protection margin has been increased by more than the 5 psi as a result of the increased releasing capacity and earlier actuation of the replacement valves. Based on the previously approved and accepted overpressure protection system design, and on our review of the information submitted in Amendment 47, we conclude that the replacement of the 2 safety valves with 2 relief/safety valves of similar design to those previously approved is acceptable.

3.2 Changes to Technical Specification, Appendix A

The staff has reviewed the changes to the Technical Specification, Appendix A, requested by the licensee's letter dated August 16, 1974, and find that the changes are consistent with the design modification to the overpressure protection system as evaluated and approved above. Further, the staff concludes that the changes do not reduce the protection afforded by the Technical Specifications as previously issued and are therefore acceptable. These changes to the Technical Specifications are indicated by the revision mark on pages 1.2-1, 1.2-3, 1.2-4, 1.2-6, 1.2-7, 3.6-9, 3.6-20 and 3.6-21; which supersede and replace the respective pages of that document. The revised pages are attached as Enclosure 1 to the license amendment.

3.3 Completion of Work Item 3 of the Temporary Restrictions

Regulatory Operations has advised the staff in writing that the relief/safety valves as described in Amendment 47 have been satisfactorily installed. The Temporary Restriction Item 3 is therefore found completed and hereby deleted from the Technical Specification, Appendix A.

4.0 CONCLUSION

The design modification to the overpressure protection system, the changes to the Technical Specifications reflecting that design change, and the installation of the two relief/safety valves described above do not compromise the safe operation of the Edwin I. Hatch Nuclear Plant Unit 1.

The staff concludes that the changes do not involve a significant hazards consideration since they do not involve a safety consideration of a type or magnitude not previously considered for the facility, do not involve a substantial increase in the probability or consequences of accidents previously considered, and do not involve a substantial decrease in the margin of safety during normal plant operation, anticipated operational occurrences, or postulated accidents previously considered. There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Spottswood B. Burwell

Spottswood B. Burwell, Project Manager
Light Water Reactors Branch 2-1
Directorate of Licensing

Dennis M. Crutchfield for

John F. Stolz, Chief
Light Water Reactors Branch 2-1
Directorate of Licensing

Dated: SEP 27 1974

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 1 to Facility Operating License No. DPR-57 issued to the Georgia Power Company which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant Unit 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment permits the replacement of two safety valves on the main steam lines within the dry well with relief/safety valves, as described in Amendment 47 to the Final Safety Analysis Report (FSAR). Because installation of the main steam relief/safety valves has been completed, this amendment removes Temporary Restriction No. 3 from the Technical Specifications.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

For further details with respect to this action, see (1) the application for amendment dated August 16, 1974, (2) Amendment 47 to the FSAR, (3) Amendment No. 1 to License No. DPR-57, with any attachments, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Appling County Public Library, Parker Street, Baxley, Georgia 31513.

A copy of items (3) and (4) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 27 day of September, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Dennis M. Crutchfield

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Light Water Reactor Projects Branch 2-1
Directorate of Licensing