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Georgia Power
the southern electric system

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Docket Nos. 50-321 HL-4561
50-366

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant
Request for Temporary Technical Specifications Revision:
<u>Allow Power Uprate Testing</u>

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Georgia Power Company (GPC) hereby requests temporary changes to Operating Licenses Nos. DPR-57 and NPF-5 and Appendices A thereto, for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, respectively. The proposed temporary changes will allow the planned testing at Plant Hatch to demonstrate the capability to operate the plant up to a core power level of 2558 MWt. This testing is part of the overall power uprate program and is intended to evaluate the physical effects of increasing the licensed plant power level. The total duration time above the current operating limit for the testing on each unit is not to exceed 30 days. Approval of this request is expected to reduce the cost of the power uprate project by approximately one million dollars.

Enclosure 1 contains a detailed description of the specific proposed changes necessary for performing the proposed test and the technical bases for the changes. Georgia Power Company will also commit to certain administrative controls while performing this test. Specifics are also contained in Enclosure 1. Enclosure 2 addresses individual topics related to power uprate utilizing the format and content contained in the Power Uprate Safety Evaluation Report issued for the Detroit Edison Fermi 2 plant. Enclosure 3 contains bases for our determination that the proposed changes do not involve a significant hazards consideration. Enclosure 4 provides an environmental assessment. Enclosure 5 contains the page change instructions for incorporating the proposed changes. The revised, proposed Technical Specification pages, follow Enclosure 5.

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Georgia Power

In order to support the testing of the plant at the most optimum conditions, GPC requests that this proposed Technical Specification change be approved no later than August 8, 1994. This date supports testing Unit 2 during a period of near limiting environmental conditions. Also, this 1994 Unit 2 test will support delivery and lead time for any high pressure turbine hardware required for power uprate in the Fall of 1995.

The schedule for Unit 1 testing has not been finalized but will occur sometime after startup from its Fall 1994 refueling outage. If the testing is not completed before Unit 1 implementation of the Improved Technical Specifications, revised Technical Specifications pages in the new format will be provided.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated State official of the Environmental Protection Division of the Georgia Department of Natural Resources.

Mr. J. T. Beckham, Jr. states he is Vice President of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

GEORGIA POWER COMPANY

BY: J. T. Beckham, Jr.

J. T. Beckham, Jr.

Sworn to and subscribed before me this 7th day of June, 1994.

Doris Delene Brown

Notary Public

MY COMMISSION EXPIRES NOVEMBER 3, 1997

GKM/cr

Enclosures: (See next page.)

Georgia Power

Enclosures:
1. Bases for Change Request
2. Additional Bases for Change Request
3. 10 CFR 50.92 Evaluation
4. Environmental Impact Evaluation
5. Page Change Instructions

cc: *Georgia Power Company*
Mr. H. L. Sumner, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch
Mr. C. Grimes, Technical Specifications Branch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

State of Georgia
Mr. J. D. Tanner, Commission - Department of Natural Resources

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request for Temporary Technical Specifications Revision:
Allow Power Uprate Testing

Bases for Change Request

Proposed Change 1

This proposed change will temporarily revise the Unit 1 and 2 operating licenses and the Unit 1 Technical Specification Bases for limiting safety system settings to allow each unit to be operated above the current license limit for the maximum steady state reactor core thermal power level of 2436 MWt. This change will allow testing of the plant to be performed up to 2558 MWt, 105% of the current maximum steady state power level. The cumulative total of time spent for each plant with the reactor core operating at a power level in excess of 2436 MWt but less than or equal to 2558 MWt is not to exceed 30 days.

Bases for Proposed Change 1

Discussion

A substantial number of analyses related to the balance of plant performance have been performed in previous "stretch power" uprate programs by other utilities. These analyses, and those which will be performed for Hatch, are generally of limited value due to the uncertainties in the model and the lack of operational data at power levels above rated conditions. By performing testing at the uprated conditions, operational data can be obtained which can be used to largely eliminate these uncertainties and allow more focused plant performance evaluations. This will substantially reduce the overall cost of the power uprate program for Plant Hatch.

Both units at Plant Hatch were originally designed and the safety analysis performed for a maximum power level of 2537 MWt, which corresponds to 105% of the rated steam flow. This power level is often referred to as "stretch power." This power level corresponds to approximately 104.2% of the current license limit or rated power level at 100% rated steam flow of 2436 MWt. Because of the significant economic advantages of operating at a higher power level, Georgia Power Company (GPC) intends to pursue a permanent amendment to the operating license for each unit at Plant Hatch, which will enable them to be operated at power levels up to 105% of the current rated power level (i.e., approximately 0.8% above the "stretch power" level).

Other plants similar in design to Plant Hatch have already received NRC approval to operate at "stretch power." For the necessary permanent license changes, including Technical Specification changes, submittals will be made which are consistent with NEDC-31897P-1 (Reference 2). All of the required safety analyses for Plant Hatch are currently being re-evaluated as a part of the request for the permanent operating license amendment to be submitted in accordance with 10 CFR 50.90.

For the proposed test program, test data will be taken at various power levels between 100% and 105% of rated. In addition, the steam dome pressure will be increased up to a maximum of 30 psi above the current steady state rated conditions. The increase in steam dome pressure is necessary to determine the turbine performance characteristics as a function of turbine inlet pressure and to accommodate the higher steamline pressure drop associated with the higher steam flow rate at the increased power level. General Electric Company, the nuclear steam supply system and turbine generator supplier for Plant Hatch, has participated in the preparation of this submittal and will aid in the formulation of the detailed test plan.

The proposed test program is designed to minimize the number of plant changes required to perform the necessary testing. In addition to modifying the operating license to operate up to 105% power, only three Technical Specification changes are being requested: (1) an increase in the limiting safety system setting for the high pressure scram (see Proposed Change 2); (2) an increase in the limiting condition for operation for the average power range monitor (APRM) rod block (see Proposed Change 3); and (3) an increase in the limiting condition for operation for the low low set safety/relief valve arming (see Proposed Change 4). A detailed review of plant operations indicates that these Technical Specification changes are sufficient to allow the plant to be temporarily operated in a manner to obtain the necessary test data. This number of requested changes is significantly less than will be necessary for the full implementation of the power uprate which will be needed to assure an acceptable margin for operational flexibility.

Previous analyses have demonstrated that the impact of "stretch power" on the relevant safety analyses is small. These analyses are documented in generic assessments (References 2 and 3) and plant specific analyses for Hatch (References 4, 5, and 6). Detailed plant-specific analyses have also been performed for several BWR/4 plants using the Reference 2 guidance, including two 218 inch BWR/4 plants with Mark I containments (References 4, 5, and 6).

Based on the assessments provided in NEDC-31897P-1 for an increase of 5% steam flow and supplementary assessments performed by GPC for the bounding licensing criteria, the proposed test program can be conducted with an acceptable margin of safety. These assessments include an evaluation of the following criteria: (1) the peak fuel cladding temperature for the postulated loss of coolant accident; (2) the operating limit minimum critical power ratio; (3) the code overpressure protection analysis; (4) the peak primary containment pressure for the postulated loss of coolant accident; (5) the peak suppression pool temperature for the postulated loss of coolant accident; (6) the calculated off-site doses for postulated design basis accidents; (7) the impact on on-site radiation exposure during normal operation; (8) the change in water discharge temperature to the environment; and (9) stability. These assessments are summarized below.

Peak Clad Temperature

NEDC-31897P-1 indicates that the maximum increase in peak fuel cladding temperature for the postulated loss of coolant accident for "stretch power" is expected to be less than 20°F. For Plant Hatch, the peak cladding temperature has been calculated using the GE Nuclear Energy SAFER/GESTR methodology at 105% steam flow , a vessel dome pressure of 1055 psia, and a 2% uncertainty (Reference 7). The calculated licensing basis peak clad temperatures were determined to be 1509°F and 1526°F for Units 1 and 2, respectively. An increase of 0.8% in thermal power to achieve 105% would have an insignificant impact on peak clad temperature. Therefore, substantial margin exists to the regulatory limit of 2200°F, and there is no need to modify the operating limit maximum average planar linear heat generation rate in the Core Operating Limits Report for the proposed test.

Minimum Critical Power Ratio

NEDC-31897P-1 indicates that the maximum change in operating limit minimum critical power ratio for power uprate conditions is less than 0.02 at "stretch power". To assure that there is an acceptably low probability of exceeding the safety limit minimum critical power ratio during the proposed test, an analysis of the limiting events that can establish the minimum critical power ratio operating limit will be performed prior to the test. This analysis will be used to establish any changes required to the Core Operating Limits Report. It is anticipated, that by taking credit for the actual test conditions, no changes to the minimum critical power ratio operating limit will be required, and therefore, no change to the Core Operating Limits Report will be necessary. In any case, the maximum change in minimum critical power ratio that may be applied to the values in the Core Operating Limits Report for periods of operation above rated power will be less than 0.02.

Overpressure Analysis

NEDC-31897P-1 indicates that there is little change in the challenge to reactor pressure vessel overpressure based on an assessment of the code overpressure protection analysis. This conclusion is based on taking a minimal change to the initial operating pressure. The proposed test program for Plant Hatch is designed to obtain turbine performance test data that is dependent on a range of turbine inlet conditions. It is anticipated that the turbine inlet pressure may be increased by as much as 30 psi during the test program. This represents about a 3% increase in the reactor operating pressure. This range of inlet conditions will allow turbine control and performance to be adequately evaluated. Sensitivity studies performed by the reactor manufacturer have demonstrated that increasing the initial operating reactor operating pressure has a small effect on the results of the code overpressure protection analysis. It should be noted, that for this test, the safety/relief valve setpoints will not be increased. Current reload evaluations at Plant Hatch demonstrate that there is at least a margin of 100 psi to the ASME Code upset limit of 1375 psig. The increase in peak pressure for the bounding upset isolation event is expected to increase less than the vessel dome pressure increase of 30 psi. Therefore, an acceptable margin to the event acceptance limit is available for the proposed test. To provide additional assurance, a code overpressure protection analysis will be performed prior to the test that covers the test conditions. It should be noted that the probability of a stuck open relief valve event does not significantly increase due to this increase in pressure since the probability of transients which result in relief valve lift events does not increase. Pressurization transients could potentially cause more safety relief valves to lift due to the 30 psi additional pressure in the vessel; however, Target Rock two-stage SRVs are highly reliable for closing.

Containment Pressure

NEDC-31897P-1 indicates that uprating to 105% steam flow is expected to cause less than a 1 psi increase in the peak primary containment pressure for the postulated loss of coolant accident. Based on engineering judgement, an additional 0.8% power increase to achieve 105% power would have negligible impact on peak containment pressure. Analyses performed for the Long Term Mark I Program resulted in a peak calculated containment pressure of about 50 psig for Plant Hatch. Considering the small increase in calculated peak containment pressure, substantial margin to the maximum allowable containment pressure of 62 psig still remains for power uprate.

Suppression Pool Temperature

NEDC-31897P-1 indicates that uprating to 105% steam flow is expected to cause less than a 4°F increase in the peak suppression pool temperature for the postulated loss of coolant accident. This is a relatively small temperature increase. An assessment of the available net positive suction head requirements, the limiting parameter with respect to the long term containment temperature increase for the emergency core cooling system pumps, indicates that acceptable margins remain with respect to this parameter. This assessment is based on the Reference 8 analysis which provided the peak suppression pool temperature and pump net positive suction head using both the conservative May-Witt decay heat values and the ANS decay heat correlations. The ANS 5.1 decay heat correlation will be used for power uprate. Reference 9 provides the NRC Safety Evaluation Report for use of the ANS 5.1 decay heat correlation for the analysis of the containment pressure and temperature response. Again, uprating to 105% thermal power is expected to insignificantly increase the calculated peak suppression pool temperature reported in the Final Safety Analysis Reports.

Offsite Doses

NEDC-31897P-1 indicates that uprating to 105% steam flow is expected to cause less than a 2% increase in calculated off-site doses for postulated accidents. Increasing an additional 0.8% power, to 105% thermal power, would insignificantly increase calculated off-site doses. The radiological safety analysis for Plant Hatch demonstrates that there is adequate margin to the values in 10 CFR 100 and 10 CFR 50, Appendix A (General Design Criteria 19) for offsite, main control room, and Technical Support Center. Therefore, substantial margin remains with respect to the radiological exposure limits for postulated accidents (see Reference 10.)

Onsite Doses

NEDC-31897P-1 indicates that uprating to 105% steam flow is expected to cause less than a 5% increase in calculated on-site radiation sources during normal operation. The impact of an additional 0.8% increase in thermal power would have minimal effect. The on-site radiological exposures to personnel are controlled by plant procedures. This change in radiation sources is well within the normal variability of radiation sources experienced at Plant Hatch. The current plant procedures are more than adequate to handle any small increase in radiation sources.

Water Discharge Temperature

NEDC-31897P-1 indicates that uprating to 105% steam flow is expected to cause less than a 2°F increase in water discharge temperature due to an increase of 5% in core power level. Again, an additional 0.8% change in thermal power would have negligible effect on discharge temperature. This magnitude of change represents a small impact on the plant performance. The temperature in the safety related portions of the service water systems will be maintained within the limits established by the plant safety analysis.

Stability

NEDC-31897P-1 indicates that stability is adequately managed in accordance with NRC Bulletin 88-07, Supplement 1. The proposed test program does not require operation at high power and low flow and does not propose a change to the region restricted on the power/flow map. Therefore, stability considerations are not impacted significantly by the proposed test program.

Requested Test Duration

To assure that there is adequate time to obtain the test data, this temporary license amendment request is for a cumulative duration of 30 days for each unit with the reactor core operating at a power level in excess of 2436 MWt but less than or equal to than 2558 MWt. Based on current assessments, it is believed that this period provides the necessary time to obtain the necessary plant performance data and to perform all testing necessary to demonstrate the plant capability for operating safety at the power uprate conditions. In the selection of the desired test period, consideration was given to the number of test points, the data needed, and possible contingencies. The selection of three to four test points provides an adequate number of points and allows the optimum steady state operating points for the permanent uprate program to be identified. To obtain steady state data, it is important to remain at each test point for a reasonable period of time to allow slowly changing parameters, such as heat sink temperatures, to reach a reasonable equilibriums. It is desired to obtain plant operating data at environmental conditions approaching the limiting design conditions. For this reason, Unit 2 testing is targeted for late summer conditions which maximizes the challenge to service water and heating, ventilating, and air conditioning systems. The test program needs to be of sufficient length to assure that data can be obtained at high environmental temperature conditions that can be impacted by changing weather conditions. Sufficient time needs to be provided to allow for contingencies such as the need to reduce power for some unforeseen reason during the testing, or equipment maintenance, and still have the time available to obtain the desired test data.

Proposed Change 2

The proposed change will temporarily revise the high pressure scram Technical Specification 2.2.A.1.a limiting safety system setting and the Technical Specification 3.1.A (Table 3.1-1, Item 4) limiting condition for operation for Unit 1 and Technical Specification 2.2.1 (Table 2.2.1-1, Item 3) limiting safety system setting for Unit 2 from a maximum of 1054 psig to a maximum 1065 psig. The cumulative total of time spent with each unit operating with the revised high pressure scram setpoint is not to exceed 35 days.

Bases for Proposed Change 2

The proposed power uprate testing includes provisions for increasing the steam dome pressure up to 30 psi above the current steady state rated conditions. As described in the Bases for Proposed Change 1, the increase in steam dome pressure is necessary to determine the turbine performance characteristics as a function of turbine inlet pressure and to accommodate the high steamline pressure drop associated with the higher steam flow rate at the increased power level. With the increased steady state operating pressure, it is desirable to increase the pressure scram setpoint to reduce the potential for a spurious scram to occur on high pressure. Operation with the higher steam dome pressure is consistent with the proposed power uprate program.

The proposed increase in the high pressure scram setpoint has been limited to 11 psi. Based on the historical plant performance, it is judged that for the higher pressure scram setpoint, the probability of a scram occurring on high pressure is acceptably low for increases in steady state operating pressure associated with the proposed test program.

In the safety analysis process, no credit is taken for the high pressure scram for the potentially limiting events being initiated at near rated conditions. For the potentially limiting events requiring scram intervention, scram will occur on either a trip scram, (e.g., turbine stop valve position switches, or turbine control valve fast closure), or on high neutron flux. For some non-limiting or low power event sequences, it may be assumed that a high pressure scram occurs. These non-limiting event sequences do not establish any core operating limits at rated conditions.

The high pressure scram setpoint increase was limited to 11 psi to preserve the current hierarchy of pressure setpoints. This means that the high pressure scram setpoint remains below the opening setpoint of the safety/relief valves. This hierarchy of setpoints provides assurance there is a very low probability of opening more than one safety/relief valve without scram intervention.

To assure that there is adequate time to obtain the test data, this temporary Technical Specification is for a cumulative duration of 35 days for each unit with the modified high pressure scram setpoint. This allows for 30 days of power uprate testing and an additional period of time for the test preparation and recovery from the test conditions.

Proposed Change 3

The proposed change will temporarily revise the average power range monitor rod block Technical Specification 3.2.G (Table 3.2-7 Item 3) limiting condition for operation for Unit 1 and Technical Specification 3.3.5 (Table 3.3.5-2 Item 1.a) limiting condition for operation for Unit 2 from a maximum of 0.58W + 50% - 0.58ΔW to a maximum of 0.58W + 53% - 0.58ΔW. The cumulative total of time spent with each unit operating with the revised average power range monitor rod block setpoint is not to exceed 35 days.

Bases for Proposed Change 3

The proposed power uprate testing includes provisions for performing tests up to 105% of the current rated power. The average power range monitor rod block is conservatively set during plant operation to assure that it is unlikely to exceed the Technical Specification setpoint. Based on an assessment of plant performance, it is believed that there is a significant likelihood that an average power range monitor rod block may be encountered due to the normal neutron noise during the plant maneuvering required to reach the elevated power levels.

The proposed increase in the average power range monitor rod block setpoint is 3%. It is judged that for this magnitude of increase in rod block setpoint, that sufficient plant maneuvering capability exists to allow the plant to reach the desired test conditions. The increase in the rod block setpoint allows control rod patterns for the power uprate testing to be established at lower power levels. In this way, control rod manipulations above the current rated power can be minimized, and the core power increase can be accomplished by increasing recirculation flow.

The average power range monitor rod block is provided to block control rod withdrawal prior to a scram on high flux. In the safety analysis process, no credit is taken for the average power range rod block monitor trip.

To assure that there is adequate time to obtain the test data, this temporary Technical Specification is for a cumulative duration of 35 days for each unit with the modified APRM rod block scram setpoint. This allows for 30 days of power uprate testing and an additional period of time for the test preparation and recovery from the test conditions.

Proposed Change 4

The proposed change will temporarily revise the low low set safety/relief valve arming Technical Specification 3.2.N (Table 3.2-14 Item 1) limiting condition for operation and Technical Specification 4.6.H.2 surveillance requirement for Unit 1 and Technical Specification 3.3.3 (Table 3.3.3-2 Item 5.a) limiting condition for operation and Technical Specification 4.4.2.2 surveillance requirement for Unit 2 from a maximum of 1054 psig to a maximum of 1065 psig. The cumulative total of time spent with each unit operating with the revised low low set safety/relief valve arming setpoint is not to exceed 35 days.

Bases for Proposed Change 4

The proposed power uprate testing includes provisions for increasing the steam dome pressure up to 30 psi above the current steady state rated conditions. As described in the Bases for Proposed Change 1, the increase in steam dome pressure is necessary to determine the turbine performance characteristics as a function of turbine inlet pressure and to accommodate the high steamline pressure drop associated with the higher steam flow rate at the increased power level. Also as described in the Bases for Proposed Change 2, with the increased steady state operating pressure, it is desirable to increase the pressure scram setpoint to reduce the potential for a spurious scram to occur on high pressure.

The low low set relief logic is provided to mitigate the postulated containment loads of subsequent safety/relief valve actuations during small or intermediate break loss of coolant accidents by extending the time between actuations. To preserve the hierarchy of pressure setpoints, the high pressure input to the low low set safety/relief valve arming logic has the same setpoint as the high pressure scram. This proposed increase in the reactor pressure input to the low low set safety/relief valve arming logic is necessary to preserve the hierarchy of high pressure setpoints. This approach minimizes the potential for a spurious relief valve opening through the low low set logic without the occurrence of a reactor scram.

To assure that there is adequate time to obtain the test data, this temporary Technical Specification is for a cumulative duration of 35 days for each unit with the modified low low set high pressure setpoint. This allows for 30 days of power uprate testing and an additional period of time for the test preparation and recovery from the test conditions.

Administrative Controls

GPC will administratively control entering Technical Specifications action statements for key safety equipment while operating above rated power. Should any of this equipment become inoperable, the unit will be reduced in power to 2436 MWt or less within 6 hours. Examples of key equipment:

- RHR, RHRSW, HPCI, LLS
- RCIC
- RPS
- Emergency Diesel Generators
- SRVs

GPC will administratively control key surveillance testing while operating above rated power. Should surveillance be required, the unit will be reduced in power to 2436 MWt or less prior to performing the surveillance. Examples of key surveillances include:

- HPCI, RCIC In-service Testing
- Turbine Stop Valve and Control Valve Testing

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request for Temporary Technical Specifications Revision:
Allow Power Uprate Testing

Additional Bases for Change Request

This enclosure provides a basis or the bases for the necessary changes to the operating license and Technical Specifications utilizing the format and content contained in the Power Uprate Safety Evaluation Report issued for the Detroit Edison Fermi 2 plant (Reference 1).

1.0 Reactor Core and Fuel Performance

The effect of power uprate testing has been evaluated for potential impact on various areas related to reactor thermal-hydraulic and neutronic performance. These included temporary setpoint changes, core stability, reactivity control, fuel design, control rod drives, and scram performance. Additionally, the impact of the power uprate test on reactor transients, anticipated transients without scram (ATWS), emergency core cooling system (ECCS) performance and peak cladding temperature has been assessed. The assessments are based on NEDC-31897P-1 (Reference 2), NEDC-31984 (Reference 3), and supplementary plant-specific evaluations.

1.1 Fuel Design and Operation

No new fuel designs are required to implement the proposed power uprate test. The current plan is to test Unit 2 in the late summer or early fall. The unit will have operated about one-third of its 18-month fuel cycle and will have sufficient reactivity to reach 105% power. Unit 1 testing would be delayed until sometime after startup from the Fall 1994 refueling/maintenance outage to assure sufficient reactivity exists.

Fuel operating limits, such as the maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR) will continue to be met for the test. The methods used for calculation of MAPLHGR and OLMCPR limits will not be changed for the test.

NEDC-31897P-1 (Reference 2) indicates that the maximum change in OLMCPR for power uprate conditions is less than 0.02 for "stretch power." To assure that there is an acceptably low probability of exceeding the safety limit MCPR during the proposed test, an analysis of the limiting events that can establish the OLMCPR will be performed prior to the test. This analysis will be used to establish any changes required to the Core

Operating Limits Report. It is anticipated, that by taking credit for the actual test conditions, no changes to the OLMCPR will be required, and therefore, no change to the Core Operating Limits Report will be necessary. In any case, the maximum change in MCPR that may be applied to the values in the Core Operating Limits Report for periods of operation above rated power is expected to be less than 0.02.

It is also anticipated that no change to MAPLHGR limits will be required for the test. Large margins exist relative to ECCS (peak clad temperature) performance and Kw/ft limits are now determined by the thermal-mechanical design of the fuel bundles, which is not being changed for the test.

1.2 Fuel Enrichment and Burnup

Fuel enrichment and design burnup are not being changed for this test.

1.3 Power/Flow Operating Map

The power-to-flow map will not be changed for test purposes, except to allow operation at power levels above 100%. Figure 1 presents the Plant Hatch power versus flow map. Enclosure 1, Proposed Change 3 discusses the temporary change to the APRM rod block equation.

1.4 Stability

NEDC-31897P-1 (Reference 2) indicates that stability is adequately managed in accordance with NRC Bulletin 88-07, Supplement 1. The proposed test program does not require operation at high power and low flow and does not propose a change to the restricted region on the power/flow map.

1.5 Control Rod Drives and Scram Performance

The power uprate test conditions are within the range of values specified in GE generic guidelines. As delineated in Reference 2, the 3% increase in reactor pressure has no detrimental effect on scram speed. A review of the CRD pump curves indicates sufficient head is available, even for a permanent program.

2.0 Reactor Coolant System and Connected Systems

This section presents information on the effect of the power uprate test on the structural and pressure boundary integrity of the piping systems and reactor vessel.

2.1 Nuclear Steam Pressure Relief

The purpose of the nuclear steam pressure relief system is to prevent overpressurization of the NSSS during abnormal operational transients. In BWRs, the main steam line safety/relief valves (SRVs) provide this protection. In Reference 3, GE evaluated the impact of uprated conditions; namely, increased temperatures, pressures, and flow rates on the SRVs. GE concluded that the function and structural integrity of the SRVs would not be compromised by power uprate. The only change to the SRVs which would result from a permanent power uprate would be an increase in the setpoints of the SRVs to accommodate an approximate 30 psi increase in reactor vessel upper head pressure. However, for the test, the SRV setpoints will not be increased.

2.2 Reactor Overpressure Protection

Reference 2 indicates that there is little change in the challenge to reactor pressure vessel overpressure based on an assessment of the code overpressure protection analysis. This conclusion is based on making a minimal change to the initial operating pressure. Current reload evaluations at Plant Hatch demonstrate that there is at least a margin of 100 psi to the ASME Code upset limit of 1375 psig. The increase in peak pressure for the bounding upset isolation event is expected to increase less than the vessel dome pressure increase of 30 psi. Therefore, an acceptable margin to the event acceptance limit is available for the proposed test. To provide additional assurance, a code overpressure protection analysis will be performed prior to the test that covers the test conditions.

2.3 Reactor Vessel and Internals

A comprehensive review for the permanent submittal is underway for reactor vessel and internal components. However, based on generic evaluations, plant-specific* evaluations for other BWR/4 plants, and feasibility studies for Plant Hatch, the reactor vessel and internals are adequate for the proposed test and will be adequate for permanent power uprate.

1) The impact of the power uprate test on reactor internal pressure differences is small and is primarily due to an increase in core average void fraction.

2) Bechtel Power Corporation has reviewed the existing annulus pressurization loads resulting from a postulated recirculation line break and concluded that the existing analysis will bound power uprate conditions.

3) Safety relief valve loads are not being increased for the test, and seismic loading will be unaffected by power uprate.

4) Fatigue and reactor vessel fracture toughness considerations are not a concern for the short duration of the test.

5) Power uprate feasibility studies for Plant Hatch have evaluated the Unit 1 and Unit 2 reactor vessel internals to assess their structural integrity at "stretch power" and concluded the reactor vessel nozzles, and internals would be adequate for uprate power without modifications.

2.4 Reactor Recirculation System

The increase in reactor power for the test will be achieved without increasing the maximum licensed recirculation pump speed or core flow of 105%. At a given core flow, a small increase in flow resistance is expected due to an increase in core average void fraction and a corresponding increase in two-phase flow resistance. Reference 6 (the Plant Hatch Feasibility Study) concluded the recirculation pumps had margin to accommodate a power uprate.

2.5 Reactor Coolant and Balance-of-Plant Piping

The piping systems which will experience increased loading due to uprated power conditions are currently being evaluated for the permanent license. The technique being utilized is the same as that used for the Fermi power uprate. Basically, the modest increases in flow, temperature, and pressure at uprate conditions are input to the ASME code equations to estimate stress increases. The revised stresses will then be compared to ASME code allowable limits for acceptability. This task is scheduled for completion by the end of 1994.

For the test, the impact on piping and pipe supports was qualitatively assessed. This assessment is based on the evaluations done to date for the Plant Hatch permanent license and the experience from similar evaluations done for other BWR/4 plants. The results of this qualitative assessment are that none of the (approximately) 4000 pipe supports per unit are expected to require modification as a result of power uprate.

a) The General Electric Piping Analysis group has been involved in nine power uprate projects to date. All of the piping evaluated by GE has been shown to be acceptable as installed. No supports or welds in GE's scope of services have required modification to satisfy power uprate requirements. (GE has evaluated power uprate to as high as 117%.)

b) No new pipe breaks have been postulated. Pipe whip restraints have been adequate as designed.

c) The effect of power uprate increases in pressure, temperature, and flow have had insignificant effects on piping displacements and fatigue.

d) In general, the piping has been demonstrated to have significant margins to the Code allowables.

e) In addition to power uprate projects, GE has managed many snubber reduction projects. In all of these projects, GE has eliminated at least 50% of the snubbers in the systems analyzed. This indicates the existence of significant margins on the piping systems. Since GPC has not performed snubber reduction at Hatch Units 1 and 2, this option remains, and indicates, that significant margins exist in most of the piping as originally designed.

Based on the qualitative information presented above, GPC believes the piping systems have adequate margin to perform the proposed test safely.

2.6 Main Steam Isolation Valves (MSIVs)

NEDC-31984P (Reference 3) indicates the changes in operating conditions associated with power uprate are small when compared to the existing normal operating conditions, and the MSIVs are designed to accommodate such small changes in operating conditions.

2.7 Reactor Core Isolation Cooling (RCIC) System

The RCIC system is designed to provide rated flow over a vessel pressure range of 150 psig up to a maximum pressure based on the lowest SRV safety setpoint. For the proposed test, no SRV setpoint is being changed. Therefore, the ability of the RCIC system to supply water to the isolated reactor is not changed.

For permanent power uprate submittals, which include small increases in the SRV setpoints, licensees have generally committed to implement GE SIL 377 recommendations. Specifically, this entails adding a small bypass around the steam admission valve of the RCIC turbine to reduce the probability of a turbine overspeed trip during system startup. Section 4.2 of NEDC-31984P (Reference 3) discusses this in detail.

Although not required for the uprate test, the low speed RCIC bypass modification discussed above has been completed on Unit 1, and is scheduled for completion on Unit 2 in 1995.

2.8 Residual Heat Removal (RHR) System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The performance of the RHR system in the suppression pool cooling mode is discussed in Section 3.1. The LPCI mode is discussed in Section 3.2.2.

If the unit is forced into the shutdown cooling mode after operating for a week or more at uprated power, the time to reach cold shutdown will be increased slightly due to higher decay heat. This should not present a problem as the RHR systems can currently bring the units to cold shutdown more quickly than specified by NRC Regulatory Guide 1.139.

2.9 Reactor Water Cleanup (RWCU) System

During the test, the RWCU operating pressure and temperature will increase slightly. References 2 and Hatch-specific feasibility studies have concluded that uprated power operation may affect the cleanup effectiveness slightly, but not its integrity. Also, current specifications for reactor water chemistry will not be changed for the test.

3.0 Engineered Safety Features

The impact of the proposed test on containment system performance, the standby gas treatment system, post-LOCA combustible gas control, the main steam isolation valve leakage control system, the control room atmosphere control system, and the emergency cooling water system has been reviewed. This review was performed to ensure that the ability of these systems to perform their safety function to respond to or mitigate the effects of design basis accidents was not impaired due to power uprate. Additionally, the effects of power uprate on high energy line breaks, fire protection, and station blackout were considered.

3.1 Containment System Performance

Primary containment temperature and pressure response following a postulated LOCA is evaluated when determining the potential for offsite release of radioactive material, in determining ECCS pump net positive suction head (NPSH) requirements, and in determining environmental qualification requirements for safety-related equipment located inside the primary containment.

As discussed in Enclosure 1, proposed change 1, significant margin exists to accommodate an increase to 105% thermal power.

In addition, General Electric has assessed the impact of the 5% power uprate on Plant Hatch both in feasibility studies and recently as part of the detailed evaluations for GPC's permanent license submittal. The containment parameters evaluated include:

a) Peak suppression pool temperature during a DBA-LOCA

b) Peak drywell pressure and temperature during a DBA-LOCA

c) DBA-LOCA hydrodynamic loads

d) SRV discharge loads

The results of this plant-specific evaluation show all parameters will be within acceptable limits and that expected changes in these parameters will be small. Existing primary containment environmental qualification (EQ) profiles are expected to be unchanged.

The expected impact on high energy line break (HELB) analysis outside containment was also reviewed by Bechtel Power Corporation as part of the effort for a permanent license submittal. The impact of power uprate conditions is small, and is expected to be bounded by the existing calculations.

3.2 Emergency Core Cooling Systems (ECCS)

As discussed above, the use of new models is expected to result in the peak suppression pool temperature following a LOCA about the same as the values reported in the FSAR (204°F for Unit 1 and 210°F for Unit 2). Therefore, ECCS NPSH requirements will be satisfied even if a LOCA occurs during the test.

3.2.1 High Pressure Coolant Injection (HPCI) System

The HPCI system design basis is to provide reactor vessel inventory make-up during small and intermediate breaks for loss-of-coolant accidents (LOCA) and reactor vessel isolation events. The HPCI system is designed to provide its rated flow over a reactor pressure range of 150 psig to a maximum pressure based on the lowest SRV safety setpoint. The SRV opening setpoints will not be increased for the power uprate test.

For permanent power uprate submittals, which include small SRV setpoint increases, licensees have generally committed to implement GE SIL 480 recommendations. Section 4.2 of NEDC-31984P (Reference 3), discusses this in detail. Although not required for the uprate test, the SIL 480 recommendations have already been implemented on both units.

3.2.2 RHR System (Low Pressure Coolant Injection, LPCI)

A generic evaluation is provided in Reference 3 for the LPCI mode of the RHR system. There are no changes in the LPCI mode of operation proposed for the test, and none are planned for the permanent submittal.

3.2.3 Low Pressure Core Spray (CS) System

A generic evaluation is provided in Reference 3 for the CS system. No changes in the system are proposed for the test and none are planned for the permanent submittal.

3.3.3 Emergency Core Cooling System Performance Evaluation

The ECCS performance, under all LOCA conditions and their analysis models, must satisfy the acceptance criteria and requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. Enclosure 1, proposed change 1, shows that the impact of operating at 105% thermal power insignificantly changes the calculated results.

3.4 Standby Gas Treatment System (SGTS)

No changes to this system are proposed for the test. The system performance should not be significantly affected by the test, as documented in Reference 2.

Design Basis Accident radiological consequences have been evaluated in References 2, 3, and 10, and discussed in Section 6.0 of this submittal.

3.5 Other ESF Systems

3.5.1 Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

This system was never installed in Unit 1, and has recently been removed from Unit 2 following NRC approval. Reference 10 is GPC's detailed submittal which provides radiological safety analyses results for removing the MSIV-LCS and increasing allowable MSIV leakage. Section 6.0 provides more detail.

3.5.2 Post-LOCA Combustible Gas Control

Both units at Plant Hatch are Mark I containments with purge and inerting systems. The primary containments are inerted during power operation. Unit 2 is equipped with a hydrogen recombiner system.

These manually initiated systems will not be changed for the test, and in all likelihood not be changed for the permanent license. The Reference 7 ECCS/LOCA analysis has resulted in low fuel peak clad temperatures following a LOCA.

Reference 1 also concluded that existing post-LOCA combustible gas control systems will function at uprate conditions.

3.5.3 Main Control Room Environmental Control Systems (MCRECS)

Changes to this system are not being proposed for the test, and in all likelihood will not be proposed for a permanent license. Heat loads in the MCR are not impacted by uprated power. The minor impact on MCR post-accident doses is discussed in Section 6.0.

4.0 Instrumentation and Control

As stated previously, the proposed test program will minimize the number of plant changes required to perform the necessary testing. The three setpoint changes being requested are detailed in Proposed Changes 2 through 4 of Enclosure 1.

Table 1 provides a comparison of the Technical Specifications setpoint changes anticipated for a permanent license versus that proposed for the test. A detailed discussion for each setpoint is provided below. Table 2 provides an estimate of the expected margins to the trips during the test.

A. Average Power Range Monitor (APRM) Flow-Biased Rod Block - Georgia Power is requesting a 3% change in this setpoint for the test; we anticipate a 5% change for the permanent license. This approach is consistent with Reference 2.

   The APRM flow-biased rod block line provides margin to the APRM simulated thermal power monitor (STPM) flow-biased scram. The 5% change which is anticipated for the permanent license will maintain the same analyzed low core flow region (87% core flow) at uprated power as exists today at rated power.

   Enclosure 1, Proposed Change 3 details the bases for the 3% change requested for the test. The request is necessary because a combination of conservative APRM rod block nominal trip setpoints and normal APRM noise alarms restrict operation at rated power below 95% core flow. This change will provide more plant maneuvering capability to reach higher power.

B. APRM Flow-Biased Scram - This setpoint will not be modified for the test. It will be increased approximately 5% for the permanent power uprate license, coincident with the APRM flow-biased rod block line.

Maintaining the current APRM flow-biased setpoints for the test is conservative, but tends to reduce the margin to scram. However, GPC believes the margin to be adequate for temporary operation for the following reasons:

a) The margin between the trip setpoint and operation will be approximately 8% at 100% core flow.

b) The flow-biased scram has a time-averaging circuit. This circuitry simulates the thermal time constant of the fuel and prevents inadvertent trips from APRM noise.

C. APRM Flux Scram - This scram on neutron flux is not being adjusted upward for the test, but will be increased 5% for a permanent license. The trip setpoint is approximately 117% which should allow adequate margin for the test.

D. SRV Setpoints - The SRV opening setpoints are not being modified for the test; they will be increased 30 psi for the permanent license to assure that there is no decrease in operating margin for long term plant operation.

The SRV mechanical opening pressures are setup with a bench-test tolerance of ± 1%. Maintaining the existing nominal setpoints for the test (as compared with raising the setpoints) is conservative relative to the safety analysis, and the 45-55 psi margin is more than adequate to prevent spurious SRV operation. It should be noted that the Plant Hatch Target Rock SRVs have been changed from the original three stage to the newer two stage design. This design change was made to specifically reduce the probability of a stuck open relief valve (SORV) due to the previously identified failure mode (steam cutting). With the new design, Plant Hatch has not experienced an SORV event. Therefore, the possibility of an SORV event during the power uprate test is considered very unlikely.

However, the expected test pressure margin of 45-55 psi is less than the margin of 75-85 psi during normal operation. This could increase the probability of SRV actuation during a pressurization transient, (e.g., turbine trip, load rejection, etc.), should it occur during the test. This could, therefore, increase the probability of a stuck-open relief valve (SORV) event. The impact of this reduced pressure margin during the test was evaluated and found to be very small.

a) A review of plant transients revealed that an average of seven SRVs lifted for turbine trips and MSIV closures from 100% power. If it is conservatively assumed that all eleven SRVs will lift for turbine trips and MSIV closures during the 30 day test, the risk of an SORV during the 30 day test is roughly equivalent to the risk of an SORV during 47 days of full power operation. Thus, the increased risk of an SORV during the 30 day test is comparable to the increased risk of an SORV attributable to a 3% increase in plant availability.

b) The Plant Hatch probabilistic risk assessment model conservatively assumed that all eleven SRVs must reclose following all turbine trips and MSIV closure events. It is concluded that stuck open SRVs were not a significant contribution to overall plant risk.

E. <u>High Pressure Scram</u> - This setpoint will be increased 11 psi (1%) for the test and 30 psi for the permanent license. Operating margin will be approximately 20 psi at 1035 psig operation which is adequate to prevent spurious scrams. This change is discussed in Enclosure 1, Proposed Change 2.

F. <u>ATWS High Pressure Recirculation Pump Trip (RPT)</u> - This setpoint is not being changed for the test. It is anticipated we will increase this setpoint 30 psi for a permanent license, although some BWR/4 submittals for uprate have left this setpoint unchanged. At any rate, the margin to this limit will be high enough to prevent a spurious RPT.

G. <u>Main Steam Line High Flow Isolation</u> - This setpoint will not be changed for the test but may be increased 6% for a permanent license. However, margins to the Group 1 isolation will be adequate, and surveillance testing, which could increase steam line flow (e.g., turbine stop valve, control valve, MSIV closure), will not be performed while the unit is at uprated power.

H. <u>HPCI/RCIC High Flow Isolation</u> - These setpoints may be changed for a permanent license, but are not being changed for the test. This setpoint is intended to isolate the HPCI or RCIC steam supply in the event of a break in this line, but set high enough to allow system operation between 150 psig and system pressures during vessel isolation. Since SRV setpoints are not being changed, there is no impact on HPCI or RCIC operation. Surveillance testing of HPCI and RCIC will not be performed above 2436 MWt.

I. Low-Low Set (LLS) Arming Pressure - This setpoint is being increased 11 psi for the test and will be increased 30 psi for the permanent license. It is one of two confirmatory signals necessary to arm LLS and is discussed in Enclosure 1, Proposed Change 4.

TABLE 1

COMPARISON OF TECHNICAL SPECIFICATIONS SETPOINT CHANGES (PERMANENT LICENSE VS. TEST)

Technical Specifications Setpoints	*Changed for Permanent License?*	*Changed for Test?*	*Comments*
APRM Rod Block	Yes - 5%	Yes - 3%	Rod block line is conservatively set. Relaxation will allow rod pattern at (100%P, 95%F)
APRM Flow Biased-Flux Scram	Yes - 5%	No	
APRM Flux Scram	Yes - 5%	No	
SRV Setpoints	Yes - 30 psi	No	
High Pressure Scram	Yes - 30 psi	Yes - 11 psi	No specific credit taken for this scram at high power. Hatch has large margin to code limits
ATWS High Pressure RPT	Yes - 30 psi	No	
Main Steam Line Hi-Flow Isolation	Yes - 6%	No	
HPCI/RCIC Isolation	Being Evaluated	No	
Low-Low Set Arming Pressure	Yes - 30 psi	Yes - 11 psi	One of two parameters to arm LLS. Moved up with high pressure scram setpoint

TABLE 2

COMPARISON OF TECHNICAL SPECIFICATIONS SETPOINT CHANGES (1)

	Expected Test Condition	*Unit 1 Trip Setpoint*	*Unit 2 Trip Setpoint*	*Estimated Margin*
APRM Flow-Biased Flux Scram (2)	105%	113.5%	113.5%	8%
APRM Flux Scram (3)	105%	117%	117%	12%
Nominal SRV Setpoints	1035 psig	1080 psig	1090 psig	45 - 55 psi
Hi Pressure Scram (4)	1035 psig	1053 psig	1053 psig	18 psi
ATWS High Pressure RPT	1035 psig	1086 psig	1086 psig	51 psi
Main Steam Line High Flow Isolation	106%	≤ 138%	≤ 138%	< 32%
HPCI/RCIC High Flow Isolation	Start-up Transient	303%/307%	303%/307%	N/A

(1) APRM rod block and LLS arming pressure not included in this table since they do not cause trips of the reactor or recirculation pumps. The columns for trip setpoints are generally lower than the Technical Specifications Allowable Values.

(2) Time-averaging circuit should limit impact of APRM noise on this trip. Estimated margin is at 100% core flow.

(3) Estimated margin does not include APRM noise.

(4) Includes 11 psi requested setpoint increases.

5.0 Auxiliary Systems

5.1 Spent Fuel Pool Cooling

This system will not be affected by the test.

5.2 Water Systems

The impact of power uprate on the various plant water systems was evaluated during feasibility studies and will be evaluated in more detail for the permanent submittal. As discussed earlier, one reason for the test is to collect plant operating data for these evaluations.

It is expected that the impact of the test on most plant water systems will be small. This conclusion is based on the following:

1) Feasibility studies for Plant Hatch.

2) Estimated NSSS heat load increases at power uprate conditions.

3) Detailed power uprate impact assessments performed for another 218 inch BWR/4 plant with a Mark I containment.

Non safety-related cooling systems, such as the circulating water/cooling tower system, will see increased heat loads at uprated test conditions, and key parameters (e.g., condenser vacuum) will be monitored during the test.

5.3 Standby Liquid Control System (SLCS)

References 2 and 3 provide generic assessments for SLCS. Although GE has not completed the Hatch-specific evaluations for the permanent license, it is anticipated that no increase in boron concentration will be needed. Reference 1 also indicates the increase in boron concentration was due to an increase in fuel enrichment. The power uprate test will not impact fuel cycle energy requirements which were established in accordance with the energy utilization plan for the entire cycle.

5.4 Power Dependent Heating, Ventilation, and Air Conditioning (HVAC)

The proposed test (and permanent uprate operation) will increase primary system water temperatures by approximately 4°F. Feasibility studies, and detailed studies on other BWR plants, indicate HVAC systems will continue to function adequately. However, data will be taken during the test on systems to determine the actual impact on system performance.

5.5 Fire Protection

The fire suppression and detection systems will not be changed for the test or for permanent power uprate implementation. There are no physical plant configuration or combustible load changes resulting from the uprate. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and will most likely be shown adequate for permanent power uprate operation. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not significantly affected by the planned power uprate test.

6.0 Radwaste Systems and Radiation Sources

Uprating to 105% thermal power is expected to cause an insignificant increase in calculated off-site doses for postulated accidents as discussed in Enclosure 1, proposed change 1.

Uprating to 105% steam flow is expected to cause minimal increase in calculated on-site radiation sources during normal operation as discussed in Enclosure 1, proposed change 1.

7.0 Reactor Safety Performance Evaluations

7.1 Reactor Transients

See Section 1.1

7.2 Design Basis Accidents

See Section 6.0

7.3 Anticipated Transients Without Scram (ATWS)

GE report NEDC-31984P, Supplement 1 (Reference 3) contains generic ATWS evaluations for BWR/4 power uprate conditions. The following initial conditions were increased to bound expected increases for power uprate:

- Reactor Power (5% increase)
- Dome Pressure (40 psi increase)
- SRV opening setpoints (80 psi increase)
- ATWS high pressure setpoint (20 psi increase)

The Reference 5 study showed significant margin to all the pertinent ATWS acceptance criteria. The initial conditions bound those expected during the proposed uprate test.

7.4 Station Blackout (SBO)

Plant response and coping capability for an SBO event will be impacted slightly by operation at uprated power due to an increase in decay heat. However, it is unlikely that a more detailed evaluation for the permanent license will cause changes to the coping period or to the systems and equipment used to respond to an SBO. This qualitative conclusion is based on the following:

a) Plant Hatch was able to take credit for its "swing" emergency diesel generator as the emergency AC power source. As such, the SBO coping period was only one hour.

b) Other BWR/4 plants with similar containment, condensate storage tank, and battery capacities have complied with the SBO rule using a 4-hour coping period.

8.0 Additional Aspects of Power Uprate

8.1 High Emergency Line Break (HELB)

Bechtel Power Corporation has reviewed HELB loads outside containment and concluded that the mass and energy release rates in the existing analysis bound the power uprate test conditions. Pipe whip, jet impingement, and moderate energy line cracks are currently under review for the permanent submittal. It is expected that no physical modifications will be required for permanent uprate operation. This qualitative assessment is based on the conservative assumptions and analytical techniques used in the existing evaluations, and the results of similar studies for other BWR/4 plants.

8.2 Equipment Qualification (EQ)

Although the EQ evaluations have not yet been completed for the permanent submittal, there is a high degree of confidence the equipment is acceptable for the proposed test.

a) Slight increases in normal process temperatures and radiation levels during this short duration test should not affect component qualified life significantly.

b) The accident pressure, temperature, and radiation profiles in primary and secondary containments will not change significantly for power uprate.

c) Seismic and dynamic qualification of mechanical and electrical equipment should not be affected significantly by power uprate operation.

References

1. "Safety Evaluation by the office of NRR Related to Amendment 87 to Facility Operating License No. NPF-43, Detroit Edison Company Fermi 2," dated September 9, 1992.

2. NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate", June 1991.

3. NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," (Volume I, II, Supplements 1,2), July 1991.

4. Safety Evaluation of the Edwin I. Hatch Nuclear Plant Unit 1 - Docket No. 50-321, Issue Date: May 11, 1973.

5. "Final Environmental Statement for the Edwin I. Hatch Nuclear Plant Units 1 and 2," October 1972.

6. NUREG-0417, "Final Environmental Statement related to operation of Edwin I. Hatch Nuclear Plant Unit No. 2," March 1978.

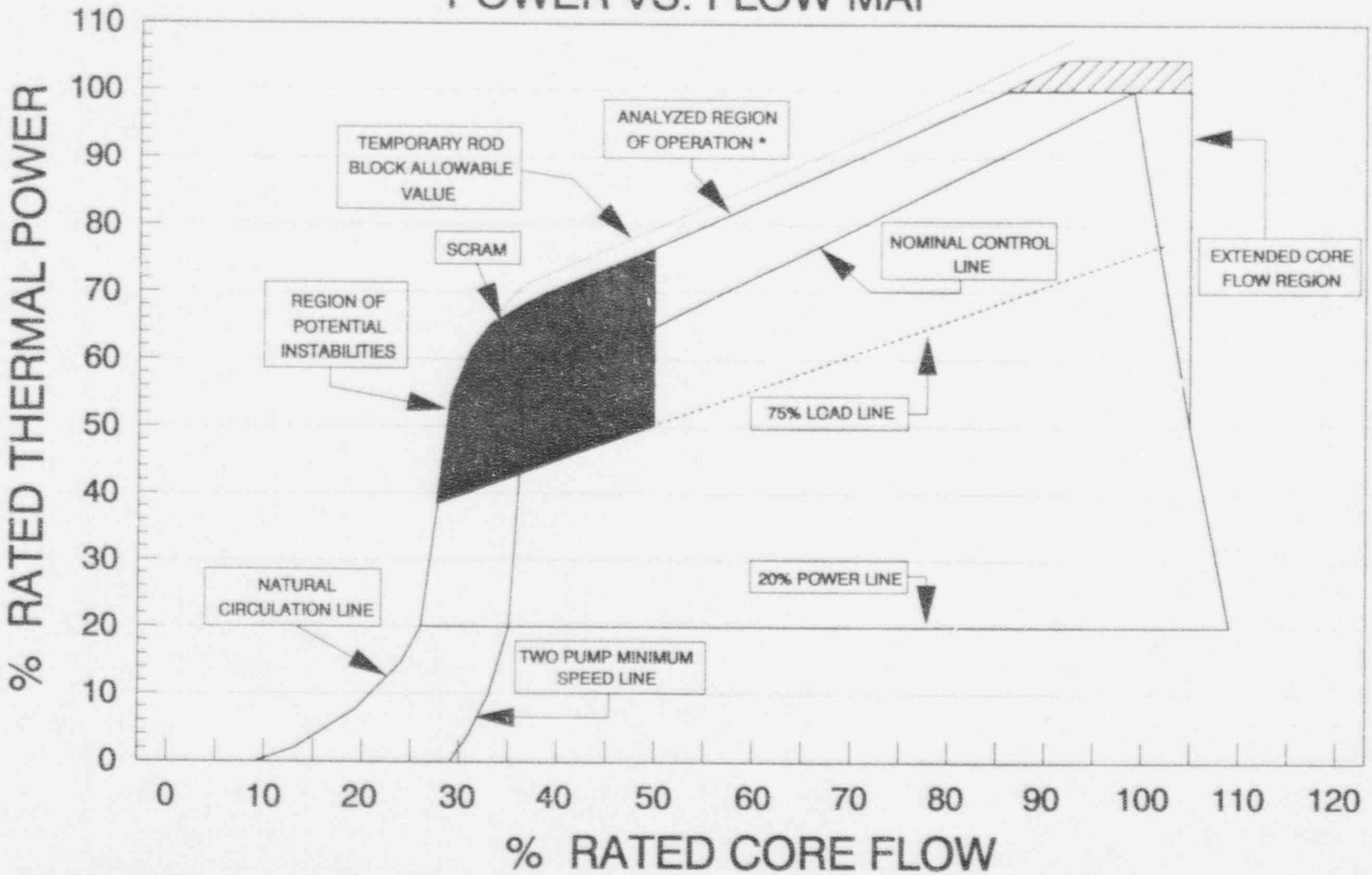
7. NEDC-31376P, Edwin I. Hatch Nuclear Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," December 1986.

8. EAS-19-0388, "Elimination of the High Suppression Pool Temperature Limit for Plant Hatch Units 1 and 2," March 1988.

9. Letter, William T. Russel to Patrick W. Marriott, "Staff Position Concerning GE BWR Power Uprate Program (TAC No. 79384)," September 30, 1991.

10. Letter HL-4468, J. T. Beckham, Jr. to NRC, "Request to Revise Technical Specifications: Increase in Allowable MSIV Leakage Rate and Deletion of the MSIV Leakage Control System," dated January 6, 1994.

FIGURE 1
PLANT E. I. HATCH
POWER VS. FLOW MAP



* TEMPORARY ROD BLOCK NOMINAL TRIP SETPOINT

E2-21

Enclosure 3

Edwin I. Hatch Nuclear Plant
Request for Temporary Technical Specifications Revision:
Allow Power Uprate Testing

10 CFR 50.92 Evaluation

The NRC has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists in a proposed license amendment. A proposed license amendment does not involve a significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in the margin of safety.

Georgia Power Company has reviewed the proposed license amendment and Technical Specification changes and has determined that its adoption would not involve a significant hazards consideration. The basis for this determination is given below.

Evaluation of the Proposed Changes

The proposed changes do not involve a significant hazards consideration for the following reasons.

1. The proposed license amendment and Technical Specification changes do not involve a significant increase in the probability or consequence of any accident previously evaluated.

   The temporary license amendment and Technical Specification changes for the proposed plant power uprate testing involve only relatively small changes in the plant operating conditions and are for a relatively short duration. The primary changes in plant operating conditions are associated with the increase in plant power, which result in an increase in steam and feedwater flow. Also, there is an increase in reactor operating pressure, which is necessary to obtain additional turbine performance data. Plant Hatch operation since the original issuance of the operating license for each unit has demonstrated that the plant has margin for increased power operation. The purpose of the proposed testing is to obtain sufficient operational data to minimize uncertainties related to plant performance capabilities and provide additional information in support of future permanent power uprate activities.

There are no specific plant modifications required for the proposed test other than some minor changes in instrument setpoints. As a result, potential event initiators remain unchanged. Therefore, the probability of an accident previously evaluated in the plant safety analysis is not significantly increased.

An assessment of the potential impact of the power uprate testing on the bounding licensing criteria contained in the GE Nuclear Energy Licensing Topical Reports NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," and NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," has been performed. This assessment demonstrates that there is only a relatively small increase in the consequences of previously evaluated accidents and all applicable safety analysis criteria and limits are satisfied for operation at the uprated power level. Therefore, the consequences of an accident previously evaluated in the plant safety analysis is not significantly increased.

2. The proposed license amendment and Technical Specification changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The temporary license amendment and Technical Specification changes for the proposed plant power uprate testing do not involve any changes to plant systems, structures, or components. The design function of all structures systems and components remains the same. The primary change associated with the proposed power uprate test is some relatively small changes in the normal operating conditions. The only plant modifications required for the proposed test are some minor changes in instrument setpoints. However, the hierarchy of instrument setpoints is preserved. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed license amendment and Technical Specification changes do not involve a significant reduction in the margin of safety.

An evaluation of the impact of the proposed power uprate testing on the bounding licensing criteria identified in NEDC-31897P-1 was performed. The evaluation covered the potentially limiting events in the plant safety analysis. If necessary, the core operating limits are to be modified to assure that the appropriate event acceptance limits will not be exceeded for any of the potentially limiting events initiated at the increased power level. Because the applicable safety analysis criteria and limits are satisfied for the proposed test conditions, the margin of safety associated with the safety limits and other limits identified in the bases for the Technical Specifications will be maintained.

Enclosure 4

Edwin I. Hatch Nuclear Plant
Request for Temporary Technical Specification Revision:
Allow Power Uprate Testing

Environmental Impact Evaluation

Background

Plant Edwin I. Hatch Units 1 and 2, NRC Operating License Nos. DPR-57 and NPF-5 are currently licensed to operate at a core thermal power level of 2436 MWt. A permanent amendment to the operating license for each unit is planned to allow operation at power levels up to 105% of the current rated power level. In support of this permanent license amendment, testing of the plant at up to 105% of the current maximum steady state power level (2558 MWt) is proposed. The cumulative amount of time spent for each unit operating at a power level in excess of 2436 MWt but less than 2558 MWt will not exceed 30 days.

Section 5.5.3 of the Edwin I. Hatch Environmental Technical Specifications (ETS), Appendix B to Facility Operating Licenses DPR-57 and NPF-5, states that "the licensee may make changes to station design and operation and conduct tests or experiments without prior NRC approval, unless the proposed change, test, or experiment involves either a change in the objectives of the ETS or an unreviewed environmental question of substantive impact. A proposed change, test, or experiment is deemed to involve an unreviewed environmental question if it concerns:

1. A matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental statement, as modified by staff's testimony at the hearing, supplements thereto, environmental impact appraisals, or in initial or final adjudicatory decisions.

2. A significant change in effluents or power level.

3. A matter not previously reviewed and evaluated in the documents specified above which may have a significant environmental impact".

Section 5.5.3 requires the licensee to prepare a written evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question of substantive impact, or does not constitute a change in the objectives of the ETS. In accordance with the above requirements, an environmental evaluation assessing the environmental impact of the proposed power level uprate (2436 MWt to 2558 MWt) test has been performed. This evaluation documents that the proposed power uprate test is not significant, relative to adverse environmental impact, and does not constitute an unreviewed environmental question.

References

1. Edwin I. Hatch Nuclear Plant - Environmental Report.

2. Edwin I. Hatch Nuclear Plant Unit 1 and Unit 2 - Final Environmental Statement.

3. Edwin I. Hatch Nuclear Plant - Environmental Technical Specifications.

Bases

The Final Environmental Statement (FES) evaluates the nonradiological impact of the two Plant Hatch units at a maximum design reactor power of 2537 MWt per unit. In support of the requirements of the Plant Hatch ETS, the parameters evaluated in the Environmental Report and the subsequent Final Environmental Statement at maximum design reactor power of 2537 MWt were re-evaluated at the 2558 MWt power level proposed for the power uprate test to determine if the proposed change in power level test is significant relative to adverse environmental impact. The engineering evaluation of cooling tower performance parameters was performed by Southern Company Services (SCS). The following environmental evaluation utilizes the information provided in the SCS evaluation and specifically considers effects on the following parameters.

River Water Intake System

Withdrawal rate
Intake canal velocity

Circulating Water System

Changes in rate of cooling tower blowdown
Changes in temperature of cooling tower blowdown
Changes in makeup to the cooling towers
Changes in the amount of cooling tower drift
Changes in cooling tower chemistry
Changes in consumptive water use

Groundwater Withdrawal System

Changes in groundwater withdrawal to supply water treatment plant
Changes in groundwater withdrawal to supply fire protection system

Radwaste Dilution System

Changes in liquid radwaste which would impact required dilution flows

River Water Discharge System

Changes in discharge flow rate or velocity
Changes in discharge temperature or thermal plume
Changes in discharge chemical composition

Based on information contained in the SCS engineering evaluation on main condenser and cooling tower performance parameters, and review of information contained in the Environmental Report and Final Environmental Statement relative to environmental impacts associated with the above systems, the following information is provided:

River Water Intake and Circulating Water System

The Circulating Water System design flowrate is the primary basis for determining makeup water for the Plant Hatch cooling towers. Other factors affecting tower makeup include tower performance and meteorological conditions. Based on review of engineering information relative to cooling tower performance parameters associated with the proposed test, the design flow rate of the cooling towers does not change. Makeup requirements may increase slightly due to increased heat load on the towers and the associated increase in evaporation. This increase in makeup due to consumptive water use (evaporation) is not significant and is enveloped by the river water withdrawal rates discussed in the FES and the rates approved under the current Georgia Surface Water Withdrawal Permit for Plant Hatch. Intake canal velocity is not significantly affected.

Changes in cooling tower blowdown rate and cooling tower chemistry as a result of the uprate are not significant. Any changes in blowdown rate and cooling tower cycles of concentration resulting from the test are enveloped by the existing design criteria discussed in the FES.

The effect of cooling tower drift increase is not significant. Cooling tower drift will not exceed the 0.2 percent criteria guaranteed by the manufacturer as discussed in the FES.

There will be a slight increase in cooling tower blowdown temperature (< 1 degree F) associated with the power uprate test. This slight increase will also produce a slight increase in river discharge temperature. This increase has been reviewed relative to the conclusions of the FES and thermal studies required to support licensing of the plant. Based on this review, the slight increase in temperature at the river discharge is not significant. The slight temperature increase does not significantly impact the size of the thermal mixing zone for the Plant Hatch thermal effluent and does not alter the conclusions of the FES relative to thermal impacts. The conclusions of additional thermal studies performed to document the thermal mixing zone impact are also not impacted.

No significant change in discharge flow rate, velocity, or chemical composition will occur due to the proposed power uprate test. The proposed test does not impact the discharge characteristics on which the NPDES Permit is based. No notification, changes, or other action relative to the NPDES Permit are required.

Other Systems

The evaluation also considered the flow rate required by the liquid radwaste system due to the proposed power level increase. No significant change in liquid radwaste quantities or activity levels which would increase the required radwaste dilution flow are expected.

Conclusions

Based on the above evaluation, the plant operating parameters impacted by the proposed power uprate test remain within the bounding conditions on which the FES was based. The FES concluded that no significant environmental impact would result from operation of Plant Hatch. This conclusion remains valid for the proposed power uprate test. Per Section 5.5.3 of the Plant Hatch ETS, a change in power level is an unreviewed environmental question if the change is determined to be significant relative to environmental impact. Based on the above evaluation, it can be concluded that no significant environmental impact will result from the proposed test to increase power level from 2436 MWt to 2558 MWt. As such, the proposed test does not constitute an unreviewed environmental question per Section 5.5.3 of the Plant Hatch ETS.

This evaluation has been prepared in accordance with the provisions of Section 5.5.3 of the Plant Hatch ETS and will be provided in summary form as part of the Annual Environmental Surveillance Report (per ETS Section 5.6.1).

Enclosure 5

Edwin I. Hatch Nuclear Plant
Request for Temporary Technical Specification Revision:
Allow Power Uprate Testing

Page Change Instructions

The proposed changes to the Plant Hatch Units 1 and 2 Operating License and Technical Specifications will be incorporated as follows:

Unit 1 - Operating License

Page	Instruction
- 3 -	Replace

Unit 1 - Technical Specifications

Page	Instruction
1.1-10	Replace
1.2-1	Replace
3.1-3	Replace
3.2-16	Replace
3.2-23d	Replace
3.6-9a	Replace

Unit 2 - Operating License

Page	Instruction
- 4 -	Replace

Unit 2 - Technical Specifications

Page	Instruction
2-4	Replace
3/4 3-29	Replace
3/4 3-40	Replace
3.4 4-4a	Replace