UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. NPF-81

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated February 28, 1992, as supplemented June 26 and August 28, 1992, and February 12, 18, 23, and 25, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-81 are hereby amended to read as follows:

   (1) Maximum Power Level

   GPC is authorized to operate the facility at reactor core power levels not in excess of 3565 magewatts thermal (100 percent power) in accordance with the conditions specified herein.

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 39 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachment:
1. Technical Specification Changes
2. License changes

Date of Issuance: March 22, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

	Remove Pages	Insert Pages
License No. NPF-68	3	3
License No. NPF-81	3	3
Appendix "A"	1-5	1-5
	2-4	2-4
	2-9	2-9
	B 3/4 6-1	B 3/4 6-1
	B 3/4 6-2	B 3/4 6-2
	B 3/4 7-1	B 3/4 7-1

(3) GPC, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) GPC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) GPC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) GPC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

GPC is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 60 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

10 CFR Part 50, to possess the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(3) GPC, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) GPC, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) GPC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) GPC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Vogtle Electric Generating Plant, Units 1 and 2.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

GPC is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in appendix A, as revised through Amendment No. 39 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.25 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.26 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3565 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be the exclusion boundary line as shown in Figure 5.1-1.

SLAVE RELAY TEST

1.32 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	≤109% of RTP#	≤111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	≤ 25% of RTP#	≤27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	≤5% of RTP# with a time constant ≥2 seconds	≤6.3% of RTP# with a time constant ≥2 seconds
4. DELETED					
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	≤25% of RTP#	≤31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	10.7	8.8	1.96 + 1.17	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.3	1.54	1.96	See Note 3	See Note 4

#RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T′	$\leq$	588.4°F Nominal T_{avg} operating temperature;
K_3	=	0.00115/psig;
P	=	Pressurizer pressure, psig;
P′	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform variable, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -32.0% and + 10.0%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -32.0%, the ΔT Trip Setpoint shall be automatically reduced by 3.25% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 10.0%, the ΔT Trip Setpoint shall be automatically reduced by 2.7% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.5% of ΔT span.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3 psig, and (2) the containment peak pressure does not exceed the design pressure of 52 psig during loss of coolant accident conditions.

The maximum peak pressure expected to be obtained from a loss of coolant accident is 36.5 psig assuming an initial containment pressure of 3.0 psig. The initial positive containment pressure will limit the total pressure to less than P_a, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a loss of coolant accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure in the event of a loss of coolant accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon strands for Unit 1, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment and the Type A leakage test for both units are sufficient to demonstrate this capability. (The tendon strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.) Lift-off testing on Unit 2 will be accompanied by detensioning of one tendon on Unit 1. This tendon will alternate between the hoop and inverted -U tendons. With regard to D-cracking, the acceptance criteria for the visual inspection of the containment concrete is that the area comprising D-cracking should not exceed 25 sq. ft.

The conditions referenced by Action statement 3.6.1.6.b do not define abnormal containment degradation. These conditions are indications of potential abnormal degradation and their existence requires an appropriate engineering evaluation and a Special Report in accordance with Specification 6.9.2.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken, or proposed.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 24-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4. Sealed closed isolation valves are isolation valves under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, the use of blind flanges, or removal of power to the valve operator.

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1304 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 18,607,220 lbs/h which is 117% of the total secondary steam flow of 15.92×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following basis:

For four loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NPF-81

GEORGIA POWER COMPANY, ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated February 28, 1992, as supplemented June 26 and August 28, 1992, and February 12, 18, 23, and 25, 1993, Georgia Power Company, et al. (the licensee), proposed license amendments to change the operating licenses and Technical Specifications (TS) for Vogtle Electric Generating Plant (Vogtle), Units 1 and 2. The proposed change would revise the value used in TS 1.27 for the definition of rated thermal power (RTP) of the core from 3411 megawatts thermal (MWt) to 3565 MWt. The core power of 3565 MWt corresponds to a Nuclear Steam Supply System (NSSS) power of 3579 MWt.

The change would represent an increase of about 4.5 percent over the presently licensed core power rating of 3411 MWt. In support of this 154 MWt uprating, the Vogtle units were reevaluated for operation at an Engineering Safety Features Design Rating of 3565 MWt core power and a 3579 MWt NSSS power. The licensee's rerating program also includes consideration of a previously approved decrease in thermal design flow from 95,700 gpm to 93,600 gpm, an increase in the steam generator tube plugging limit to 10 percent, and plant operation with a reduced hot leg temperature (T_{hot}).

Since operation at the revised power level would result in a slight increase in the steam flow at RTP, the licensee also proposed that the second sentence of the second paragraph of TS Bases 3/4.7.1.1 "Safety Valves" be changed to state: "The total relieving capacity for all valves on all of the steam lines is 18,607,220 lbs/h which is 117 percent of the total secondary steam flow of 15.92×10^6 lbs/h at 100% RATED THERMAL POWER." The same relieving capacity is currently stated to be "123% of the total secondary steam flow of 15,135,453 lbs/h" at 100% RTP.

Operation at the proposed RTP level would also require a change to the value of the statistical summation of errors assumed in the setpoint calculation for the overtemperature delta temperature (OTDT) reactor trip function, and a change to the value of the power distribution reset function for OTDT. Specifically, the value of "Z" for the OTDT function given in TS Table 2.2-1 "Reactor Trip System Instrumentation Trip Setpoints" reflects the statistical summation of errors and would change from 7.04% of span to 8.8% of span.

Note 1 to TS Table 2.2-1 currently states, in part, that "For each percent that the magnitude of $q_t - q_b$ exceeds + 11%, the Delta T Trip Setpoint shall be automatically reduced by 1.97% of its value at RATED THERMAL POWER." The proposed amendments would change these Note 1 values of 11.0% and 1.97% to 10.0% and 2.7%, respectively. Note 2 of TS Table 2.2-1, which is referenced for the allowable value of the trip setpoint for the OTDT function, states that "The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.1% of Delta T span." This percentage of Delta T span would be reduced from 3.1% to 1.5%.

TS Bases 3/4.6.1.4 "Primary Containment Internal Pressure," 3/4.6.1.5 "Primary Containment Air Temperature," and 3/4.6.1.6 "Primary Containment Structural Integrity," would be changed to indicate that the maximum peak containment pressure was calculated to occur as a result of a loss-of-coolant accident (LOCA) instead of a main steam line break (MSLB) accident. The change would indicate that the new analyses assumed an initial containment pressure of 3 psig instead of 0.3 psig and resulted in a peak calculated pressure of 36.5 psig instead of 41.9 psig.

On August 28, 1992, the licensee proposed a corresponding change to Section C.(1) of the Vogtle Operating Licenses that would increase the maximum power level from 3411 MWt to 3565 MWt.

On June 26, 1992, and February 12, 18, 23, and 25, 1993, the licensee provided additional information in support of the above proposed changes. The letters in February 1993 did not change the NRC staff's proposed no significant hazards consideration determination as published in the Federal Register (57 FR 47132 dated October 14, 1992).

2.0 EVALUATION

2.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) AND SUPPORTING SAFETY SYSTEMS

The scope of the licensee's review to support the proposed power uprating encompassed all aspects of the Vogtle NSSS design and operation affected by the increase. NSSS designs were reviewed to verify compliance at the increased power rating with licensing criteria and standards currently specified in the Vogtle operating licenses. In addition, the licensee conducted a review to identify any potential unreviewed safety questions that might occur as a result of the increased power rating in accordance with 10 CFR 50.59. The structural design of NSSS equipment was reviewed to assure that compliance would be maintained at the increased power rating with industrial codes and standards that applied when the equipment was originally built. In addition, the review encompassed the verification that NSSS components and systems will continue to meet functional requirements specified in the Final Safety Analysis Report (FSAR) at the increased power rating. Currently approved analytical techniques were used for analyses at the increased power rating.

The licensee also reviewed the definition of NSSS/Balance of Plant (BOP) safety related interfaces for any impact at the increased power rating. Based on the scope of the review as outlined above, the licensee finds that each Vogtle unit is capable, in its present design configuration, of operating at the proposed core power rating of 3565 MWt and an NSSS power rating of 3579 MWt without violating any of the design criteria or safety limits specified in the Vogtle FSAR and as currently required in Facility Operating Licenses NPF-68 and NPF-81 for Vogtle Units 1 and 2.

In addition to evaluating the ability of the plants to perform at the new power level under steady state conditions, the licensee reevaluated all the design basis transients and accidents which the NRC staff utilizes to determine that adequate safety margins are maintained. These analyses were performed by Westinghouse using computer codes which have been previously reviewed and approved by the NRC staff. Those events which might challenge the departure from nucleate boiling ratio (DNBR) limits were evaluated using the approved Westinghouse Revised Thermal Design Procedure (RTDP). Steady state instrument errors were considered in establishing the initial conditions, including the addition of 2 percent to the initial power to account for calorimetric error.

By License Amendments Number 51 for Vogtle Unit 1 and Number 30 for Vogtle Unit 2, the NRC staff authorized operation with a reduced minimum design flow for the reactor coolant system (RCS). This reduced flow is associated with an increase in the steam generator tube plugging limit to 10 percent. This lower flow was assumed in the licensee's analyses in support of the power uprate request. We conclude that the lower flow is acceptable for the uprated power because it is used in the technical justifications for the power uprating and, as discussed herein, does not cause existing safety criteria to be exceeded.

The licensee's analyses supporting this power uprating also assumes a reduction in vessel outlet temperature (T_{hot}) from 618.2 °F to 603.2 °F during normal full power operation. The analyses also assume a corresponding reduction in vessel inlet temperature (T_{cold}) from 558.8 °F to 538.3 °F. These revised values are considered in this staff evaluation.

Core Design

On September 19, 1991, the NRC issued License Amendments Numbers 43 and 44 for Vogtle Unit 1 and Numbers 23 and 24 for Vogtle Unit 2, authorizing the use of Westinghouse VANTAGE-5 reload fuel for Vogtle cores. The supporting analyses for these previous amendments accounted for incremental replacement of the resident Westinghouse Low Parasitic fuel, considering the varied mixtures of the fuel types during the transitional period. The descriptions, analyses, and technical specifications for these previous amendments were based upon operation at a core power of 3565 MWt with reduced RCS flow and, except technical specifications changed by this uprating amendment, remain applicable and acceptable for the uprated power. On October 4, 1991, the NRC issued License Amendments Number 47 for Vogtle Unit 1 and Number 26 for Vogtle Unit 2 which authorized limited use of ZIRLO™ fuel cladding in Vogtle cores. The amounts authorized by those amendments (two fuel assemblies, each containing no more than twelve fuel rods clad with ZIRLO™) are sufficiently small to be insignificant to safety and the NRC staff's previous conclusions are not changed by the proposed small increase in power level. Moreover, the NRC staff finds that use of ZIRLO™ cladding in the amounts previously authorized continues to be acceptable for the uprated core power of 3565 MWt.

Overpressure Protection

Pressurizer safety valves are required to be designed with sufficient capacity to prevent the pressurizer pressure from exceeding 110 percent of design pressure following the worst RCS pressure transient. For purposes of analytical justification, this event is specified to be a 100 percent load rejection resulting from a turbine trip with concurrent loss of main feedwater. No credit is taken for operation of RCS relief valves, steam line relief valves, steam dump system, pressurizer level control system, pressurizer spray, or direct reactor trip on turbine trip. Reactor scram is initiated by the second safety-grade signal from the reactor protection system.

By letter of February 12, 1993, the licensee provided the results of overpressure analyses similar to those referenced in the Vogtle FSAR, but performed specifically for Vogtle operating at 3650.7 MWt (102% of the uprated Engineered Safety Features Design Rating of 3579 MWt). The results of these analyses demonstrate that 81 percent of the total safety valve capacity would be needed to meet the above 110 percent criterion if any of three safety-grade trips (overtemperature delta-T, high pressurizer level, or low-low steam generator water level) from the reactor protection system are assumed to cause the reactor trip instead of the initial safety-grade trip (high pressurizer pressure). The analyses also indicate that steam generator safety valve flow would not exceed nominal plant steam flow at any time during the event.

Based on the above analyses, we conclude that, for operation at powers up to the proposed uprated power, the pressurizer safety valve capacity at Vogtle is adequate to meet the requirements to which Vogtle was originally licensed.

In a letter of February 12, 1993, the licensee indicated that it had reviewed the Vogtle low temperature overpressure (LTOP) system, and had concluded that the LTOP design and operation is not affected by the power uprating or by the reduced RCS operating temperature. This is because arming of the LTOP system is based on RCS conditions (300 °F) which is independent of previous operation. Supporting analyses and LTOP administrative requirements are also insensitive to operating history. The NRC staff agrees with the licensee's conclusion.

Cooldowns with the Auxiliary Feedwater and Residual Heat Removal Systems

The NRC staff's review and approval of the Vogtle auxiliary feedwater (AFW) system design is given in NUREG-1137, Safety Evaluation Report (SER) Section 10.4.9. The transients that identify worst single-failure assumptions and minimum flow requirements for the Vogtle AFW system design are station blackout, loss of normal feedwater, and feedwater line break events. These analyses for Vogtle, which assumed the uprated power conditions, were previously reviewed and approved in support of the licensee's program to relocate the instrumentation tap for narrow-range level on the steam generators (License Amendment Number 34 for Unit 1 and Number 14 for Unit 2).

The licensee stated that, as a result of these analyses, identification of limiting scenarios for single failure and required flow were not changed. The NRC staff concludes that, since the Vogtle AFW flow capacity exceeds cooling requirements for the uprated power, cooldown time to reach conditions for initiation of the residual heat removal (RHR) system would not be significantly affected.

The licensee indicated that the original 16-hour cooldown time from RHR initiation at 350 °F to 140 °F using both RHR trains would be increased to less than 20 hours for the uprated power; a cooldown using one RHR train would take 36 hours for the uprated power. The licensee also stated that initial cooling water temperatures of interfacing systems, flow rates, and heat removal characteristics of heat exchangers for the uprated power remain consistent with the original Vogtle cooldown analysis and existing Vogtle capabilities. The RHR cooldown times continue to comply with the guidance provided in Branch Technical Position RSB 5-1 and 10 CFR 50 Appendix R. Therefore, the NRC staff concludes that the impact of the power uprating on RHR performance would not be significant.

Emergency Core Cooling System (ECCS)

From the licensee's study, no adverse impact to ECCS operability or vulnerability to single failure due to the power uprating was identified. As noted earlier, ECCS performance analyses were previously performed at the uprated power and lower RCS flow and accepted by the NRC staff for amendments regarding use of VANTAGE-5 reload fuel. The licensee also performed large and small break analyses to demonstrate that similar cases, using the same evaluation models and assuming the lower T_{hot} and T_{cold} values are bounded by the previous analyses. The NRC staff has reviewed the licensee's analyses and concludes that the ECCS analyses referenced in support of the power uprate continue to be in compliance with 10 CFR 50.46 and Appendix K. The Vogtle ECCS design is, therefore, acceptable for operation at the uprated power conditions.

Accident Analyses

The licensee states that all events in FSAR Chapter 15 were reanalyzed or reevaluated considering the uprated power as part of the documentation to support use of Westinghouse's VANTAGE-5 reload fuel and to support relocation of the narrow range instrumentation taps on the Vogtle steam generators. The accident analyses supporting the power uprating also assumed a reduced minimum RCS design flow associated with a steam generator plugging level of 10 percent. The NRC staff reviewed these analyses and concluded that appropriate safety criteria are met.

The licensee's uprating submittal assesses the impact of lowering T_{hot} and T_{cold} on the results of the existing approved analyses of FSAR Chapter 15. For events that are limited by the calculated DNBR, the licensee found that, at the lower operating temperature, the initial margin to the DNBR criterion was greater and the calculated DNBR was, therefore, not as limiting. Thus, the licensee concluded that the previously analyzed and approved cases bound the reduced RCS temperature cases. Similarly, for cases in which maximum pressure or inventory is the limiting parameter, the lower temperature cases provided greater margins to limiting criteria. From its assessment the licensee concluded that, except for FSAR depressurization events, the existing FSAR Chapter 15 analyses continue to be bounding when the reduced RCS temperatures and the uprated power are considered. Two depressurization events, which were reanalyzed using methodologies previously approved for application to Vogtle, are (1) Inadvertent Opening of a Steam Generator Relief or Safety Valve and (2) Steam Line Break. From these reanalyses, performed at the uprated power with reduced RCS flow and temperature, the licensee found that results for both events continue to meet all applicable criteria, including DNBR.

The NRC staff has reviewed the licensee's analyses and conclusions and finds them acceptable to support operation at the uprated power with reduced RCS temperatures and flows and a limit of 10 percent for steam generator tube plugging.

2.2 SAFETY-RELATED COOLING WATER SYSTEMS

Nuclear Service Cooling Water System and Ultimate Heat Sink

Each Vogtle unit is served by two nuclear service cooling water (NSCW) basins and associated induced draft cooling towers, which function as the ultimate heat sink (UHS). The NSCW system transfers heat from the diesel generators (DGs), containment coolers, control building essential chiller condensers, various engineered safety feature (ESF) pump coolers, and the component cooling water (CCW) heat exchangers to the UHS.

The licensee evaluated the NSCW system and the UHS under the most limiting postulated post-accident conditions to determine the effects of the proposed changes on the NSCW basin inventory and peak NSCW basin temperature. For this evaluation, the licensee selected a model representing the cooling water systems for Vogtle Unit 2 because the postulated peak heat load for Unit 2, including the heat load resulting from the storage of spent fuel in the high-density spent fuel racks in Unit 2's spent fuel pool, bounds the peak postulated heat load for Unit 1. The model also conservatively included heat input resulting from operation of the NSCW pumps.

The licensee evaluated the effect on basin inventory of a design basis loss-of-coolant accident (LOCA), which maximizes heat rejection to the UHS under the uprated conditions. The decay heat load used in the analysis for the period greater than 3904.1 seconds after the LOCA was revised using the methodology of NRC Branch Technical Position ASB 9-2, Revision 2, dated July 1981, to reduce the degree of conservatism in the analysis. Meteorological data which results in the greatest evaporative loss of UHS inventory was used in the analysis, consistent with Vogtle's original design basis. The heat loads used for the analysis included the DGs, containment coolers, control building essential chiller condensers, CCW system heat exchangers, and various ESF pump coolers. The operation of two NSCW trains was assumed for the first 24 hours post-LOCA, followed by single train operation for the remainder of the analysis. A transfer pump was assumed to operate to transfer water from the idle train basin to the operating basin after the first 24 hours. A cooling tower basin initial temperature of 90°F was used, consistent with technical specification 3/4.7.5.

The results of the analysis that maximizes NSCW basin inventory usage indicates that the basin contains sufficient inventory to support post-LOCA operation for at least 30 days. Therefore, the NSCW basin satisfies the guidance of Regulatory Guide 1.27 and Section 9.2.5 of the Standard Review Plan (SRP), NUREG-0800, with regard to UHS inventory. Accordingly, the staff concludes that the UHS inventory is sufficient to support operation at the uprated power level.

The licensee also evaluated the maximum NSCW basin temperature. For this analysis, the licensee assumed heat loads and meteorological conditions that maximized the NSCW basin temperature following a design basis LOCA. To maximize the calculated basin temperature, the licensee assumed the operation of a single NSCW train with no transfer of water between basins . An initial temperature of 90°F for the cooling tower basin was also used for this analysis.

The maximum calculated NSCW basin temperature was found to be 92.1°F, which is less than the design basis value of 95°F. Thus, the analysis demonstrates that the UHS has adequate cooling capability to satisfy the guidance of Section 9.2.5 of the SRP and the requirements of General Design Criterion (GDC) 44 of Appendix A to 10 CFR 50 with regard to UHS operation under uprated conditions.

The NRC staff has reviewed the licensee's evaluation of the cooling water system performance in transferring the heat released during a design basis LOCA, in addition to heat loads resulting from normal operation, to the UHS. The NRC staff concludes that the NSCW system is capable of performing its cooling function under both normal and accident conditions under conditions associated with the proposed change. Moreover, the NSCW system design would continue to satisfy the guidance of Section 9.2.1 of the SRP and the requirements of GDC-44 with regard to system functional performance, and is, therefore, acceptable for operation as proposed.

Component Cooling Water System

The CCW system transfers heat to the NSCW system from the RHR system, the spent fuel pool cooling (SFPC) system, reactor coolant pump thermal barriers, certain chemical and volume control system (CVCS) heat exchangers, and various other components.

The licensee evaluated the effects of the proposed power uprate and hot leg temperature reduction on the CCW system. The licensee found that the primary side inlet temperatures to the CVCS heat exchangers and the reactor coolant pump thermal barriers were bounded by the design outlet temperature (primary side) of the steam generator, 560°F, for conditions resulting from the proposed change. Since the rate of heat transfer is directly related to the temperature difference across a heat exchanger, the heat loads on the CCW system from the CVCS heat exchangers and reactor coolant pump thermal barriers were also bounded by the design values determined from the design outlet temperature (primary side) of the steam generator.

Based on the results of licensee's evaluations of the performance of the UHS, the SFPC system, and the RHR system during RCS cooldown, the NRC staff finds that the CCW system is capable of transferring the normal operating and accident heat loads from the RHR heat exchangers and the SFPC system to the NSCW system under conditions resulting from the proposed changes. The NRC staff agrees with the licensee's conclusion that changes in heat load from the remaining components cooled by the CCW system are negligible due to operation at the proposed power level. Therefore, the NRC staff concludes that the CCW system design would continue to satisfy the guidance of Section 9.2.2 of the SRP and the requirements of GDC-44 with regard to system functional performance at the uprated conditions, and is, therefore, acceptable for operation as proposed.

Spent Fuel Pool Cooling System

The licensee evaluated the SFPC system to determine the effects of the power uprate on the capability of the SFPC system to maintain fuel pool temperatures within acceptable limits. Although the SFPC systems for Unit 1 and 2 are of the same design, the Unit 2 spent fuel pool has a greater storage capacity than the Unit 1 pool. The high-density storage racks in Unit 2's spent fuel pool are designed to contain 2098 fuel assemblies, compared to a storage capacity of 936 fuel assemblies for Unit 1's spent fuel pool. Therefore, the licensee determined that an evaluation of the effects of the uprate on the Unit 2 spent fuel pool cooling system would bound the Unit 1 design.

The licensee evaluated the Unit 2 spent fuel pool peak temperature assuming conservative values for spent fuel pool inventory, NSCW basin temperature, and heat loads on the various cooling water systems. The licensee's evaluations included three scenarios: (1) a normal refueling offload, (2) a maximum normal refueling offload, and (3) an emergency core offload. In each scenario, one train of spent fuel pool cooling was assumed to be operating with the remaining train unavailable. The total heat load was determined based on the uprated power level for the entire inventory of fuel assemblies assuming a burnup equivalent to 1145 days of full power operation and using the methodology of Branch Technical Position ASB 9-2.

The licensee's analysis indicated that the bulk spent fuel pool temperature would remain below 140°F for the normal refueling offload, and below the temperature corresponding to bulk boiling conditions for the emergency core offload.

The NRC staff has reviewed the licensee's analyses and finds them to be acceptable. The results of the analysis satisfy the guidance of Section 9.1.3 of the SRP and the requirements of GDC-44 with regard to the functional capability of the spent fuel pool cooling system. Therefore, the staff concludes that the spent fuel pool cooling system has adequate capacity to support the additional heat loads that would result from operation as proposed.

2.3 BALANCE OF PLANT SYSTEMS

Auxiliary Feedwater System

The AFW system, in conjunction with two condensate storage tanks (CSTs), provides a means of supplying feedwater to the steam generators for cooling the RCS under emergency conditions or situations where the main feedwater system is not available whenever RCS temperature is sufficiently high to produce steam. The licensee evaluated the adequacy of the AFW system for the proposed changes. This included an evaluation of the CST capacity and the AFW system flow rate.

As noted in Section 10.4.9.2.1 of the Vogtle FSAR, the CST is designed to maintain the plant at hot standby for four hours, followed by a five-hour cooldown to 350°F at an average rate of 50°F per hour. Vogtle TS 3.7.1.3 requires that a minimum volume of 340,000 gallons of water be retained in the CST for use by the AFW system. The licensee finds that 235,000 gallons of water would be needed to hold the plant in hot shutdown for two hours and then accomplish a four-hour cooldown to the RHR system initiation condition, assuming initial operation at the proposed power level and that only one reactor coolant pump operated throughout the hot shutdown and cooldown periods. Based on minimum flow and heat removal requirements, the NRC staff finds the available water volume to be consistent with the stated assumptions. The NRC staff finds that the CST capacity required by the TS is sufficient to provide the additional heat removal necessary to meet the CST design basis as presented in the FSAR for operation at the uprated power level. Based on these conclusions, the NRC staff finds the present AFW storage requirements to be consistent with the guidance of Sections 9.2.6 and 10.4.9 of the SRP, and the requirements of GDC-34 with regard to residual heat removal. Therefore, The NRC staff concludes that the AFW storage capacity is acceptable for operation at the proposed power level.

The licensee's accident analyses to support operation at the uprated power level assume total AFW system flow rates equal to those presented in the FSAR for operation at the originally licensed power level. Since the steam generator safety valve setpoints will not be changed for the proposed operating conditions, the staff concludes that the AFW pumps will continue to be capable of delivering the minimum flow assumed in the accident analyses to the steam generators while at their highest anticipated pressure.

Turbine Generator

The turbine overspeed protection system reduces the risk of generation of turbine missiles that could impact operation of safety-related structures, systems or components.

The licensee's turbine heat balance indicates that the predicted maximum steam generator outlet steam flow is not significantly greater than that calculated for the turbine control valves wide-open condition used in the turbine overspeed protection system analysis. Therefore, the analysis that formed the basis for the NRC staff's original acceptance of the turbine overspeed protection system would not be affected significantly by the proposed changes.

Since the probability of turbine missile generation would not significantly change for the rerate, the NRC staff concludes that the turbine overspeed protection system would continue to satisfy the guidance of Section 10.2 of the SRP. Therefore, the NRC staff finds the turbine generator to be acceptable for operation as proposed.

Main Steam Supply System

The main steam supply system (MSSS) directs steam from the steam generators to the turbine generator and various auxiliary loads over a range of operating conditions from system warmup to full power operation. The system also dissipates energy generated in the NSSS to the atmosphere through power-operated atmospheric relief valves (ARVs) or spring-loaded main steam safety valves (MSSVs), or, if the condenser is available, to the main condenser through the steam dump valves or the turbine-generator. In addition, the main steam isolation valves (MSIVs) and main steam bypass isolation valves perform the system isolation function.

The licensee evaluated the capability of the MSSS components to perform their design functions with the proposed changes. The licensee found that the existing setpoints for the ARVs and the MSSVs are adequate for the uprated conditions. The ARVs were found to have adequate capacity to achieve a 50°F per hour cooldown when the main condenser is unavailable. The licensee's engineering evaluations indicate that the MSSVs have adequate capacity since the quantity of steam for which they are designed to pass would prevent an overpressure condition during operation at the proposed power level. The licensee also found the dynamic loading on the MSSS piping due to rapid closure of the turbine stop valves, or operation of the ADVs or MSSVs, to be within acceptance limits. The licensee reviewed the MSIV and main steam bypass isolation valve capability and determined that, with the calculated maximum differential pressure under the proposed conditions acting across the valve, these valves would be capable of performing their containment isolation function within the five seconds assumed in the safety analysis.

The NRC staff has reviewed the licensee's engineering evaluation for operation at uprated conditions and finds it acceptable. The staff concludes that the design features of the MSSS are adequate to satisfy the guidance of Section 10.3 of the SRP. Therefore, the NRC staff finds the MSSS to be acceptable for operation as proposed.

Main Feedwater System

A safety function of the main feedwater system includes feedwater isolation. The system also provides a flow path for the AFW system flow to the steam generators.

The licensee's analyses show that isolation capability of the main feedwater isolation valves and the main feedwater bypass isolation valves would be unaffected by changes in feedwater flow and temperature resulting from the proposed changes. The maximum feedwater temperature that was used in the pipe stress analysis for the safety-related portion of the feedwater system envelopes the feedwater temperature calculated for operation with the proposed changes. The licensee also found that the existing analysis for the dynamic loading due to fast closure of a main feedwater check valve enveloped expected conditions for the proposed changes.

The NRC staff has reviewed the licensee's analyses and finds them to be acceptable. The staff concludes that the main feedwater system design features are adequate to continue to satisfy the guidance of Section 10.4.7 of the SRP with regard to the capability to withstand postulated dynamic effects and to isolate non-essential portions of the system. Therefore, the NRC staff finds the main feedwater system to be acceptable for operation as proposed.

Power Conversion System

The licensee's analyses indicate that power conversion systems and components (e.g. the steam dump valves, extraction steam system, turbine generator, main condenser, condensate system, feedwater heaters, heater drain system, and circulating water system) satisfy their power generation design bases for operation as proposed. Since the power generation design bases do not include safety functions, these systems and components are acceptable without review for operation as proposed.

2.4 AUXILIARY PLANT SYSTEMS

Radwaste Systems

The Vogtle radwaste systems were originally evaluated and accepted by the NRC staff based upon a core power level of 3565 MWt. This power level was used to determine the source term for gaseous and liquid effluents and the waste volume. Therefore, operation at the uprated power level would not change the analysis results or the NRC staff's conclusions in the Vogtle SER (NUREG-1137). Accordingly, the Radwaste systems continue to be acceptable for control of radioactive wastes for operation as proposed

Control Room Emergency Filtration System

The control room emergency filtration system is equipped with high efficiency particulate and activated charcoal filters to limit the dose to control room operators following a design basis accident and to maintain a habitable environment in the control room.

The impact of the proposed uprated power level on control room habitability results only from changes in the assumed radioactive source term and activated charcoal filter iodine loading. The originally licensed source term and activated charcoal filter iodine loading were computed based upon the uprated power level. Therefore, the NRC staff's conclusions in the Vogtle SER (NUREG-1137) remain valid for the proposed changes. Operation at the uprated power level continues to be acceptable with regard to dose to control room operators.

2.5 VOGTLE PROGRAM REVIEWS

Equipment Qualification

The licensee evaluated the effects of the proposed changes on qualified equipment. The radiological dose assumptions used in the present Vogtle equipment qualification program bound the calculated dose, based on the uprated power level, for equipment located both inside and outside containment. The calculated environmental temperature and pressure profiles for the limiting LOCA and MSLB cases based on the proposed changes are enveloped by the current Vogtle equipment qualification program for equipment inside containment. The licensee's evaluation of mass and energy releases in the MSIV area, including blowdown of superheated steam under conditions associated with the proposed changes, determined that the maximum calculated temperature is bounded by the value assumed in the current Vogtle equipment qualification program.

The NRC staff has reviewed the licensee's evaluation and finds it to be acceptable. Since the equipment qualification parameters affected by the proposed changes remain bounded by the values assumed in the licensee's current program, the NRC staff concludes that the present equipment qualification program for Vogtle is acceptable for operation as proposed.

High and Moderate Energy Pipe Breaks

The licensee evaluated the effects of the proposed changes upon current analyses for breaks in pipes of high or moderate energy. The evaluation determined that calculated jet thrust and impingement forces, and postulated pipe break location analyses for the main steam and feedwater system piping, would not be affected by the proposed changes, because the operating pressures of these systems would not increase. The licensee also found that other systems for which the effects of pipe breaks were previously evaluated would not be affected by the proposed changes.

The NRC staff reviewed the licensee's evaluation and finds it to be acceptable. Therefore, the staff concludes that existing pipe break analyses are acceptable for operation under the conditions associated with the proposed changes, and that the conclusions in the SER (NUREG-1137) regarding the effects of high and moderate energy pipe breaks remain valid.

Internal Flooding

The licensee evaluated the piping systems with increased flow rates as a result of the proposed changes to determine the effects of the power increase on the analysis of flooding outside of containment. These piping systems include the condensate, main feedwater, and main steam systems. The maximum temperatures and pressures for this piping, as indicated in the turbine cycle heat balances for the proposed increase in power, were used for the flooding evaluation. Based upon a review of pipe break outflows under the increased power conditions, the evaluation indicated that the current flood levels in the flooding analyses would be unaffected.

Fluid releases inside containment are based upon the present zero-load steam generator inventory, which bounds the fluid releases that would result from the proposed changes.

Therefore, the NRC staff concludes that the present flooding analyses continue to be valid and that flood levels would not increase due to operation as proposed.

2.6 PLANT STRUCTURAL DESIGN

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed changes upon the structural integrity of the NSSS and BOP pressure boundary systems, including the system piping, components, their supports, the reactor vessel and internal components, steam generators, control rod drive mechanisms, reactor coolant pumps, and pressurizer.

Reactor Vessel and Internals

The licensee assessed the adequacy of the reactor vessel by modifying the original stress report such that increases in both maximum stresses and fatigue cumulative usage factors associated with the transient changes for the Vogtle uprated power conditions are added to the original calculated values for various critical locations. The calculated stresses and fatigue usage factors for reactor vessel components were found to be within the allowable limits of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1971 Edition with Addenda through the Summer of 1972.

The licensee assessed the adequacy of the reactor internal components for the power uprated conditions. The assessment included analyses for a LOCA, fluid-induced vibration, thermal transients, and stress and fatigue analyses for critical reactor internal components such as core support components and the baffle-former-barrel bolts. The results show that the structural adequacy of the reactor internals would not be affected by the proposed uprated conditions.

The NRC staff has reviewed the licensee's assessment for uprated power conditions and the design bases analyses. The NRC staff agrees with the licensee's conclusion that the original design bases analyses of the reactor internal components and reactor vessel would not be affected by the proposed uprate conditions, and therefore, would remain within the Code-allowable limits as defined in the FSAR.

The licensee has also addressed reactor vessel structural integrity with respect to fracture toughness requirements for protection against pressurized thermal shock events. In its letter of June 25, 1992, the licensee provided results of calculations for both Vogtle vessels that were performed by Westinghouse using the guidance of Regulatory Guide 1.99, Revision 2. The results indicate that the Charpy upper shelf energy is expected to be above 50 foot pounds, even after 48 effective full power years of operation. The calculations were based on fluence values that were developed assuming that the Vogtle uprating from 3411 to 3565 MWt would occur during calendar year 1992. The projected reference temperature, calculated for end of vessel life by the method given in 10 CFR 50.61(b)(2), for the materials in the reactor vessel beltline are shown to be less than the screening criterion in 10 CFR 50.61(b)(2). Based upon these calculations, the NRC staff concludes that the Vogtle reactor vessels possess adequate fracture toughness properties for operation as proposed.

Control Rod Drive Mechanisms

The licensee evaluated the adequacy of the Control Rod Drive Mechanisms (CRDMs) by comparing the design bases input parameters with the operating conditions for the proposed power uprate. The results show that the uprate operating conditions would have no impact on the original design bases analyses for the CRDMs. The NRC staff has reviewed the licensee's evaluation and concurs with the licensee's conclusion that the current design of CRDMs would not be impacted by the power uprate.

Steam Generators

The licensee performed analyses of Vogtle's steam generators for the proposed power uprate conditions. Results of the original design analyses were multiplied by the primary-to-secondary pressure difference ratio of the uprate conditions to the design bases conditions, in order to obtain the uprate condition results. The calculated stress intensities were found to be within the allowable limits at all locations. The fatigue usage factors were also found to be acceptable. However, the increased cumulative usage in the secondary manway bolts shortened their fatigue life from 20 years to 14.5 years. Accordingly, in its letter of February 23, 1993, the licensee stated that the secondary manway bolts would be replaced prior to 14.5 years of plant service.

The licensee also performed structural evaluations for the critical components of the Vogtle steam generators. The evaluations performed considered two cases: (1) power uprate accompanied by a T_{hot} reduction, and (2) power uprate with no T_{hot} reduction. The licensee performed evaluations on the effect of the power uprate on the minimum required steam generator tube wall thickness, the number of steam generator tubes susceptible to anti-vibration bar (AVB) wear, and the propensity of the steam generator tubing to various forms of corrosion degradation.

The Westinghouse Model F steam generators at Vogtle Units 1 and 2 have thermally treated Inconel-600 tubes with an outside diameter of 0.688 inches and a wall thickness of 0.040 inches. As a result of the proposed changes for power uprating, several design parameters of interest to steam generator tube structural performance would change. These include T_{hot}, the differential pressure across the steam generator tubing, and the flow rate through the steam generators. The licensee has analyzed these design parameter changes associated with the power uprating for the effect on the structural integrity of the steam generator tubing.

The licensee determined the minimum acceptable steam generator tube wall thickness for the rerated conditions to be 0.016 inches (40% of the nominal wall thickness) using the criteria of Regulatory Guide 1.121, "Bases for

Plugging Degraded PWR Steam Generator Tubes." The present minimum acceptable tube wall thickness for Vogtle is 0.014 inches (35% of the nominal wall thickness). This increase in minimum steam generator tube wall thickness is due to the increased differential pressure across the tubes for the rerated conditions.

The licensee also performed an analysis of the number of additional steam generator tubes that may be affected by wear at the AVB supports as a result of the power uprating. This analysis showed that the number of tubes requiring plugging due to AVB wear could increase by approximately 10% to 16% over the currently projected 0.6% of the tubes; therefore, the upper limit of projected tube plugging for AVB wear at the uprated conditions would be approximately 0.7%. The methodology used by the licensee to determine the increased susceptibility of steam generator tubes to AVB wear included the following steps:

1. Simulating various AVB conditions using a Monte Carlo technique. The tube bundle was modeled with seven different row groups.

2. Determining the tube natural frequency and stability at the current operating conditions from a finite element structural analysis model using the AVB conditions determined in the Monte Carlo analysis.

3. Calculating a relative stability ratio (a ratio of the stability ratio at the rerated conditions to the stability ratio at the current design conditions).

4. Using the relative stability ratio (step 3 above) and the tube stability ratio (step 2 above) to determine the proportion of tubes in each of the seven row groups that are stable at the current operating conditions but become unstable at either of the rerated conditions.

5. Using field data to determine the percentage of the total number of tubes in each of the seven groups that have observed wear at different times.

6. Calculating the expected increase in tube plugging as a result of the rerate from steps 4 and 5.

The licensee has determined that approximately 25% of the nominal tube wall thickness is required to demonstrate acceptable structural performance of the steam generator tubing for AVB wear based on ASME Code criteria. The NRC staff performed a statistical analysis of the wear rate at the AVBs for Vogtle Unit 1 from data supplied by the licensee's letter of February 12, 1993. The observed wear rate, as estimated from this data, is consistent with the rate assumptions that have been historically used in the development of steam generator tube plugging criteria. The licensee does not expect the uprating will have a significant effect on the wear rate of individual steam generator tubes. The licensee has concluded that the power uprating would not require a change to the steam generator tube plugging criteria in the Vogtle Technical Specifications.

The licensee performed an evaluation on the effect of the rerating on the corrosion propensity for the steam generators. The analysis technique did not calculate absolute corrosion rates, but it did calculate the relative rate change resulting from operating features, conditions, or chemistry changes. The analysis showed that the steam generators are most susceptible to pitting corrosion and least susceptible to denting corrosion. As a result of the rerating, the corrosion propensity without a T_{hot} reduction increases for primary water stress corrosion cracking (PWSCC), outside diameter stress corrosion cracking (ODSCC), and denting. As a result of the rerating, the corrosion propensity with a T_{hot} reduction increases for denting but decreases for PWSCC and ODSCC. It should be noted that these propensities do not indicate that corrosion will or will not occur.

Based on the above review of Vogtle steam generator tube integrity, the NRC staff finds that the wear rates at Vogtle are consistent with the growth rates typically assumed in plugging criteria calculations. The staff does not expect that these wear rates would change significantly as a result of the power uprate. The power uprate would result in a slight increase in the minimum acceptable tube wall thickness; however, the tube plugging criterion is still appropriate when the growth rate, minimum acceptable wall thickness, and measurement uncertainty are combined. Therefore, the NRC staff concludes that the power uprate will not cause a significant adverse affect on steam generator tube integrity.

Accordingly, the NRC staff agrees with the licensee's conclusion that the Vogtle steam generators are acceptable for operation as proposed.

Reactor Coolant Pumps

The licensee has evaluated the impact of the conditions resulting from the proposed power uprate on the design analyses of the reactor coolant pumps. The impact was found to be insignificant because (1) the reactor coolant system pressure would not change, and (2) the decrease of the steam generator outlet temperature would reduce the overall thermal transient on the reactor coolant pumps.

The NRC staff has reviewed the licensee's assessment and agrees with the licensee's conclusion that the uprated conditions for the reactor coolant pumps are bounded by the original analyses. Therefore, the existing reactor coolant pumps are adequate for operation as proposed.

Pressurizer

The licensee evaluated the pressurizer and its components, such as the pressurizer safety valves and pressurizer relief valves, for the proposed uprate conditions. The licensee found that the uprate conditions are bounded by those used in the original pressurizer analyses. Thus, the pressurizer components would continue to satisfy the ASME Code, Section III stress limits.

The NRC staff has reviewed the licensee's evaluation and agrees that the original pressurizer analyses are applicable to the uprated conditions. Therefore, the staff concurs with the licensee's conclusion that the existing pressurizer and components are adequate for operation as proposed.

NSSS Piping, Supports and Primary Component Supports

The licensee evaluated the following components and supports for the uprated operating conditions: the RCS piping and supports, the primary equipment nozzles, the primary equipment supports, and the Class 1 auxiliary (branch) lines connected to the RCS piping. The evaluation compared the existing design bases with the performance requirements at the uprated power level, with respect to the design system parameters, transients, LOCA forcing functions, and the dynamic LOCA reactor vessel movements used in the original structural analyses. The evaluation verified that the power uprated conditions are bounded by those used in the original design analyses.

Based on its review of the design bases analyses, the NRC staff agrees with the licensee's conclusion that the existing NSSS piping and supports, primary equipment nozzles, primary equipment supports, and branch lines would remain in compliance with the design bases criteria given in the FSAR. These components and supports are, therefore, acceptable for operation as proposed.

The licensee also evaluated the Vogtle auxiliary NSSS components supplied by Westinghouse. For this evaluation, the licensee compared the original design bases and qualification requirements with those for the power uprate conditions. The licensee found that, for each component evaluated, the original design enveloped those for the power uprate.

The NRC staff reviewed the plant operating parameters used for the evaluation of auxiliary NSSS components supplied by Westinghouse. The NRC staff agrees with the licensee's conclusion that the uprate conditions associated with the proposed changes would not impact the design bases qualification of these components. These auxiliary NSSS components are, therefore, acceptable for operation as proposed.

Balance Of Plant Piping Systems

The licensee evaluated the adequacy of the BOP piping systems by comparing the existing design bases conditions with those for the power uprate. From this review, the licensee determined that (1) the system design pressures and temperatures would not change for the power uprate and (2) the changes in flow rate for the uprate are insufficient to impact pipe stresses and support loads of systems for which transient analyses were performed.

Based on its review of the main steam and feedwater systems, the NRC staff finds that the pressures and temperatures used for the original design analyses bound those for the uprate operating conditions. For other secondary plant systems, the changes in operating conditions due to the power uprate would be negligible. Therefore, the NRC staff concurs with the licensee's conclusion that the changes in system operating conditions due to the uprate would not have any significant impact on the pipe stresses and pipe support loads. The existing design bases piping analyses would remain within the Code-allowable limits for plant operation under uprate conditions.

The licensee also reviewed the design bases pipe break analyses to evaluate the effects of the uprate conditions upon the pipe break locations, jet thrust and jet impingement forces that were used in the plant safety analyses and the design of pipe whip restraints. The review verified that the existing postulated pipe break locations would not be changed by the power uprate since the design bases piping analyses will not change due to the power uprate. The current design bases for jet thrust and jet impingement forces due to postulated pipe breaks for these systems would not be affected by the uprate because the operating pressures remain within the original design basis. Therefore, the NRC staff agrees with the licensee's conclusion that the original design analyses for the pipe break locations, jet thrust, jet impingement and pipe whip restraints would be unaffected by the power uprate.

Based on the above evaluation, the NRC staff concludes that the BOP piping systems are acceptable for operation as proposed.

2.7 CONTAINMENT INTEGRITY ANALYSES

The licensee has performed containment integrity analyses at uprated power to ensure that the maximum pressure inside the containment would remain below the containment building design pressure of 52 psig if a design bases LOCA or MSLB inside containment should occur during plant operation. The analyses also established the pressure and temperature conditions for environmental qualification and operation of safety related equipment located inside the containment. The peak pressure is also used as a basis for the containment leak rate test pressure to ensure that dose limits would be met in the event of a release of radioactivity to containment.

The licensee indicated that the containment functional analyses included the assumption of the most limiting single active failure and the availability or unavailability of offsite power, depending on which resulted in the highest containment temperature and pressure. Bounding initial temperatures and pressures for analyses were selected to envelope the limiting conditions for operation. The licensee indicated that although the current licensed power level is 3411 MWt, all safety related systems were designed for operation at a higher core power level of 3565 MWt.

LOCA Containment Integrity Analyses

As in the current licensing basis FSAR, the rerating analyses were performed at uprated power to determine the peak pressure and temperature response for the containment atmosphere for the postulated double ended pump suction break (DEPSG) for minimum and maximum safety injection cases and the double ended hot leg break (DEHLG) case which results in maximum mass and energy release rates.

For the DEHLG break, the Vogtle rerating analyses calculated a containment peak pressure of 36.54 psig. The current FSAR resulted in a peak pressure of 38.24 psig. For the DEPSG break, the Vogtle rerating analyses provided a containment peak pressure of 34.61 psig. The current FSAR resulted in a break pressure of 35.62 psig. The peak containment temperature during LOCA was 250°F for the DEHLG break. The licensee indicated that the essential differences between the FSAR and rerating analyses are as follows:

1. The mass and energy release evaluations in the FSAR used Westinghouse report WCAP-8264 for analyses; whereas, the rerate evaluation used the current updated Westinghouse report WCAP-10325 for analyses. The analyses used an RCS temperature uncertainty of 6°F rather than the 4°F in the FSAR; an RCS pressure of 2250 + 50 psi rather than the 2250 + 30 psi in the FSAR; the 1979 ANS decay heat model rather than the 1971 model in the FSAR; and a revised method of calculating mass and energy releases from a broken loop post-LOCA to include steam and water mixing.

2. Revised containment structural heat sinks were modeled in the rerate analyses. These revised heat sinks were based upon a more accurate but conservative accounting which better reflect the current Vogtle plant design as compared to the data available when the FSAR analyses were performed. In the rerate analyses, an initial containment pressure of 3.0 psig was used versus 0.3 psig in the FSAR. The licensee was asked to add 2.7 psig to the peak pressure for containment test pressure due to the lower initial pressure used in the FSAR.

3. The current FSAR analyses assumed two fan coolers available. The rerate analyses assumed four fan coolers available in accordance with the requirements of the Vogtle Technical Specifications.

The licensee indicated that for the Vogtle uprate analyses, Westinghouse used the same methodology and assumptions (except the Vogtle plant specific data) for mass and energy release calculations as was approved by the NRC staff on the Dockets for Catawba, McGuire, Sequoyah, Watts Bar, Surry, Millstone 3 and Beaver Valley 2 . The calculation of containment pressure and temperature transients was accomplished by use of the Westinghouse computer code COCO, which the NRC staff has accepted for use in many plants.

MSLB Containment Integrity Analysis

As in the current licensing basis FSAR, the rerating analyses were performed for a spectrum of MSLBs at 102%, 70%, 30% and 0% of uprated power level to determine the peak containment pressure and temperature. The Vogtle rerating limiting case was the 0.22 ft^2 double ended rupture without entrainment at 30% power; this case provided a peak containment pressure of 32.7 psig. The current FSAR limiting case was the 0.14 ft^2 split at hot shutdown which provided a peak pressure of 41.9 psig. The rerating analyses determined a peak containment temperature of 303°F for the MSLB cases.

The essential difference in assumptions between the rerate and FSAR analyses for the containment model are the same as indicated in the LOCA analyses. For MSLB mass and energy release assumptions, the essential differences were:

1. Main feedwater flow rate - the reanalysis generally used greater flow rates than was used in the FSAR analysis

2. Auxiliary feedwater enthalpy - the current FSAR used 429.5 BTU/lb mass and the rerate analysis assumed 101 BTU/lb mass

3. Decay Heat Model - the FSAR used the 1971 model whereas the rerate analysis used the 1979 model.

For the MSLB analyses, the LOFTRAN computer code was used to generate the mass and energy released to the containment and the computer code COCO was used to generate the containment pressure and temperature response. These codes have been used by Westinghouse for other similar plants.

The LOCA peak pressure of 36.54 calculated previously for the DEHLG case bounds the peak containment pressure response to the MSLB and remains below the containment design pressure of 52 psig at the rerated power conditions. Also, the newly calculated pressure and temperature curves for LOCA and MSLB cases remain enveloped by the curves used for equipment qualifications and for containment leak rate test pressure.

The licensee indicated that the reductions in the calculated peak pressure and temperature for the rerate power analyses were due to the use of revised methods for calculating the mass and energy releases to the containment, more accurate accounting of heat sinks, and by taking credit for all fan coolers required to be operable by current technical specifications.

Based on the above discussion, the NRC staff finds the licensee's analyses for determining the containment peak pressure and temperature for design basis LOCA and MSLB acceptable as the methodology and assumptions used for calculating mass and energy release and for calculating pressure and temperature transients have been previously used for plants of similar design to meet the requirements of Standard Review Plan (SRP) Section 6.2.1.3 for mass and energy analyses and Section 6.2.1.1.A for dry PWR containment integrity peak pressure analyses. The proposed change for power uprate will not affect the containment integrity as the calculated peak containment pressure of 36.5 psig remains below the containment design pressure of 52.0 psig.

Containment Sump Temperature

The licensee indicated that mass and energy release rates for the double-ended pipe suction break with minimum safety injection were found to be limiting for sump temperature calculations. The COCO containment evaluation model was used to calculate the sump temperature profile. The containment model did not take credit for the maximum engineered safety features recirculation equipment being available. This assumption maximizes sump temperature. Also, assumptions were made to concentrate as much energy as possible in the containment sump. The results of these assumptions is a minimum steam phase heat inventory and a corresponding minimum heat removal by the fan coolers.

The maximum sump temperature at the time of switchover to recirculation was found to be 244°F, which is about 8°F different from the current licensing basis calculation. This number was used in the reverification of piping stress calculations and concluded that the piping stresses remain within their original acceptance limits. The licensee also indicated that this change in sump temperature does not affect the minimum net positive suction head requirements for the ECCS system.

Based on this review, the staff finds the rerate acceptable with regard to containment sump temperature as it does not adversely affect the safety related equipment.

LOCA Short-Term Subcompartment Analysis

The licensee indicated that Westinghouse has evaluated the current short-term LOCA mass and energy releases used for the steam generator compartment, the reactor vessel cavity, and the pressurizer subcompartment pressure and temperature analyses. From these analyses, the licensee finds that the current short-term LOCA mass and energy releases as identified in the Vogtle FSAR remain bounding for the rerating conditions. Based on the licensee's evaluation, the staff concludes that the rerating is acceptable as the subcompartment pressure loading analyses from high-energy-line ruptures remain bounded by the current FSAR analyses.

Post-LOCA Containment Hydrogen Generation

The licensee indicated that the Vogtle containment post-LOCA hydrogen generation analyses was reviewed to determine any impact due to rerate. The current hydrogen generation analysis is based on a total core thermal power of 3565 MWt, which is the uprated power level. For corrosion sources, the hydrogen generation analysis is based on corrosion rates corresponding to the temperature profile in the containment under post-LOCA conditions. Since the temperature profile in the current hydrogen generation analysis envelopes the temperature profile in the containment analysis for the rerate, the corrosion rates used in the current hydrogen generation analysis are conservative. Therefore, the staff concludes that the current hydrogen generation analysis is not affected by the rerating power level.

2.8 TECHNICAL SPECIFICATION AND LICENSE CHANGES

The licensee proposed the following changes to the Vogtle Technical Specifications and license related to the power uprating:

1.a TS Page 1-5, DEFINITIONS - Change the value of definition number 1.27, "RATED THERMAL POWER," from 3411 MWt to 3565 MWt. Also, in operating license paragraph 2.C.(1), change the authorized reactor core power level from 3411 MWt to 3565 MWt. These changes reflect the proposed uprated power.

1.b TS Page B 3/4 7-1, BASES Section 3/4.7.1.1, SAFETY VALVES - Change the total relief valve capacity statement in the second sentence of the second paragraph from "... 123% of the total secondary steam flow of 15,135,453 lbs/h..." to "...117% of the total secondary steam flow of 15.92×10^6 lbs/h..." This change reflects the change in steam flow for the uprated thermal power.

2. TS Page 2-4, TABLE 2.2-1, REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - Change the value of the Z factor in the Overtemperature Delta-T trip function from 7.04 to 8.8 percent of span. This change reflects a change in the calculation of the Z value and retains consistency with the safety analyses performed for uprated power. The licensee verified the new value using methods described in the approved Westinghouse topical report WCAP-8745-P-A.

3.a TS Page 2-9, TABLE 2.2-1, TABLE NOTATIONS, NOTE 1: - In discussions within note 1, replace the upper value of the q_t - q_b decision point from +11.0% to +10.0% in statement (1). In statement (3), also change this value from 11.0% to 10.0%, and change the delta-T Trip reduction factor from 1.9% to 2.7%. These changes are made to the reset function to reflect the power uprate and retain consistency with the safety analyses performed for uprated power. The licensee verified the new value using methods described in the approved Westinghouse topical report WCAP-8745-P-A.

3.b Page 2-9, TABLE 2.2-1, TABLE NOTATIONS, NOTE 2: - Change the limiting value of the amount that the maximum trip setpoint may exceed the computed setpoint from 3.1% of delta-T span to 1.5% of delta-T span. This change is made to the reset function to reflect the power uprate and to retain consistency with the safety analyses performed for uprated power. The licensee verified the new value using methods described in the approved Westinghouse topical report WCAP-8745-P-A.

The NRC staff finds the above TS changes acceptable because they are appropriate for the uprated power and are supported by acceptable analyses approved herein.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on March 15, 1993 (58 FR 13806).

In this finding, the Commission determined that issuance of these amendments would not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: March 22, 1993