December 7, 1987

Docket Nos.: 50-269, 50-270
and 50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment Nos. 164 , 164, and 161 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 - Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. 63068, 63069, 63070)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 164, 164, and 161 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated August 13, 1986, as supplemented May 14, 1987.

The amendments revise the TSs to raise the reactor protection system high reactor coolant system pressure trip setpoint from 2300 psig to 2355 psig.

A copy of our Safety Evaluation is also enclosed. Notice of issuance of the enclosed amendments will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/S/

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II



Enclosures:
1. Amendment No. 164 to DPR-38
2. Amendment No. 164 to DPR-47
3. Amendment No. 161 to DPR-55
4. Safety Evaluation

cc w/enclosures: See next page

| | | |
|---|---|---|
| PDII-3/DRP-I/II | PDII-3/DRP-I/II | KNJ PDII-3/DRP-I/II |
| MDuncan/rad | HPastis | Acting Director KJABBOUR |
| 11/9/87 | 11/12/87 | 12/02/87 |

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

cc:
Mr. A. V. Carr, Esq.
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell & Reynolds
1200 Seventeenth Street, N.W.
Washington, D.C. 20036

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. Paul Guill
Duke Power Company
Post Office Box 33189
422 South Church Street
Charlotte, North Carolina 28242

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated August 13, 1986, as supplemented May 14, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

   D. The issuance of this license 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 amended by changes to the Technial Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 164, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: December 7, 1987

| | | | |
|---|---|---|---|
| | | | KNJ |
| PDII-3/DRP-I/II | PDII-3/DRP-I/II | OGC-Bethesda | PDII-3/DRP-I/II |
| MDuncan/rad | HPastis | | Acting Director |
| 11/9/87 | 11/12/87 | 11/13/87 | 12/02/87 |

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated August 13, 1986, as supplemented May 14, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

   D. The issuance of this license 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 amended by changes to the Technial Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 164 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: December 7, 1987

| | | | |
|---|---|---|---|
| PDII-3/DRP-I/II | PDII-3/DRP-I/II | OGC-Bethesda | PDII-3/DRP-I/II |
| MDuncan/rad | HPastis | G.S.M. | Acting |
| ~~Director~~ | | | KNJ |
| 11/9/87 | 11/12/87 | 11/18/87 | 12/02/87 |

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated August 13, 1986, as supplemented May 14, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

   D. The issuance of this license 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 amended by changes to the Technial Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

\S\

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: December 7, 1987

| | | | |
|---|---|---|---|
| PDII-3/DRP-I/II | PDII-3/DRP-I/II | OGC-Bethesda | KNJ<br>PDII-3/DRP-I/II |
| MDuncan/rad | HPastis | G.S.M. | Acting |
| ~~Director~~ | | | |
| 11/9 /87 | 11/12/87 | 11/18/87 | 12/02/87 |

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 164 TO DPR-38

AMENDMENT NO. 164 TO DPR-47

AMENDMENT NO. 161 TO DPR-55

DOCKET NOS. 50-269, 50-270, AND 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised page are identified by amendment number and contain vertical lines indicating the areas of change.

| Remove Page | Insert Page |
|---|---|
| 2.2-1 | 2.2-1 |
| 2.3-3 | 2.3-3 |
| 2.3-5 | 2.3-5 |
| 2.3-7 | 2.3-7 |

## 2.2 SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE

### Applicability

Applies to the limit on reactor coolant system pressure.

### Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

### Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1967.

### Bases

The reactor coolant system [1] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110% of design pressure. [2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under USAS Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established. [3] The settings, the reactor high pressure trip (2355 psig) and the pressurizer safety valves (2500 psig) [4] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the Reactor Coolant pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.

### REFERENCES


(1) FSAR, Section 5

(2) FSAR, Section 5.2.3.10.1

(3) FSAR, Section 5.2.2.3, Table 5.4-7

(4) FSAR, Section 5.4.6, Table 5.1-1

## Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the reactor coolant system (RCS) high pressure setpoint is reached before the nuclear overpower trip setpoint. The high RCS pressure trip setpoint (2355 psig) ensures that the pressure remains below the safety limit (2750 psig) for any design transient.[2] The low pressure (1800 psig) and variable low pressure (11.14 $T_{out}$ - 4706) trip setpoints shown in Figure 2.3-1 ensure that the minimum DNBR is greater than or equal to the minimum allowable DNBR for those accidents that result in a reduction in pressure.[3,4] The limits shown in Figure 2.3-1 bound the pressure-temperature curves calculated for 4, 3, and 2 pump operation.

Accounting for calibration and instrumentation errors, the safety analyses used a variable low RCS pressure trip setpoint of (11.14 $T_{out}$ - 4756).

## Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setpoint (618°F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures. Accounting for calibration and instrumentation errors, the safety analyses used a trip setpoint of 620°F.

## Reactor Building Pressure

The high reactor building pressure trip setpoint (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

## Shutdown Bypass

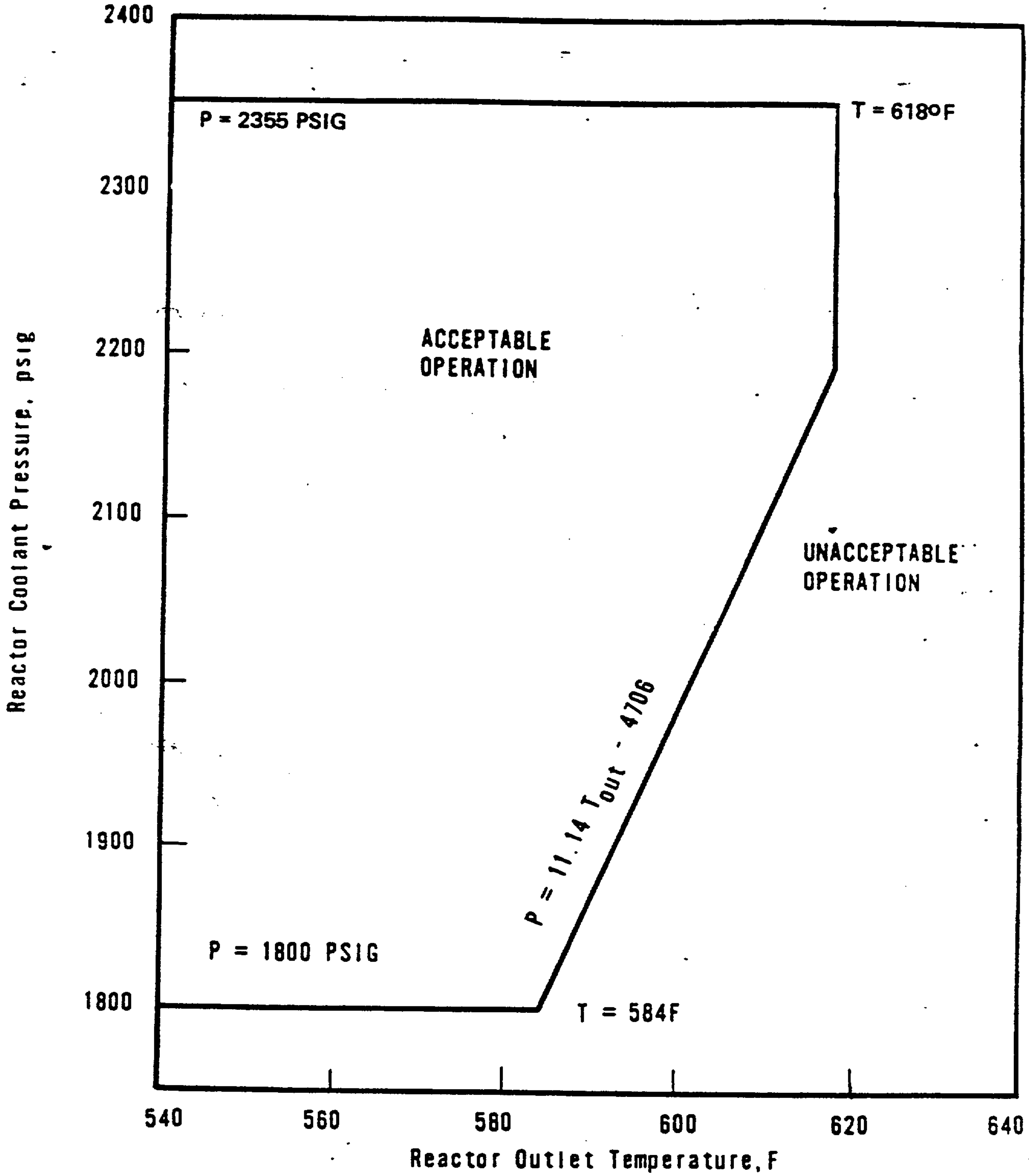
In order to startup the reactor and to be able to perform control rod drive tests and zero power physics tests (see Technical Specification 3.1.9), there is provision for bypassing certain segments of the reactor protective system (RPS). The RPS segments which can be bypassed are given in Table 2.3-1. Two conditions are imposed when the RPS is bypassed:

1. By administrative control the nuclear overpower trip setpoint is reduced to a value of ≤ 5.0% of rated power.

2. The high reactor coolant system pressure trip setpoint is automatically lowered to 1720 psig.

The high RCS pressure trip setpoint is reduced to prevent normal operation with part of the RPS bypassed. The reactor must be tripped before the bypass is initiated since the high pressure trip setpoint is lower than the normal low pressure trip setpoint (1800 psig).

The overpower trip setpoint of ≤ 5.0% prevents any significant reactor power from being produced when performing physics tests. If no reactor coolant pumps are operating, sufficient natural circulation would be available to remove 5.0% of rated power.[5]




PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS

UNITS 1, 2, AND 3

OCONEE NUCLEAR STATION

Figure 2.3-1

Amendment No. 164 (Unit 1)
Amendment No. 164 (Unit 2)
Amendment No. 161 (Unit 3)

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

| | RPS Trip | | RPS Trip Setpoint | Shutdown Bypass |
|---|---|---|---|---|
| 1. | Nuclear Overpower | | 105.5% Rated Power | 5.0% Rated Power[(1)] |
| 2. | Flux/Flow/Imbalance | | 1.07 | Bypassed |
| 3. | Pump Monitors | a. | > 0% Rated Power loss of two pumps in one reactor coolant loop | Bypassed |
| | | b. | > 55% Rated Power loss of two pumps | |
| | | c. | > 0% Rated Power loss of one or two pumps during two pump operation | |
| 4. | High Reactor Coolant System Pressure | | 2355 psig | 1720[(2)] |
| 5. | Low Reactor Coolant System Pressure | | 1800 psig | Bypassed |
| 6. | Variable Low Reactor Coolant System Pressure | | P (psig) = (11.14 $T_{out}$ - 4706)[(3)] | Bypassed |
| 7. | High Reactor Coolant Temperature | | 618°F | 618°F |
| 8. | High Reactor Building Pressure | | 4 psig | 4 psig |

(1) Administratively controlled reduction set only during reactor shutdown.

(2) Automatically set when other segments of the RPS are bypassed.

(3) $T_{out}$ is in degrees Fahrenheit (°F).



Amendment No. 164 (Unit 1)
Amendment No. 164 (Unit 2)
Amendment No. 161 (Unit 3)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 164TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 164TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 161TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270 AND 50-287

## 1. INTRODUCTION

By application dated August 13, 1986 (ref. 1) as supplemented on May 14, 1987 (Ref. 4), Duke Power Company (the licensee) requested amendments to Facility Operating License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. The proposed amendments would raise the reactor protection system (RPS) high reactor coolant system (RCS) pressure trip setpoint from 2,300 psig to 2,355 psig. These proposed revisions would improve the operational performance of the plant by reducing reactor trips. The May 14, 1987 letter provided supplemental information. It does not significantly alter the action noticed in the Federal Register on February 26, 1987, and does not affect the staff's proposed no significant hazards determination.

The licensee, in providing justification for the requested changes, referred to studies made in 1985 by the Babcock and Wilcox Owners Group (B&WOG). As part of the Transient Assessment Program the B&WOG had studies performed to improve operational safety and performance through a reduction in the frequency of reactor trips. This effort is described in the following report submitted for review to the NRC; "Justification for Raising Setpoint for Reactor Trip on Pressure," BAW-1890, September 1985 (Ref. 2). The staff accepted the above listed report by evaluation issued on April 22, 1986 (Ref. 3.)

## 2. EVALUATION

The staff was assisted in the evaluation of the Oconee Technical Specification changes by their consultants at EG&G. The EG&G evaluation included a review of the Duke Power Company request for changes (Ref. 1) and comparison with the B&WOG report (Ref. 2) which was approved by the staff (Ref. 3). We have enclosed our consultant's report. The staff agrees with the EG&G evaluation which concludes that the proposed changes meet NRC positions established in the review of B&W topical reports. Also, it was found that the accidents analyzed in the Final Safety Analyses Report (FSAR) for Oconee bound the proposed Technical Specification change. The staff, therefore, concludes that the RCS high pressure trip setpoint can be increased from 2,300 psig to 2,355 psig.

TECHNICAL SPECIFICATION CHANGES

The Technical Specification changes for Oconee Units 1, 2, and 3 are as follows:

1. TS 2.2 Bases, Page 2.2-1. Safety Limits - Reactor System Pressure. The value of the setting for the reactor high pressure trip is changed from 2,300 psig to 2,355 psig as found acceptable in Section 2 above.

2. TS 2.2 Bases, Page 2.3-3. Reactor Coolant System Pressure. The value of the setting for the reactor high pressure trip setpoint is changed from 2,300 psig to 2,355 psig as found acceptable in Section 2 above.

3. Table 2.3-1 of Reactor Protection System Trip Setting Limits, Page 2.3-7. The value for the high reactor coolant system pressure was changed from 2,300 psig to 2,355 and is acceptable as explained in Section 2 above.

4. Figure 2.3-1 - Protective System Maximum Allowable Setpoints. The reactor coolant pressure was raised from 2,300 psig to 2,355 psig. This is acceptable as explained in Section 2 above.

The staff with assistance from EG&G has reviewed the proposed changes to TS 2.2, Table 2.3-1, and Figure 2.3-1 for Oconee Units 1, 2, and 3. The proposed change would increase the setpoint for trip of the reactor on high pressure in the reactor coolant system from 2,300 psig to 2,355 psig. As discussed in Section 2, the proposed changes meet the NRC positions established in the review of the B&W topical report and, therefore, meet the applicable regulatory guidance and requirements and are, therefore, acceptable.

3. ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 5853) on February 26, 1987, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

References:

1. Letter from H. B. Tucker, Duke Power Company, to J. F. Stolz, NRC, dated August 13, 1986.

2. "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, September 1985.

3. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, Babcock and Wilcox Company, April 22, 1986.

4. Letter from H. B. Tucker, Duke Power Company, to Document Control Desk, NRC, dated May 14, 1987.

Principal Contributors: H. Pastis, PDII-3/DRPI/II
H. Balukjian

Dated: December 7, 1987

ENCLOSURE

EG&G Idaho Review
of Technical Specification Change Request for
High Pressure Reactor Trip
by Duke Power Company for Oconee Units 1, 2 and 3
Docket Numbers 50-269, -270, 287
Operating License Numbers DPR-38 and DPR-47

## 1.0 INTRODUCTION

In letter from H. B. Tucker, Duke Power Company (DPC), to H. R. Denton, Nuclear Regulatory Commission (NRC), dated August 13, 1986 (Reference 1), DPC proposed a license amendment to Facility Operating Licenses Nos. DPR-38 and DPR-47 for the Oconee Nuclear Station. The proposed amendment would raise the setpoint for trip (i.e., shutdown) of the reactor on high pressure in the reactor coolant system from 2300 psig to 2355 psig. In response to NRC requests, DPC provided additional supporting information by letter H. B. Tucker, DPC, to the NRC dated May 14, 1987 (Reference 2) and by a telephone discussion on October 5, 1987. The EG&G Idaho review of the proposed amendment and the supporting information is presented in the following report.

## 2.0 BACKGROUND

The Babcock and Wilcox (B&W) Nuclear Steam Supply System (NSSS) was designed with the capability to adjust to minor plant upsets and certain anticipated events such as feedwater transients, rapid load changes and turbine trips without a reactor trip. The system was designed to initiate a plant runback, upon detection of an upset or equipment malfunction, to a power level consistent with the plant condition and to limit the rise in Reactor Coolant System (RCS) pressure to less than the reactor trip setpoint by opening the pressurizer Power Operated Relief Valve (PORV). Subsequent to the TMI-2 accident, the NRC, by IE Bulletin 79-05B (Reference 3), required licensees for all B&W Pressurized Water Reactor

(PWR) facilities to make modifications to reduce the number of automatic actuation of the PORVs. The modifications proposed by B&W on behalf of the owners group and accepted by the NRC included (1) raising the PORV setpoint from 2255 psig to 2450 psig, (2) decreasing the reactor trip on high RCS pressure from 2355 psig to 2300 psig and (3) providing an Anticipatory Reactor Trip (ART) for turbine trips above 20% power. In addition, the NRC required that B&W demonstrate that these modifications: (1) limited the frequency of PORV openings to less than 5% of the total number of overpressure transients (NUREG 0737 Item II.K.3.7 Reference 4) and (2) limited the probability of a small-break Loss of Coolant Accident (LOCA) caused by a stuck-open PORV to less than .001 per reactor-year (NUREG 0737 Item II.K.3.2 Reference 4). B&W submitted a report (Reference 5) to demonstrate that the modifications did meet the requirements. The NRC issued a Safety Evaluation Report (Reference 6) which concluded that the requirements were met.

Although these modifications have met the objectives of reducing challenges to and opening of the PORV, they have increased the frequency of reactor trips. Each reactor trip results in a challenge to plant safety systems and any reduction in reactor trip frequency will contribute to overall plant safety as well as plant availability. On behalf of the B&W Owners Group, B&W submitted a report "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890 September 1985 (Reference 7). This report presented the B&W justification for raising the high pressure reactor trip setpoint from 2300 psig to 2355 psig. Raising the setpoint for the reactor trip on high pressure would allow successful runbacks with an increase in the arming threshold for anticipatory reactor trip on turbine trip. Eventhough the DPC proposed amendment only requested a change in the setpoint for reactor trip on high pressure, B&W has treated the raising of the arming threshold for anticipatory reactor trip on turbine trip as a closely related topic and has submitted a report "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985 (Reference 8) justifying an increase. The B&W report justifying the increase of the high pressure reactor trip setpoint, (Reference 7), included, in the discussion of the reduction of reactor trips that might be expected, those that would result from raising the arming threshold as well as those resulting from the increase in high pressure reactor trip setpoint. The report states with both changes the number of reactor trips per year would be reduced by approximately 10%. With only raising the high pressure reactor trip setpoint, the number of reactor trips per year would be expected to only be reduced by 5%. The report justifying the increase of the high pressure reactor trip setpoint (Reference 7) states that with the setpoint at 2355 psig, the NRC requirements for limiting the frequency of PORV openings and limiting the probability of a small-break LOCA due to a stuck open PORV would still be met. The report justifying raising the arming threshold for reactor trip on turbine trip (Reference 8) states that with the high pressure reactor trip setpoint at 2355 psig and the arming threshold for reactor trip on turbine trip set at 45% the NRC requirement for limiting the frequency of PORV openings would still be met.

An important parameter in limiting the RCS pressure rise and providing for a successful runback is the available steam bypass. The evaluations in the two B&W reports (References 7 and 8) considered the bypass that is available to be that provided by opening the turbine bypass valves and that provided by the lifting of the first bank of Main Steam Safety Valves (MSSVs). Table 5-3 of the B&W report justifying raising the arming threshold for the ART (Reference 8) lists the available bypass for the eight B&W plants. The Oconee units have the least available bypass of the eight plants with 39% available. The significance of the lower bypass needs to be considered in evaluating the proposed changes and may be of more significance if in the future DPC request raising the arming threshold for the ART.

In April 1986, the NRC staff completed its review of both B&W reports. In their Safety Evaluation Reports (SERs), the staff: (1) reviewed the basis for the proposed changes; (2) reviewed B&W's method of analysis of the effect of the proposed high pressure trip setpoint on PORV openings; (3) compared the results of Monte Carlo simulation for PORV openings with the NRC requirements contained in NUREG-0737 (Reference 4); and (4) reviewed the results of B&W's analysis of the arming threshold for ART. The NRC requirements include: (1) the PORV will open in less than 5% of all anticipated overpressure transients

(NUREG 0737, Item II.K.3.7 Reference 4); and (2) the probability of a small-break loss of coolant accident caused by a stuck-open PORV will be less than 0.001 per reactor-year (NUREG 0737, Item II.K.3.2 Reference 4). In the SERs (References 9 and 10), the staff concluded on a generic basis that the proposed changes met the NRC requirements, and should benefit plants by potentially reducing the reactor trip frequency. Accordingly, the NRC concluded that the B&W reports could be referenced in licensing submittals by the B&W Owners Group members.

## 3.0 EVALUATION

The licensee is a member of the B&W Owners Group and their August 13, 1986 proposal (Reference 1) included the B&W Report BAW-1890 (Reference 7) as an attachment for justification for raising the high pressure reactor trip setpoint from 2300 psig to 2355 psig. EG&G Idaho concluded that, in general, the report is applicable to the Oconee Units 1, 2, and 3 because the units are B&W 177FH plant types for which the report applies and the historical data used in the report was taken from the operation records of the Oconee units and other similar B&W plants.

During the telephone discussion of October 5, 1987, the NRC inquired if the Power Train Analysis in the B&W report (Reference 7) is valid for the Oconee units because the steam bypass flow, including the first bank of main steam safety valves, used in the analysis was 43% while the comparable available bypass flow for the Oconee units is 39%. The licensee reported that the purpose of the analysis was to make a comparison to demonstrate that the pressure overshoot above the high pressure reactor trip setpoint does not increase if the setpoint is raised from 2300 psig to 2355 psig. The Licensee contended that the absolute value of the results are not important for the conclusions and, therefore, the analysis using the slightly higher bypass flow is applicable to the Oconee units.

EG&G Idaho concluded that because DPC has only proposed to raise the high pressure reactor trip setpoint and has not proposed to raise the arming threshold for the ART for turbine trip, the Power Train results are only used for comparison and the absolute values are not important in developing the conclusions. The conclusion that the pressure overshoot will not increase if the high pressure reactor trip setpoint is raised from 2300 psig to 2355 psig is, therefore, valid for the Oconee units.

The acceptability of the analysis with the small difference in available steam bypass is further justified in that the analysis that evaluated the probability of the PORV opening used the pressure overshoot frequency distribution of the historical data with the 2300 psig high pressure reactor trip setpoint without including the approximately 10 psi reduction in overshoot predicted by the analysis for the 2355 psig setpoint.

The important conclusion of the B&W report (Reference 7) which justifies raising the high pressure trip is that even though the change will result in a small increase in the probability of opening the PORV, the increase is insignificant compared to the total openings of the PORV from all events. The estimate made in the B&W report (Reference 7) for PORV openings from high pressure events with the reactor high pressure trip set at 2355 psig is $1.8 \times 10^{-5}$ PORV openings per reactor year. The estimate made in the B&W report (Reference 7) for PORV openings from all events is $8.06 \times 10^{-2}$ PORV openings per reactor year with the major contributor to the PORV openings identified as actions by the operators in carrying out plant operation in accordance with the Abnormal Transient Operating Guidelines (ATOG). The $1.86 \times 10^{-5}$ PORV openings per reactor year from high pressure events applies to the Oconee as discussed in the preceeding paragraph. The $8.06 \times 10^{-2}$ PORV openings per reactor year from all events applies to the Oconee units because the Licensee confirmed, during the telephone discussion of October 5, 1987, that the abnormal transient procedures for the Oconee units are based on the ATOG. Therefore, for the Oconee units, the PORV openings from high pressure events with the proposed setpoints are insignificant compared with the openings from all events.

DPC states (Reference 1) that the proposed changes are within the bounds of current Final Safety Analysis Report (FSAR) because the original high pressure reactor trip setpoint was 2355 psig and this value was used in the FSAR. Based on the information provided by the licensee, EG&G Idaho concurs that the proposed change is within the bounds of the current FSAR.

The licensee in their letter of May 14, 1987 (Reference 2) confirmed that with the proposed setpoint change more MSSVs may open during an overpressure transient that causes a reactor trip on high pressure compared to those that open with the current setting. However, they contend that with the proposed changes there would be fewer such transients and the net number of openings would be expected to be reduced. Also, the licensee reports (Reference 2) that the MSSVs have a good record of performance. In over twelve years of operation of the Oconee units there have been approximately 1500 liftings of the MSSVs without a single failure to reseat. The lifting of the MSSVs, therefore, presents no significant concern. Based on the information provided by the licensee, EG&G Idaho concurs that the anticipated openings of the MSSVs are acceptable.

## 4.0 CONCLUSIONS

EG&G Idaho has reviewed the proposed changes to TS 2.2 and 2.3 for Oconee Units 1, 2, and 3. The proposed changes would increase the setpoint for trip of the reactor on high pressure in the reactor coolant system from 2300 psig to 2355 psig. As discussed in the preceding section, EG&G Idaho finds the proposed changes meet the NRC positions established in their review of the B&W topical reports and are within the bounds of the current FSAR. They therefore meet the applicable regulator guidance and requirements and are, therefore, acceptable.

## 5.0 REFERENCES

1. Letter from H. B. Tucker, Duke Power Company, to H. R. Denton, Nuclear Regulatory Commission, Oconee Nuclear Station Docket Nos. 50-269, -270, -289, dated August 13, 1986.

2. Letter from H. B. Tucker, Duke Power Company to the Nuclear Regulatory Commission, Oconee Nuclear Station, Docket Nos. 50-269, -270, -289, High Pressure Reactor Trip Setpoint, dated May 14, 1987.

3. IE Bulletin 79-05B, "Nuclear-Incident at Three Mile Island - Supplement". April 2, 1979.

4. NUREG-0737, "Clarification of TMI Action Plan Requirements". November 1980.

5. Babcock & Wilcox Report 12-1122779 Rev. 1, "Report on PORV Opening Probability and Justification of Present Systems and Setpoints". January 1981.

6. NRC Memorandum from F. H. Rowsome to G. C. Laines, "Safety Evaluation of the B&W Licenses' Responses to TMI ACtion II.K.3.2". August 24, 1983.

7. BAW-1890, "Justification for Raising Setpoint for Reactor Trip on High Pressure." September 1985.

8. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip". October 1985.

9. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, B&W, Acceptance for Referencing of Licensing Topical Report BAW-1890, "Justification for Raising Setpoint for Reactor Trip on High Pressure". April 27, 1986.

10. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, B&W, Acceptance for Referencing of Licensing Topical Report BAW-1893 "Basis for Raising Threshold for Anticipatory Reactor Trip on Turbine Trip". April 25, 1986.

50-269

DATED: December 7, 1987

AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE DPR-38 - Oconee Nuclear Station, Unit 1
AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE DPR-47 - Oconee Nuclear Station, Unit 2
AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE DPR-55 - Oconee Nuclear Station, Unit 3

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