Template NRR-058

January 18, 2000

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

DISTRIBUTION:
File Center
PUBLIC
PDII-1 R/F
WBeckner,TSB
HBerkow
OGC
ACRS T-6 E26
GHill(6)
COgle,RII

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA5153, MA5154, AND MA5155)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 309 , 309 , and 309 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications in response to your application dated April 5, 1999, as supplemented May 27, July 6, October 7, and November 22, 1999.

The amendments revise the Technical Specifications to incorporate Topical Report DPC-NE-3005-P, "Thermal Hydraulic Transient Analysis Methodology" and requirements for the Atmospheric Dump Valves.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 309 to DPR-38
2. Amendment No. 309 to DPR-47
3. Amendment No. 309 to DPR-55
4. Safety Evaluation

NRC FILE CENTER COPY

cc w/encls: See next page

DOCUMENT NAME: G:\PDII-1\OCONEE\A5153 AME.WPD
To receive a copy of this document, indicate in the box C=Copy w/o attachment/enclosure E=Copy with attachment/enclosure N = No copy * See previous concurrence

OFFICE	PDII-1/PM	PDII-1/LA	RTSB*	OGC*	PDII-1/SC*
NAME	DLaBarge:cn	CHawes CMH	WBeckner		REmch
DATE	1 /18/2000	1 /18/2000	12 / 14 /99	01 /13/2000	01 / 14 /2000

OFFICIAL RECORD COPY

DF01

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

January 18, 2000

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA5153, MA5154, AND MA5155)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 309, 309, and 309 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications in response to your application dated April 5, 1999, as supplemented May 27, July 6, October 7, and November 22, 1999.

The amendments revise the Technical Specifications to incorporate Topical Report DPC-NE-3005-P, "Thermal Hydraulic Transient Analysis Methodology" and requirements for the Atmospheric Dump Valves.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 309 to DPR-38
2. Amendment No. 309 to DPR-47
3. Amendment No. 309 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

Oconee Nuclear Station

cc:

Ms. Lisa F. Vaughn
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne W. Cottingham, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

Mr. Rick N. Edwards
Framatome Technologies
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852-1631

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory
Commission
7812B Rochester Highway
Seneca, South Carolina 29672

Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. L. E. Nicholson
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
7800 Rochester Highway
Seneca, South Carolina 29672

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

L. A. Keller
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. Steven P. Shaver
Senior Sales Engineer
Westinghouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 309
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 Energy Corporation (the licensee) dated April 5, 1999; supplemented May 27, July 6, October 7, and November 22, 1999, comply 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 Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309 , 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 18, 2000

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.309
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 Energy Corporation (the licensee) dated April 5, 1999; supplemented May 27, July 6, October 7, and November 22, 1999, comply 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 Paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309 , 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 18, 2000

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 309
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 Energy Corporation (the licensee) dated April 5, 1999; supplemented May 27, July 6, October 7, and November 22, 1999, comply 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 Paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309 , 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 18, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 309

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 309

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 309

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove	Insert	Remove	Insert
TS TOC iii	TS TOC iii	B 3.4.1-2	B 3.4.1-2
TS TOC iv	TS TOC iv	B 3.4.1-4	B 3.4.1-4
3.4.1-2	3.4.1-2	B 3.4.10-1	B 3.4.10-1
3.4.10-1	3.4.10-1	B 3.4.10-2	B 3.4.10-2
3.4.10-2	3.4.10-2	B 3.4.10-3	B 3.4.10-3
3.7.4-1	3.7.4-1	B 3.4.10-4	B 3.4.10-4
----	3.9.7-1	B 3.7.2-1	B 3.7.2-1
----	3.9.7-2	B 3.7.4-1	B 3.7.4-1
5.0-29	5.0-29	----	B 3.7.4-2
5.0-30	5.0-30	----	B 3.7.4-3
5.0-31	5.0-31	----	B 3.9.7-1
B TOC iii	B TOC iii	----	B 3.9.7-2
B TOC iv	B TOV iv	----	B 3.9.7-3
B 3.1.4-6	B 3.1.4-6		
B 3.3.1-13	B 3.3.1-13		
B 3.3.1-14	B 3.3.1-14		
B 3.3.15-1	B 3.3.15-1		

TABLE OF CONTENTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- With three RCPs operating, the limits are applied to the loop with the highest pressure. -------------------- Verify RCS loop pressure is within limits specified in the COLR.	12 hours
SR 3.4.1.2	----------NOTE---------- With three RCPs operating, the limits are applied to the loop with the lowest loop average temperature for the condition where there is a 0°F ΔTc setpoint. -------------------- Verify RCS loop average temperature is within limits specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow is within limits specified in the COLR.	12 hours
SR 3.4.1.4	----------NOTE---------- Not required to be performed until 7 days after stable thermal conditions are established in the higher power range of MODE 1. -------------------- Verify by measurement RCS total flow rate is within limit specified in the COLR.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2425 psig and ≤ 2575 psig.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 325°F.

-----NOTE-----
The lift settings are not required to be within the LCO limits for entry into the applicable portions of MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into the applicable portions of MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 3 with any RCS cold leg temperature ≤ 325°F.	12 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

LCO 3.7.4 The ADV flow path for each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3, and MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both ADV flow path(s) inoperable.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 4 without reliance upon steam generator for heat removal.	12 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Cycle the valves that comprise the ADV flow paths.	18 months

3.9 REFUELING OPERATIONS

3.9.7 Unborated Water Source Isolation Valves

LCO 3.9.7 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE----------- Required Action A.3 must be completed whenever Condition A is entered. ------------------------------ One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Initiate actions to secure valve in closed position.	Immediately
	<u>AND</u>	
	A.3 Perform SR 3.9.1.1.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify each valve that isolates unborated water sources is secured in the closed position.	31 days

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) DPC-NE-1002A, Reload Design Methodology II, Rev. 1, (SER dated October 1, 1985);

(2) NFS-1001A, Reload Design Methodology, Rev. 4, (SER dated July 29, 1981);

(3) DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, (SER dated July 19, 1989);

(4) DPC-NE-1004P-A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, (SER dated November 23, 1992);

(5) DPC-NE-2008P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3, (SER dated April 3, 1995);

(6) BAW-10192-PA, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, (SER dated February 18, 1997);

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, Rev. 1, (SER dated November 7, 1996); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, Rev. 1, (SER dated May 25, 1999).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6 Reporting Requirements (continued)

5.6.8 Steam Generator Tube Inspection Report

The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

   3. Identification of tubes plugged or repaired.

   4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC shall be reported prior to resumption of plant operation. The written report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

TABLE OF CONTENTS

BASES

ACTIONS (continued)

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.2% Δk/k at RPT, 0.4% Δk/k at 80% RPT, or 0.8% Δk/k at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. RCS HIGH PRESSURE (continued)

The RCS High Pressure trip has been credited in the transient analysis calculations for slow positive reactivity insertion transients (rod withdrawal transients and moderator dilution). The rod withdrawal transient covers a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection in the main steam line break analysis. The setpoint allowable value does not include errors induced by the harsh environment, because the trip actuates prior to the harsh environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a turbine or reactor trip signal.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped control rod events and control rod withdrawal events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Position Limits," LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.3, "QUADRANT POWER TILT (QPT)."

The normal operating band for RCS pressure is between 2125 psig and 2155 psig as measured at the hot leg pressure tap. The safety analyses assume a core exit pressure that is based on the measured pressure and concurrent pressure losses between the two locations. The pressure losses are a function of the loop flow rate, thus different values are allowed for 4 or 3 RCP operation.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump and four pump operation. These limits are specified in the COLR. The flow limits for three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops–MODES 1 and 2").

Another set of limits on DNBR related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS loop average temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔTc setpoint

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

restored to a normal operation, steady state condition following load changes and other expected transient operations. The RCS pressure value specified in the COLR is dependent on the number of pumps in operation and has been adjusted to account for the pressure loss difference between the core exit and the measurement location. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions. A Note has been added to indicate the pressure limits for three pumps operating is applied to the loop with the highest pressure.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for loop average temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions. A Note has been added to indicate the temperature limits for three pumps operating are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔTc setpoint.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODE 3, MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 1). The setpoint of the pressurizer code safety valves is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The required lift pressure is 2500 psig ± 3%. The upper and lower pressure limits are based on the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessels which they protect to 10% above the design pressure. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 3%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The valves will be tested per ASME Section XI requirements and returned to service with as-left setpoints of 2500 psig ± 1%. The upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessel which they protect, to 10% above the design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and portions of MODE 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. Portions of MODE 3 are conservatively included, although the listed accidents may not require both safety valves for protection.

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4 and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 3 with any RCS cold leg temperature ≤ 325°F within 18 hours. The 12 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the 18 hours allowed is reasonable, based on operating experience, to reach MODE 3 with any RCS cold leg temperature ≤ 325°F without challenging unit systems. With any RCS cold leg temperature at or below 325°F, overpressure protection is provided by LTOP. Reducing the RCS temperature to ≤ 325°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves setpoint is ± 3% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. These values include instrument uncertainties.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.2 Turbine Stop Valves (TSVs)

BASES

BACKGROUND	The TSVs partially isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). TSV closure partially terminates flow from the unaffected (intact) steam generator. Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators. TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds. A discussion of the TSV's function is found in the UFSAR, Section 10.3 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the TSVs is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 2). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips. Another failure considered is the loss of one switchgear. The accident analysis compares several different MSLB events. The main SLB outside containment upstream of the TSV is limiting for offsite dose. The MSLB with ICS low level control and no operator action prior to ten minutes is the limiting case for a post-trip return to power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

BASES

BACKGROUND

The ADV flow paths provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the UFSAR (Ref. 2). This is done in conjunction with the secondary cooling water from the Emergency Feedwater (EFW) System.

The steam generator tube rupture (SGTR) analysis (Ref. 3) credits operator action to depressurize the steam generators by opening each of the ADV flow paths.

For each steam generator, the ADV flow path is comprised of the atmospheric dump block valve bypass (1" bypass), the atmospheric vent valve (a 12" block valve), the atmospheric dump control valve (i.e., throttle valve), and the atmospheric vent block valve (i.e., isolation valve). The throttle valve and the isolation valve are in parallel and are located downstream of the atmospheric vent valve.

The atmospheric vent valve should be opened prior to opening the throttle valve or isolation valve. This is accomplished by first opening the atmospheric dump block valve bypass.

This equalizes the differential pressure across the atmospheric vent valve. Once the atmospheric vent valve is opened, the cool down rate is controlled using the throttle valve. If additional relief capacity is needed, the isolation valve can be opened. The capacity of the throttle or isolation valve exceeds decay heat loads and is sufficient to cool down the plant.

BASES

APPLICABLE SAFETY ANALYSIS

The SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

The ADV flow paths satisfy Criterion 3 of 10 CFR 50.36 (Ref.1).

LCO

The ADV flow path for each steam generator is required to be OPERABLE. The failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

APPLICABILITY

The ADV flow path for each steam generator is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 4, steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.5, "RCS Loops - MODE 4" or available to transfer decay heat to satisfy LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." The steam generators do not contain a significant amount of energy in MODE 4 when the unit is not relying upon a steam generator for heat transfer, and MODES 5 and 6; therefore, the ADV flow paths are not required to be OPERABLE in these MODES and condition.

BASES

ACTIONS

A.1 and A.2

With one or both of the ADV flow path(s) inoperable, the Unit must be placed in a condition in which the LCO does not apply. To achieve this status, the Unit must be paced in at least MODE 3 within 12 hours, and at least MODE 4 without reliance on a steam generator for heat removal within 24 hours. The Completion Times are reasonable, based on operating experience, to reach the required Unit conditions from full power conditions in an orderly manner and without challenging Unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cool down of the RCS, the valves that comprise the ADV flow path for each steam generator must be able to perform the following functions:

a) the atmospheric dump block valve bypass and the atmospheric vent valve must be capable of being opened and closed; and

b) the atmospheric dump control valve and atmospheric vent block valve must be capable of being opened and throttled through their full range.

This SR ensures that the valves that comprise the ADV flow path for each steam generator are cycled through the full control range at least once per 18 months. Performance of inservice testing or use of an ADV flow path during a unit cool down satisfies this requirement. This surveillance does not require the valves to be tested at pressure. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.36.

2. UFSAR, Section 10.3.

3. UFSAR, Section 15.9.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Unborated Water Source Isolation Valves

BASES

BACKGROUND

During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Coolant Storage System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources when in MODE 6, a boron dilution event as analyzed in the UFSAR is prevented.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

BASES

APPLICABILITY (continued)

For all other applicable MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated. The boron dilution event is applicable in MODES 1 and 6.

ACTIONS

The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the fuel transfer canal and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.1.
2. 10 CFR 50.36.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 309 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 309 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 309 TO FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated April 5, 1999, as supplemented May 27, July 6, October 7, and November 22, 1999, Duke Energy Corporation (Duke or the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TSs). The requested changes would incorporate design-bases assumptions, limitations, and results of the methodology described in Topical Report DPC-NE-3005-P, "Thermal Hydraulic Transient Analysis Methodology." The staff approved the use of the new Chapter 15 transient analysis methodology described in the topical report by letter dated May 25, 1999. Since the analysis had not yet been completed for Units 1 and 3, the TS change would reflect that the methodology would apply only to a unit when the analysis was completed for that unit(s).

The submittal also contained the corresponding changes to the Oconee Updated Final Safety Analysis Report (UFSAR) and the Core Operating Limit Report (COLR). The UFSAR and the COLR presented background information and documented the new thermal-hydraulic methodology. The initial proposed no significant hazards consideration that was included in the April 5, 1999, letter was published in the *Federal Register* on June 30, 1999.

A supplement dated July 6, 1999, provided clarifying information that did not change the scope of the April 5, 1999, application and the initial proposed no significant hazards consideration determination.

The submittal dated May 27, 1999, supplied dose analysis information and a revision to the original no significant hazards consideration. The submittal dated October 7, 1999, responded to staff questions and reported that the analysis had been completed for all three units. As a result, the submittal was a major revision to the original application. It extended the proposed amendment provisions to all three units and revised the proposed no significant hazards consideration determination. The revised proposed no significant hazards consideration was published in the *Federal Register* on November 3, 1999.

By letter dated November 22, 1999, the licensee submitted the steam generator post-accident steaming rates used in the dose analysis calculation related to the proposed amendment and corrected radiological information that had been supplied in the original amendment application. This submittal did not change the November 3, 1999, proposed no significant hazards consideration determination.

2.0 STAFF EVALUATION

The proposed changes to the TS Limiting Conditions for Operation (LCO) 3.4.10, 3.7.4, and 3.9.7, TS Section 5.6.5, surveillance requirement (SR) 3.4.1, and corresponding Bases sections are discussed below.

2.1 Changes to SR 3.4.1

2.1.1 Licensee's Proposal and Justification

LCO 3.4.1 specifies the initial conditions used in the departure from nucleate boiling transient (DNB) analyses. The safety analysis includes operation of three or four reactor coolant pumps (RCPs), as well as the pressure, temperature and flow rates assumed in the analyses that are specified in the COLR. SR 3.4.1 ensures that the DNB parameters are monitored and remain within the limits specified in the COLR for three or four RCP operation.

In the current TS, a note in SR 3.4.1 states, "With three RCPs operating, the limits are applied to the loop with the lowest loop averaged temperature." Duke proposed to add "for the condition where there is a 0°F ΔTc setpoint," to the end of the SR 3.4.1.2 note. The actual ΔTc temperature is the difference between the reactor vessel loop A and loop B inlet temperature. The operator supplies the ΔTc setpoint and dials it into the integrated control system (ICS), which is then compared with the actual ΔTc input. The ΔTc control is designed to equalize the loop cold-leg temperatures to prevent unequal radial flux distribution.

If four RCPs are operating, both loops would have to meet the temperature limit given in the COLR. When three RCPs are operating, under the current SR 3.4.1.2 requirements, only the loop with the lower loop averaged temperature would have to meet the Reactor Coolant System (RCS) temperature limits. However, according to the licensee's analysis, if three RCPs are operating, the loop with the lowest loop averaged temperature is required to meet the loop averaged temperature limit only if the 0°F ΔTc setpoint is dialed into the ICS. If three RCPs are operating and the setpoint is not at 0° ΔTc, or if four RCPs are operating regardless of the ΔTc setpoint, then both RCS loops must be within the loop Tave limits specified in the COLR. Duke determined that the proposed note is more restrictive than the current requirement.

2.1.2 Staff Evaluation

The proposed change to SR 3.4.1.2 limits the applicability of the SR 3.4.1.2 note. With three RCPs operating, the licensee would have to ensure that both loops meet the temperature limits specified in the COLR, except when the 0°F ΔTc setpoint is dialed into the ICS. Therefore, the note in SR 3.4.1.2 is applicable for a special case only. In general, for both loops, the loop averaged temperature would have to be compared to the temperature limit specified in the COLR whether three or four RCPs are operating. The staff agrees with the licensee that the proposed changes are more restrictive and finds the proposed change to SR 3.4.1.2 acceptable.

2.2 Changes to LCO 3.4.10, "Pressurizer Safety Valves"

2.2.1 Licensee's Proposal and Justification:

A. Pressurizer Safety Valves Lift Setting

The TS LCO 3.4.10 currently states, "Two pressurizer safety valves (PSVs) shall be OPERABLE with lift setting of ≥2475 psig and ≤2525 psig." The licensee has proposed to change the lift setpoints to 2425 psig and 2575 psig, respectively. The difference in the lift setpoints results from the proposed setpoint tolerance increase from ± 1 percent to ± 3 percent of the nominal lift pressure of 2500 psig.

In the submittal, the licensee stated that the upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in system pressure to 10 percent above the design pressure. The licensee has performed transient overpressure analyses to support the proposed increase in the PSV setpoint tolerance. To address the effect of possible inaccuracy in setpoint testing, the licensee stated that the current setpoint testing instrument uncertainty is approximately 3 psi, which will be accounted for in the revised testing procedures.

The licensee also evaluated the effect of the increased PSV setpoint tolerance on the performance of safety-related valves. The licensee determined that the increased TS PSV setpoints will not preclude the ability of the applicable valves to perform their function in the event of an accident or transient in which the PSV setpoints are challenged.

Duke pointed out that the higher allowable pressure for the normal lift setting is consistent with the licensing basis analyses documented in DPC-NE-3005-PA. The licensee stated that increasing the pressurizer lift setpoint to 3 percent affects the reactor coolant peak pressure during pressurizing transients. For Oconee, the start-up accident and rod ejection transients are the two most limiting overpressure transients. For these two limiting pressurization transient analyses, the pressurizer safety valves were modeled to actuate at +3 percent above the nominal lift pressure. The licensee stated that the resulting peak pressure remained within the design pressure limit of 2750 psig.

In justifying the lowered trip setpoint pressure for the PSVs from 2475 to 2425 psig, the licensee stated that the nominal high-pressure reactor trip setpoint is 2355 psig. Therefore, with the -3 percent allowance, the PSV setpoint would remain above the nominal reactor high pressure trip setpoint and the PSVs would not actuate before the high pressure scram had occurred. The licensee also stated that, "Although there is an appearance of an increase in the probability of a PSV to open, based on analysis results it is expected that the PSVs will be challenged to the same extent regardless of whether the setpoint drift is -1 percent (2475 psig) or -3% (2425 psig)." Duke also stated that, "increasing the PSV drift to ±3% will not delay reactor trip, nor will it in effect increase the probability of opening a PSV."

For the rod ejection transient and the uncontrolled rod withdrawal from power transient, reactor scram does not prevent the PSVs from lifting. Duke evaluated the impact of the lowered lift-off pressure on these two DNB transients and concluded that the analyses showed that for both events the DNB limit is not violated.

B. SR 3.4.10.1

The licensee reported that, even though the PSV setpoint tolerance requirement would be changed to ±3 percent for the purpose of demonstrating operability, SR 3.4.10.1 will still require the PSV lift settings to be within ± 1 percent of the nominal value following testing. Duke concluded that, even though the proposed change will increase the allowable PSV setpoint drift, the transient analyses take this into account. In addition, the higher tolerance value will provide relief from unnecessary reporting requirements.

2.2.2 Staff Evaluation

The licensee evaluated the impact of the increased PSV lift tolerance (from ± 1 percent to ± 3 percent), and the results of the reactor transient analyses remained within the applicable acceptance criteria. Duke also determined that the likelihood of the PSV opening would not increase significantly as a result of the lower tolerance of -3 percent. Since the licensee performed the relevant transient analyses using the ± 3 percent tolerance range for the PSV lift setpoint, the staff finds the proposed change will still ensure that the PSVs will perform their intended function. The licensee is also proposing to reset the PSVs to within ±1 percent of nominal setpoints after testing. This will reduce the possibility of setpoint drift outside the allowable tolerance. The staff finds the TS changes acceptable.

2.3 <u>LCO 3.7.4 "Atmospheric Dump Valves (ADVs) Flow Paths" and SR 3.7.4.1</u>

2.3.1 Licensee's Proposal and Justification:

The licensee proposed to add a new LCO 3.7.4 to the current Oconee TSs. The new LCO would require the atmospheric dump valve (ADV) flow path to be operable for each steam generator during MODES 1, 2, 3, and during Mode 4 when the steam generator is relied upon for heat removal. There is one ADV flow path associated with each steam generator. The ACTION statement would require that if one or more ADV flow paths are inoperable: (1) the reactor must be in MODE 3 in 12 hours, and (2) the reactor must be in MODE 4 in 24 hours without the steam generator providing heat removal. In addition, the corresponding surveillance SR 3.7.4.1 would require cycling "the valves that comprise the ADV flow paths" every 18 months.

The licensee stated that, in the current thermal-hydraulic methodology described in topical report DPC-NE-3005-PA, both ADV flow paths are credited in the steam generator tube rupture (SGTR) analysis. In the SGTR analysis, the ADVs are credited to depressurize the reactor and control water level in the ruptured steam generator within 40 minutes after the event. According to the licensee, the turbine bypass valves (TBVs) are not credited in the new SGTR analysis methodology.

All of the valves in the ADV flow paths are positioned locally using a chain-operated device. The licensee credits operator action in the steam generator tube rupture analysis to depressurize the steam generators by opening each of the ADV flow paths. For each steam generator, the ADV flow path is comprised of the atmospheric dump valve bypass (1" bypass), the atmospheric vent valve (a 12" block valve), the atmospheric dump control valve (i.e., throttle valve), and the atmospheric vent block valve (i.e., isolation valve). The throttle valve and isolation valve are in parallel and are located downstream of the atmospheric vent valve. The atmospheric vent valve is opened first and the throttle valve is then used to control the plant cool down rate. If additional relief capacity is needed, the isolation valve can also be opened. The bypass and block valves and their associated piping form the main steam line pressure boundary, and they are designated as seismically qualified QA-1 Category. However, the throttle and isolation valves and their associated piping in the ADV flow paths are not seismically qualified and are not part of the safety system category. In a December 16, 1999, submittal, Duke committed to include the remainder of the ADV flow path in their QA-5 program. According to the licensee, this quality assurance program will apply testing and maintenance quality controls consistent with the testing and maintenance requirements for QA-1 safety systems and components. Duke pointed out that the ADVs are located on the turbine deck just outside the control room. The ADVs are easily accessible and clearly visible and, therefore, can be manually operated to function if needed.

2.3.2 Staff Evaluation

Since the components in the ADV flow paths will be required to perform safety functions, the proposed ADV TS LCO 3.7.4 will enhance and contribute to the reliability of the systems, structures, and components (SSC) in the ADV flow path. In topical report DPC-NE-3005-PA, the staff reviewed the SGTR analysis and approved the thermal-hydraulic methodology that credited the ADVs for plant depressurization during the SGTR event.

Parts of the ADV flow path are seismically qualified and the licensee is committed to including the components in the ADV flow path to a quality assurance program level developed for the testing and maintenance of SSCs that are required to perform safety functions. If the QA-1 program (QA-5 for Oconee) is properly executed or implemented, the improved monitoring, testing, and maintenance will increase the reliability of the ADVs. In the SGTR analysis, it is conservative to credit the ADVs for depressurization when performing dose analysis. Therefore, the staff finds the proposal to add LCO 3.7.4 to the Oconee TSs acceptable. Duke confirmed that the use of the ADVs is conservative and ADVs will be credited only for the SGTR analysis.

The proposed SR would cycle the valves in the ADV flow paths every 18 months. The staff finds that the licensee's proposed TS 3.7.4 for the ADV flow paths is generally consistent with the Standard Technical Specifications (STS) outlined in NUREG-1430. The proposed change would allow 6 more hours than the 6 hours provided in the STS to be in Mode 3 and allows 6 more hours than the 18 hours provided in the STS to be in Mode 4 without reliance upon the steam generator for heat removal. However, the STS allows up to 7 days to restore a single inoperable valve, while the licensee's proposed ACTION allows no additional time to restore inoperable valves. The staff finds the licensee's proposed TS 3.7.4 for the ADV flow paths to be adequate and, therefore, acceptable.

2.4 LCO 3.9.7 "Unborated Water Source Isolation Valves"

2.4.1 Licensee's Proposal and Justification

Duke has proposed to add LCO 3.9.7 to the current Oconee TSs. The new TS LCO would require that each valve used to isolate an unborated water source be secured in a closed position. The LCO is applicable during MODE 6 and separate condition entry is allowed for each unborated water source isolation valve. The ACTION statements are shown in the table below.

Condition	Required ACTION	Completion Time
A.______NOTE__________ Required Action A.3 must be completed whenever Condition A is entered. ____________________ One or more valves not secured in closed position	A.1 Suspend CORE ALTERATIONS. <u>AND</u> A.2 Initiate actions to secure valve in closed position <u>AND</u> A.3 Perform SR 3.9.1.1	Immediately Immediately 4 hours

The corresponding surveillance, SR 3.9.7.1, requires that each valve that isolates an unborated water source be verified closed every 31 days. The licensee stated that the LCO prevents an unplanned boron dilution during MODE 6, and the SR ensures that the subject valves are in the closed position.

2.4.2 Staff Evaluation

The proposed LCO will formalize the administrative procedure that currently ensures that unborated water sources are isolated from the reactor cavity during MODE 6. The proposed change increases the safety of the refueling process because it reduces the likelihood that an inadvertent boron dilution event will occur during MODE 6. In addition, the proposed LCO 3.9.7 is consistent with the Standard Technical Specification (STS) LCO 3.9.2 for Westinghouse reactors. Therefore, the staff finds the proposed LCO 3.9.7 acceptable since it will enhance reactivity management during refueling.

2.5 TS Section 5.6 "Reporting Requirements"

2.5.1 Licensee's Proposal and Justification

The licensee proposed to revise TS Section 5.6.5, " Core Operating Limits Report (COLR)." TS Section 5.6.5 lists the COLR reference documents that specify the analytical methods used to determine the COLR parameters. Duke proposed to: (1) update the revisions of some of the listed reference documents, (2) replace incorrectly referenced documents with the applicable references, and (3) add the date of the approving safety evaluation reports (SERs) in the referenced documents.

2.5.2 Staff Evaluation

Duke has reviewed and revised the COLR reference list by deleting erroneous references and substituting more relevant references for them. The staff finds the proposed changes in TS Section 5.6.6 acceptable because the changes are administrative, the COLR references have been previously approved by the staff and, for the new references, the licensee addressed NRC limitations and restrictions for the approved methodologies.

2.5.3 Summary

Duke has revised the Chapter 15 transient analysis methodology in the UFSAR for Oconee and used the NRC-approved new methodology in Duke's Topical Report DPC-NE-3005-PA. The license amendment request proposes to incorporate the assumptions and results from the new thermal-hydraulic transient analyses into the Oconee TSs. The staff has reviewed the licensee's submittal and found the proposed TS changes acceptable.

Duke also amended the TS Bases for Oconee Units 1, 2, and 3. The staff finds the proposed Bases changes acceptable, since the new or revised Bases discuss the reasoning behind the TS requirements proposed in this amendment request.

2.6 Radiological Evaluation

By letter dated May 27, 1999, the licensee submitted the radiological consequences analyses done in support of the April 5, 1999, Oconee Unit 2 Cycle 18 reload analysis submittal. By letter dated November 22, 1999, the licensee submitted corrections and additional information to support NRC staff review of the analyses. The information was also provided to support the use of the methodology provided in Topical Report DPC-NE-3005-PA for operation of Oconee Units 1, 2, and 3.

2.6.1 Evaluation

The staff reviewed the licensee's revised radiological consequences analyses for the design basis accidents that are affected by the change in fuel. The licensee revised the radiological consequences analyses for the following accidents:

a. Locked rotor accident
b. Steam generator tube rupture accident (SGTR)
c. Rod ejection accident (REA)
d. Main steam line break (MSLB)
e. Small steam line break

a. Locked Rotor Accident

The licensee determined that no fuel failures would occur in the event of a reactor coolant pump locked rotor and, therefore, the radiological consequences are bounded by the consequences of a main steam line break accident. The staff finds this conclusion acceptable.

b. Steam Generator Tube Rupture (SGTR)

The licensee analyzed the SGTR to assure that the offsite radiological consequences are within the 10 CFR Part 100 acceptance criteria of 25 rem to the whole body or 300 rem to the thyroid from iodine exposure. No fuel cladding ruptures or fuel melting occur during the accident, and offsite power remained available for the duration of the accident. The licensee is not committed to the Standard Review Plan (NUREG-0800); therefore, it did not analyze the effects of iodine spiking as outlined there. The licensee analyzed two postulated cases:

Case 1: Equilibrium RCS iodine concentrations consistent with 1 percent failed fuel exist at the time of the accident, which bounds the TS RCS equilibrium iodine concentration limit. No iodine release rate spiking is assumed.

Case 2: Pre-existing iodine spike at the time the accident occurs. The reactor coolant iodine concentrations are the maximum permitted for full power operation (50 times the equilibrium TS limit).

Other assumptions used in the SGTR analysis are presented in the licensee's letter dated May 27, 1999. The staff reviewed the licensee's assumptions and determined they are acceptable. The licensee's calculated offsite dose results are within the acceptance criteria as shown in Table 1. NRC staff performed a calculation to confirm licensee results. The staff has determined that the licensee's SGTR dose analysis is acceptable.

Table 1
Licensee Calculated Offsite Dose
Steam Generator Tube Rupture

	Dose (rem)		10 CFR 100
	EAB	LPZ	Acceptance Criteria (rem)
Thyroid			
Eq I*	39.5	11.3	300
Max I**	262	66.1	300
Whole Body	0.4	0.08	25

* Eq I = RCS equilibrium iodine concentration for 1 percent failed fuel
**Max I = 50 times TS limit RCS equilibrium iodine concentration

c. Rod Ejection Accident

The licensee analyzed the REA to assure that the offsite radiological consequences are within the 10 CFR Part 100 acceptance criteria of 25 rem to the whole body or 300 rem to the thyroid from iodine exposure. The licensee assumed that the cladding fails on 50 percent of the fuel in the core. One hundred percent of the gap fission gas activity is assumed to be instantaneously released and mixed homogeneously in the entire reactor coolant system volume. The staff performed a dose analysis to confirm licensee results. Other assumptions for the REA analysis are contained in the licensee's letter dated May 27, 1999. The staff reviewed the licensee's assumptions and determined they are acceptable. The licensee's calculated offsite dose results are within the acceptance criteria as shown in Table 2. The staff performed a calculation to confirm licensee results and determined that the licensee's REA dose analysis is acceptable.

Table 2
Licensee Calculated Offsite Dose
Rod Ejection Accident

	Dose (rem)		10 CFR 100
	EAB	LPZ	Acceptance Criteria (rem)
Thyroid	84.6	11.5	300
Whole Body	0.4	0.04	25

d. Main Steam Line Break

The licensee analyzed the MSLB to assure that the offsite radiological consequences are within the 10 CFR Part 100 acceptance criteria of 25 rem to the whole body or 300 rem to the thyroid from iodine exposure. No fuel cladding ruptures or fuel melting occur during the accident, and offsite power remained available for the duration of the accident. The licensee is not committed to the Standard Review Plan. Therefore, it did not analyze the effects of iodine spiking as outlined there. The licensee analyzed two postulated cases:

Case 1: TS limit RCS equilibrium iodine concentrations exist at the time of the accident. No iodine release rate spiking is assumed. The RCS activities for non-iodine isotopes are consistent with 1 percent failed fuel.

Case 2: Pre-existing iodine spike at the time the accident occurs. The reactor coolant iodine concentrations are the maximum permitted for full power operation (50 times the equilibrium TS limit).

Other assumptions used in the MSLB analysis are presented in the licensee's letter dated May 27, 1999. The staff reviewed the licensee's assumptions and determined they are acceptable. The licensee's calculated offsite dose results are within the acceptance criteria as shown in Table 3. The staff performed a calculation to confirm licensee results and determined that the licensee's MSLB dose analysis is acceptable.

Table 3
Licensee Calculated Offsite Dose
Main Steam Line Break

	Dose (rem)		10 CFR 100
	EAB	LPZ	Acceptance Criteria (rem)
Thyroid			
Eq I*	9.80	1.08	300
Max I**	11.4	1.58	300
Whole Body	0.01	0.003	25

* Eq I = TS limit RCS equilibrium iodine concentration
**Max I = 50 times TS limit RCS equilibrium iodine concentration

e. Small Steam Line Break

The licensee analyzed the small steam line break outside containment to assure that the offsite radiological consequences are within the 10 CFR Part 100 acceptance criteria of 25 rem to the whole body or 300 rem to the thyroid from iodine exposure. No fuel cladding ruptures or fuel melting occur during the accident, and offsite power remained available for the duration of the accident. The licensee is not committed to the Standard Review Plan; therefore, it did not analyze the effects of iodine spiking as outlined there. The licensee analyzed two postulated cases:

Case 1: TS limit RCS equilibrium iodine concentrations exist at the time of the accident. No iodine release rate spiking is assumed. The RCS activities for non-iodine isotopes are consistent with 1 percent failed fuel.

Case 2: Pre-existing iodine spike at the time the accident occurs. The reactor coolant iodine concentrations are the maximum permitted for full power operation (50 times the equilibrium TS limit).

Other assumptions for the small steam line break analysis are presented in the licensee's letter dated May 27, 1999. The staff reviewed the licensee's assumptions and determined they are acceptable. The licensee's calculated offsite dose results are within the acceptance criteria as shown in Table 4. The staff has determined that the licensee's small steam line break outside containment dose analysis is acceptable.

Table 4
Licensee Calculated Offsite Dose
Small Steam Line Break

	Dose (rem)		10 CFR 100
	EAB	LPZ	Acceptance Criteria (rem)
Thyroid			
Eq I*	4.97	0.62	300
Max I**	6.69	1.52	300
Whole Body	0.02	0.004	25

* Eq I = TS limit RCS equilibrium iodine concentration
**Max I = 50 times TS limit RCS equilibrium iodine concentration

2.6.2 Summary

The staff has reviewed the licensee radiological consequences analyses performed in support of the Oconee Unit 2 Cycle 18 reload and has found the analyses and results to be acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 35202, 64 FR 59801). Accordingly, the 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 the amendments.

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.

Principal Contributors: Z. Abdullahi
G. Hammer
M. Hart

Date: January 18, 2000