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

September 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. MA5348, MA5349, AND MA5350)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 315 , 315 , and 315 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 26, 1999, as supplemented May 15, July 26, and August 23, 2000.

The amendments revise various provisions contained in the Technical Specifications and Updated Final Safety Analysis Report related to (1) the steam generator tube loads (related to re-rolling) following a main steam line break, and (2) crediting the main steam line break detection and feedwater isolation instrumentation as a means for providing runout protection for the turbine-driven emergency feedwater pump.

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. 315 to DPR-38
2. Amendment No. 315 to DPR-47
3. Amendment No. 315 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

September 18, 2000

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

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The amendments revise various provisions contained in the Technical Specifications and Updated Final Safety Analysis Report related to (1) the steam generator tube loads (related to re-rolling) following a main steam line break, and (2) crediting the main steam line break detection and feedwater isolation instrumentation as a means for providing runout protection for the turbine-driven emergency feedwater pump.

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/RA/

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4. Safety Evaluation

incorporated DEL 9/23/2000

cc w/encls: See next page
DOCUMENT NAME: G:\PDII-1\OCONEE\MA5348 AMD.WPD

OFFICE	PDII-1/PM	PDII-1/LA	EEIB	OGC	PDII-1/SC
NAME	DLaBarge	CHawes	JCalvo	MYoung	REmch
DATE	8/24/00	8/23/00	8/30/00	9/11/00	9/18/00

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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. 315
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 26, 1999; supplemented May 15, July 26, and August 23, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and 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 Appendix A, as revised through Amendment No. 315 , 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 60 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: September 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. 315
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 26, 1999; supplemented May 15, July 26, and August 23, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and 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 Appendix A, as revised through Amendment No. 315 , 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 60 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: September 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.315
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 26, 1999; supplemented May 15, July 26, and August 23, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and 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 Appendix A, as revised through Amendment No. 315 , 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 60 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: September 18, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 315

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 315

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 315

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix A Technical Specifications and associated Bases 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 LOEP1	TS LOEP1	B 3.3.11-1	B 3.3.11-1
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OCONEE NUCLEAR STATION
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3.3 INSTRUMENTATION

3.3.11 Main Steam Line Break (MSLB) Detection and Feedwater Isolation Instrumentation

LCO 3.3.11 Three MSLB Detection and Feedwater Isolation instrumentation channels per steam generator (SG) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure ≥ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each SG (Feedwater Isolation Function).

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Feedwater Isolation Functions with one channel inoperable.	A.1 Place channel(s) in trip.	4 hours
B. One or more Feedwater Isolation Functions with two or more channels inoperable. OR Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2.1 Reduce main steam header pressure to < 700 psig. OR B.2.2 Close all MFCVs and SFCVs.	12 hours 18 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.11.2	----------NOTE---------- Only applicable when modifications are implemented that allow online testing. ---------- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.11.3	Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.12 Main Steam Line Break (MSLB) Detection and Feedwater Isolation Manual Initiation

LCO 3.3.12 Two MSLB Detection and Feedwater Isolation manual initiation switches shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure ≥ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One manual initiation switch inoperable.	A.1	Restore manual initiation switch to OPERABLE status.	72 hours
B. Two manual initiation switches inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2.1	Reduce main steam header pressure to < 700 psig.	18 hours
	OR		
	B.2.2	Close all MFCVs and SFCVs.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.12.1	Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.13 Main Steam Line Break (MSLB) Detection and Feedwater Isolation Logic Channels

LCO 3.3.13 Two MSLB Detection and Feedwater Isolation Logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure ≥ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One logic channel inoperable.	A.1	Restore channel to OPERABLE status.	72 hours
B. Two logic channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2.1	Reduce main steam header pressure to < 700 psig.	18 hours
	OR		
	B.2.2	Close all MFCVs and SFCVs.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	Perform CHANNEL FUNCTIONAL TEST.	18 months

OCONEE NUCLEAR STATION
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OCONEE NUCLEAR STATION
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B 3.3 INSTRUMENTATION

B 3.3.11 Main Steam Line Break (MSLB) Detection and Feedwater Isolation Instrumentation

BASES

BACKGROUND

The MSLB Detection and Feedwater Isolation instrumentation is designed to address containment overpressurization and steam generator tube load concerns by isolating main feedwater (MFW) and stopping the turbine-driven emergency feedwater pump (TDEFW) from delivering feedwater to both steam generators during an MSLB and to mitigate core overcooling concerns. In addition, the MSLB Detection and Feedwater Isolation instrumentation provides runout protection for the TDEFW pump in the event of a MSLB and certain large MFW line breaks with the pump in the automatic mode of operation.

Steam generator header pressure is used as input signals to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. These functions are credited for mitigating an MSLB. The function of closing the main and startup feedwater block valves is not credited in the MSLB analysis. However, the MSLB Detection and Feedwater Isolation circuitry performs this function.

There are three pressure transmitters per steam generator with each feeding a steam pressure signal to a signal isolator (when used) and bistable. These bistables are calibrated to provide an ON/OFF signal at the desired setpoint for actuation of the feedwater isolation circuitry. A pressure transmitter and its associated signal isolator(s) and bistable(s) constitute a MSLB detection analog channel.

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the

BASES

BACKGROUND (continued)

digital channels will not result in an in advertent feedwater isolation. If either digital channel is actuated, a feedwater isolation (i.e., MFW pumps trip, MFCVs and SFCVs close, and TDEFW pump inhibited from auto-starting or auto-stopped if running) will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. The two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, time delay, and test/manual actuation pushbutton are considered a feedwater isolation digital channel.

The feedwater isolation digital channels are enabled and disabled administratively rather than automatically. Appropriate operating procedures contain provisions to enable/disable the digital channels.

APPLICABLE SAFETY ANALYSES

The MSLB Detection and Feedwater Isolation instrumentation is utilized to isolate main feedwater and stop the TDEFW pump from supplying feedwater to both steam generators in the event a MSLB occurs. This function is credited in the MSLB analyses regarding containment response and steam generator tube loads.

Steam generator header pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs are closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. All of these functions are credited for mitigating a MSLB inside containment.

The MSLB Detection and Feedwater Isolation Instrumentation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

This LCO requires that instrumentation necessary to initiate a feedwater isolation (i.e., trip the MFW pumps, close the MFCVs and SFCVs, inhibit the auto-start of or auto-stop the TDEFW pump) shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the Function.

BASES

LCO (continued)

Three analog channels per SG are required to be OPERABLE to ensure that no single failure prevents actuation of the MSLB Detection and Feedwater Isolation instrumentation. Each MSLB Detection and Feedwater Isolation instrumentation channel includes the sensor and measurement channel.

APPLICABILITY

The MSLB Detection and Feedwater Isolation Function shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure ≥ 700 psig, because there is significant mass and energy in the RCS and steam generators. Once the steam header pressure has decreased below 700 psig, additional time is available for the operator to manually isolate main and emergency feedwater to the affected steam generator. Thus, the Feedwater Isolation Function can be bypassed to avoid actuation during normal unit cooldowns. Also during MODE 3, the Feedwater Isolation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since the function of the instrumentation is already fulfilled. In MODE 3 when the turbine header pressure is < 885 psig, automatic actuation of the TDEFW pump is blocked. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent. In MODES 4, 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy and MSLB Detection and Feedwater Isolation instrumentation is not required to be OPERABLE.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or any of the transmitter or signal processing electronics, are found inoperable, then the Function provided by that channel must be declared inoperable and the unit must enter the appropriate Conditions.

A Note has been added to the ACTIONS indicating that a separate Condition entry is allowed for instrumentation channels associated with each SG (feedwater isolation function).

A.1

Condition A applies to failures of a single MSLB Detection and Feedwater Isolation instrumentation channel in one or more Feedwater Isolation Functions.

BASES

ACTIONS

A.1 (continued)

With one channel inoperable in one or more MSLB Detection and Feedwater Isolation Function, the channel(s) must be placed in trip within 4 hours. Tripping the affected channel places the Function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the MSLB Detection and Feedwater Isolation Function is capable of actuating in the presence of any single random failure. The Completion Time of 4 hours is adequate to perform Required Action A.1.

B.1, B.2.1, and B.2.2

With two channels in one or more MSLB Detection and Feedwater Isolation Function inoperable or the Required Action and associated Completion Time of Condition A not met, the unit must be placed in MODE 3 within 12 hours and main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Time is 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 3.3.11.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1 (continued)

required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified, where practical, to be reading at the bottom of the range and not failed downscale.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.11.2

A CHANNEL FUNCTIONAL TEST is performed on each required instrumentation channel to ensure the channel will perform its intended function.

The Frequency of 31 days is based on operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel in any 31 day interval is a rare event.

This SR is modified by a Note indicating that it is only applicable when modifications are implemented that allow online testing.

SR 3.3.11.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The Frequency is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

BASES (continued)

REFERENCES

1. 10 CFR 50.36.
2. UFSAR Section 6.2.1.4.
3. UFSAR Section 5.2.3.4.

B 3.3 INSTRUMENTATION

B 3.3.12 Main Steam Line Break (MSLB) Detection and Feedwater Isolation Manual Initiation

BASES

BACKGROUND

The MSLB Detection and Feedwater Isolation manual initiation capability provides the operator with the capability to actuate the isolation function from the control room. This Function is provided in the event the operator determines that the Function is needed and does not automatically actuate. This is a backup Function to the automatic Feedwater isolation.

The MSLB Detection and Feedwater Isolation manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The Feedwater Isolation Function credited in the safety analysis is automatic. However, the manual initiation Function is required by design as backup to the automatic Function and allows operators to actuate Feedwater Isolation whenever the Function is needed. Furthermore, the manual initiation of Feedwater Isolation may be specified in unit operating procedures.

The MSLB Detection and Feedwater Isolation manual initiation function satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

One manual initiation switch per actuation channel (A and B) is required to be OPERABLE. The Feedwater Isolation function, has two actuation or "trip" channels, channels A and B. Within each channel actuation logic there is one manual trip switch. When the manual switch is depressed, a full trip of actuation channel A or B occurs.

APPLICABILITY

The Feedwater Isolation manual initiation Function shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure ≥ 700 psig because there is significant mass and energy in the RCS and steam generators. Once the steam header pressure has decreased below 700 psig, additional time is available for the operator to manually isolate main and emergency feedwater to the affected steam generator. Thus, the

BASES

APPLICABILITY (continued)

Feedwater Isolation Function can be bypassed to avoid actuation during normal unit cooldowns. During MODE 3, the Feedwater Isolation manual initiation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled. In MODE 3 when the turbine header pressure is < 885 psig, automatic actuation of the turbine-driven emergency feedwater pump is blocked. In MODES 4, 5, and 6, the SG energy level is low and secondary side feedwater flow rate is low or nonexistent.

ACTIONS

A.1

With one manual initiation switch inoperable, the manual initiation switch must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and Feedwater Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and Feedwater Isolation Function.

B.1

With both manual initiation switches inoperable or the Required Action and associated Completion Time of Condition A not met, the unit must be placed in MODE 3 within 12 hours and the main steam header pressure reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

BASES (continued)

REFERENCES

1. IEEE-279-1971, April 1972.
2. 10 CFR 50.36.
3. UFSAR Section 6.2.1.4.
4. UFSAR Section 5.2.3.4.

B 3.3 INSTRUMENTATION

B 3.3.13 Main Steam Line Break (MSLB) Detection and Feedwater Isolation Logic Channels

BASES

BACKGROUND

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an inadvertent feedwater isolation. If either digital channel is actuated, a Feedwater isolation will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. Each of the two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, and time delay is considered a feedwater isolation digital channel.

APPLICABLE SAFETY ANALYSES

The MSLB Detection and Feedwater Isolation instrumentation is utilized to isolate main feedwater and stop the TDEFW pump from supplying feedwater to both steam generators in the event a MSLB occurs. This function is credited in the MSLB analyses regarding containment response and steam generator tube loads.

Steam generator outlet pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started and the switch for MS-93 is in the AUTO position. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for heat removal. All of these functions are credited for mitigating a MSLB inside containment.

The MSLB Detection and Feedwater Isolation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

BASES (continued)

LCO

Two channels of MSLB Detection and Feedwater Isolation automatic actuation logic shall be OPERABLE. There are only two channels of automatic actuation logic. Therefore, violation of this LCO could result in a complete loss of the automatic Function assuming a single failure of the other channel.

APPLICABILITY

The MSLB Detection and Feedwater Isolation automatic actuation logic channels shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure ≥ 700 psig because there is significant mass and energy in the RCS and steam generators. Once the steam header pressure has decreased below 700 psig, additional time is available for the operator to manually isolate main and emergency feedwater to the affected steam generator. Thus, the Feedwater Isolation Function can be bypassed to avoid actuation during normal unit cooldowns. Also, during MODE 3, the Feedwater Isolation function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled. In MODE 3 when the turbine header pressure is < 885 psig, automatic actuation of the TDEFW pump is blocked. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent.

ACTIONS

A.1

With one automatic actuation logic channel inoperable, the channel must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and Feedwater Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and Feedwater Isolation Function.

B.1, B.2.1, and B.2.2

With both logic channels inoperable or the Required Action and associated Completion Time not met, the unit must be placed in MODE 3 within 12 hours and the main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. This test verifies Feedwater Isolation automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

REFERENCES

1. 10 CFR 50.36.
2. UFSAR Section 6.2.1.4.
3. UFSAR Section 5.2.3.4.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 315 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 315 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 315 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 26, 1999, as supplemented May 15, July 26, and August 23, 2000, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee) Technical Specifications (TS). The amendments would revise various provisions contained in the TS and Updated Final Safety Analysis Report (UFSAR) related to (1) the steam generator tube loads (related to re-rolling) following a main steam line break (MSLB), and (2) crediting the MSLB detection and feedwater isolation instrumentation as a means of providing runout protection for the turbine-driven emergency feedwater pump (TDEFWP). The supplements dated May 15, July 26, and August 23, 2000, provided clarifying information that did not change the scope of the April 26, 2000, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

On November 21, 1997, the staff issued Amendment Nos. 227, 227, and 224 to the Oconee Units 1, 2, and 3, respectively, TS regarding the use of re-rolling criteria to repair steam generator (SG) tube defects in the upper tube sheet roll area. This was based in part on information and analysis results supplied by the licensee in Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," supplied by Framatome Technology Incorporated (FTI). The topical report contained a change to the MSLB SG tube design load from the previous generic load of 3140 pounds-force (lbf) to the Oconee-specific loading of 2376 lbf, which was calculated using more recent analysis and analytical tools. Staff approval of these amendments referenced Regulatory Guide 1.121 and NB-3225 of Section III of the American Society of Mechanical Engineers (ASME) Code, but did not specifically address the tube load details documented in the submittal.

This value of SG tube load, 2376 lbf, was contained in Reference 1 as the SG design basis tube load. However, to resolve an earlier FTI Preliminary Safety Concern regarding SG tube enhanced analysis and evaluation for Oconee, the revised tube loads and the supporting information were provided to the staff for review in a supplement dated May 15, 2000,

(Reference 3) submitted by the licensee. It resulted in a new design basis load of 2870 lbf, which is the value used in this safety evaluation.

Reference 1 and related supplements provide descriptions and technical justifications for proposed changes to the TS, TS Bases, and Chapters 3, 5, and 15 of the Oconee UFSAR related to the re-analysis criteria and results presented in the submittal. The supporting information for the UFSAR Chapter 15 revision is related to the thermal-hydraulic transient analysis methodology and the need to retain the details of this assessment from the UFSAR. The licensee requested that the proposed analytical details, conclusion resulting from staff evaluations, and various other changes, be reviewed by the staff and incorporated into the TS and licensing basis for Oconee as detailed in the submittal dated April 26, 1999, and supplements dated May 15 and July 26, 2000, as described below.

3.0 PROPOSED FSAR CHANGES

3.1 Section 3.9.3.1.1.1.5

The description, related to the rupture of a secondary pipe that would result in an overcooling transient in which the SG tubes cool down faster than the SG shell but would not result in a loss of reactor coolant, would be replaced with a statement regarding the analyses that have shown that the resulting stresses would not result in tube rupture.

3.2 Section 5.2.3.4

The description, related to the rupture of a secondary pipe that would result in an overcooling transient in which the SG tubes cool down faster than the SG shell but would not result in a loss of reactor coolant, would be replaced with information regarding the analyses that have shown the resulting stresses would not result in tube rupture. In addition, details of the method and criteria used to calculate and evaluate the SG tube load resulting from an MSLB are described. The supplement dated May 15, 2000, revised the axial tube load used in an update to the analysis to 2870 lbf.

3.3 Section 15.13.4

This section would be deleted. The revised MSLB SG tube analysis would be provided in UFSAR Section 5.2.3.4.

3.4 Other proposed changes include the addition of references and administrative changes to various sections.

4.0 EVALUATION OF FSAR CHANGES

4.1 Steam Generator Tube Loading Thermal Analysis

4.1.1 Scope

Staff review of the licensee's submittals with respect to the thermal analysis for SG tube loads related to re-rolling following an MSLB was limited to the use of the RETRAN-02 computer code (Reference 4) that was used to generate primary and secondary system pressures and the

tube-to-shell temperature difference as inputs to the SG tube load structural and mechanical analysis. The staff has approved Oconee's use of the methodology described in Topical Report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient and Accident Analysis Methodology," (Reference 5) by letter dated May 25, 1999.

4.1.2 Evaluation

The rupture of a main steam line, an MSLB, would result in an overcooling transient in which the SG tubes cool down faster than the steam generator shell. As a result, the tubes are subjected to a tensile load that might cause tube deformation. The licensee analyzed the MSLB accident using the industry standard RETRAN-02 computer code (Reference 4), in accordance with Chapter 15 methodology (Reference 5), which, as stated above, was previously reviewed and approved by the staff for use at Oconee. This analysis was performed to determine the appropriate input for the SG tube stress analysis, including the primary and secondary system pressures and the tube-to-shell temperature difference. The revised analysis reported in the supplement resulted in an axial load of 2870 lbf for the MSLB conditions.

The revised MSLB SG tube load thermal analysis results in a greater tube-to-shell temperature difference (and a larger tube load) and is conservative relative to the previous value. In addition, since the RETRAN-02 computer model is applied consistent with the approved topical report DPC-NE-3005-PA methodology and the SG tubes have been shown to meet the acceptance criteria for the MSLB accident load, the revised thermal analysis methodology is acceptable.

4.2 Revised Thermal-Hydraulic Analysis

4.2.1 Scope

The licensee has proposed changes in the emergency plant operating procedures following an MSLB event to more effectively manage the core reactivity. However, these changes also result in higher SG tube loads. The thermal-hydraulic analysis information submitted in support of the proposed changes is intended to demonstrate that even though the SG tube loads are higher than those resulting from the original emergency plant operating procedures, they are still within acceptable limits.

4.2.2 Evaluation

The revised MSLB thermal-hydraulic analysis assumes a full-power initial condition and a double-ended rupture of the largest main steam line. The main feedwater (MFW) system is isolated on low steam line pressure by the MSLB detection and feedwater isolation instrumentation. This circuit also inhibits auto-start of the TDEFW pump and auto-stops the TDEFW pump if it is running. The analysis assumptions were selected to maximize the tube-to-shell temperature difference, which results in bounding SG tube loads. The SG MSLB thermal-hydraulic analysis was performed using the RETRAN computer code. The pressure and temperature data obtained from this analysis were used as input to the structural analysis and adjusted for features not specifically modeled. The structural analysis was performed with an ANSYS finite element model of the SG to calculate the axial tube loads and tube sheet deflections. The SG finite element model is an axisymmetric model of the entire SG. The

thermal-hydraulic analysis resulted in a maximum tube-to-shell temperature difference of 256°F and a maximum tensile tube loading of 2870 lbf, which was determined based on this temperature differential. Although these results exceeded the loads described in the current UFSAR Section 15.13.4, the revised loads meet the applicable acceptance criteria as described in the revised UFSAR Section 5.2.3.4 and discussed later in this safety evaluation (SE).

In the original MSLB SG analysis, the Instrument Control System was used to control SG levels to the post-trip minimum level. The MFW system remained in operation, and the Emergency Feedwater (EFW) system was not actuated without the need for operator action to mitigate the event.

The revised MSLB analysis uses a methodology different from that employed in the original analysis and is summarized below. The SG depressurization following the MSLB initiates a rapid reactor coolant pump (RCP) cooldown leading to a reactor trip on low RCP pressure within the first few seconds. The reactor trip causes the turbine stop valves to close, isolating the SG with the ruptured main steam line from the unaffected SG. The steam line break detection and mitigation circuitry isolates the MFW and the TDEFW pumps in about 40 seconds. The SG with the ruptured main steam line continues to depressurize while the pressure in the isolated SG increases, which is controlled by the turbine bypass valves and the main steam safety valves. The analysis assumes a single failure that results in a maximum uncontrolled flow from one of the two motor-driven EFW pumps to the SG with the ruptured main steam line. This flow contributes to an increase in the SG tube-to-shell temperature differential. Operator action to isolate the motor-driven EFW flow to the SG with the ruptured main steam line is credited at 10 minutes. A penalty is also taken for the operators tripping the reactor coolant pumps on loss of subcooled margin at two minutes. The reduced RCP flow results in a colder tube bundle temperature since the primary inventory resides longer in the tube bundle. In addition, by assuming that maximum flow of cold water occurs from the emergency core cooling system, the SG tube-shell temperature differential is further maximized.

The EFW is directed onto the peripheral SG tubes near the top of the SG, where it boils on contact with the tubes and support plates as it flows downward and inward. The resulting steam flows upward and out the break, and interacts with the downward flowing liquid. This local three-dimensional thermal-hydraulic process is not specifically modeled in the analysis, which models the average SG tube region. The steam and feedwater interaction is assumed to be sufficient to justify the approximate average tube modeling approach. MFW and EFW isolation are the dominant factors in the analysis.

The results of the thermal-hydraulic analysis are provided in Reference 6. The average tube-to-shell temperature differential and a number of properties such as pressures, temperatures, and mass flow rates are tabulated in the various internal regions of the SG such as the hot and cold legs, inlet and outlet plenums, downcomer regions during the transient. These properties are used as input for the finite element structural analysis model to determine the axial tube loads and tube sheet deflections.

A loss of offsite power does not impact the ability of the MSLB detection and feedwater isolation circuitry to perform its intended safety function. Even though the station air compressors would not be available during such an event, a sufficient inventory of air will continue to be available, via the instrument air system reservoirs, to operate the main and startup feedwater control

valves. The feedwater control valves are supplied by two redundant subsystems, the instrument air system and the service air system. Both of these systems have reservoirs of sufficient capacity to provide air to close the MFW control valves within 25 seconds of the initiation of the MSLB detection and feedwater isolation instrumentation.

The total load on the SG tubes is the resultant of loads produced from a number of sources. These loads include:

- Installed preload on the tube.
- Thermal load due to differential expansion between the tube bundle and the shell. This is the largest component of the total load and is determined at the maximum differential temperature that occurs during the overcooling transient. Based on the thermal hydraulic analysis discussed earlier, this differential temperature was determined to be 256°F.
- Compressive loads on the tubes resulting from the bowing of the tubesheet.
- Loads on the tubes resulting from the rotation of the SG head, shell and tubesheet ring under varying pressures and temperatures.
- Axial loads on the tubes resulting from the Poisson's effect of the pressure on tubes and the shell. These loads are relatively small in magnitude.

The net tensile tube loads vary across the tubesheet due to the deflection of the tubesheet and other factors discussed earlier. Thus, for example, the maximum tube loads due to the MSLB accident have been determined to be 2870 lbf at the periphery and varying to 1840 lbf at the center of the bundle. All of the loads discussed above and the resultant load combinations are obtained as output from the finite element structural analysis using the ANSYS computer code.

The licensee has indicated that the thermal stresses due to these loads are secondary stresses in accordance with the description of such stresses in Subsection NB-3213.13(a), Section III of the ASME Code. As such, the licensee indicated that these stresses are not subject to the stress limits for faulted conditions set forth in Subsection NB-3225 and Appendix F of Section III of the ASME Code that are intended for the evaluation of primary stresses. The staff does not fully support the licensee's contention because a large percentage of the thermal load is axial throughout the entire length of the tube. However, since the licensee has conservatively evaluated them as primary stresses, the staff considers the licensee's determination acceptable.

The allowable stress intensity for a general primary membrane stress (P_m) in accordance with Appendix F-1331.1(a) of the ASME Code, is the lessor of $2.4S_m$ (Code specified stress intensity) or $0.7S_u$ (ultimate tensile strength). The values of S_m and S_u were obtained from a statistical study of actual SG tube material test data obtained by FTI. While the staff has not reviewed the complete test data or the testing methodology in detail for this safety evaluation, the staff has reviewed the calculated results of the lower 95/95 tolerance limits for the ultimate and yield strengths at 600°F. The results indicate that adequate margin exists between the allowable values and the highest calculated stress intensity during the overcooling transient. Based on its review, as discussed above, the staff finds the structural evaluation of the SG

tube loads reasonable, acceptable, and in compliance with the ASME Code Section III requirements.

4.3 Conclusion

The results of the revised SG tube load analysis indicate that a maximum tube-to-shell temperature differential of 256°F occurs during the MSLB accident. This results in a maximum tensile load of 2870 lbf on the SG tubes, which is greater than, and more conservative than, that previously calculated, which were found to be acceptable.

The staff has reviewed the SG load methodology and thermal-hydraulic analysis information submitted in support of proposed changes to the UFSAR Chapter 15 transient and accident analysis methodology, the Oconee TS Bases and other sections of UFSAR, and the methodology used to determine the axial load for MSLB conditions. The staff has determined that the methodology and analysis described in the UFSAR section is consistent with the previously-approved Topical Report DPC-NE-3005-PA and demonstrates that the applicable safety limits will be maintained to prevent fuel failure since SG integrity is maintained even with the increased SG tube load. Therefore, the staff finds the analysis acceptable. Also, the proposed UFSAR changes are consistent with the licensing basis documented in the licensee's topical report and are acceptable.

4.4 Operator Actions

4.4.1 Scope

By letter dated July 26, 2000, the licensee supplied information concerning operator response times for manual actions necessary to trip the reactor coolant pumps (RCPs) and to isolate EFW flow following a main steam line break event. Operator action to trip the RCPs upon loss of subcooled margin was added as part of Oconee's action plan to address issues related to the 1979 Three Mile Island 2 accident. Action was added for mitigating small break loss of coolant accident (SBLOCA) events and a time of two minutes is credited for SBLOCA analysis. Since subcooled margin will also be lost very quickly for large MSLB events, the same action - tripping the RCPs - will occur for either event within the same time limit, per the emergency procedure. For the MSLB event, tripping the RCPs at two minutes is conservative in that the calculated tube loads are maximized.

The staff normally uses American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 58.8 (1984), "Time Response Design Criteria for Safety-Related Operator Actions," for operator response time estimates. ANSI/ANS 58.8 provides estimates of reasonable response times for operator actions, but allows the use of time intervals derived from independent sources, provided they are based on task analyses that consider such factors as specific events, plant design, human factors, procedures, and independent data bases.

4.4.2 Evaluation

In the submittal (Reference 1), the licensee stated that for the MSLB event, manual tripping of the RCPs is not a credit, but a penalty. The loss of forced circulation by tripping the pumps causes the SG tubes to cool to a greater extent, increasing the temperature difference between tubes and the SG shell, thereby increasing the tube loading. Thus, the most conservative time

for operator action is the minimum time. The licensee further stated that the control room indications are similar to those of an SBLOCA, namely depressurization of the primary system, actuation of engineered safeguards equipment, and loss of subcooled margin. Consequently, the staff concludes that it is reasonable to include the same operator action for an MSLB event as that defined for an indication of an SBLOCA.

In a letter dated May 29, 1986, to all applicants and licensees with Babcock and Wilcox designed nuclear steam supply systems, the staff addressed implementation of Three Mile Island Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Generic Letter 86-05). In the accompanying safety evaluation, the staff endorsed the concept of using subcooled margin as an indication for the need to manually trip the RCPs. In addition, the staff indicated that if the RCPs are tripped within ten minutes during an SBLOCA, the peak clad temperature will not exceed the safety limit. This action of tripping the RCPs in response to SBLOCA conditions was also evaluated and approved in the March 15, 1988, letter (Reference 7).

Therefore, staff concludes that operator action to trip the RCPs is appropriately based on loss of subcooling margin, not in diagnosing an SBLOCA or an MSLB, a philosophy that has been in place in the emergency operating procedures for a number of years. In addition, the staff finds two minutes to be an acceptable minimum time estimate for the MSLB calculation.

The second manual action related to the MSLB analysis is isolating EFW flow from the motor-driven EFW pumps to the affected SG within ten minutes. The staff questioned whether ANSI/ANS 58.8 criteria had been used to determine the time estimate. The licensee indicated that they have not committed to ANSI/ANS 58.8 and that the assumed ten-minute operator action time to isolate EFW flow was an established time-critical operator action reflected in Section 6.2.1.4.4 of the Oconee UFSAR, which was approved by the staff in the December 7, 1998, amendment (Amendment Nos. 234, 234, and 233). The licensee also indicated that all associated instrumentation is located within a twelve-foot area of the main control room horseshoe and that the operators are trained to perform this action within the ten minute time limit. The licensee also described the specific manipulations required to isolate EFW flow. The staff finds that this explanation is a satisfactory resolution of the guidance contained in ANSI/ANS 58.8. Based on the reasonableness of the action time and the and previous approval of this action time by the staff in the May 29, 1986, and the December 7, 1998, SEs, the staff finds the proposed ten-minute operator action time for isolating EFW flow acceptable.

4.4.3 Conclusion

As explained above, the staff finds the two minutes minimum time estimate for the MSLB calculation and the proposed ten-minute operator action time for isolating EFW flow to be acceptable.

4.5 Equipment Qualification

As explained in the May 15, 2000, submittal, the MSLB detection and feedwater isolation instrumentation is qualified as Quality Assurance (QA)-1. However, some of the equipment operated by this instrumentation is not fully qualified or single failure proof. This equipment includes the Integrated Control System, main feedwater pumps, main feedwater pump trip circuitry, main feedwater control valve operators and associated power supplies, startup main feedwater control valve operators and associated power supplies, TDEFWP steam admission

valve, and air supply paths to air operated valves. The staff approved this design in the letter dated December 7, 1998, for Oconee Units 1, 2, and 3 Amendment Nos. 234, 234, and 233, respectively. The SE accompanying these amendments concluded that the potential for the main feedwater control valve failing to close upon receipt of a close signal from the redundant MSLB circuitry is remote. In addition, because the main feedwater control valve is necessary for normal SG water level control, failure to function for any reason would be detected quickly during normal plant operation by loss of level control. In the event that all sources of compressed air are lost, sufficient air remains in the air reservoirs (air receivers) to close the valves within 25 seconds of the initiation of the MSLB circuitry. The safety evaluation also pointed out that the licensee indicated there is no single active failure to the electrical/electronic controls that would prevent closure of a main feedwater valve. Therefore, the safety evaluation concluded that the design of the MSLB isolation system, although not single-failure proof, was acceptable because the design basis and most limiting MSLB for Oconee does not rely on automatic main feedwater isolation.

Therefore, the staff concludes that the present QA design and designation of the above listed systems is acceptable for the purposes of the described changes.

5.0 PROPOSED TS AND BASES CHANGES

5.1 TS 3.3.11, 3.3.12, and 3.3.13 and associated Bases reference to the MFW isolation instrumentation function would be replaced with references to the feedwater isolation instrumentation function. This would clarify that the MSLB detection and feedwater isolation instrumentation also inhibits the auto-start of or auto-stops the TDEFWP.

5.2 Proposed changes to the LCO and Actions Sections of the Bases for TS 3.3.11:

a) clarify that a feedwater isolation involves the trip of the MFW pumps, closure of the main feedwater control valves and the startup feedwater control valves, and inhibiting the auto-start of or auto-stopping the TDEFWP;

b) clarify that the three channels per SG that are required to be operable are analog channels; and

c) clarify the single failure design regarding the MSLB detection and feedwater isolation instrumentation.

5.3 Proposed changes to the background and applicable safety analysis sections of the Bases for TS 3.3.11:

a) reflect that the MSLB detection and feedwater isolation instrumentation is also used to address SG tube load concerns during an MSLB;

b) clarify that a credited function of the MSLB detection and feedwater isolation instrumentation is to stop the TDEFWP from supplying feedwater to both SGs; and

c) clarify that a feedwater isolation involves the trip of the MFW pumps, closure of the main feedwater control valves and the startup feedwater control valves, and inhibiting the auto-start of or auto-stopping the TDEFWP.

5.4 Applicability sections of the Bases for TS 3.3.11, 3.3.12, and 3.3.13: replace the statement describing the reason that the MSLB detection and isolation instrumentation is required to be operable in Modes 1 and 2, and Mode 3 with main steam header pressure ≥700 psig from "...the SG inventory can be at a high energy level and contribute significantly to the peak pressure with a secondary side break" to "...there is significant mass and energy in the RCP [Reactor Coolant System] and steam generators." The licensee also proposed other changes to the Bases that provide additional justification and amplification related to this applicability change.

5.5 Reference section of Bases for TS 3.3.11, 3.3.12, and 3.3.13: add references to UFSAR 5.2.3.4 and 6.2.1.4, which address the MSLB analyses regarding SG tube loads and containment response, respectively.

5.6 Applicable safety analysis section of the Bases for TS 3.3.13:

a) reflect that the MSLB detection and feedwater isolation instrumentation is also used to address SG tube load concerns during an MSLB; and

b) clarify that a credited function of the MSLB detection and feedwater isolation instrumentation is to stop the TDEFWP from supplying feedwater to both SGs.

5.7 Evaluation

The proposed changes to TS 3.3.11, 3.3.12, and 3.3.13 would change the term "MFW" to "Feedwater." This change ensures that the TS instrumentation requirements are clearly applicable to the entire feedwater system, including the EFW system, not just the Main Feedwater System. Thus, the proposed changes would clarify that a feedwater isolation signal trips the MFW pumps, closes the main feedwater control valves and the startup feedwater control valves, and inhibits auto-start of (or auto-stops) the TDEFWP, changes that are consistent with the analysis contained in Amendment Nos. 234, 234, and 233 that was approved for Oconee Units 1, 2, and 3, respectively, on December 7, 1998.

In addition to the TS changes described above, the licensee proposed changes to the TS Bases consistent with the proposed changes to the TS and FSAR changes that are described in Sections 3.0 and 4.0 of this SE.

5.8 Summary

The staff has performed a technical review of the application for an amendment regarding the proposed changes to the Oconee TS and Bases regarding MFW, MSLB, isolation instrumentation, and TDEFWP runout protection. The staff finds that the proposed changes are adequately supported by the licensee's submittals, previously-issued documents, and/or our analysis of the related documentation that clarify the feedwater system requirements. Therefore, the staff finds the proposed changes to the TS and Bases to be acceptable. In addition, the revisions to the FSAR accurately reflect the revised analysis of SG tube loading following an MSLB and are acceptable.

6.0 References

1. Letter from Duke Energy Corporation to the NRC, "Proposed Amendment to the Facility Operating License Regarding Methodology for Determining Steam Generator Tube Loads Following a Main Steam Line Break and Runout Protection for the Turbine-Driven Emergency Feedwater Pump (TSC -99-01)," April 26, 1999.

2. Letter from Framatome Technologies to the NRC, "Report of Preliminary Safety Concern Related to Design Steam Generator Tube Tensile Loads (PSC 2-98)," October 19, 1998.

3. Letter from Duke Energy Corporation to the NRC, "Supplemental Information - Proposed Amendment to the Facility Operating License Regarding Methodology for Determining Steam Generator Tube Loads Following a Main Steam Line Break and Runout Protection for the Turbine-Driven Emergency Feedwater Pump (TSC -99-01)," May 15, 2000.

4. EPRI NP-1850-CCM, Revision 4, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI, November 1988.

5. Submittal DPC-NE-3005-PA, Revision 1, "UFSAR Chapter 15 Transient and Accident Analysis Methodology," Duke Energy Corporation, Oconee Nuclear Station, dated February 1, 1999.

6. Letter from Duke Energy Corporation to Framatome Technologies, Incorporated, dated November 6, 1998, detailed Oconee main steam line break data.

7. Letter, Safety Evaluation, and Technical Evaluation Report from NRC to Duke Power Company dated March 155, 1988, detailed evaluation of Duke's response to Generic Letter 86-05 and NUREG-0737, Item II.K.3.5, "Automatic Tripping of Reactor Coolant Pumps."

7.0 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.

8.0 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. 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 27320). 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.

9.0 CONCLUSION

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

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