NOTICE OF VIOLATION

Duke Energy Corporation
Oconee Nuclear Station

Docket Nos. 50-269, 50-270, 50-287
License Nos. DPR-38, DPR-47, DPR-55
EA 98-552

During an NRC inspection conducted on November 2-6, November 16-20, and December 11, 1998, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

A. Technical Specification 6.4.1 requires that the station shall be operated in accordance with approved procedures. Written procedures with appropriate instructions shall be provided for emergency procedures involving potential release of radioactivity and for actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety.

Updated Final Safety Analysis Report (UFSAR) Section 3.6, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, incorporated by reference MDS Report OS-73.2, Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station Units 1, 2, and 3, dated April 25, 1973. MDS Report OS-73.2 described a potential event involving an auxiliary steam or main feedwater line break in the turbine building, and a strategy to mitigate that event. The MDS Report strategy included establishing emergency feedwater flow to a steam generator within 15 minutes and then beginning a plant cooldown. Prior to plant cooldown, operators must manually connect temporary power cables to an HPI pump. The MDS Report stated that these actions can be accomplished within a 30 minute time period.

IP/O/A/0050/001, Procedure to Provide Emergency Power to an HPI Pump Motor from the ASW Switchgear, Rev. 7, dated September 8, 1998, was the written procedure used to mitigate an auxiliary feedwater or main steam line break, and included to manually connecting temporary cables to power an HPI pump from the ASW switchgear. The procedure required first racking out the electrical breaker to the HPI pump at the safety-related 4160-volt switchgear.

Contrary to the above, on November 5, 1998, written procedures with appropriate instructions were not provided for emergency procedures involving potential release of radioactivity or for actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety. Specifically, the procedure was inadequate in that in the event of an auxiliary steam or main feedwater line break as described in MDS Report OS-73.2, the safety-related 4160-volt switchgear would be inaccessible because it would be in a steam environment. (01014)

This is a Severity Level IV violation (Supplement I).

B. 10 CFR 50.59 allows a licensee to make changes in procedures as described in the safety analysis report (SAR), without prior NRC approval, unless the proposed change involves a change in the technical specifications (TS) or an unreviewed safety question (USQ). A proposed change shall be deemed to involve a USQ if the probability of

occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, or if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created, or if the margin of safety as defined in the basis for any technical specification is reduced. The licensee shall maintain records of changes in procedures made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the SAR. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

Updated Final SAR Section 3.6, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, incorporated by reference MDS Report OS-73.2, Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station Units 1, 2, and 3, dated April 25, 1973. MDS Report OS-73.2 described a potential event involving a main feedwater or auxiliary steam line break in the turbine building that, by direct impingement, would make the three trains of safety-related 4160-volt switchgear inoperable. As a consequence of this event, all EFW and ES pumps would become inoperable. The MDS Report strategy included establishing emergency feedwater flow to a steam generator within 15 minutes and then beginning a plant cooldown. Prior to plant cooldown, operators must manually connect temporary power cables to an HPI pump. The MDS Report stated that these actions can be accomplished within a 30 minute time period.

Contrary to the above, the licensee made changes to a procedure as described in the SAR, and failed to perform a required written safety evaluation which provided the basis that the change did not involve a USQ. Specifically, the licensee revised Procedure IP/O/A/0050/001, Procedure to Provide Emergency Power to an HPI Pump Motor from the ASW Switchgear, Rev. 8, on November 20, 1998, by adding steps to go to the blockhouse and isolate electrical power to the 4160 volt switchgear, if that switchgear was inaccessible. The added steps included pulling two fuses and racking out six breakers in the blockhouse, which could take additional time to accomplish. The 10 CFR 50.59 evaluation failed to reference UFSAR Section 3.6, and lacked an adequate basis to support the conclusion that the change would allow timely connection of the HPI pump motor to the ASW switchgear. A licensee simulation of the revised procedure, after it was approved and issued, determined that performance of the procedure would take approximately 37 minutes. The 37 minutes exceeded the 30 minute time referenced in the UFSAR, and thus, represented a potential adverse affect on the ability to mitigate an auxiliary steam line or main feedwater line break in the turbine building. (02014)

This is a Severity Level IV violation (Supplement I).

The NRC has concluded that information regarding the reasons for Violations A and B, the corrective actions taken and planned to correct the violations and prevent recurrence and the date when full compliance was achieved has been adequately addressed on the docket as discussed in the letter transmitting this Notice of Violation (NOV), and in Inspection Report Nos. 50-269/98-15, 50-270/98-15, 50-287/98-15. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to

respond, submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001 with a copy to the Regional Administrator, Region II, U.S. Nuclear Regulatory Commission, Atlanta Federal Center, 23T85, 61 Forsyth Street S.W., Atlanta, Georgia, 30303-3415 and a copy to the NRC Resident Inspector at Oconee, within 30 days of the date of the letter transmitting this Notice.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you chose to respond, your response will be placed in the NRC Public Document Room (PDR). Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days after receipt.

Dated this 12th day of February 1999

LIST OF ATTENDEES

Nuclear Regulatory Commission:

L. Reyes, Regional Administrator, Region II (RII)
B. Mallett, Director, Division of Reactor Safety (DRS), RII
C. Casto, Deputy Director, Division of Reactor Projects (DRP), RII
V. McCree, Deputy Director, DRS, RII
A. Boland, Enforcement Officer, Enforcement and Investigations Coordination Staff (EICS), RII
A. Jones, Enforcement Specialist, EICS, RII
C. Evans, Regional Counsel, RII
C. Ogle, Chief, DRP Branch 1, RII
K. Landis, Chief, Engineering Branch, DRS, RII
R. Schin, Senior Reactor Inspector, DRS , RII
S. Freeman, Resident Inspector, DRP, RII
M. Thomas, Senior Reactor Inspector, DRS, RII
R. Bernhard, Senior Reactor Analyst, DRS, RII
B. Westreich, Senior Enforcement Specialist, Office of Enforcement (video conference)
H. Berkow, Director, Project Directorate, Office of Nuclear Reactor Regulation (NRR) (video conference)
D. LaBarge, Project Manager, NRR, (video conference)
C. Jackson, NRR, (video conference)
G. Galletti, NRR, (video conference)
J. Tatum, NRR, (video conference)
N. Saltos, NRR, (video conference)

Duke Energy Corporation:

W. McCollum, Site Vice President
J. Forbes, Station Manager
M. Nazar, Manager of Engineering
W. Foster, Safety Assurance Manager
E. Burchfield, Regulatory Compliance Manager
L. Azzarello, Design Basis Manager
G. Hamrick, Chemistry Manager
D. Brewer, Senior PRA Engineer
L. Vaughn, Assistant General Counsel

Enclosure 2

OPEN PREDECISIONAL ENFORCEMENT CONFERENCE AGENDA

OCONEE NUCLEAR STATION

JANUARY 26, 1999, 10:30 A.M.
NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS AND INTRODUCTIONS
   L. Reyes, Regional Administrator

II. SUMMARY OF THE ISSUES
   L. Reyes, Regional Administrator

III. NRC ENFORCEMENT POLICY
   A. Boland, Director
   Enforcement and Investigations Coordination Staff

IV. STATEMENT OF CONCERNS AND APPARENT VIOLATIONS
   V. McCree, Deputy Director
   Division of Reactor Safety

V. LICENSEE PRESENTATION

VI. BREAK / NRC CAUCUS

VII. NRC FOLLOWUP QUESTIONS

VIII. CLOSING REMARKS
   L. Reyes, Regional Administrator

Enclosure 3

STATEMENT OF APPARENT VIOLATIONS

A. TS 6.4.1 requires that the station shall be operated in accordance with approved procedures. Written procedures with appropriate instructions shall be provided for emergency procedures involving potential release of radioactivity and for actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety.

UFSAR Section 3.6, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, incorporated by reference MDS Report OS-73.2, Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station Units 1, 2, and 3, dated April 25, 1973. MDS Report OS-73.2 described a potential event involving an auxiliary steam or main feedwater line break in the turbine building that, by direct impingement, would make the three trains of safety-related 4160-volt switchgear inoperable. As a consequence of this event, all emergency feedwater (EFW) and engineered safeguards (ES) pumps would become inoperable. The licensee's mitigation strategy relied on manually connecting temporary cables and powering a high pressure injection (HPI) pump from the auxiliary service water (ASW) switchgear within 35 minutes of the event.

On November 5, 1998, written procedures with appropriate instructions were not provided for emergency procedures involving potential release of radioactivity or for actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety. IP/O/A/0050/001, Procedure to Provide Emergency Power to an HPI Pump Motor from the ASW Switchgear, Rev. 7, dated September 8, 1998, was the written procedure to manually connect temporary cables to power an HPI pump from the ASW switchgear. The procedure required first racking out the electrical breaker to the HPI pump at the safety-related 4160-volt switchgear. However, in the event of an auxiliary steam or main feedwater line break as described in MDS Report OS-73.2, the safety-related 4160-volt switchgear would be inaccessible because it would be in a steam environment. Instructions in the procedure were not appropriate and would not have enabled plant personnel to power an HPI pump within 35 minutes of the event.

Note: The apparent violations discussed in this PREDECISIONAL enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

B. 10 CFR 50.59 allows a licensee to make changes in procedures as described in the safety analysis report (SAR), without prior NRC approval, unless the proposed change involves a change in the technical specifications (TS) or an unreviewed safety question (USQ). A proposed change shall be deemed to involve a USQ if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, or if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created, or if the margin of safety as defined in the basis for any technical specification is reduced. The licensee shall maintain records of changes in procedures made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the SAR. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

Updated Final SAR Section 3.6, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, incorporated by reference MDS Report OS-73.2, Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station Units 1, 2, and 3, dated April 25, 1973. MDS Report OS-73.2 described a potential event involving a main feedwater or auxiliary steam line break in the turbine building that, by direct impingement, would make the three trains of safety-related 4160-volt switchgear inoperable. As a consequence of this event, all EFW and ES pumps would become inoperable. The licensee's mitigation strategy relied on starting a high pressure injection (HPI) pump, with alternate power from the auxiliary service water (ASW) switchgear, within 35 minutes of the event. This strategy included manually connecting temporary power cables to the HPI pump within 30 minutes.

The licensee made changes to procedures as described in the SAR, without prior NRC approval, that involved USQs and also failed to perform a required safety evaluation for a procedure change, as described in the following examples:

Note: The apparent violations discussed in this PREDECISIONAL enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

1. EP/O/A/1800/16, Loss of Power, Rev. of June 4, 1981, revised the loss of power procedure so that it would be entered on a loss of the 4160 volt main feeder bus, and not on a loss of the 4160 volt switchgear TC, TD, and TE. As a result of this change, emergency operating procedures no longer directed operators to power an HPI pump from the ASW switchgear to mitigate the auxiliary steam or main feedwater line break event described in MDS Report OS-73.2. The safety evaluation, dated March 28, 1981, incorrectly stated that the change may not increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR.

2. EP/O/A/1800/16, Loss of Power, Rev. of July 15, 1985, revised the loss of power procedure so that it no longer directed operators to power an HPI pump from the ASW switchgear on a loss of the 4160 volt main feeder bus. Instead, the procedure directed operators to start the standby shutdown facility (SSF) reactor coolant (RC) makeup pump (MUP). However, this event was outside of the licensing basis of the SSF RC MUP. At the time, the licensee had no analysis to demonstrate that this change would not increase the consequences of an accident. The safety evaluation, dated June 19, 1985, incorrectly stated that the change may not increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR.

Note: The apparent violations discussed in this PREDECISIONAL enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

3. IP/O/A/0050/001, Procedure to Provide Emergency Power to an HPI Pump Motor from the ASW Switchgear, Rev. 8, dated November 20, 1998, revised the procedure by adding steps to go to the blockhouse and isolate electrical power to the 4160 volt switchgear, if that switchgear was inaccessible. (The 4160 volt switchgear would be inaccessible in the event of an auxiliary steam or main feedwater line break in the turbine building as described in MDS Report OS-73.2.) The added steps included pulling two fuses and racking out six breakers in the blockhouse, which could take additional time to accomplish. However, the licensee failed to perform a required 50.59 safety evaluation of this change because the 50.59 screening incorrectly concluded that the change could not adversely affect any system, structure, or component necessary to operate the plant in accordance with the SAR. The 50.59 screening incorrectly failed to reference UFSAR Section 3.6, and incorrectly stated that the change would allow timely connection of the HPI pump motor to the ASW switchgear. However, a licensee simulation of the revised procedure, after it was approved and issued, determined that performance of the procedure would take approximately 37 minutes. The 37 minutes exceeded the 30 minute time described in the SAR and thus represented an adverse affect on the ability to mitigate an auxiliary steam line or main feedwater line break in the turbine building.

Note: The apparent violations discussed in this PREDECISIONAL enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

CHRONOLOGY OF OCONEE ABILITY TO MITIGATE A MAIN FEEDWATER OR AUXILIARY STEAM LINE BREAK THAT FAILS ALL THREE TRAINS OF SAFETY-RELATED 4160 VOLT POWER

Approximate Dates	Equipment Available	Procedures	Comments
7/73	U1 licensed, with **station ASW pump** & alternate power to an **HPI pump**.	Start station ASW pump & align alternate power to HPI pump on loss of the three 4160V switchgear.	Station ASW pump alone, started "within 15 minutes" can mitigate for an extended time. HPI pump alone, started "within 35 minutes," can also mitigate. (BUT an RCP seal LOCA was not considered)
12/73	**EFW unit cross-ties** installed & EFW line rerouted to make **TDEFWP** available (w/ local manual start).	Locally start TDEFWP or align unit cross-ties or use station ASW pump, & align alternate power to HPI pump on loss of the three 4160V switchgear.	Secondary cooling can now withstand a single failure and still be started "within 15 minutes." HPI can be started "within 35 minutes" for plant cooldown.
1974	U2 & U3 licensed.		
1979	Post-TMI EFW upgrade. Two MDEFWPs installed, but would be disabled by this event. EFW pumps designed to automatically start (but this event would disable auto start of TDEFWP).		
1981		EOPs changed align alternate power to HPI pump on loss of the 4160V feeder bus (but feeder bus is not lost in this event). **NOTE: FIRST EXAMPLE OF 50.59 APPARENT VIOLATION**	
1985	SSF installed, including **SSF ASW pump** and 30 gpm **SSF RC makeup pump**.	EOPs changed to start SSF RC makeup pump (& not HPI pump) on loss of the 4160V feeder bus. **NOTE: SECOND EXAMPLE OF 50.59 APPARENT VIOLATION**	SSF is licensed as a backup to other equipment (SSF not single failure-proof). SSF RC makeup pump is licensed for maintaining plant in hot standby while preventing an RCP seal LOCA. (Not licensed for mitigation of an auxiliary steam line break)

11/98		Procedure to align alternate power to an HPI pump was inadequate - it required first racking out HPI pump breaker at the inaccessible 4160V switchgear. **NOTE: INADEQUATE PROCEDURE APPARENT VIOLATION**	In response to this issue, the licensee performed an analysis showing that, if OTSG secondary cooling water was started within 15 minutes, and RCP seal leakage increased to 25 gpm each, then at least eight hours would have been available in which to rewrite the procedure and start an HPI pump.
11/98		Licensee revised procedure to align alternate power to an HPI pump, without V&V. **NOTE: THIRD EXAMPLE OF 50.59 APPARENT VIOLATION**	Using revised procedure, licensee walkdown determined that more than 35 minutes were needed to align alternate power to an HPI pump and start it.
1/99	Licensee identified some equipment / testing deficiencies: - EFW unit cross-tie valves were difficult to operate (two chain operators fell off, two valves were stuck closed) - HPI pump had never been tested with suction from SFP (tornado lineup) Also, NRC requested and licensee has been unable to locate records of testing HPI pumps when powered from ASW switchgear.	Licensee identified an operational deficiency: - In simulations, operators were unable to start OTSG cooling within 15 minutes by using TDEFWP local start or by using EFW unit cross-ties. Also, licensee agrees that starting OTSG cooling using the station ASW pump may well take even longer. Before the enforcement conference, the licensee plans to determine how long it would take.	In response to this information, the licensee performed an analysis showing that, if OTSG secondary cooling water was started within 30 minutes, then at least eight hours would have been available in which to rewrite the procedure and start an HPI pump.

Oconee Nuclear Station

Predecisional Enforcement Conference

January 26, 1999

Agenda

- Opening Remarks
- Apparent Violations
- Overview of High Energy Line Breaks (HELBs)
- Apparent Violation on Inadequate Procedure
- Apparent Violation on Inadequate 10 CFR 50.59 Safety Evaluations
- Assessment of Procedure Issues with Respect to Tornado Mitigation
- Regulatory Significance
- Closing Remarks

Apparent Violations

- Maintenance procedure IP/0/A/0050/001 was not adequate to accomplish the event mitigation as described in MDS Report OS-73.2
- Three examples of inadequate 10 CFR 50.59 Safety Evaluations:
  - » Revision to Loss of Power procedure in 1981 no longer directed operators to power an HPI pump from the ASW switchgear to mitigate the 300 psig auxiliary steam line break described in MDS Report OS-73.2
  - » Revision to Loss of Power procedure in 1985 directed operators to start SSF RC makeup pump instead of HPI pump, which was outside the licensing basis of the SSF RC makeup pump
  - » Revision to Maintenance procedure IP/0/A/0050/001 in November 1998 used a 10 CFR 50.59 screening evaluation as opposed to a 10 CFR 50.59 safety evaluation

HELB Risk Perspective

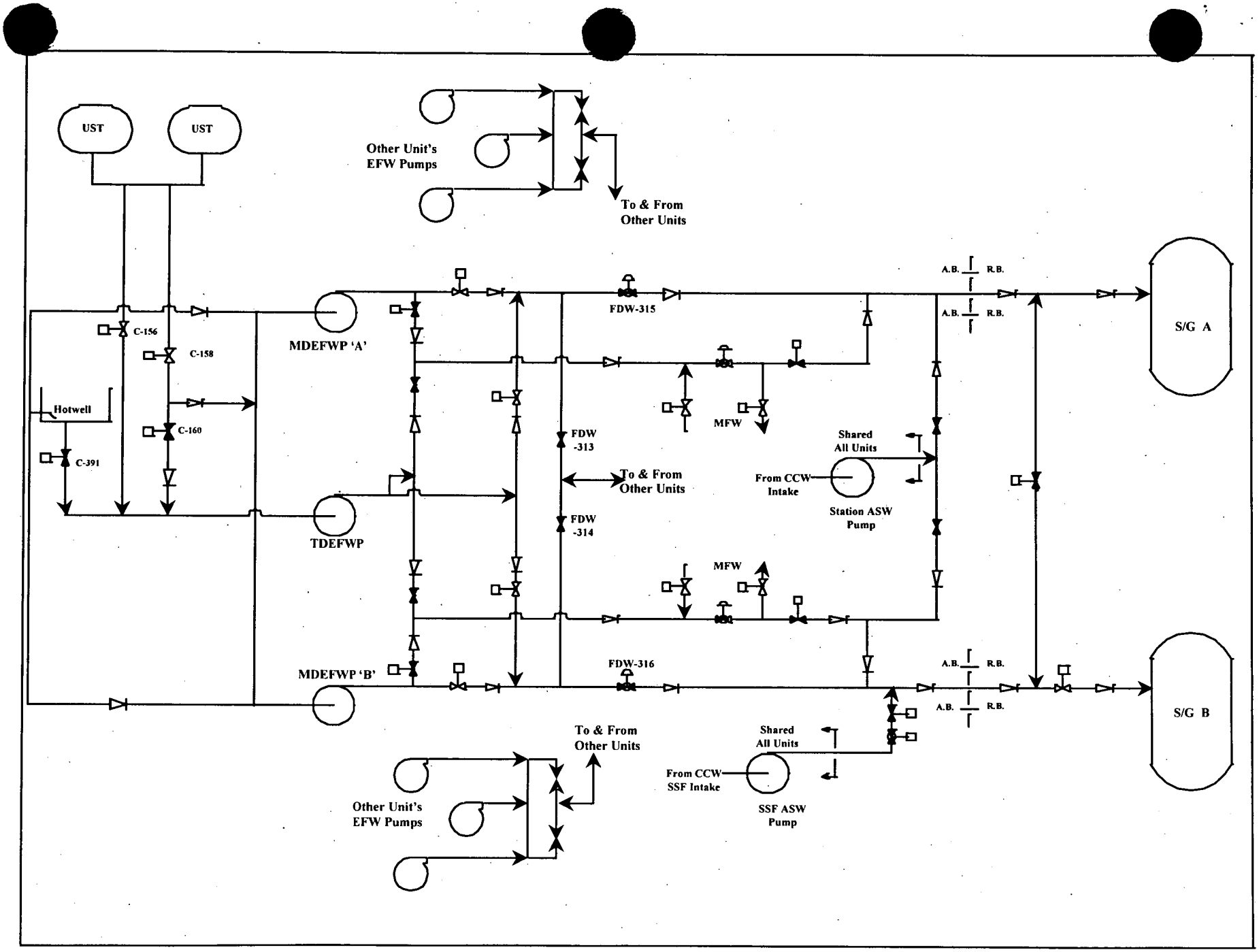
- Low likelihood of MFW or AS line failure near 4 kV switchgear
- Secondary Side Decay Heat Removal
  - » Turbine driven EFW pump on affected unit
  - » EFW from unaffected units (6 pumps)
  - » SSF ASW
- RCP seal cooling
  - » SSF RC Makeup pump (minimizes potential for seal leakage)
- Primary System Makeup
  - » HPI pump A or B from ASW switchgear
- Impact on core damage frequency for MFW or AS line break is estimated to be on the order of 1E-8

HELB Design Basis

- Scenarios of interest are a break of MFW line or auxiliary steam line near 4 kV switchgear TC, TD, and TE
- Initiating event leads to:
  - » Unit blackout from loss of 4kV switchgear TC, TD, and TE
  - » Loss of all feedwater
- Final safe shutdown approach in HELB Report MDS OS-73.2
  - » Restoration of emergency feedwater allows the plant to be maintained in hot shutdown condition for an extended period of time
  - » Prior to initiating plant cooldown, HPI is restored from ASW switchgear for coolant makeup and boron control
    - – Restoration of HPI within 30 minutes of event is not required to mitigate the event

Evolution of HELB Mitigation Strategies

Time Period	Change	Secondary Side Heat Removal	Primary System Makeup
7/73 to 11/73 (Unit 1 only)	Original design	Station ASW Pump	HPI pump A or B from ASW SG within 35 minutes
11/73 for Unit 1 and prior to startup for Units 2 and 3	Modifications implemented to route EFW piping through Turbine Building basement and cross-connect EFW between units	Affected unit's TDEFWP OR EFW cross-connect	HPI pump A or B from ASW SG prior to plant cooldown
6/81	Changed entry condition for Loss of Power procedure from loss of 4kV power to loss of 4kV main feeder bus power	Affected unit's TDEFWP OR EFW cross-connect	HPI pump A or B from ASW SG prior to plant cooldown
1/84	SSF operational	Affected unit's TDEFWP OR EFW cross-connect OR SSF ASW	SSF RC makeup pump within 10 minutes (Procedure directs start of HPI pump A or B from ASW SG regardless of SSF RC makeup pump status)
7/85	Loss of Power procedure revision (did not direct start of HPI pump from ASW SG if SSF RC makeup pump operated)	Affected unit's TDEFWP OR EFW cross-connect OR SSF ASW	SSF RC makeup pump within 10 minutes (Procedure did not direct start of HPI pump A or B from ASW SG unless SSF RC makeup pump fails)
11/98	• Loss of Power procedure revision (directed concurrent actions to start SSF RC makeup pump and HPI pump from ASW SG) • Revised HPI procedure to include contingency actions if 4kV SG TC, TD, and TE are not accessible	Affected unit's TDEFWP OR EFW cross-connect OR SSF ASW	SSF RC makeup pump within 10 minutes (Procedure directs HPI pump A or B from ASW SG regardless of SSF RC makeup pump status)


UST
UST
Other Unit's
EFW Pumps
To & From
Other Units
C-156
C-158
C-160
C-391
Hotwell
MDEFWP 'A'
TDEFWP
MDEFWP 'B'
FDW-315
FDW-316
FDW
-313
FDW
-314
MFW
MFW
To & From
Other Units
Shared
All Units
From CCW
Intake
Station ASW
Pump
A.B.
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A.B.
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R.B.
A.B.
R.B.
S/G B
To & From
Other Units
Shared
All Units
From CCW
SSF Intake
SSF ASW
Pump
Other Unit's
EFW Pumps

Summary of Recent HELB Analyses

- Questions were raised by Region II regarding the potential for excessive RCP seal leakage during a HELB
- Additional seal cooling capability during loss of power events added in mid 1980s with Standby Shutdown Facility (SSF) reactor coolant makeup pumps
- HELB analyses revised in November 1998 to address potential for excessive RCP seal leakage
  - » Seal leakage assumption of 25 gpm/pump
  - » Restoration of EFW assumed at 15 minutes
  - » Restoration of HPI assumed at one hour
  - » SSF RC makeup pump assumed unavailable
- Analyses demonstrated substantial margin to core uncovery

Summary of Recent HELB Analyses

- Validation work identified potential delay in restoring EFW within 15 minutes
  - » Initiated a PIP
  - » Operability evaluation successfully completed assuming EFW restoration at 30 minutes and HPI at 8 hours
  - » Implemented corrective actions to revise procedure to ensure restoration of EFW within 15 minutes
  - » Completed simulator and field validations to confirm 15 minute operator response
- Analyses confirm original design basis that hot shutdown can be maintained with EFW for an extended period of time, even with the assumption of increased seal leakage

HELB Summary

- Limiting HELB scenarios have been reanalyzed and operator action times have been validated for emergency feedwater
- HELB licensing and design bases will be updated to reflect the SSF and the results of ongoing work
- Safe shutdown capability for MFW line break or auxiliary steam line break dependent on restoration of feedwater
  - » Restoration of primary system makeup not time critical if feedwater is restored

Overview of Procedure Issues

- Procedural issues described in Inspection Report 98-15 did not impact the ability of ONS to safely shut down the plant in the event of a high energy line break
- Improvements in validation of operator actions being pursued
- Corrective actions being pursued to update Oconee HELB design and licensing basis

Apparent Violation for Inadequate Procedure

- Maintenance procedure for repowering HPI pump did not include steps to address the potential for 4kV switchgear TC, TD, and TE being inaccessible during a HELB
- Cause
  - » Failure to identify and address potential impact of HELB environmental conditions in original procedure and subsequent revisions

Apparent Violation for Inadequate Procedure

- Completed Corrective Actions
  - » Revised IP/0/A/0050/001 to remove power from 4 kV main feeder bus if 4 kV SG TC, TD, and TE are inaccessible
  - » Performed walkdown to validate procedure steps
  - » Performed HELB analyses to verify substantial time exists to establish HPI flow

Apparent Violation for Inadequate Procedure

- Planned Corrective Actions
  - » Periodic testing of HPI pump from ASW switchgear
  - » Comprehensive Program for Validation of Event Mitigation Operator Actions
    - – Initiative on risk significant operator actions
    - – Identification of operator actions in the licensing basis
    - – Validation of operator actions in the licensing basis accident analyses
    - – Improve technical review process for procedures used for event mitigation

10 CFR 50.59 Safety Evaluations (First Example)

- 1981 change to Loss of Power procedure: loss of 4kV main feeder bus power vs. loss of 4kV switchgear TC, TD, and TE
- Reason:
  - » Believe change attempted to enable operator to diagnose based on voltage indication
- Duke does not believe this change would have impacted the operators' response to a loss of power
- This procedure would have directed the operators to power an HPI pump from the ASW switchgear
- Therefore, Duke believes this change did not involve an unreviewed safety question

10 CFR 50.59 Safety Evaluations (First Example)

- Completed Corrective Actions:
  - » Procedure was clarified to reflect potential for unit blackout without a loss of main feeder bus
  - » Simulator validation exercises confirmed procedure will successfully direct actions necessary to mitigate HELBs

10 CFR 50.59 Safety Evaluations (Second Example)

- 1985 procedure change (Loss of Power): if successful with SSF RC makeup pump, not directed to rewire HPI pump
- If SSF successful: minimizes potential for excessive RCP seal leakage
  - » "Overcooling" from 300 psig auxiliary steam line break
- If SSF is not successful: procedure directs to rewire HPI pump
  - » At least 8 hours to perform

10 CFR 50.59 Safety Evaluations (Second Example)

- 1985 50.59 did not identify specific basis related to an AS line break scenario that might have supported the safety evaluation
- Entire industry has worked to improve documentation of 50.59 safety evaluations
- Oconee believes this change did not involve an unreviewed safety question

10 CFR 50.59 Safety Evaluations (Second Example)

- Completed Corrective Actions:
  - » Loss of power procedure was revised to direct starting both the SSF RC makeup pump and the HPI pump from the ASW switchgear
- Planned Corrective Actions:
  - » Update design and licensing basis to reflect the use of the SSF in HELB mitigation

10 CFR 50.59 Safety Evaluations (Third Example)

- Revision to Maintenance procedure IP/0/A/0050/001 in November 1998 used a 10 CFR 50.59 screening evaluation as opposed to a 10 CFR 50.59 safety evaluation
- Cause:
  - » Process does not require engineering review for all changes to event mitigation procedures
    - – Elements of HELB safe shutdown actions were not described in UFSAR

10 CFR 50.59 Safety Evaluations (Third Example)

- Completed Corrective Actions:
  - » Performed field walkdown validations of procedure change
  - » Performed simulator validations of HELB mitigation strategy

10 CFR 50.59 Safety Evaluations (Third Example)

- Planned Corrective Actions
  - » Performing 10 CFR 50.59 safety evaluation
  - » Update UFSAR to include relevant HELB information
  - » Identifying operator actions in the licensing basis that have time constraints
  - » Validate operator actions through integrated simulator/field validations
  - » Improve technical review process for procedures used for event mitigation

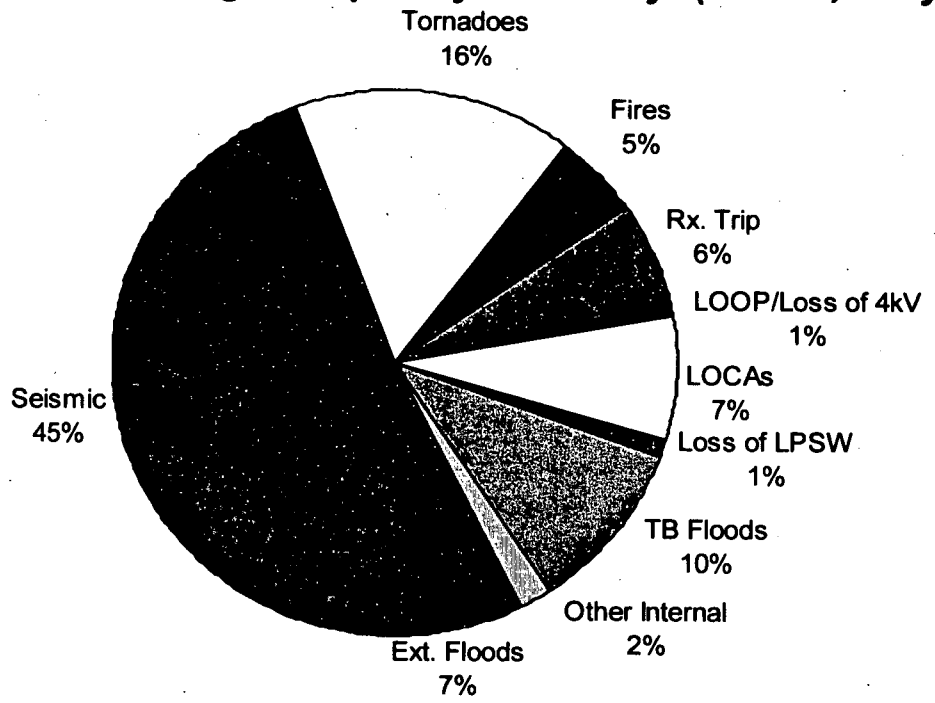
Tornado

- Role of HPI Pump in Tornadoes
  - » The *Natural Disaster* Procedure has steps which will:
    - – Pre-stage operators during tornado to activate SSF in a timely manner
    - – Pre-stage I & E personnel to configure ASW switchgear during tornado
  - » IP/0/A/0050/001, *Providing Emergency Power to an HPI Pump Motor From the ASW Switchgear*, is performed either per *Natural Disaster* Procedure or *Loss of Power* Procedure.
  - » HPI pump may be required for heat removal during certain PRA tornado scenarios

Tornado

- Risk Perspective of Tornado
  - » Tornado risk is about 16% of Oconee CDF (based on Rev. 2 of Oconee PRA)

Total Core Damage Frequency = 8.9E-5/yr (1 in 11,000 yrs)



Tornado

- Risk Perspective of using HPI Pump powered from ASW Pump Switchgear during a tornado:
  - » Current PRA assumes:
    - – 10% chance of failure to complete action for all cases (with secondary side heat removal (SSHR) or without SSHR)
  - » However, the time available to complete the action is significantly longer for sequences with SSHR. Therefore, a sensitivity study was performed.
  - » Assumptions:
    - – 10% chance of failure to start HPI pump for cases with SSHR
    - – 100% chance of failure to start HPI pump for cases without SSHR

Tornado

- Risk Perspective of using HPI Pump powered from ASW Pump Switchgear during Tornado Mitigation
  - » Results:
    - – 1% (1E-6/yr) change in Oconee's Core Damage Frequency
  - » Conclusion: "Very small change" per Regulatory Guide 1.174
  - » Impact of HPI pump procedure issues on tornadoes is not significant

Regulatory Significance

- Oconee has a strong focus on improving the design and licensing basis
  - » Numerous initiatives in Recovery Plan Design Basis focus area
  - » UFSAR Chapter 15 accident analyses reanalyzed and under staff review
- Safety significance has been thoroughly evaluated and does not appear to meet criteria for escalated enforcement
  - » Analyses confirm original design basis that EFW can maintain plant in a safe shutdown condition for an extended period of time
  - » Restoration of HPI is not a time critical action and procedural errors are not risk significant
- Prompt and comprehensive corrective actions to address issues
  - » Validation of operator actions in licensing basis accident analyses
  - » Updating Oconee HELB design and licensing basis
- Escalated enforcement not necessary to assure improvement of the design and licensing basis of Oconee

Closing Remarks

Duke Power Company

10 CFR 50.59 USQ EVALUATION

(NSD 209-11-1)

(1) STATION(s):	(2) UNITS(s):	(3) TYPE OF ACTIVITY:	
☒ Oconee Nuclear Station	☒ Unit 1	☐ Nuclear Station Modification	☐ Operability Evaluation
☐ McGuire Nuclear Station	☒ Unit 2	☐ Minor Modification	☐ Test or Experiment
☐ Catawba Nuclear Station	☒ Unit 3	☒ Procedure	☐ UFSAR Change
☐ _____	☐	☐ Other _____	☐ Temporary Modification

(4) DOCUMENT NUMBER, REV. NUMBER and DESCRIPTION: IP/0/A/0050/001 Rev. 8 Provide Emerg. Power to an HPI Pump Motor from ASW Switchgear

(5) SCREENING FOR INCREASED MANAGEMENT INVOLVEMENT (NSD 209-11-2 & 11-3)

1. Is the activity being evaluated a procedure, test, experiment, or evolution? If "No," proceed to Part (6). If "Yes", continue to the next question. ☒ Yes ☐ No

2. Does the item involve infrequently performed tests or evaluations that have the potential to significantly degrade the level of nuclear safety? If "Yes," consult with the Superintendent of Operations to determine if additional controls are necessary. ☐ Yes ☒ No

Procedure Qualified Reviewer _____ Date: _____

Superintendent of Operations _____ Date: _____

(6) SAFETY ANALYSIS REPORT DOCUMENT REVIEW (NSD 209-11-3)

1. Will technical specification changes be required? * If the answer is "Yes," then the part of the activity requiring a change to the Technical Specifications cannot be performed under 10 CFR 50.59 regulations nor implemented without prior NRC approval. ☐ Yes ☒ No

2. TECHNICAL SPECIFICATIONS AND ASSOCIATED BASES CONSULTED: See Attached Sheets

3. UFSAR SECTIONS CONSULTED: See Attached sheets

4. OTHER SAR DOCUMENTS CONSULTED: See Attached Sheets

SAR DOCUMENT SECTIONS WHICH NEED REVISION: NONE, See conclusion

(7) SAFETY REVIEW (NSD 209-11-4)

Safety Review performed and documented as required per Section 209.11.4 and 209.12? ☒ Yes

(8) EVALUATION OF UNREVIEWED SAFETY QUESTIONS (NSD 209-11-5)

May the proposed activity:

1. Increase the probability of occurrence of an accident previously evaluated in the SAR? ☐ Yes ☒ No
2. Increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR? ☐ Yes ☒ No
3. Increase the consequences of an accident previously evaluated in the SAR? ☐ Yes ☒ No
4. Increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR? ☐ Yes ☒ No
5. Create the possibility for an accident of a different type than any evaluated previously in the SAR? ☐ Yes ☒ No
6. Create the possibility for a different type of malfunction of equipment important to safety than any evaluated previously in the SAR? ☐ Yes ☒ No

Does the proposed activity:

7. Reduce the margin of safety as defined in the basis for any Technical Specification? ☐ Yes ☒ No

* If the answer to any of the above seven (7) unreviewed questions in Part (8) is "Yes," the change cannot be performed under 10CFR 50.59 regulation nor implemented without prior NRC approval.

The Design and Safety Considerations in NSD 209 Table 209-2 have been considered, as appropriate. ☒ Yes

(9) DOCUMENTATION (NSD 209-11-6 & 209-12)

Activity Description, Safety Review, Justification of Answers to the 7 USQ Questions in Part 8, Conclusion, Summary for Annual Report, & References attached? ☒ Yes

(10) APPROVAL (NSD 209-11-7)

Preparer: Doug Ripko Date: 1-19-98

Qualified Reviewer: Stephen L Meden Date: 2-3-98

The Qualified Reviewer is responsible for assuring a copy of the completed evaluation is sent to Site Regulatory Compliance and the Nuclear General Office NSRB Staff (NSRB Staff mail code - EC05N). Date Sent 2-3-98

Enclosure 5

FOR INFORMATION ONLY - Design and Safety Considerations **

_____ QA condition of SSCs
_____ Containment integrity
_____ Seismic analysis and mounting
_____ Seismic qualification of equipment
_____ Environmental qualification
_____ Materials compatibility
_____ Single failure criteria
_____ Separation criteria
_____ Equipment accessibility
_____ Control room habitability
_____ Fire protection and fire loads
_____ Release of radioactive gases and liquids
_____ Potential for normal effluents to become radioactive
_____ Possibility of operator error
_____ Design bases, assumptions, and values used in the SAR
_____ Missile protection (internal and external)
_____ Effects of natural phenomena (flood, wind, lightning)
_____ Postulated pipe breaks and new spray zones
_____ Potential for internal plant flooding
_____ Electrical failure
_____ Mechanical failure
_____ Control signal failure
_____ Plant security
_____ 10CFR 50 Appendix R review
_____ Overpressure protection
_____ Pipe class breaks
_____ Heavy loads (NUREG-0612)
_____ Fuse and breaker protection and coordination
_____ Power system and cable loading
_____ Electrical penetration protection
_____ Diesel generator loading
_____ Diesel generator load sequencing
_____ HVAC air flow restrictions
_____ Safety/nonsafety circuit isolation
_____ Adequate pneumatic pressure to a device
_____ Valve motor torque requirements
_____ Fuel movement considerations
_____ Mode change considerations
_____ Effect on the other unit(s) or train(s)
_____ Valve types due to Type C reverse flow testing
_____ Human factors considerations (e.g., control room)
_____ Reactivity Management (See NSD 304)
_____ SQUG Review
_____ New surveillance testing requirements
_____ Common cause failures (analog-to-digital replacements)
_____ Valve pressure locking/thermal binding
_____ Instrument grounding due to test equipment
_____ Test instrument compatibility
_____ SAR specified testing requirements
_____ Procedure step sequence
_____ New sources of debris for the containment sump

**Note: This list is not all inclusive. This information is from NSD 209 Table 209-2 and is provided to aid in the thought process for evaluating an activity for screening or USQ Evaluations. The list does not have to be included in the final documentation of the Evaluation and is not considered as part of the evaluation forms.

10CFR50.59 USQ Evaluation
IP/0/A/0050/001 Revision 8

Activity Description

Revision 8 to procedure IP/0/A/0050/001 is adding guidance for isolating the normal power path of the HPI pump motors prior to connecting the motor to the Auxiliary Service Water (ASW) switchgear. The additional guidance is to isolate the normal circuit by opening the feeders for 4160VAC switchgear if the HPI pump motor breaker is not accessible. If the feeder breakers for the 4160 VAC switchgear are not accessible, steps were added to isolate the main feeder bus. The isolations are a safety precaution if power is restored, since the existing HPI pump motor cable must be disconnected prior to connection of the HPI pump motor to the ASW switchgear. This procedure is to connect a HPI pump motor to the ASW switchgear when the 4160 VAC switchgear power sources normally powering the HPI pump motors are not available. Adding the additional guidance of alternate isolation points would only be performed in an emergency situation and therefore does not affect the normal operation of plant equipment.

Safety Review

In a High Energy Line Break (HELB) scenario, as outlined in MDS report OS-73.2 and Supplement 1 to MDS report OS-73.2, a main feedwater or auxiliarly steam line break could result in a loss of main and emergency feedwater along with 4160 volt switchgears TC, TD, and TE. In this scenario, original analysis relied on restoration of High Pressure Injection (HPI) to prevent the core from uncovering without the aid of secondary cooling until a redundant emergency feedwater flow path could be installed. Once the redundant emergency feedwater flow path was installed, analysis showed that emergency feedwater would be established to a steam generator and would be sufficient for decay heat removal to allow the core to remain covered and maintain the reactor at hot shutdown conditions for an extensive period of time. Once power was restored to a high pressure injection pump, the reactor coolant system then would be cooled. The restoration of a high pressure injection pump would require manually restoring power by connecting the motor to the ASW switchgear. As part of the response it was stated that the actions could be accomplished in 30 minutes.

Adding the additional isolation points in procedure IP/0/A/0050/001 if the HPI pump breakers are not accessible adds additional time in performing the task of connecting a high pressure injection pump motor to the ASW switchgear. The time to accomplish the additional isolations to ensure personnel safety makes the time to accomplish the task approximately 38 minutes. The 38 minutes was the most restrictive time required to perform the activity during actual plant walk downs of the procedure with the additional isolation steps.

Although the time required to complete the connection of the HPI pump to the ASW switchgear exceeds the time stated in the MDS report OS-73.2, the completion of the activity in thirty minutes is not required for mitigation of a HELB. After design changes ensured alternate sources of secondary side cooling, the analysis relies on emergency feedwater for mitigation. By establishing emergency feedwater to the steam generator, additional time can elapse until HPI is started and the core would not be uncovered. The additional analysis performed under calculation OSC-7299 concluded the HPI pump did not have to be started until 1 hour and preliminary analysis performed under PIP 99-0057 concluded HPI could wait to be initiated up to 8 hours.

Connection of a HPI pump to the ASW switchgear is part of the emergency procedures in response to a tornado to provide RCS makeup. However, it is the redundancy and diversity of Oconee systems which is relied upon in place of extensive tornado protection requirements. The time to establish secondary cooling dictates when HPI is required. The main avenue to isolate the HPI pump is still at the switchgear cubical for the pump. If secondary cooling is established sooner, HPI makeup can wait. Since in a tornado it is not possible to predict all scenarios, all equipment availability is not known. During mitigation it may be prudent to isolate the HPI pump at the alternate locations based on the damage and time when secondary cooling was establishing. But as stated earlier, the redundancy and diversity of systems for secondary cooling is relied upon in a tornado scenario. The additional steps for isolation of the HPI pump motors do not affect the ability to supply water to the steam generators via the other Unit's Emergency Feedwater, Station ASW, or SSF ASW.

Evaluation of unreviewed safety questions

1. May the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

   No. The actions taken in the procedure are in response to mitigation of an accident that has occurred. The additional actions would not increase the probability of occurrence of an accident.

2. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

   No. The additional isolations added are for personnel safety. The actions to isolate the switchgears would not cause a malfunction of equipment. The addition of steps in the procedure for isolation if the 4160 switchgears are not accessible are not performed unless HPI is lost. TC, TD, and TE switchgears are not available. The additional isolations isolate the switchgears versus the pump itself when connecting the HPI pump to the ASW switchgear. Since the switchgears are not available in the HELB scenario, isolation of the switchgears does not affect any other equipment. Depending on the extent of the tornado damage, actions to isolate the pumps may or may not be accomplished at the pump cubicle. If additional isolations are performed, the actions would not affect other systems for mitigation such as other Unit Emergency Feedwater, the Unit's ASW, or SSF systems. Therefore the additional isolations added do not increase the probability of occurrence of an accident.

3. May the proposed activity increase the consequences of an accident previously evaluated in the SAR?

   No. The additional steps for the isolations do add time for connection of the HPI pump to the ASW switchgear and therefore increases the time for HPI initiation. However, an analysis was performed under calculation OSC-7299 which shows that increasing the time for establishing HPI does not uncover the core for an HELB. The additional isolation points for the HPI pumps do not affect the timing for providing secondary water via the other Unit's Emergency Feedwater, Station ASW, or SSF ASW.

4. May the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

   No. The activity of performing additional isolation when connecting an HPI pump to the ASW switchgear in the event the switchgears are not accessible would not increase the consequences of a malfunction of equipment.

5. May the proposed activity create the possibility for an accident of a different type than any evaluated previously in the SAR?

   No. As stated above, the TC, TD, and TE switchgears are assumed to be lost during a HELB. Therefore isolating the switchgears would not affect equipment since it is already assumed not to be available. In a tornado event, depending on the extent of the tornado damage, the isolation may or may not be performed in the same manner at the pump breaker as was done prior to the change. If additional isolations are performed, the actions would not affect other systems for mitigation such as other Unit Emergency Feedwater, the Unit's ASW, or SSF systems. Therefore the additional isolations added do not create the possibility for an accident of a different type not evaluated in the SAR.

6. May the proposed activity create the possibility for a different type of malfunction of equipment important to safety than any evaluated previously in the SAR?

   No. The only actions added to this procedure are for isolation at different locations if the 4160 switchgears are not accessible. The actions of connecting the HPI pump to the ASW switchgear is not being revised. Since all the actions for physical connection of the HPI pump to the ASW switchgear is the same, no malfunction of equipment would be created.

7. May the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

   No. The actions would be performed in a HELB scenario if the switchgear is not accessible. The additional isolation of equipment assumed not to be available does not reduce the margin of safety defined in the basis of any Technical Specification.

Conclusion

A unreviewed safety question does not exist for this procedure change. No changes to Technical Specifications or UFSAR is required. The UFSAR does not discuss time requirements of the activity. Only in correspondence to the NRC is the time required for connection of an HPI pump to the ASW switchgear mentioned.

Although this procedure change may increase the time before an HPI pump is available for certain scenarios, the ability to keep the core cool and covered is unaffected. Initiation of feedwater is the more time critical function. It is unaffected by this procedure change.

10CFR50.59 USQ Evaluation
IP/0/A/0050/001 Revision 8

Summary for 10CFR50.59 Annual Report

The evaluation performed for IP/0/A/0050/001 revision 8 was to determine if there were any unreviewed safety questions associated with adding the additional isolation points for the High Pressure Injection (HPI) pumps. In a High Energy Line Break (HELB) scenario, a main feedwater or auxiliary steam line break could result in a loss of TC, TD, and TE switchgears by direct water or steam impingement. Therefore, alternate points for isolation were added since accessing the switchgears would not be possible.

Although the time required to complete the connection of the HPI pump to the Auxiliary Service Water (ASW) switchgear exceeds the time stated in the MDS report OS-73.2, the completion of the activity in thirty minutes is not required for mitigation of a HELB. Analysis relies on emergency feedwater for mitigation. With establishing emergency feedwater to the steam generator, additional time can elapse until HPI is started and the core would not be uncovered. The analysis performed under calculation OSC-7299 concluded the HPI pump did not have to be started until 1 hour and preliminary analysis performed under PIP 99-0057 concluded HPI could wait to be initiated up to 8 hours.

Connection of a HPI pump to the ASW switchgear is part of the emergency procedures in response to a tornado to provide RCS makeup. However, it is the redundancy and diversity of systems which is credited for tornado mitigation. The time establishing secondary cooling dictates when HPI is required. The main avenue to isolate the HPI pump is still at the switchgear cubical for the pump. If secondary cooling is established sooner, HPI makeup can wait. Since in a tornado it is not possible to predict all scenarios, all equipment availability it is not known. During mitigation it may be prudent to isolate the HPI pump at the alternate locations based on the damage and time when secondary cooling was establishing. But as stated earlier, the redundancy and diversity of systems for secondary cooling is the mitigation strategy for a tornado scenario. The additional isolation points for the HPI pumps do not affect the ability to supply water to the steam generators via the other Unit's Emergency Feedwater, Station ASW, or SSF ASW.

A unreviewed safety question does not exist for this procedure change. No changes to Technical Specifications or UFSAR is required. The UFSAR does not discuss time requirements of the activity. Only in correspondence to the NRC is the time required for connection of an HPI pump to the ASW switchgear mentioned.

Although this procedure change may increase the time before an HPI pump is available for certain scenarios, the ability to keep the core cool and covered is unaffected. Initiation of feedwater is the more time critical function which is unaffected by this procedure change.

References

Technical Specifications 3.2, 3.3, 3.4, 3.7, 3.18 9/24/98 Revision

UFSAR 3.1.2, 3.2.2, 3.6, 8, 9.6.3.1, 9.6.3.3, 9.6.4.2, 10.4.7, 13.5.2.1.2, 15.8 12/31/1997

SLC 16.9.9 10/14/98

Letter from Leonard A Wiens (NRC) to H.B. Tucker (Duke Power) Dated July 28,1989. Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System

Letter from A. C. Thies (Duke Power) to Angelo Giambusso (NRC) Dated April 25, 1973. Report OS-73.2 Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment

Letter from A. C. Thies (Duke Power) to Angelo Giambusso (NRC) Dated June 22, 1973. Supplement 1 to Report OS-73.2 Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment

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DBD OSS-254.00-00-4005, Design Basis Events, Rev. 4 October 5, 1995

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