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

February 23, 1999

LICENSEE: Duke Energy Corporation

FACILITY: Oconee Nuclear Station, Units 1, 2, and 3

SUBJECT: SUMMARY OF THE EMERGENCY FEEDWATER SYSTEM MEETING ON FEBRUARY 8, 1999, RELATED TO NRC INSPECTION REPORT 50-269/99-10, 50-270/99-10, AND 50-287/99-10

On February 8, 1999, the NRC met with representatives of the Duke Energy Corporation (DEC) staff at the NRC headquarters in Rockville, Maryland, to discuss certain design and licensing issues related to the Emergency Feedwater (EFW) System. These issues were described as unresolved items in NRC Inspection Report 50-269/99-10, 50-270/99-10, and 50-287/99-10, which was issued on January 26, 1999. Enclosure 1 is a list of the individuals who attended the meeting, Enclosure 2 is the handout material that was supplied by DEC, and Enclosure 3 is the handout material that was prepared by the NRC.

The topics discussed included such issues as documentation history of the EFW system; single failure considerations; design, operation, and failure consequences of the hotwell makeup valve from the upper surge tank; modifications implemented in the past related to the EFW system; EFW pump runout protection; overview of the design of the main feedwater, Standby Shutdown Facility, auxiliary service water (ASW) system, and station ASW systems as they relate to the reliability of the EFW system; implementation of post-TMI action plan items that were designed to improve the EFW system; ability to cross-connect EFW systems between units if needed; EFW probabilistic risk assessment review; and Final Safety Analysis Report changes that are needed to clarify the design and licensing bases of the EFW system.

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

Enclosures:
1. Attendance List
2. DEC Handout
3. NRC Handout
4. Miscellaneous Information

cc w/enclosures:
See next page

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ATTENDANCE RECORD

PURPOSE: Meeting with Duke Energy Corp. to discuss Oconee EFW issues

DATE: 2/8/99

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Vonna Ordaz	NRR/DSSA/SPLB
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MIKE BARRETT	DUKE - PRA GROUP
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Lenny Azzarello	Duke/ONS Engineering
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ATTENDANCE RECORD

PURPOSE: Meeting with Duke Energy Corp. to discuss Oconee EFW issues

DATE: 2/8/99

NAME (PLEASE PRINT)	AFFILIATION
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Victor M. McCree	RII / Dep Dir DRS
DAVID E. LABARGE	NRR / PD II-2
* Walt Rogers	RII / Sr. Rx Analyst
* Bruce Mallett	RII / Dir / DRS
* Chuck Casto	RII / Dep Dir DRP

* Via video connection

Oconee Nuclear Station

EFW Meeting with NRC

February 8, 1999

Agenda

- Opening Remarks
- Inspection Report 99-10 Issues
- Overview of Oconee Feedwater Systems
- Emergency Feedwater (EFW) Reliability
- Licensing Evolution of EFW System
- Engineering Perspective
- Closing Remarks

Inspection Report 99-10 Issues

- EFW should withstand any single-active failure
- Valve C-187 failing to close prevents EFW from performing its intended function
- Previous modifications to correct specific issues represented missed opportunities to resolve the C-187 design deficiency
- EFW runout protection

Duke Perspective

- EFW system is reliable
- EFW system not required to withstand all single-active failures
- Risk significance of single failure issue is low
- Oconee is evaluating increasing the design margin of the system
- EFW runout being addressed

Overview Of Oconee Feedwater Systems

- Main Feedwater (MFW)
  - » Two turbine driven MFW pumps per unit
  - » Three motor driven hotwell pumps per unit
  - » Three motor driven condensate booster pumps per unit
- Emergency Feedwater (EFW)
  - » Two motor driven pumps and one turbine driven pump per unit
- EFW from other units (cross-connect)
- Standby Shutdown Facility Auxiliary Service Water (SSF ASW)
  - » One motor driven pump capable of feeding all three units
- Station Auxiliary Service Water
  - » One motor driven pump capable of feeding all three units

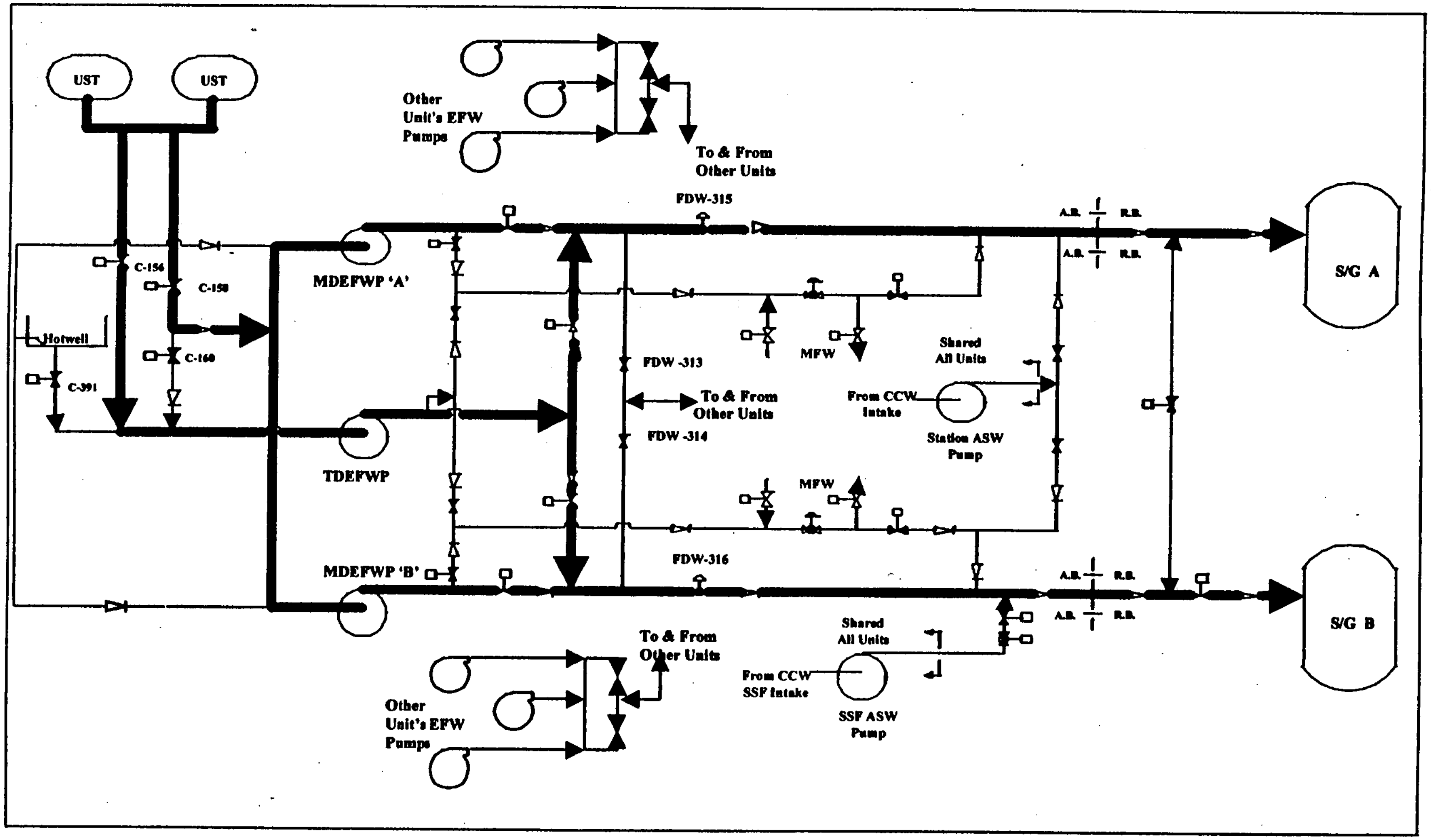
Main Feedwater

- Two turbine driven MFW pumps per unit
- Suction provided by hotwell pumps and condensate booster pumps
- Suction source is hotwell
- MFW pumps trip on loss of offsite power
- MFW system usually available on reactor trip

Emergency Feedwater

- Two motor driven pumps and one turbine driven pump per unit
- Safety-related, non load shed power for motor driven EFW pumps
- Turbine driven EFW pump is independent of AC power
- Auto-start on:
  - » Low MFW pump hydraulic control oil pressure
  - » Low steam generator level (motor driven pumps)
  - » AMSAC
- Upper Surge Tank (UST) is initial suction source

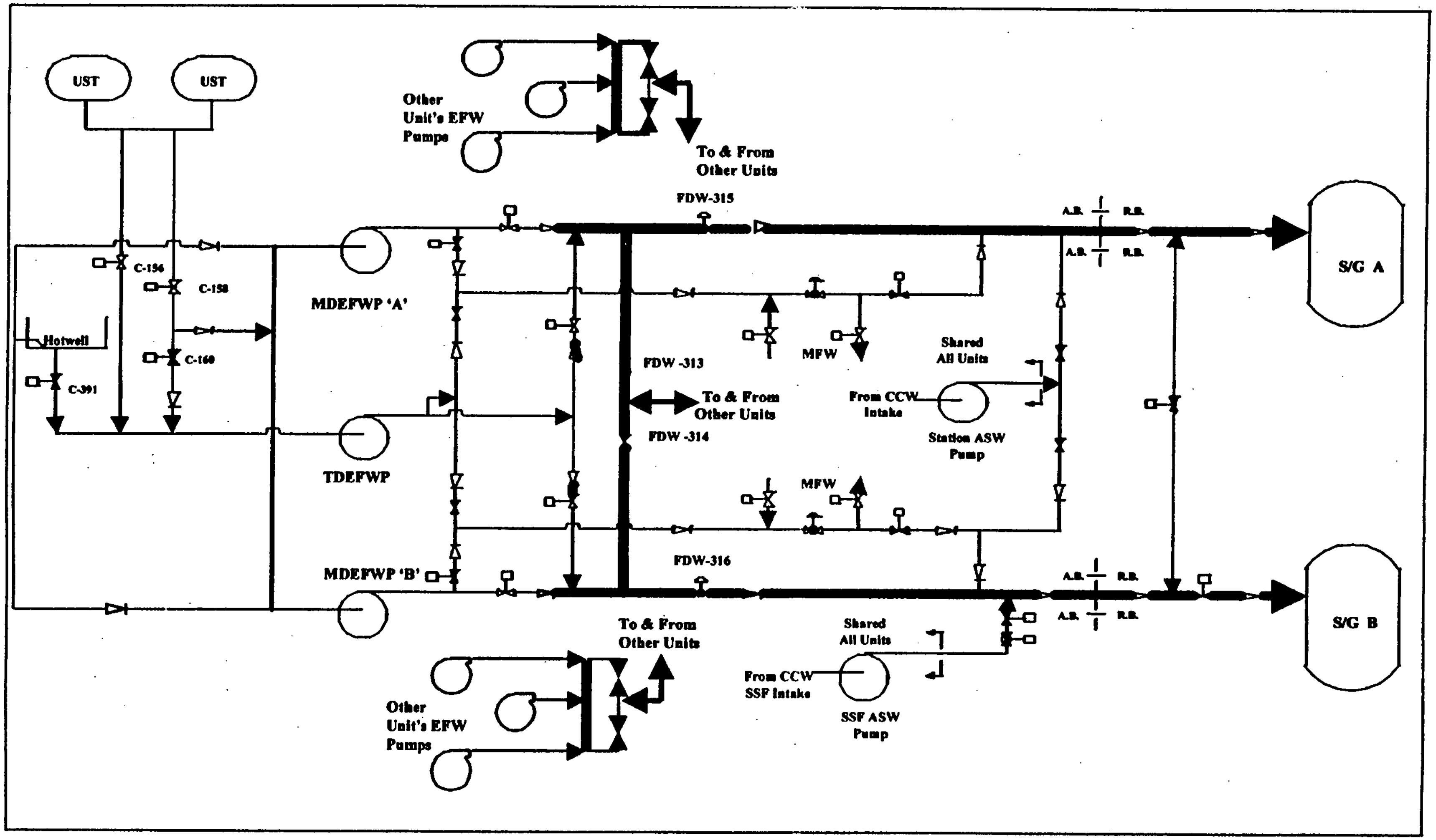
Emergency Feedwater System



Emergency Feedwater

- Plant response following a typical loss of MFW event:
  - » All three EFW pumps start
  - » Feedwater injected through SG upper nozzle
  - » SG level controlled by EFW control valves
  - » Secure turbine driven EFW pump if both motor driven pumps are running
  - » Makeup to UST is initiated by operators
  - » If UST makeup is unavailable, manual alignment to hotwell on low UST level
- Cross-connects capable of providing EFW flow to either SG from alternate units

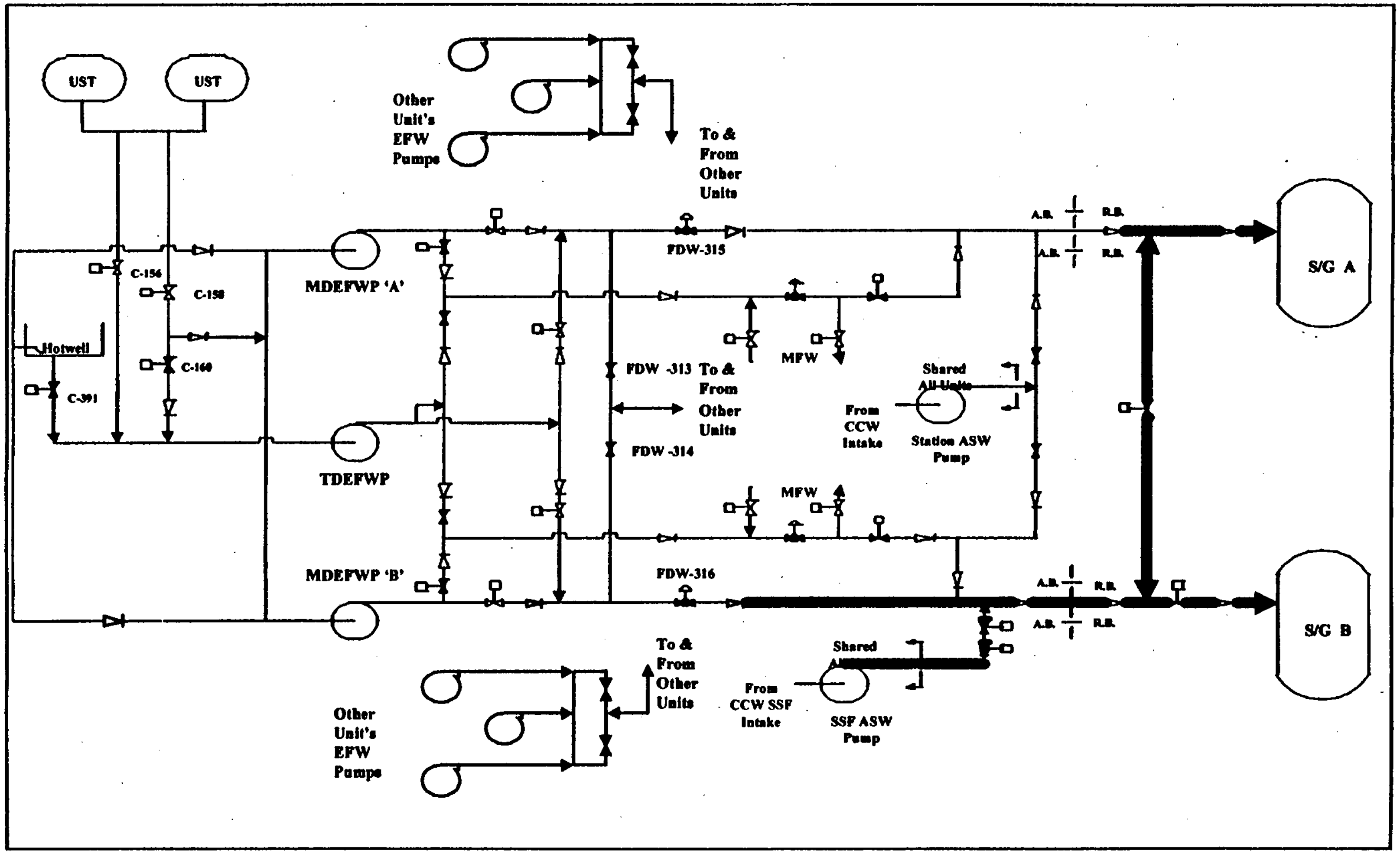
EFW System Cross-Connects



SSF ASW

- One motor driven pump capable of feeding all three units
- High head pump can supply all three units at full SG pressure
- Dedicated power supply from SSF diesel generator
- Manually aligned within 14 minutes from SSF control room
- Suction source is lake water in Unit 2 CCW piping
- Capable of maintaining hot shutdown for 72 hours
- Seismically designed
- Designed for tornado wind loads and missiles (portions in penetration rooms not fully tornado proof)

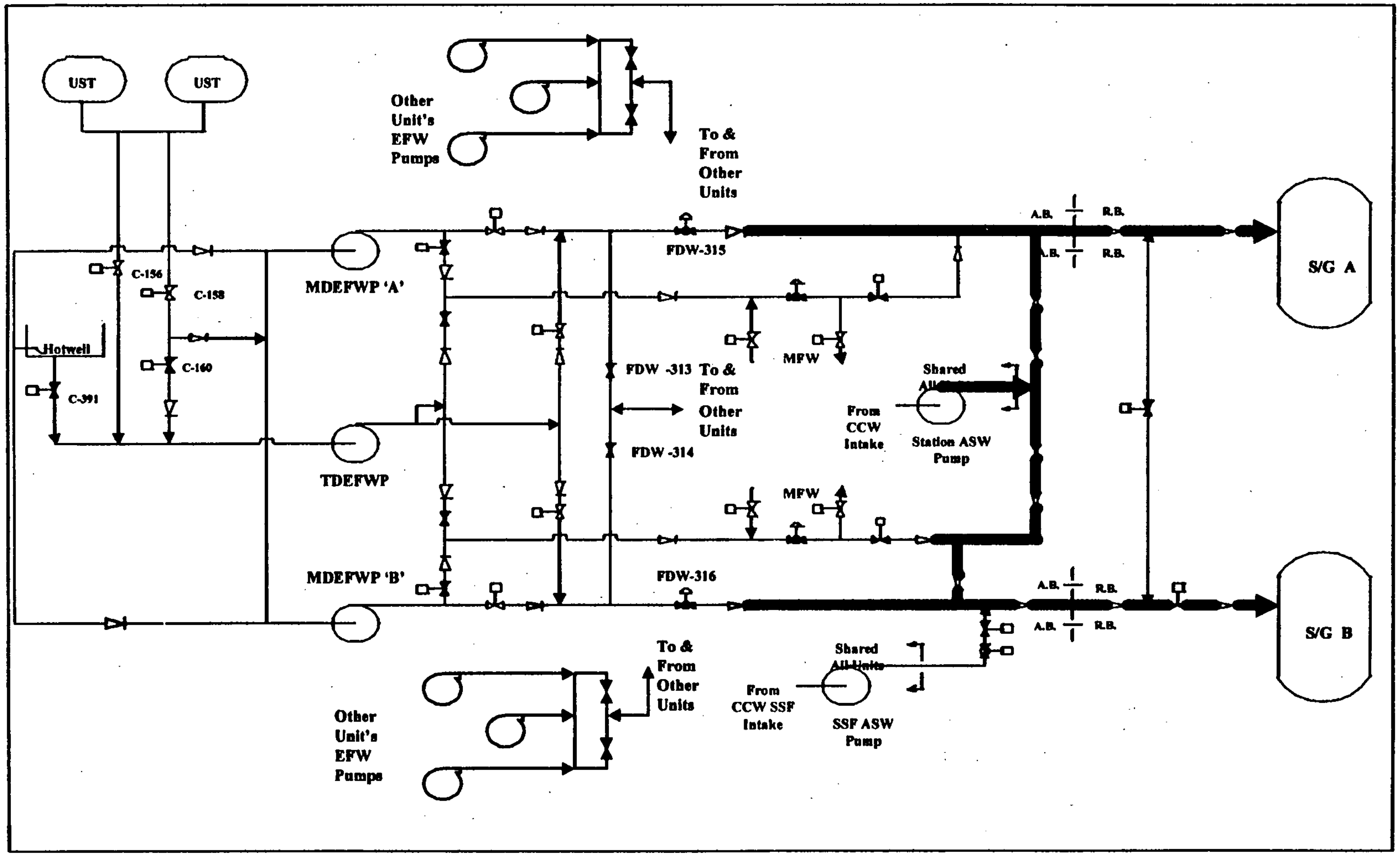
SSF ASW System



Station ASW

- Tornado-protected pump in basement of Auxiliary Building
- Powered from Keowee underground path or CT-5 from Lee via the standby bus
- Capable of feeding all three units at low SG pressures
- Requires manual start and system alignment in Auxiliary Building

Station ASW System



Secondary Side Cooling Prioritization

- MFW (two pumps)
- EFW (three pumps)
  - » Auto start or manual start from control room
- Condensate booster pump/hotwell pump combination can be used if SG pressure is less than 500 psig
- EFW cross-connected to alternate unit (six pumps)
  - » Requires valve alignments in Turbine Building basement
- SSF ASW (one pump)
  - » Started at SSF control room
- Station ASW (one pump)
  - » Started locally in Auxiliary Building basement

Evolution of EFW System

Time Frame	Change	Impact
1973	One TDEFWP per unit	Original Design
1973-1974	Added EFW cross-connects Rerouted EFW piping through Turbine Building basement	Resolved HELB vulnerability Allowed EFW to be fed from alternate unit
1979-1980	Added two motor driven pumps per unit	Improved redundancy and diversity of design
1979-1980	Implemented auto-actuation circuitry and safety-grade control system	Improved automatic response of system
1984	SSF operational	Improvement in overall reliability of SG heat removal function
1986	Lowered elevation of suction source from hotwell for motor driven EFW pumps	Improved NPSH and increased available hotwell inventory for motor driven EFW pumps
1989	GL 81-14 seismic modifications for seismic boundary valves	Improves seismic design/boundaries for EFW System
1990	Added AMSAC	Added diverse actuation circuitry for motor driven EFW pumps
1991-1992	Further improved hotwell suction source for motor driven EFW pumps	Improved NPSH and increased available hotwell inventory for motor driven EFW pumps
1991-1992	Added SG dryout protection	Added diverse actuation circuitry for motor driven EFW pumps
1993	Auxiliary Instrument Air modification	Increased reliability of several key air operated valves
1994	C-187 auto-closure on low UST level	Reduces vulnerability associated with hotwell emergency makeup line
1994-1996	MSLB mod	Improves runout protection for turbine driven EFW pump

EFW Reliability

- Primary objective of post-TMI EFW action plan was to improve reliability of emergency feedwater
- Oconee implemented changes to improve emergency feedwater reliability, meeting objectives of action plan

EFW Reliability

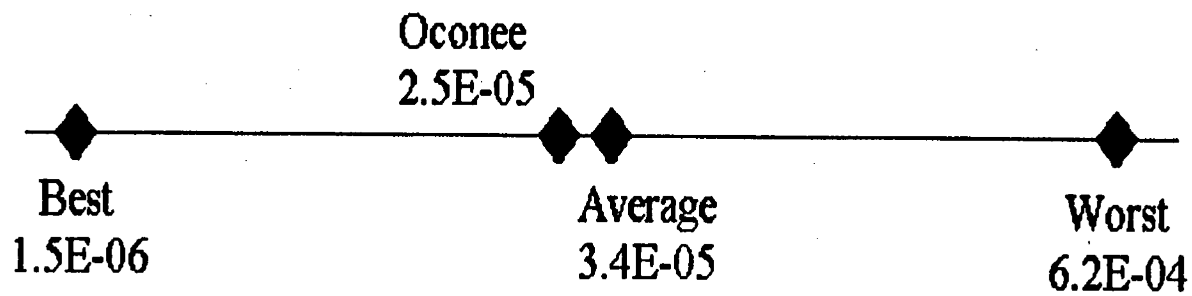
- Data indicates EFW System is reliable
  - » Plant operating experience
  - » NUREG/CR-5500
  - » Duke EFW Reliability Calculations
    - – Assessment of C-187

EFW Reliability

- EFW operational reliability from 1980 through 1998:
  - » 47 EFW demand events on three units
  - » Affected unit's EFW System successfully provided secondary side heat removal 47 times
  - » Reliance on diverse backup means of decay heat removal was not needed for any of these events

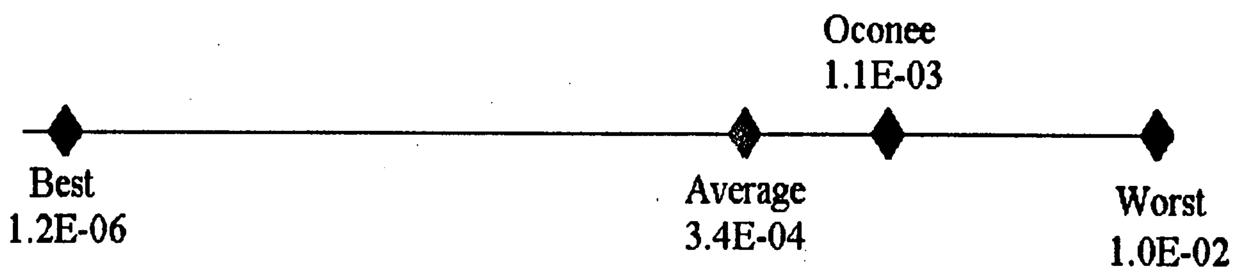
NUREG/CR-5500

- Relative ranking of Oconee EFW unreliability versus other plants AFW systems
- Operational unreliability based on 1987-1995 experience



NUREG/CR-5500

- Relative ranking of Oconee EFW unreliability versus other plants AFW systems
- Plant-specific estimates (PRA-based) calculated with IPE failure rates



NUREG/CR-5500

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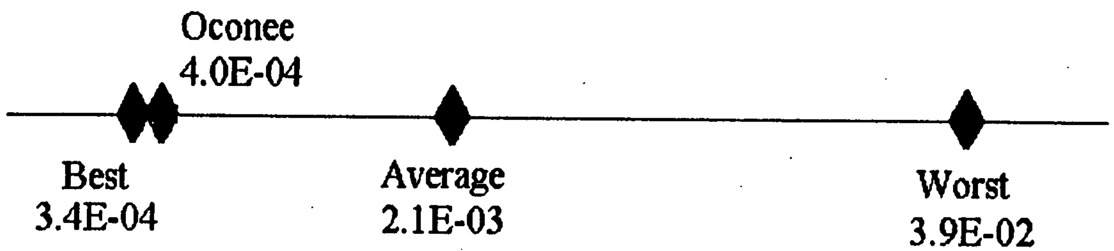


Current Oconee PRA EFW Reliability Estimates

- The current EFW system failure probability is estimated to be 2.1E-03 for loss of MFW events
- When the EFW cross-connect capability and the SSF are included the failure probability of this function improves to 9.7E-05
- This is a substantial improvement in the reliability (reduction in the failure probability by a factor of approximately 20)

NUREG/CR-5500 Summary

- The industry experience captured in the NUREG suggests that the Oconee EFW system reliability is comparable to other plants
- Diversity in systems that support the secondary side heat removal function make the overall reliability of the function even better

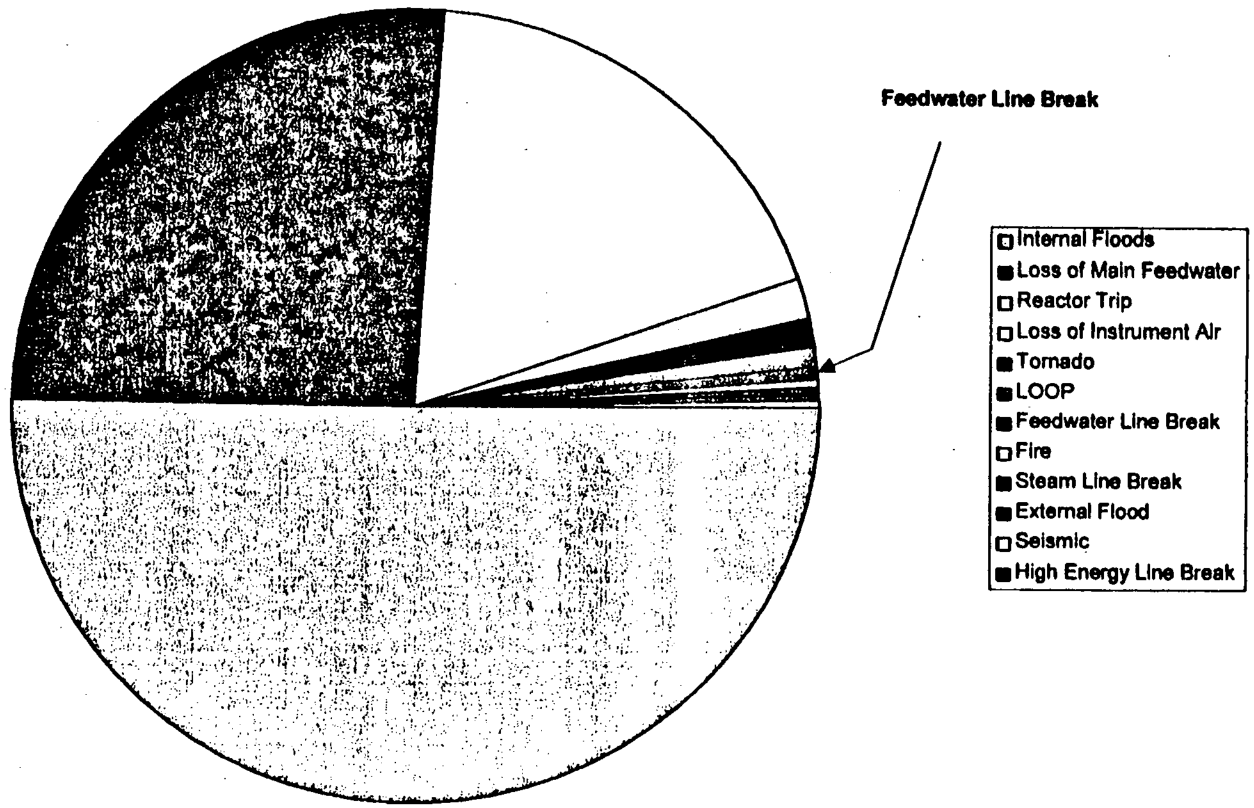
Overall EFW Reliability

- Sequences of Interest
  - » Reactor Trip
  - » Loss of Main Feedwater
  - » Loss of Offsite Power
  - » Loss of Instrument Air
  - » Feedwater Line Break
  - » Internal Floods
  - » Steam Line Break
  - » External Flood
  - » Tornado
  - » Seismic
  - » Fire
  - » *High Energy Line Break*

Overall EFW Reliability

- Is different for various initiators
- Overall EFW failure probability (frequency weighted) is estimated to be 4.5E-03
- Overall secondary side heat removal failure probability is estimated to be 2.7E-04
  - » Reflects impact of:
    - – EFW cross-connect from alternate unit
    - – SSF ASW
    - – Station ASW

Overall EFW Reliability



Effect of Valve C-187 on EFW Reliability

- Potential single failure point
  - » Potential failure to close on some feed line breaks
  - » Potential to transfer open for other initiators
- Makes essentially no contribution to the base case Oconee CDF
- Makes essentially no contribution to the Oconee CDF with an arbitrary increase in failure probability by a factor of 10

Conclusions

- Nominal EFW reliability is good
- Diversity in systems that support secondary side decay heat removal makes SG cooling function very reliable
- Addition of redundant valve to C-187 has an insignificant impact on overall performance measures
  - » EFW performance
  - » Secondary side decay heat removal performance
  - » Core damage frequency

Licensing Evolution of EFW

- Original Licensing Basis
- HELB
- Post-TMI
- GL 81-14 Seismic Qualification
- Updating of UFSAR

EFW Original Licensing Basis

- System as originally designed was not single failure proof
  - » Only one turbine-driven pump
  - » No cross-connects between units
- "Redundancy" considered in context of entire steam conversion system
  - » Main feedwater, hotwell, condensate booster, emergency feedwater pumps & Station ASW
- Main feedwater line breaks not considered in original design

HELB Influence on EFW

- AEC (Giambusso) letter dated December 15, 1972 requested that Duke address HELBs
  - » Focused on dynamic effects
- Duke HELB analysis identified secondary side cooling vulnerabilities
- EFW modifications addressed vulnerabilities:
  - » Rerouted EFW piping through Turbine Building basement
  - » Installed EFW cross-connects between units
- AEC Safety Evaluation for operating license, dated 7/6/73, accepted Duke's HELB strategy
  - » Relied upon cross-connects between units to address single failure criterion

TMI Influence on EFW

- Order issued on 5/7/79 after TMI-2 accident
- Duke submitted conceptual design for EFW upgrade on 5/17/79
- Key system improvement was the installation of two motor driven EFW pumps and associated piping on each unit

TMI Influence on EFW

- NRC letter (5/18/79) finds satisfactory compliance with immediate actions of order
- Duke letter (7/23/80) responding to NUREG-0667 recommendation to upgrade EFW to meet safety grade requirements stated: "***The Oconee emergency feedwater system coupled with the dedicated Standby Shutdown Facility, currently under construction, meet this recommendation and no additional modifications to the system are necessary.***"
- NRC SER (8/25/81) accepts Duke submittal (4/3/81) which credited EFW unit cross-connects and SSF capability
- NRC SER (12/29/81) revising the TS out-of-service times for the motor-driven EFW pumps recognizes and credits cross-connect, SSF and station ASW as means of providing EFW

Result of Post-TMI Effort

- Significant modification to system
- Significant correspondence, meetings and dialogue
- Mutual agreement in overall direction
- Licensing review focused on failure of pumps and specific valves, not entire system
- No change in feedwater line break response strategy

Seismic Qualification of EFW (GL 81-14)

- Duke original and subsequent responses repeated intent to utilize the dedicated Standby Shutdown Facility (SSF) as an alternate means of feedwater supply (1982)
- NRC requests further information relating to SSF (1982, 1984)
- Duke identifies and corrects issues involving seismic qualifications of certain EFW valves and piping (1985 - 1986)
- NRC SER (1/14/87) approving Duke's response based in part on the availability of alternate means of decay heat removal

UFSAR Wording

- 1982 update of FSAR by Duke used wording from various submittals out of context
- Recent UFSAR review:
  - » Inappropriately dispositioned statements as acceptable since wording matched docketed correspondence
- 2nd pass UFSAR review ongoing

UFSAR Wording

- **Example 1:**
  - » *"Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."*

UFSAR Wording: Example 1

- Wording taken from 5/17/79 Duke submittal describing system concept associated with adding motor-driven EFW pumps
  - » Sentence taken from paragraph describing capability of pumps to provide sufficient flow
  - » EFW cross-connect valves listed as example of valving "*provided to select and isolate water sources and assure system function in the event of various failures*"
  - » Safety evaluation in submittal contained listing of diverse methods for providing feedwater inventory.

UFSAR Wording: Example 1

- From the Safety Evaluation contained in the Duke 5/17/79 submittal:
  - » Safety Evaluation
    - – "Feedwater inventory is maintained in the steam generators following reactor shutdown by one of the following methods listed:
      - Either of the two main feedwater pumps...
      - The two EFW motor driven pumps…
      - The single EFW turbine driven pump…
      - Alternate EFW supplies may be available from the EFW Systems of the other Units…
      - The hotwell and condensate booster pumps…
      - The Auxiliary Service Water System…
    - – A sufficient depth of backup measures is provided to allow steam generator water inventory to be maintained by any of the diverse methods listed above."

UFSAR Wording

- **Example 2:**
  - » *"In the event of a postulated break in the Main Steam or Main Feedwater system inside or outside containment coupled with a single active failure, the EFW system provides sufficient flow to ensure adequate core cooling."*

UFSAR Wording: Example 2

- UFSAR wording taken directly from NRC SER dated 8/25/81, which accepted Duke's submittal of 4/3/81
- Duke 4/3/81 submittal in response to NRC request for information dated 11/14/80
- NRC request focused specifically on EFW flow characteristic during accidents
  - » Enclosure 3, question 3 asked: "*Verify that the AFW pumps in your plant will supply the necessary flow to the steam generators as determined by items 1 and 2 above considering a single active failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.*"

UFSAR Wording: Example 2

- Duke's 4/3/81 response addressed single failure by evaluating pumps and EFW flow control valves:
  - » "*In the event of a postulated break in the main steam or main feed system, coupled with a single active failure of either one of the three emergency feedwater pumps, sufficient flow will occur to provide adequate core cooling. Similarly, if the active failure occurs with the flow control valve (FDW-316), emergency feedwater flow can be aligned through the main feedwater startup control valves to either the main or auxiliary nozzles.*"
  - » "*Additional assurance to maintain core cooling for very low probability events will be provided by the Standby Shutdown Facility once it is operable.*"

UFSAR Wording Conclusions

- With regard to EFW single failure, UFSAR language did not adequately reflect design basis
- Understanding of licensing evolution essential
  - » 4 volume UFSAR
  - » Diverse secondary side heat removal features
  - » Duke's 50.59 guidance requires review and consideration of all licensing documents
- UFSAR revision to clarify did not involve an USQ

Licensing Basis Conclusions

- EFW System was not designed to withstand all single-failures
- NRC has accepted diverse and redundant methods of supplying feedwater to address EFW limitations
- FSAR does not appropriately reflect EFW single failure design basis
- Feedwater line break accident addressed by High Energy Line Break submittals

Licensing Basis - Going Forward

- Revise UFSAR to clarify EFW licensing basis
- UFSAR 2nd pass review
- Enhance administrative controls to ensure availability of redundant sources

Engineering Perspective (C-187)

- In compliance with design basis
- Failure contribution to plant risk insignificant
- Vulnerability of failure significantly reduced
  - » Modified to fail close, with signal to close on low UST level
  - » Included in IST program
  - » Monitored via maintenance rule

Engineering Perspective (C-187)

- Overall Engineering direction to increase design margins of plant systems
- Duke evaluating improved design for UST to hotwell flowpath
  - » Build in design margin
  - » Reduce operator burden

Engineering Perspective (EFW Runout)

- Licensing basis relies on prompt operator action to throttle flow
- Licensing submittal to credit MSLB modification for runout protection of turbine driven EFW pump
- EFW System modifications planned to further reduce operator burden
- EFW full flow test planned for Unit 1 outage this June to determine if pumps actually experience runout conditions

Engineering Perspective - Going Forward

- Validation of Operator Actions
- Engineering focus on reducing operator burden
  - » EFW modifications for control system enhancements
- Initiative on risk significant operator actions
- Continue with program for system reviews
- Secondary Side Decay Heat Removal Reliability Study
  - » Complements Keowee PRA and HPI Reliability Studies

Closing Remarks

EFW HISTORY		COMMENTS	
1973-74	Units 1, 2, & 3 licensed • Initial EFW system design included only one pump (turbine-driven) and was not safety-related.	1973-94	EFW System as designed would fail on MFW line break or non-seismic pipe break in that C-187 (and C-176) would open on low Hotwell level, dump UST to Hotwell in about 2 minutes, and consequently remove EFW pumps' suction source. C-187 (and C-176) were also susceptible to opening and failing to close during normal plant operation.
1974	• EFW cross-ties to other units installed.		
1979	EFW Modified post TMI to be 2 trains (3 pumps). • Added 2-MDEFW pumps • EFW system automatically initiated on loss of main feedwater pumps	1979-94	Most of EFW system was made safety-related, but above design vulnerability not recognized until 1989 and corrective modification not installed until 1993-94.
1981	Post TMI EFW SER issued by NRC: • EFW system required to mitigate MFW line break • EFW system required to withstand single active failure and secondary pipe break and loss of AC power. TS Modified to require three operable EFW pumps, two flowpaths, and automatic initiation circuitry.	1981-94	Post-TMI EFW system as design was not able to mitigate MFW line break or other non-seismic pipe break. EFW system was not designed to withstand a single failure.
1982	FSAR changed to state new requirements above.	1982	EFW design vulnerabilities not recognized.
1986-87	EFW was redesigned to address Seismic qualification of AFW systems (GL-81-14) - Mod not installed until 1989.	1986-87	EFW design vulnerabilities not recognized.
1989	C-187 (and C-176) modification was installed. • C-187 (and C-176) became seismic boundary valves for EFW • C-187 (and C-176) designed to be safety related (SR)	1989	Mod left C-187 (and C-176) to automatically open on low Hotwell level (i.e., MFW line break or non-seismic pipe break).
6/30/89	Problem Investigation Report identified low Hotwell level effect on C-187 and C-176 causes EFW system failure.	6/30/89	Licensee isolated C-176 and left C-187 in service. Licensee evaluated PIR as not effecting EFW operability.
1993-94	C-187 modified to close on UST low level (7ft) based on corrective actions for 6/30/89 PIR.	1993-94	C-187 Mod made EFW system able to withstand MFW line break or other non-seismic pipe break but not with single active failure of C-187.
11/18/98	UFSAR review: • For a secondary pipe break coincident with a single failure, the emergency feedwater function may be provided by another unit. • No longer requires EFW system to withstand single failure other than EFW pumps or FCVs.	11/18/98	The 10 CFR 50.59 safety evaluations did not identify a USQ. For the UFSAR change.

Enclosure 3

OCONEE EFW LICENSING AND DESIGN BASES STATEMENTS

(NOTE: Underlined statements have been in the UFSAR since 1982)

May 17, 1979 - Duke Ltr responding to NUREG-0737 Item II.E.1.1, Auxiliary Feedwater Evaluation

- "Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

- "Each reactor unit is provided with a separate EFW system..." "Each EFW system is provided with two full capacity motor-driven pumps and one full capacity turbine-driven pump." "Although redundancy and diversity is provided in the listed measures (including EFW systems of the other units, condensate booster pumps, and auxiliary service water pump), the EFW system has been designed with special considerations to enable it to function when conventional means of feedwater makeup may be unavailable. Redundancy is provided with separate, full capacity, motor and turbine driven pump subsystems. Separate piping subsystems include redundant hotwell, upper surge tank, and condensate supply piping, aligned individually to the separate pump trains. Cross-connection is provided, however, to allow a subsystem to supply all pumps in the event of a single failure of a suction piping subsystem. The same design philosophy is included in the discharge piping subsystems."

April 3, 1981 - Licensee Ltr responding to NUREG-0737 Item II.E.1.1, Auxiliary Feedwater Evaluation

- "In the event of a postulated break in the main steam or main feed system, coupled with a single active failure of either one of the three emergency feedwater pumps, sufficient flow will occur to provide adequate core cooling." "Any single failure in the three pump-two flowpath EFW system design will not result in only one motor-driven EFW pump available."

1981 - 1998 - TS 3.4, Secondary System Decay Heat Removal

- Three EFW pumps (one steam-driven and two motor-driven), two flowpaths, and the automatic initiation circuitry shall be operable.

? - 1998 - TS 3.4 Bases

- The EFW system consists of a turbine-driven pump (880 gpm), two motor-driven pumps (450 gpm each), and associated flow paths to the steam generators.

August 25, 1981 - NRC SER on NUREG-0737 Item II.E.1.1, Auxiliary Feedwater

- The NRC accepted the Oconee post-TMI EFW design and stated that, with respect to a main feedwater line break, "The system is designed so that a single active failure of any of the emergency feedwater pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator."

- The NRC stated that the licensee should lock open single valves or multiple valves in series in the EFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. The NRC further stated: "As evidenced by the piping and instrumentation diagram for the Oconee EFW system, there are no single valves or multiple valves in series in the system pump suction or other single or multiple valves in series that could interrupt all EFW flow."

- The NRC stated: "...low water level in the primary water source tanks is not ever expected to be a cause for suction water to be unavailable to the EFW pumps. The availability of the primary water source is assured ..."

? - 1998 - Licensee EFW Design Basis Specification Document

- "The EFW system shall be capable of withstanding any credible single failure during certain of the system design basis events." It further stated that the EFW system shall be designed for a main feedwater line break event, and that the main feedwater line break scenario requires consideration of any single active failure.

May 7, 1986 - Licensee Letter Responding to GL 81-14, Seismic Qualification of Auxiliary Feedwater Systems

- The licensee stated that valves relied upon as single EFW seismic boundary valves would meet the existing EFW design criteria; in particular that the EFW system can "<u>perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power</u>." The licensee also stated: "Two hotwell make-up line isolation valves are normally open (C-186, C-191). Modifications at this boundary will be made to protect EFW against single failure." (NOTE: In 1986 this boundary became C-187.)

? - 1998 - UFSAR 3.2

- Section 3.2 description of seismic classifications states that the following equipment and portions of systems can withstand the maximum hypothetical earthquake: Upper surge tanks and piping to the emergency feedwater pump; and Emergency feedwater pump and turbine and auxiliary feedwater piping to the steam generators.

January 14, 1987 - NRC SER on Seismic Qualification of the EFW System

- "The licensee has demonstrated adequate post-seismic event decay heat removal capability in accordance with the criteria of Generic Letter 81-14 by committing to correct identified deficiencies in the seismic qualification of the EFW system itself, and by demonstrating adequate seismically qualified alternative capability utilizing the SSF ASW pump and HPI pump (feed-and-bleed) in the event of the loss of the AFW system as a result of seismically induced flooding."

1989 - PIR 4-089-0111

- Operability evaluation for this PIR stated: "This PIR points out a licensing commitment regarding the seismic boundary of the EFW suction water supply which is not fully satisfied..." "In the event the seismically non-qualified hotwell piping is postulated to fail during an earthquake, hotwell level would be lost and valves would automatically open and drain the UST. Since the valves are not remotely operable, credit is not allowed for operator action to stop the loss of UST volume." "Engineering analysis has determined ...by SQUG/seismic margin techniques...the non-qualified portion of the EFW suction (hotwell & associated piping) can withstand a seismic event. As such, there is no realistic seismic/non-seismic boundary between the UST and the hotwell. Based on the above, the EFW system is considered operable for all three units."

NRC PRELIMINARY CONCLUSIONS ABOUT THE EFW LICENSING BASIS AND LICENSEE DISSENTING COMMENTS

As described in Inspection report 50-269,270,287/99-10, the NRC reached preliminary conclusions about the Oconee EFW licensing basis. The licensing basis issues involved the EFW system ability to function during a main feedwater line break, a non-seismic pipe break, or a single active failure. The NRC preliminary conclusions were not changed after consideration of the licensee's dissenting comments. The NRC preliminary conclusions and licensee dissenting comments included:

A. <u>Main Feedwater Line Break</u>

From initial plant licensing in 1973 through today, the EFW system has been licensed to be able to function during a main feedwater line break. That requirement was reinforced in 1979 by NUREG-0737 Item II.E.1.1, Auxiliary Feedwater System Evaluation. However, from 1973 through 1994, the EFW system was designed so that it would fail during a main feedwater line break. The failure would occur as follows: 1) Water from the condenser hotwell would be lost out the break, lowering the hotwell level. 2) The low hotwell level would cause hotwell makeup valves C-187 and C-176 to open, draining the upper surge tank (UST) until it was empty (in about two minutes). 3) The EFW pumps would automatically start and take a suction from the UST; however, when the UST became empty, they would no longer have water to pump and consequently all EFW pumps would likely be quickly damaged.

References supporting this licensing interpretation include:

1. UFSAR Section 10.4.7, Emergency Feedwater System (Tab 4)
2. TS 3.4, Secondary System Decay Heat Removal (Tab 4)
3. Licensee Letter of May 17, 1979, Describing the Post-TMI EFW System (Tab 5)
4. Licensee Letter of April 3, 1981, Further Describing the Post-TMI EFW System (Tab 6)
5. NRC SER of August 25, 1981, on the Oconee EFW System (Tab 8)
6. Licensee Design Basis Specification for the EFW System (Tab 7)

B. <u>Non-seismic Pipe Break</u>

The EFW system was licensed to be able to function following a seismic event. This included requirements for seismic boundary valves that were assured of remaining closed to protect the seismically designed UST from a break in the the non-seismically designed condenser hotwell and related condensate and feedwater piping. The seismic design requirements were stated in 1986 and 1987 correspondences. However, the design of valves C-187 and C-176 through 1994, as previously described, prevented them from being acceptable EFW system seismic boundary valves.

Supporting references include:

1. UFSAR Section 3.2, Seismic Classification (Tab 9)
2. Licensee Letter of May 7, 1986, Describing Seismic Qualification of the EFW System (Tab 10)

3. NRC SER of January 14, 1987, on Seismic Qualification of the EFW System (Tab 9)
4. Problem Investigation Report 4-089-0111 (Tab 14)

C. Single Failure

From 1981 through today, the EFW system was licensed to function while sustaining a single active failure during a main feedwater line break or a non-seismic pipe break. This was stated in 1979-1981correspondences related to NUREG-0737 Item II.E.1.1, Auxiliary Feedwater System Evaluation. Also, EFW seismic boundary valves were not to be susceptible to a single active failure, as stated in 1986-1987 correspondences. In 1989 the licensee isolated valve C-176 and in 1994 the licensee modified valve C-187 so that it would be overridden closed on a low UST level to protect the EFW pumps' initial water source. This modification enabled the EFW system to function during a main feedwater line break or non-seismic pipe break. However, valve C-187 was designed so that an active failure of the valve during a main feedwater line break or a non-seismic pipe break could fail the EFW system. Supporting references include: A 1, 3, 4, 5, and 6 above; B 2 and 3 above; and the Oconee PRA on the EFW System (Tab 14).

D. Licensee Dissenting Comments

The licensee did not agree with the NRC preliminary conclusions regarding the licensing basis of the EFW system, for the following reasons:

1. The diversity of the Oconee design includes alternate methods of providing emergency cooling water to the once-through steam generators (OTSGs), including EFW from other units, lake water from the "tornado" station auxiliary service water (ASW) pump, and lake water from the standby shutdown facility (SSF) ASW pump. Therefore, the safety function of secondary cooling was designed to withstand a single failure.

2. The August 25, 1981, NRC SER on the upgraded EFW system focused on the new EFW flowpaths, from the two new motor-driven EFW pumps to the OTSGs, and did not require that the old EFW suction sources be designed against a single failure. The SER stated that, in the event of a main feedwater line break, "The (EFW) system is designed so that a single active failure of any of the emergency feedwater pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator." The licensee contended that the word "valves" in that statement referred only to the two EFW flow control valves on the discharge side of the EFW pumps.

3. The NRC approved the EFW system design with recognition that it was not designed to be single failure proof for three events: high energy line break, turbine building flood, and tornado.

EFW HISTORY		ACTUAL PLANT CONDITION	
1973-74	Units 1, 2, & 3 licensed • Initial EFW system design included only one pump (turbine-driven) and was not safety-related.	1973-94	EFW System as designed would fail on MFW line break or non-seismic pipe break in that C-187 (and C-176) would open on low Hotwell level, dump UST to Hotwell in ~~it~~ about 2 minutes, and consequently remove EFW pumps' suction source. C-187 (and C-176) were also susceptible to opening and failing to close during normal plant operation.
1979	EFW Modified post TMI to be 2 trains (3 pumps). • Added 2-MDEFW pumps	1979-94	Most of EFW system was made safety-related, but above design errors not recognized until 1989 and corrective modification not installed until 1993-94.
1981	Post TMI EFW SER issued by NRC: • EFW system required to mitigate MFW line break • EFW system required to withstand single active failure and secondary pipe break and loss of AC power. TS Modified to require three operable EFW pumps, two flowpaths, and automatic initiation circuitry.	1981-94	EFW system as designed was not able to mitigate MFW line break or other non-seismic pipe break and therefore was inoperable.
1982	FSAR changed to state new requirements above.	1982	EFW design errors not recognized.
1986-87	EFW was redesigned to address Seismic qualification of AFW systems (GL-81-14) - Mod not installed until 1989.	1986-87	EFW design errors not recognized.
1989	C-187 (and C-176) modification was installed. • C-187 (and C-176) became seismic boundary valves for EFW • C-187 (and C-176) designed to be safety related (SR)	1989	Mod left C-187 (and C-176) to automatically open on low Hotwell level (i.e., MFW line break or non-seismic pipe break).
6/30/89	Problem Investigation Report identified low Hotwell level effect on C-187 and C-176 causes EFW system failure.	6/30/89	Licensee isolated C-176 and left C-187 in service. Licensee inappropriately evaluated PIR as not effecting EFW operability; failed to report to NRC and failed to initiate timely corrective action. C-187 was not modified until 1993-94.
1993-94	C-187 modified to close on UST low level (7ft) based on corrective actions for 6/30/89 PIR.	1993-94	C-187 Mod made EFW system able to withstand MFW line break or other non-seismic pipe break but not with single active failure of C-187.
11/18/98	UFSAR revision approved to no longer require EFW system to withstand single active failure other than for EFW pumps or FCVs (excluding support system effects such as C-187). UFSAR revision also approved EFW system no longer required to withstand a secondary pipe break coincident with any single failure.	11/18/98	Both UFSAR changes constitute USQs not recognized by the licencee. EFW system is nonconforming in that single failure and seismic criteria are not met. Licensee does not agree that EFW design or revised FSAR represent nonconforming conditions or USQs.

OCONEE EFW LICENSING AND DESIGN BASES STATEMENTS

(NOTE: Underlined statements have been in the UFSAR since 1982)

DATE	DOCUMENT	STATEMENTS
5/17/79	Licensee Letter in response to NUREG-0737 Item II.E.1.1, Auxiliary Feedwater Evaluation	"Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."
4/3/81	Licensee Letter in response to NUREG-0737 Item II.E.1.1, Auxiliary Feedwater Evaluation	"In the event of a postulated break in the main steam or main feed system, coupled with a single active failure of either one of the three emergency feedwater pumps, sufficient flow will occur to provide adequate core cooling. Any single failure in the three pump-two flowpath EFW system design will not result in only one motor-driven EFW pump available.
8/25/81	NRC SER on NUREG-0737 Item II.E.1.1, Auxiliary Feedwater	NRC accepted the Oconee post-TMI EFW design and stated that, with respect to a main feedwater line break, "The system is designed so that a single active failure of any of the emergency feedwater pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator."
1981 - 1998	TS 3.4, Secondary System Decay Heat Removal	Three EFW pumps (one steam-driven and two motor-driven), two flowpaths, and the automatic initiation circuitry shall be operable. (NOTE: TS 3.4 does not credit other sources of OTSG water, such as another unit's EFW pumps, the SSF ASW pump, or the station ASW pump.)
? - 1998	TS 3.4 Bases	The EFW system consists of a turbine-driven pump (880 gpm), two motor-driven pumps (450 gpm each), and associated flow paths to the steam generators.
1982-1998	UFSAR 10.4.7	Section 10.4.7 description of the EFW system has included the above and below underlined licensee statements about the system design from 1982 through November 18, 1998.
? - 1998	Licensee EFW Design Basis Specification document	"The EFW system shall be capable of withstanding any credible single failure during certain of the system design basis events." It further stated that the EFW system shall be designed for a main feedwater line break event, and that the main feedwater line break scenario requires consideration of any single active failure.

? - 1998	UFSAR 3.2	Section 3.2 description of seismic classifications states that the following equipment and portions of systems can withstand the maximum hypothetical earthquake: Upper surge tanks and piping to the emergency feedwater pump; and Emergency feedwater pump and turbine and auxiliary feedwater piping to the steam generators.
5/7/86	Licensee Letter in response to GL 81-14, Seismic Qualification of Auxiliary Feedwater Systems	The licensee stated that valves relied upon as single EFW seismic boundary valves would meet the existing EFW design criteria; in particular that the EFW system can "perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power." The licensee also stated: "Two hotwell make-up line isolation valves are normally open (C-186, C-191). Modifications at this boundary will be made to protect EFW against single failure." (NOTE: In 1986 this boundary became C-187.)
1/14/87	NRC SER on Seismic Qualification of the EFW System	"The licensee has demonstrated adequate post-seismic event decay heat removal capability in accordance with the criteria of Generic Letter 81-14 by committing to correct identified deficiencies in the seismic qualification of the EFW system itself, and by demonstrating adequate seismically qualified alternative capability utilizing the SSF ASW pump and HPI pump (feed-and-bleed) in the event of the loss of the AFW system as a result of seismically induced flooding"
1989	PIR 4-089-0111	Operability evaluation for this PIR stated: "This PIR points out a licensing commitment regarding the seismic boundary of the EFW suction water supply which is not fully satisfied..." "In the event the seismically non-qualified hotwell piping is postulated to fail during an earthquake, hotwell level would be lost and valves would automatically open and drain the UST. Since the valves are not remotely operable, credit is not allowed for operator action to stop the loss of UST volume." "Engineering analysis has determined ...by SQUG/seismic margin techniques...the non-qualified portion of the EFW suction (hotwell & associated piping) can withstand a seismic event. As such, there is no realistic seismic/non-seismic boundary between the UST and the hotwell. Based on the above, the EFW system is considered operable for all three units."
1998	Oconee PRA Rev. 2	The list of top cut sets for EFW system failure includes "Air-operated valve C-187 transfers open," with a probability of 9.72 E-05. The PRA also states: "If a main feed line break is assumed, the UST could be drained into the hotwell, thereby failing EFW's initial suction source."

Design Basis Specification for the Emergency Feedwater and the Auxiliary Service Water Systems
Spec. OSS-0254.00-00-1000
Rev. 18

May 11, 1989

Duke Power Company
Oconee Nuclear Station
Units 1, 2 and 3

Revision Log

Revision 1	August 29,1989 (Incorporates all previous revisions)
Revision 2	February 1, 1990
Revision 3	September 5, 1990
Revision 4	October 16, 1990
Revision 5	February 13, 1991
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Revision 10	April 14, 1993
Revision 11	February 10, 1994
Revision 12	May 1, 1995
Revision 13	August 3, 1995
Revision 14	December 4, 1995
Revision 15	May 1, 1996
Revision 16	February 17, 1997
Revision 17	February 23, 1998
Revision 18	May 19, 1998

20. SYSTEM DESIGN BASES

The system design bases are documented in this section. Specific system functional design bases are described in Section 20.1, "SYSTEM FUNCTIONAL DESIGN BASES." Section 20.2, "SYSTEM SPECIFIC DESIGN CRITERIA" on page 8 includes descriptions of generic system design criteria.

20.1 SYSTEM FUNCTIONAL DESIGN BASES

The information presented in this section constitutes requirements that the EFW system shall meet to ensure that the system is capable of performing its required functions. Required functions are the functions necessary to produce either a parameter or condition assumed or bounded by the reactor transient analysis.

20.1.1 EMERGENCY FEEDWATER SYSTEM

This portion of the EFW System shall deliver feedwater to one or both steam generators during the listed design basis events when the Main Feed Water System (MFW) is incapable of doing so. This provides an adequate heat sink for reactor heat removal.

20.1.1.1 FLOWRATE REQUIREMENTS

The following flowrates to the steam generator have been assumed in various accident analyses. Design calculations shall demonstrate that these flowrates can be achieved when the pumps are operable as described on TAC drawings, the most recent system tests described on TAC drawings have been met, and the valves in the system are operable. These flowrates are flow delivered to the steam generator (not pump flowrate) and already consider credible single failures if appropriate.

1. Any of the motor driven emergency feedwater pumps shall be capable of delivering at least 400 GPM at or below 130°F to any single steam generator that is at a pressure of 1060.5 psig or below. These conditions apply to loss of main feedwater during full power operation. They do not apply to startup following a reactor trip or a refueling outage. Although not a licensing basis, this should include steam generators of other units via pump discharge piping cross-connections. (Reference20.5.2.1.1, "Duke Calc OSC-3578, Rev.1, 6-9-89; Justification of a 130°F EFW System Temperature" on page 29 and 20.5.2.1.9, "Duke Calc OSC-4549, Rev. 0, 11-04-93; FSAR Section 10.4.7 .1.1-Loss of Main Feedwater" on page 30.)
2. Each turbine driven EFW Pump, when delivering flow to both steam generators of any unit, shall supply at least 450 GPM total to the two steam generators at 1100 psig. (Reference 20.5.2.1.2, "Duke Calc OSC-2624, Rev.1, 12-9-87; FSAR Loss of All AC Power" on page 29). This is required by the loss of all AC power analysis.
3. Flow shall not exceed 1098 GPM to any one steam generator through the emergency feedwater header. This limit is imposed to protect the steam generator tubes from the effects of flow induced vibration. (Reference 20.5.2.1.3, "Duke Calc OSC-2569, Rev.0, 7-30-87; Evaluation of FIV Potential of Flow Into the OTSG Thru AFW Nozzles" on page 29).
4. When aligned to the UST, the MDEFWP flowrate shall be maintained below 850 gpm and the TDEFWP flowrate shall be maintained below 1500 gpm to assure adequate NPSHa for the pumps. (Reference 20.5.2.1.5, "Duke Calc OSC-2155, Rev.1, 7-21-86; Motor Driven and Turbine Driven EFW Pump NPSHa from the USTs" on page 29).
5. When aligned to the hotwell under vacuum the TDEFWP flowrate shall be maintained below 500 gpm.

6. The combined flow from each unit's three EFW pumps to both its steam generators shall be at least 1350 GPM when the pressure of the steam generators is 1010 psig or less. (References 20.5.2.4.1, "Letter from PM Abraham(Duke) to HA Hammond(Duke), dated 12-10-87" on page 30 and 20.5.2.5.2, "Analysis of ATWS, BAW-1099, Rev.1, 5-77, Babcock and Wilcox; Analysis of B&W NSSS Response to ATWS Events, BAW-1610, Re v.1, 1-80, Babcock and Wilcox" on page 30). This requirement is necessary for the ATWS analysis.

20.1.1.2 DESIGN BASIS EVENTS

The following paragraphs discuss the events for which the EFW System shall be designed. These events are all assumed to occur at full power since this maximizes heat removal requirements for the system.

1. LOSS OF MAIN FEEDWATER WITH OFFSITE POWER AVAILABLE

   This event can be caused by a number of failures that result in a loss of main feedwater flow. An anticipatory circuit senses the loss of main feedwater and trips the reactor. Low hydraulic control oil pressure on both Main Feedwater Pump Turbines shall result in auto-initiation of the EFW System. EFW shall maintain steam generator water level and remove decay heat until normal feedwater is restored or until decay heat removal (using LPI) can begin. (Reference 20.5.1.2.1, "FSAR Sections 10.4.7 and 7.4.3" on page 28). This scenario requires consideration of any single active failure and the possibility of a seismic event.

2. LOSS OF MAIN FEEDWATER WITHOUT OFFSITE POWER

   This scenario is very similar to the event described above. The major difference is that the recovery shall be accomplished without the reactor coolant pumps. Steam generator level shall be maintained at a high level by the EFW System in order to promote natural circulation. (Natural circulation is further promoted by the EFW injection nozzle location near the top of the steam generators.) (Reference 20.5.1.2.1, "FSAR Sections 10.4.7 and 7.4.3" on page 28). This scenario requires consideration of any single active failure and the possibility of a seismic event.

3. MAIN FEEDWATER LINE BREAK

   A main feedwater line break not only results in a loss of main feedwater flow to the steam generator(s), but also could result in the complete blowdown of one steam generator. The EFW System shall be designed to terminate, limit, or minimize the fraction of EFW flow which is delivered to the faulted loop to ensure that sufficient flow will be delivered to the intact steam generator. Operator action is required to avoid pump runout, potential primary system overcooling, and flow-induced vibration of steam generator tubes. (References 20.5.1.2.1, "FSAR Sections 10.4.7 and 7.4.3" on page 28 and 20.5.2.1.3, "Duke Calc OSC-2569, Rev.0, 7-30-87; Evaluation of FIV Potential of Flow Into the OTSG Thru AFW Nozzles" on page 29). This scenario requires consideration of any single active failure but not a seismic event.

4. STEAM LINE BREAK

   A steam line break results in plant cooldown and, for breaks inside containment, increasing containment pressure and temperature. Emergency feedwater is not needed during the early phase of this accident, and flow to the faulted loop will contribute to an excessive release of mass and energy to the containment. To limit the amount of emergency feedwater being delivered to the steam generators, the main steam line break (MSLB) detection/mitigation circuitry will automatically stop the turbine driven EFW pump. Eventually, the reactor coolant system will begin to heat up, at which time emergency feedwater shall be delivered to the intact steam generator. Operator action is required to limit, control, or terminate flow to the faulted steam generator in order to prevent containment overpressurization, pump runout, primary system overcooling, flow-induced tube vibration, and to ensure adequate flow to the intact steam generator. (Reference 20.5.1.2.1, "FSAR Sections 10.4.7 and 7.4.3" on page 28). This scenario requires consideration of any single active failure but not a seismic event.

5. LOSS OF ALL AC POWER

The loss of all AC Power is the hypothetical case where all onsite and offsite AC Power is lost. The event is known as the Station Blackout (SBO). Decay heat is removed following a station blackout using the SSF Auxiliary Service Water System. The turbine driven emergency feedwater pump may also be used instead of SSF-ASW, but it is not required as part of the station blackout licensing basis (Reference 20.5.1.2.9, FSAR Section 8.3.2.2.4). All equipment associated with the turbine driven pump's ability to feed the steam generators that require compressed air to perform their function during this event shall have a two hour supply of bottled nitrogen (Reference 20.5.1.9.1, Letter from WO Parker to HR Denton(NRC), dated 4-3-81). Station blackout is not a design basis event. Therefore, the SBO event is not concurrent with any design basis event or single failures (Reference 20.5.1.2.9, FSAR Section 8.3.2.2.4).

6. LOSS OF COOLANT ACCIDENT

During a small break LOCA, the principal function of the Emergency Feedwater System shall be to maintain proper steam generator water level to promote heat transfer and an orderly cooldown of the Reactor Coolant System. Manual operator action may be required for proper EFW control. (Reference 20.5.1.2.1, "FSAR Sections 10.4.7 and 7.4.3" on page 28). This scenario requires consideration of any single active failure and the possibility of a seismic event.

7. FIRE/FLOOD/SABOTAGE

For the scenario where both the Main Feedwater and Emergency Feedwater Systems are inoperable due to an Appendix R fire, Turbine Building flood, or sabotage, the Standby Shutdown Facility Auxiliary Service Water System shall provide an alternate, totally independent means of secondary side decay heat removal for achieving and maintaining hot shutdown. See the SSF Design Bases Specification for further details. (Reference 20.5.1.2.4, "FSAR Section 9.6" on page 28).

8. ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

For this event credit shall be taken for the operation of three EFW pumps. The ATWS Mitigation System Actuation Circuitry (AMSAC) initiates the start of these three pumps (Reference 20.5.2.2.1, "Design Basis Document OSS-0254.00-00-2001,Rev.0; ATWS Mitigation System Actuation Circuitry (AMSAC) and Diverse Scram System (DSS) Design Bases Specification" on page 30). The licensing basis does not require consideration of a single failure or a seismic event.

20.1.1.3 SUPPORT/INTERFACING SYSTEM REQUIREMENTS

1. CONDENSATE SYSTEM

The upper surge tanks are the assured, safety-grade water source for the EFW System. An assured, safety-grade supply of condensate quality water shall be normally aligned as the primary suction source for the motor driven and turbine driven emergency feedwater pumps. It shall have sufficient capacity to allow the operator time to manually align alternate sources. Sufficient water supply should be available to cool the reactor coolant system to the decay heat removal operating conditions after any of the design basis transients.

The hotwell (non-safety) provides alternate capacity. Additional makeup may be available from the condensate storage tanks and demineralized water system. Raw (lake) water can be used as a last resort (via either the ASW Pump or the SSF ASW System).

All manually operated valves in the piping from the USTs to the suction of the EFW pumps shall be locked open. (Reference 20.5.1.9.1, "Letter from WO Parker to HR Denton(NRC), dated 4-3-81" on page 29).

In a Loss of Lake Keowee event, the CCW recirculation mode is the preferred method for shutdown cooling. Concerns were raised by the Atomic Energy Comission (AEC) that the underwater weir wall may not withstand the ensuing rapid drawdown of Lake Keowee by a dam failure. The AEC decided, however, that adequate cooling water was available in the water trapped in the CCW System intake and discharge piping, assuming complete failure of the weir wall. The inventory trapped in the CCW System intake and discharge piping has been evaluated to supply approximately 37 days of emergency cooling using the ASW system (Reference 20.5.2.1.6). Therefore, the ASW system is required to mitigate a loss of Lake Keowee event with a loss of the CCW recirculation capability.

20.1.2.3 SUPPORT/INTERFACING SYSTEM REQUIREMENTS

The CCW System shall provide a source of water for the ASW pump. The buried CCW pipe provides the source of water. The Unit 2 intake piping below the elevation of the turbine building floor has been evaluated to supply approximately 11 hours of SG makeup to the three units. (Reference 20.5.2.1.6, "Duke Calculation OSC-864, Rev.0; RC System DH Removal Following Loss of Intake Structure" on page 29).

20.2 SYSTEM SPECIFIC DESIGN CRITERIA

NONE

20.3 SYSTEM GENERIC DESIGN CRITERIA

20.3.1 SINGLE FAILURE

The EFW System shall be capable of withstanding any credible single failure during certain of the system design basis events described in Section 20.1.1.2, "DESIGN BASIS EVENTS" on page 4. The description of each event indicates that a single failure either needs to be considered or need not be considered in addition to the initiating event.

For Oconee, spurious operation of a powered component need not be considered when designing a system to withstand a single failure. For example, a normally open EMO valve that is required to remain open to perform its safety function is not assumed to be closed either by single failure (assuming the valve controls are 1E) or by failure of non-safety control components.

For Oconee, passive failures are only assumed credible in the Emergency Core Cooling System (ECCS) systems. Since the EFW System is not considered an ECCS system, only active failures need to be considered in the EFW System.

A single failure of either EFW control valve to open is considered a credible event. Such a failure would result in no adverse effects for initiating events other than secondary side piping failures since the redundant EFW train would be unaffected by the failure. Since breaks larger than a certain size require isolation of the faulted steam generator, the ability to feed a steam generator via the EFW headers cannot be assured for secondary side breaks that require steam generator isolation if the control valve in the other EFW header fails to open. This limitation was recognized when the present system was designed and licensed. It has been determined to be acceptable based on the low probability of the scenario and the existance of several non-safety grade, alternative paths of supplying water to the steam generators. These alternative methods include:

1. Licensing documentation explicitly identifies bypassing the failed valve via the Main Feedwater System startup control valve while acknowledging that this alignment is not fully safety grade (e.g., the control valve is positioned via the ICS using instrument air without a backup supply, use of non-active valves, etc.). (Reference 20.5.1.2.6, "FSAR Section 10.4.7.3" on page 28).
2. Restablishing Main Feed Water System operation to supply makeup water to the intact steam generator. (The system is not safety-grade but is designed to supply low flowrates at high steam generator pressure during unit startup and shutdown.)
3. Supplying makeup water to the intact steam generator using a hotwell pump and condensate booster pump. This would require lowering the pressure in the steam generator by venting steam.
4. The SSF Auxiliary Service Water Pump (which did not exist when the EFW System was extensively modified and relicensed) can supply the intact steam generator at full pressure.
5. Supply the intact steam generator via the Auxiliary Service Water Pump. This would require lowering steam generator pressure by venting steam.

20.3.2 FIRE PROTECTION SAFE SHUTDOWN CONSIDERATIONS

For a fire anywhere in the plant except the West Penetration Room or the SSF, the assured method of establishing and maintaining hot shutdown shall be from the SSF. The SSF ASW Pump discharge piping ties into the EFW piping in the West Penetration Rooms. There shall be check valves in the EFW systems that are located within suitable boundaries which prevent significant flow from going in the wrong direction. There shall be no equipment in the EFW systems that is required to be repaired through damage control procedures after postulated fire events to cool its reactor coolant system to cold shutdown conditions.

For a fire in the West Penetration Room or the SSF, the assured method of establishing and maintaining hot shutdown and to cool the reactor to cold shutdown conditions shall be from the control room using available normal plant equipment. The location of the equipment and its supporting power and controls shall assure that at least one train of EFW is free from fire damage and available to supply makeup to at least one steam generator.

20.3.3 SYSTEM CLASS

The Auxiliary Service Water System and portions of the EFW are Oconee Class F. The Emergency Feedwater System was originally designed to the requirements of ANSI B31.1 (July 1967). Since the system is required to mitigate the consequences of an accident, portions of the piping were analyzed and qualified to survive a design basis seismic event. These portions are shown on the flow diagrams as Oconee Class F. The piping material shall be compatible with secondary side chemistry.

20.3.4 SEISMIC

The Emergency Feedwater System shall be designed to mitigate the consequences of the accidents listed below coincident with a design basis earthquake:

1. Loss of normal feedwater with offsite power available
2. Loss of normal feedwater without offsite power available
3. Small break loss of coolant accident

The Upper Surge Tanks and the piping connecting them to the EFW pumps have been analyzed and qualified to withstand a design basis seismic event. This includes piping that supplies other systems up to the first normally closed valve. The hotwell and connected piping which contain the secondary EFW water supply have been evaluated using a "seismic experience" approach and found to be capable of withstanding a seismic event. Although the evaluation methodology isn't formally recognized for licensing bases, Duke considers this secondary water supply a "seismically assured" source of water.

The Auxiliary Service Water Pump and piping shall be qualified to withstand a design basis seismic event.

20.3.5 TORNADO/WIND

The EFW System is vulnerable to the effects of a tornado. The UST, the primary source of water for the EFW Pumps, is located on the sixth floor of the Turbine Building. The UST is partially shielded by the sixth floor of the Auxiliary Building, but is still vulnerable to the effects of high winds. The EFW pumps are physically located in the Turbine Building basement. Since the pumps are located below ground level, the pumps are essentially protected from the effects of a tornado. The 4160 VAC switchgears that provide power to the motor driven EFW pumps are vulnerable to the effects of tornado damage. The EFW pump discharge piping passes into the Auxiliary Building and rises up into the east and west penetration rooms before entering the Reactor Building. The piping is vulnerable to Auxiliary Building damage.

The Standby Shutdown Facility Auxiliary Service Water System (SSF-ASW) provides additional capability to remove core decay heat following a tornado. However, the SSF-ASW piping and cabling located in the West Penetration is vulnerable to tornado damage. Collapse of any portion of the West Penetration Room structure may render the SSF-ASW system unavailable (Reference 20.5.2.4.3).

Another source of feedwater is the auxiliary service water system (ASW). The ASW Pump portion of the EFW system shall be designed to withstand the effects of a tornado and perform as described in Section 20.1.2. The piping and valves downstream of the isolation valves to each SG are not protected from the effects of a tornado. However, due to physical separation it is credible for only the east or the west penetration room equipment on all units to be damaged by a tornado. Thus, as a minimum, the piping in the undamaged penetration room for each unit shall be available for supplying water to a steam generator following a postulated tornado.

A tornado is assumed capable of damaging a variety of locations in the plant. Mitigation of tornado damage relies upon the capability of providing decay heat removal from any of several different systems. The NRC's acceptance relies on probabilistic risk using a combination of EFW, SSF-ASW, and ASW to assure the decay heat removal function (Reference 20.5.1.3.2).

20.3.6 MISSILES

The only portion of the EFW system that shall be required to be protected from the effects of tornado missiles is the ASW pump and discharge piping (including necessary isolating valves) up through the valves that isolate the ASW piping from each steam generator. Protection of the EFW system from postulated turbine missiles is not part of the design bases.

20.3.7 PIPE RUPTURE/SUBCOMPARTMENT PRESSURIZATION

Licensing documents (Reference 20.5.1.2.7, "FSAR Section 3.6.1.1" on page 28) state the basic design criteria for pipe whip protection. These criteria are:

1. All penetrations shall be designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.

FORM SPD-1001-2

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) STATION: Oconee Nuclear UNIT: 1 Yes 2 Yes 3 Yes
OTHER: ______

(2) CHECK LIST APPLICABLE TO: NSM - ON-1275

(3) SAFETY EVALUATION - PART A

The item to which this evaluation is applicable represents:

Yes ✓ No ____ A change to the station or procedures as described in the FSAR; or a test or experiment not described in the FSAR?

If the answer to the above is "Yes", attach a detailed description of the item being evaluated and an identification of the affected section(s) of the FSAR.
The description of and procedures for the present emergency FD system must be changed to include the new motor driven E pumps. FSAR should be revised to show new equipment and a new evaluation of emerg. FDW sys. should be described in FSAR.

(4) SAFETY EVALUATION - PART B

Yes * No ____ Will this item require a change to the station Technical Specifications?
* A change to the station Tech. Spec may be required.
If the answer to the above is "Yes," identify the specification(s) affected and/or attach the applicable pages(s) with the change(s) indicated.

(5) SAFETY EVALUATION - PART C

As a result of the item to which this evaluation is applicable:

Yes ____ No ✓ Will the probability of an accident previously evaluated in the FSAR be increased?
Yes ____ No ✓ Will the consequences of an accident previously evaluated in the FSAR be increased?
Yes ____ No ✓ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
Yes ____ No ✓ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
Yes ____ No ✓ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
Yes ____ No ✓ May the possibility of a malfunction of equipment important to the safety different than any already evaluated in the FSAR be created?
Yes ____ No ✓ Will the margin of safety as defined in the bases to any Technical Specifications be reduced?

If the answer to any of the preceding is "Yes", an unreviewed safety question is involved. Justify the conclusion that an unreviewed safety question is or is not involved. Attach additional pages as necessary.
This modification will greatly improve the margin of safety for Oconee Nuclear Station by providing a new emergency FDW sys. for each unit.

(6) PREPARED BY: L. E. Summerlin DATE: 5-3-79

(7) REVIEWED BY: R E Hall DATE: 5/3/79

(8) Page 1 of 1

FINAL DESIGN SUMMARY

This modification provides the design for the addition of a separate and redundant emergency feedwater system for each of the three units at Oconee. Two (2) motor driven emergency feedwater pumps are to be added for each unit. Each of these pumps will take suction from the unit's upper surge tank with the backup capability to take suction from the unit's hotwell. The discharge piping of each pump has been tied into the existing emergency feedwater piping to allow each of the new pumps to serve either "A" or "B" steam generator or to serve "A" and "B" steam generator of each unit. The existing piping will also allow the motor driven EFPs of one unit to supply feedwater to another unit's steam generators should the need arise.

The new motor driven EFPs for each unit are to be located in the turbine building basement east of the hotwell pumps. To provide a suction path to the upper surge tanks requires that the control valve C-181 be removed. A tee will be made up to replace this valve and bolt up to the flanges of valves C-180 and C-183. The branch path of this tee will be piped into the suctions of each of the motor driven EFPs for that unit. An additional suction for the new pumps will be provided from the hotwell by a connection into this same line between C-183 and C-184 valves. This connection will require that the unit be out of service. A check valve and a manual valve will be provided at each of these connections. A manual valve will also be provided in the suction piping for isolation of either pump from the system for maintenance.

The discharge piping for each pump will have a minimum flow recirculation line to the condensate storage tank. The discharge of each pump will have a motor operated valve and a check valve. The two normal discharge paths per pump will be to the normal emergency feedwater line to "B" OTSG and to the emergency emergency feedwater line to "B" OTSG for the "B" pump and the same "A" lines for the "A" pump. Each of these pumps can also serve the other OTSG for that unit or serve the OTSGs for another unit through the existing piping. See the PO drawings for each unit to determine the actual flow paths available.

All of the piping for the motor driven EFPs will be Class "F". The route of this piping has been checked in the station. The gravity, thermal and seismic analysis of this piping was made using this piping configuration. The hanger/restraints will be designed for this piping configuration and the piping material has been ordered. Revisions to this piping configuration should not be made without the prior approval of the Station Support Section of Design Engineering.

This addition was designed to allow installation (with exception of one suction to pumps) with the unit in operation. All piping should be installed in a clean condition as it will be very difficult to flush this piping with the unit in operation. Flushing of the system will also create large quantities of waste water to be disposed of. Hydro testing of this piping can also be done with the unit in service.

The electric motors for the new pumps are water cooled. Low Pressure Service Water (LPSW) will be used for cooling water. The connection to the LPSW will be made in the 3" strainer back wash line. This connection can be made with the LPSW in service. A line will be run to each unit and that line will be divided into two lines, one for each motor. The cooling water from the two motors will be recombined into a single line. This single line will be tied into the Condenser Circulating Water (CCW) return from the condensate coolers. This will require a "wet tap" if the unit is in service. A valve is to be provided at the connection in the CCW line.

The cooling water piping for the motors will be Class "F". This piping is to be field routed. The hangers/restraints for all of the piping involved in this modification are to be designed by Design Engineering personnel.

The suction piping from the hotwell to the motor driven EFPs is to be installed when the unit is off the line. The permanent valves or piping blanks can be used to allow use of the system until the unit can be removed from service to install the suction pipe to the hotwell.

NSM 22640

PROJECT DESCRIPTION & FUNCTIONAL DESCRIPTION

REVISION 1

The purpose of this NSM is to bring the Oconee Emergency Feedwater System into full compliance with seismic design requirements. DNC-0061 first documented noncompliance with seismic criteria. Modifications required to bring the system into compliance were identified in Design Study ONDS-146. These modifications, revised on several occasions throughout the design process, are detailed below.

The completion of the NSM will also satisfy all remaining items of NRC concern documented in their Safety Evaluation Report of 1-14-87 regarding Duke's response to NRC Generic Letter 81-14.

Project Description:

a. Replace the following valves with Oconee Class F, QA Condition 1 valves:

   2C-187
   2C-192
   2C-176

   Note: Existing valves C-176 and C-187 have limit switches. The replacements for these valves do not. Associated cabling is being removed. These valves are being revised to fail closed. 1

b. Extend the Class F, QA Condition 1 boundary as follows: (Piping layouts will not change, but Class G piping will require replacement with piping qualified for Class F.)

FROM	THRU
2C-175	2C-176
2C-186	2C-187
2C-191	2C-192

c. Extend the Class F, QA Condition 1 boundary thru the second restriction orifice beyond valve 2FDW-89. (Piping layout will not change, but Class G piping will require replacement with piping qualified for Class F.)

d. Obtain vendor certification of the seismic capability of valve 2FDW-318.

e. Downgrade the TD EFW Pump seal injection lines from Class F to Class G. Install a flow restriction orifice at the new F/G break upstream of valve 2FDW-85.

f. Remove approximately 600 feet of piping, with valves, associated valve controls, and supports, from the plant heating system which presently ties into the Upper Surge Tanks (4 lines per unit). Cap the cut lines.

FSD PAGE 7

g. Replace the flange bolts in 2C-152, -153 with thru bolts that extend thru both pairs of flanges.

h. Perform mechanical system calculations demonstrating the adequacy of LPSW even with failure of non-seismic valves.

Note: Items c. and e. above concern piping identified as needing support/restraint repair during Duke's review in response to NRC GL 81-14. (Reference T D Brown's Memo To File dated 9-1-87, File OS-161) The necessary repairs and/or upgrade is performed within the scope of the stress analysis and support-restraint review required for items c. and e.

Functional Description:

The only functional changes to the plant concern items a, e, and f.

Item a: Valves 2C-176 and 2C-187 are being revised to fail-closed so that a failure will not result in drainage of the Upper Surge Tank contents into the hotwell. Loss of the Upper Surge Tank contents must be prevented since this volume is the assured source of water for EFW following a seismic event. 1

Additionally, limit switches on valves C-176 and C-187 are being removed. The function of the limit switches was to close C-192 if valves C-176 and/or C-187 opened. (Since flow thru C-192 is very small compared to C-176 or C-187, eliminating this interlock will have a negligible impact on hotwell level control.)

Item e: The seal injection lines for the Turbine-Driven EFW pump are not required for EFW system operability, but loss of these lines must not result in loss of excessive amounts of EFW flow. A flow restriction orifice is being added in the seal supply line to prevent excessive fluid loss due to potential line breaks following a seismic event.

Item f: The plant heating lines, designed for heating the Upper Surge Tanks, are not used and are unnecessary. This heating function with the associated lines is being eliminated.

HAH
3-17-89

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26173 (R10-86) FORM 101.3 REVISION 1

CERTIFICATION OF ENGINEERING CALCULATION - CONTINUATION

STATION AND UNIT NUMBER Oconee Nuclear Station Units 1,2, and 3

TITLE OF CALCULATION USQ review of NSMs ON-1,2,3-2640-EFW seismic upgrade (keywords: S/E*,EFDW*)

CALCULATION NUMBER OSC-3047

REVISION/ADDENDA LOG:

REV NO	CALCULATION PAGES (VOL)			ATTACHMENTS (VOL)			VOLUMES		ORIG	CHKD	APPR	ISSUE DATE
	REVISED	DELETED	ADDED	REVISED	DELETED	ADDED	DELETED	ADDED	DATE	DATE	DATE	REC'D DATE
6	1a, 2, 3, 5, 7, 8a	—	1b, 3a,	—	—	—	—	—	KW Sandel 10/27/89	R Paul Colaiano 10/27/89	[signature] 10/30/89	KWS 10/31/89 CEB 11/6/89

FOR INFORMATION ONLY

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Form 01077 (R3-88)

FORM 101.1	REVISION 11

CERTIFICATION OF ENGINEERING CALCULATION

STATION AND UNIT NUMBER Oconee Nuclear Station Units 1,2, and 3

TITLE OF CALCULATION USQ Review of NSMs ON-1,2,3-2640 - EFW Seismic Upgrade (Keywords: S/E*, EFDW*)

CALCULATION NUMBER OSC-3047

ORIGINALLY CONSISTING OF:

PAGES 1 THROUGH 8

TOTAL ATTACHMENTS 0 TOTAL MICROFICHE ATTACHMENTS 0

TOTAL VOLUMES 1

THESE ENGINEERING CALCULATIONS COVER QA CONDITION 1 ITEMS. IN ACCORDANCE WITH ESTABLISHED PROCEDURES, THE QUALITY HAS BEEN ASSURED AND I CERTIFY THAT THE ABOVE CALCULATION HAS BEEN ORIGINATED, CHECKED OR APPROVED AS NOTED BELOW:

ORIGINATED BY KW Sandel DATE 6/1/88

CHECKED BY R.V. Foster DATE 6/2/88

APPROVED BY RCB P.M. Abraham DATE 6/3/88

ISSUED TO GENERAL SERVICES DIVISION L.C. Deaver DATE 6/16/88

RECEIVED BY GENERAL SERVICES DIVISION PC Fishback DATE 6/16/88

MICROFICHE ATTACHMENT LIST: ☐ Yes ☒ No SEE BACK OF FORM

REV. NO.	CALCULATION PAGES (VOL)			ATTACHMENTS (VOL)			VOLUMES		ORIG	CHKD	APPR	ISSUE DATE
	REVISED	DELETED	ADDED	REVISED	DELETED	ADDED	DELETED	ADDED	DATE	DATE	DATE	REC'D DATE
1	2,3,4, 5,6,7, 8	-	1a						KWS 10/19/88	RWD 10/19/88	RCB PMA 10/19/88	—
2	1a,3,7	-	8a						KWS 10/28/88	RWD 11/1/88	RCB PMA 11/2/88	RCB 11-16-88 SGB 11/16/88
3	1a,2,3, 4,5,6, 7,8a	-	-	-	-	-	-	-	KW Sandel 3/22/89	R Paul Colaianni 3/22/89	— —	— —
3	—	—	—	—	—	—	—	-	—	R Paul Colaianni 3/27/89	[illegible] 3/28/89	KWS 3/28/89 CEB 3/28/89
4	1a,2, 4,6, 8a	-	-	-	-	-	-	-	KW Sandel 4/19/89	R Paul Colaianni 4/19/89	[illegible] 4/19/89	KWS 4/19/89 CEB 4/19/89
5	1a,3, 4,5,7, 8a	-	-	-	-	-	-	-	KW Sandel 9/19/89	R Paul Colaianni 9/20/89	[illegible] 9/21/89	KWS 9/21/89 CEB 9/21/89

Form 101.2 Revision 0
26164 (12-82)

Sheet 1a of 8

Calculation No. OSC-3047

Added in Revision 1

REVISION DOCUMENTATION SHEET

REVISION NUMBER	REVISION DESCRIPTION
1	Incorporated PCA 1230 which deleted limit switches on valves C-176 and C-187. Also stated stress analysis and S/R work was completed on Units 1 and 2. By KWS Date 10/19/88 Ckd RmD Date 10/19/88
2	Incorporated PCA 1235 which deleted limit switches on Unit 2 valves C-176 and C-187. Also added final scope document (Revision 1) for Unit 1 to the list of references. By KWS Date 10/28/88 Ckd RmD Date 11/1/88
3	Incorporated PCA 1239 which deleted limit switches on Unit 3 Valves C-187 and C-176. Added final scope documents for Units 2 and 3 to list of references. Revised to state some class G pipe, which could not be qualified as class F would be replaced with class F pipe. By KWS Date 3/22/89 Ckd. RPC Date 3/22/89
4	Incorporated Revision 2 to final scope document for ON-22640/0 and Revision 1 to final scope document for ON-32640/0. Both revisions deleted statement that valve C-187 would be revised to fail closed since the controls and failure modes of these valves are not being changed. By KWS Date 4/19/89 Ckd. RPC Date 4-19-89
5	Revised to incorporate Revision 2 to final scope document for NSM ON-32640/0. This revision changed proposed rerouting of LPSW piping. By KWS Date 9/19/89 Ckd RPC Date 9/20/89
6	Revised to incorporate Unit 3 VNs OC-3560 and OC-3565 which removed solenoid valve BY KWS DATE 10/27/89 CKD RPC DATE 10/27/89

Statement of Problem

The purpose of this calculation is to determine if any unreviewed safety questions exist due to NSMs ON-12640/0, ON-22640/0 and ON-32640/0. The criteria of 10CFR50.59a(2) will be applied to determine if any unreviewed safety questions exist. In determining the presence or absence of an unreviewed safety question, this calculation is classified as QA Condition 1.

Description of Modification

The following is a description of this modification, which upgrades portions of the EFW System:

1. Extend Class F, QA Condition 1 boundary from C-186 thru C-187. Replace valve C-187 with a Class F, QA Condition 1 valve. Revise piping stress analysis. Hangers will not need revising, per Design Study ONDS-146 (Units 1, 2, and 3). See notes 4 and 5. 6

2. Extend Class F, QA Condition 1 boundary from C-191 thru C-192. Replace valve C-192 with a Class F, QA Condition 1 valve. Revise piping stress analysis. Hangers will not need revising, per Design Study ONDS-146 (Units 1, 2, and 3).

3. Extend Class F, QA Condition 1 boundary thru the second restriction orifice beyond valve FDW-89. Piping layout will not change. Revise piping stress analysis (Units 1, 2, and 3).

4. Obtain vendor certification of seismic capability of valve FDW-318 (Units 1, 2, and 3).

5. Replace valves 1FDW-433, -434 with Class F, QA Condition 1 valves. "Vent and Drain Criteria" will apply (Unit 1 only).

6. Replace the flange bolts in C-152, -153 with thru bolts that extend thru both pairs of flanges (Units 1, 2, and 3).

7. Remove approximately 600 feet of piping, with valves and supports, from the plant heating system which presently ties into the Upper Surge Tanks (4 lines per unit). Cap the cut lines (Units 1, 2, and 3).

8. Extend Class F, QA Condition 1 boundary from C-175 thru C-176. Replace valve C-176 with a Class F, QA Condition 1 valve. Valve C-176 will be changed to fail closed. Revise piping stress analysis. Hangers will not need revising, per Design Study ONDS-146 (Units 1, 2, and 3). See notes 4 and 5. 6

Rev. 6 - Page Retyped

9. Perform mechanical system calculations demonstrating the adequacy of LPSW even with failure of non-seismic valves (Units 1, 2, and 3). See Note 3.

10. Downgrade turbine-driven EFW Pump Seal Injection lines from class F to class G. Install a flow restriction orifice at the new class F/G break upstream of valve FDW-85. The orifice will be located past the turbine-driven EFW minimum flow line to the Upper Surge Tank (Reference 7).

11. Revise the routing of LPSW supply to several air handling units. The new routing will require a new 8" supply line and new valve 3LPSW-844 downstream of valve 3LPSW-45. The existing 8" supply line upstream of 3LPSW-45 will have valves 3LPSW-260 and -403 removed and the line will be isolated by a blind flange and a pipe cap (Unit 3 only).

Notes:

1. For items 1, 2, 3, and 8, piping layouts will not change, but Class G piping will be replaced with Class F qualified piping.

2. Deleted.

3. The calculation to be performed to determine the adequacy of LPSW has not been completed. This part of the modification does not require any changes to the plant. If any plant changes are required due to the calculation results, the changes will be performed under a revision to this modification or a new modification. Therefore since no changes to the plant will be a part of this modification, Item 9 will not be addressed under the criteria of 10CFR50.59.

4. Existing valves C-176 and C-187 have limit switches. The replacements for these valves will not. Associated cabling will be removed. The function of the limit switches was to close C-192 if valves C-176 and C-187 opened. Flow thru C-192 is very small compared to flow thru C-176 or C-187, so allowing C-192 to remain open while the others are open will have a negligible impact on level control (Units 1, 2, and 3).

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By KWS Date 10/27/89

Ckd RPC Date 10/27/89

Rev. 6 - Page Added

5. The Unit 3 solenoid valve (and its associated wiring) that closed valve C-192 upon receiving a signal from the limit switches on valves C-176 and C-187 will also be removed. This solenoid valve also receives a signal to close from valve C-196 but the line that has valve C-196 in it is not used (Unit 3 only).

(References 1,8,12,13,27,28,30,33, and 34)

Reasons for each of the respectively numbered changes are as follows:

1. Per FSAR Section 3.7.3.9, seismic boundaries must be at a remotely operable valve or a normally closed manual valve. Valve C-186 is a normally open manual valve. Valve C-187 is remotely operable.

2. Similar to reason 1.

3. A break before the second orifice could possibly "starve" the turbine driven EFW pump cooling line. A break past the second orifice will not "starve" the pump.

4. Present valve is not class F qualified but is in a class F line.

5. Similar to reason 4.

6. Similar to reason 4. Bolts will seismically qualify valves (References 24 and 25).

7. Piping is not used and is a non-seismically qualified line off a seismic qualified tank.

8. Similar to reason 1. Valve C-176 will fail closed to prevent the Upper Surge Tank from draining to the condenser hotwell.

9. Calculation will show that failure of the non-seismic boundary LPSW valves will not prohibit LPSW from achieving its safety related design function.

10. Valves FDW-86, -87, -129, and -218 are supposed to be class F but are not seismically qualified. There are also uncertainties with the "seismic" support/restraint design.

11. Manual normally open valves 3LPSW-260 and -403 are not qualified as seismic boundary valves. Valve 3LPSW-45 is qualified as a boundary valve. The reroute will have the class F branch line connect to a qualified class F line and boundary valve.

(Reference 7) | 5

Safety Review and USQ Evaluation

The EFW System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the main feedwater system to remove energy stored in the core and primary coolant. The portion of the condensate system

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that this NSM involves is the EFW pumps' supply water and the turbine driven EFW pump's cooling water. The LPSW System provides cooling water for normal and emergency services throughout the station, including EFW pump motor air coolers and the turbine driven EFW pump turbine bearing oil cooler.

The modification is QA Condition 1 and safety related. The system upgrades are being made to meet original seismic design criteria. Valve power/air supplies will not be changed for valve replacements. The only electrical changes will be the removal of cabling to limit switches on the present valves C-176 and C-187 and the removal of the solenoid valve and its wiring that was controlled by the limit switches. The replacement valves will not have limit switches. Appendix R criteria will not be affected since no cabling is being rerouted. The function of the limit switches was to close C-192 if valves C-176 or C-187 opened. Flow thru C-192 is very small compared to flow thru C-176 or C-187, so allowing C-192 to remain open while the others are open will have a negligible impact on level control. These valves are to add water to the condenser hotwell from the Upper Surge Tank when the hotwell reaches certain levels. The solenoid valve also receives a signal from valve C-196 but the line that has valve C-196 in it is not used. (References 7, 8, and 34). The valve changes and minor piping changes were reanalyzed by the pipe stress analysis group for Units 1, 2, and 3. Hangers will not need to be redesigned per Design Study ONDS-146 for some changes. Support changes not covered under the design study were reviewed for Units 1, 2, and 3 (References 13, 26, 27, 28, 30, 31, 32). 6

The criteria of 10CFR50.59 will be applied to determine if any unreviewed safety questions exist. The criteria and responses are as follows:

1. Will the probability of an accident previously evaluated in the FSAR be increased?

   No. This modification is not related to any of the conditions or events which lead to accidents analyzed in the FSAR.

2. Will the consequences of an accident previously evaluated in the FSAR be increased?

   No. Most parts of this NSM upgrade portions of the EFW System so that they are qualified seismically. These portions are presently supposed to be qualified but documentation to verify the seismic qualification is not available. The safety function for the pneumatic valves being replaced is for boundary isolation (seismic) of the Upper Surge Tank, which is the assured source of EFW. The removal of the plant heating system piping will eliminate a potential seismic qualified/non-seismic qualified interaction. The plant heating system for the Upper Surge Tank is not used. The plant heating lines were evidently provided to protect against the freezing of the tanks contents. It is now felt that the freezing of the tank's contents is unrealistic, especially due to the high

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improbability of a temporary drop of air temperature in the Turbine Building below 32°F. Also the freezing of other components important for continued plant operation is much more apt to happen long before this tank's contents could freeze (Reference 10). The turbine driven EFW seal injection lines will have an orifice installed and will be downgraded from class F to G on the downstream side of the orifice. If this line fails in a seismic event, it is judged that the pump's performance will not be adversely affected and significant loss of EFW water supply thru the seals will not occur. Also the orifice will prevent excessive EFW water loss from the pumps discharge in the event of a break in the class G pipe (Reference 11). The rerouting of the LPSW piping (Unit 3 only) will have a motor operated valve as the class F/G break with only class F pipe on one side of the valve and only class G pipe on the other side. These changes will all better qualify the EFW System to mitigate the effects of a seismic event and to shut the plant down safely during or following a seismic event. The removal of limit switches on valves C-176 and C-187 will have negligible impact on hotwell level control.

3. May the possibility of an accident which is different than already evaluated in the FSAR be increased?

   No. The EFW System will still be able to perform its intended function and no unusual conditions will be created to lead to the possibility of an accident scenario different than those already evaluated in the FSAR.

4. Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

   No. The affected piping, valves, orifice, equipment, and operation of the EFW system will be qualified for seismic conditions. This qualification will increase the reliability of these components for seismic events and not introduce other failure modes previously evaluated in the FSAR.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

   No. Valve C-176 will now fail closed on loss of air. This change in failure mode will prevent the safety related EFW source in the Upper Surge Tank from draining to the condenser hotwell on loss of air to these valves. 4

6. May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

   No. The additional orifice in the turbine-driven EFW pump seal injection line may clog up but the pumps performance is not judged to be adversely affected and significant loss of EFW water supply thru the seals will not occur (Reference 11). The new orifice is not in the turbine-driven EFW minimum flowline so the minimum flowpath is not degraded (Reference 7). The thru bolts in valves C-152 and -153 do not add any new failure modes. Other equipment that is affected by the modification is already addressed in questions 4 and 5.

7. Will the margin of safety as defined in the bases to any Technical Specification be reduced?

   No. No safety/design limits are adversely affected so margins of safety as defined in the bases to Technical Specifications will not be reduced.

Conclusion

There are no unreviewed safety questions associated with this NSM.

References

1. Letter No. MBOE-87-79 dated 2/13/87 from H. A. Hammond to R. L. Cope, et. al, File: OS-161 sending NSM Scope documents.

2. NSMS ON-12640/0, ON-22640/0, and ON-32640/0 Request, Approved 12/2/86.

3. Memo to File dated 3/11/86 from P. F. Guill on seismic qualification of EFW System.

4. Letter dated 3/11/86 from P. F. Guill to R. D. Groux on seismic qualification of EFW System, File: OS-802.01.

5. Oconee Nuclear Station FSAR, 1988 Update, Sections 3.2, 3.7, 9.2.2, 10.4.6, 10.4.7, and 15.8.

6. Oconee Nuclear Station Technical Specifications, as amended to 9/12/89, Sections 3.3, 3.4, and 4.9. 6

7. Conversations with Hugh Hammond, MBOE.

8. PCAs 716, 717, 718, 751, 752, 753, 802, 803, 804, 825, 829, 840, 842, 843, 852, 853, 913, 927, 915, 914, 926, 916, 943, 944, 945, 1230, 1235, 1239, 1396.

9. Letter dated 4/28/87 from R. C. Bucy to H. A. Hammond, File: ON-1,2,3-2640, requesting additional information on initial installation of plant heating lines.

10. Letter No. MBOE-87-280 dated 5/28/87 from H. A. Hammond to R. C. Bucy, File: ON-1,2,3-2640/0 giving information of plant heating lines.

11. Memo to File No. MBOE-87-389 dated 8/10/87 by H. A. Hammond stating loss of seal injection to the turbine driven EFW pump is judged to not adversely affect pump performance or not cause loss of water thru the seals.

12. Letter No. MBOE-88-159 dated 3/24/88 from H. A. Hammond to O. J. Gilstrap, File: ON-1,2,3-2640/0 sending updated scope document.

13. Letter No. MBOE-88-246 dated 5/23/88 from H. A. Hammond to O. J. Gilstrap, File: ON-12640/0 sending final scope document.

14. Flow Diagram OFD-121A-1.7, Revision 6.

15. Flow Diagram OFD-121A-1.8, Revision 4.

16. Flow Diagram OFD-121D-1.1, Revision 4.

17. Flow Diagram OFD-121A-2.7, Revision 6.

18. Flow Diagram OFD-121A-2.8, Revision 2.

19. Flow Diagram OFD-121D-2.1, Revision 4.

20. Flow Diagram OFD-121A-3.7, Revision 6.

21. Flow Diagram OFD-121A-3.8, Revision 1.

22. Flow Diagram OFD-121D-3.1, Revision 5.

23. Flow Diagram OFD-124A-3.1, Revision 2.

24. Memo to File No. MBOE-86-146 dated 2/27/86 by H. A. Hammond, File: OS-27-G stating criteria for seismically qualifying cast iron valves with thru bolts.

25. Letter dated 2/24/72 from R. E. Miller to R. L. Dick, File: OS-27B stating solution for seismically qualifying cast iron valves with thru bolts.

26. Telephone conversation on 10/14/88 with Eileen Robinson (CSPF) and David Perry (CSPA) stating Units 1 and 2 stress analysis and support/ restraint work was complete. 1

27. Letter No. MBOE-88-445 dated 10/21/88 from H. A. Hammond to O. J. Gilstrap, File: ON-12640/0 sending revision 1 of final scope document.

28. Revision 2 to final scope document for NSM ON-22640/0, dated 4/17/89.

29. EFW Seismic Upgrade Design Inputs (Mech.), NSM 12640, 22640, 32640, Calculation No. OSC-2554, Revision 0.

30. Revision 2 to final scope document for NSM ON-32640/0, dated 9/18/89.

31. Telephone conversations on 3/13/89 with Eileen Robinson (COPA) and Steve Crews (COPS) which concluded that pipe stress analysis and support/restraint work was completed.

32. Telephone conversations on 9/19/89 with Joe Herrick (COPA) and Rusty Childs (COPS) which concluded that Unit 3 pipe stress analysis was reviewed and support/restraint work was completed for PCA 1396 and VN OC-3509.

33. Variation Notices OC-3560, Approved 10/4/89 and OC-3565, Approved 10/4/89. 6

34. Conversation between Hugh Hammond, MOSE, and Ken Sandel, MONE, on 10/25/89 concerning input signals to solenoid valve and the input signals' functions.

12/1/98 ~~HRC~~ NRC INSPECTOR NOTE:
THE ABOVE LIST OF REFERENCES
DOES NOT INCLUDE
— DUKE LTR OF 5/7/86 TO NRC
RE: SEISMIC QUALIFICATION OF EFW
— NRC SER OF 1/14/87
RE: SEISMIC QUALIFICATION OF EFW

Form 45077 (10-92)

See Attached Calculation: OSC - 5317, Rev. 0

Duke Power Company
10 CFR 50.59 EVALUATION

(1) Station: Oconee Nuclear Station Unit(s): 3

(2) Evaluation for: NSM ON-32911/0, Upper Surge Tank Makeup to Hotwell Control Valves

(3) FSAR sections consulted: 3.0, 7.4.3, 7.5.2.11, 9.2.3, 10.4, 15.0 and Figure 10-6 1992 update

(4) Technical specifications consulted: 3.4, 3.5.6, and 4.9 9-16-93 update

Will technical specification changes be required? ☐ Yes ☒ No

- Technical specifications affected:
- Station Regulatory Compliance personnel contacted:

(5) USQ EVALUATION APPLICABILITY

Does the modification involve a Structure, System, or Component (SSC) that is evaluated in the FSAR or a smaller SSC that is part of an SSC evaluated in the FSAR, and does the modification do more than replace components with equivalent components? ☒ Yes ☐ No

Will the modification degrade the effectiveness of an SSC important to safety in any design basis accident or event? ☐ Yes ☒ No

Does the modification appear to require inclusion in the FSAR due to the installation of a new system significant to plant operation, or installation of a significant addition to an existing system? ☒ Yes ☐ No

(6) USQ EVALUATION

USQ EVALUATION NOT APPLICABLE ☐

May the modification:

Increase the probability of an accident evaluated in the SAR? ☐ Yes ☒ No

Increase the consequences of an accident evaluated in the SAR? ☐ Yes ☒ No

Create the possibility for an accident of a different type than any evaluated in the SAR? ☐ Yes ☒ No

Increase the probability of a malfunction of equipment important to safety evaluated in the SAR? ☐ Yes ☒ No

Increase the consequences of a malfunction of equipment important to safety evaluated in the SAR? ☐ Yes ☒ No

Create the possibility for a malfunction of a different type than any evaluated in the SAR? ☐ Yes ☒ No

Will the modification:

Reduce the margin of safety as defined in the basis for any technical specification? ☐ Yes ☒ No

PROVIDE AN A[illegible]MENT TO SUBSTANTIATE ALL YES AND NO ANSWERS

Prepared by/date: DV Beyson 12-30-93

Reviewed by/date: E [illegible] 12-30-93

Form 01077 (R7-82)

FORM 101.1 | REVISION 14

CERTIFICATION OF ENGINEERING CALCULATION

STATION AND UNIT NUMBER Oconee Nuclear Station, Unit 3

TITLE OF CALCULATION USQ Review of Nuclear Station Modification 32911/0

Upper Surge Tank Makeup to Hotwell Control Valves (S/E*, C*, UST*)

CALCULATION NUMBER OSC - 5317

ORIGINALLY CONSISTING OF:

PAGES 1 THROUGH 5

TOTAL ATTACHMENTS 0 TOTAL MICROFICHE ATTACHMENTS 0

TOTAL VOLUMES 1 TYPE I CALCULATION/ANALYSIS YES ☐ NO ☒

TYPE I REVIEW FREQUENCY

THESE ENGINEERING CALCULATIONS COVER QA CONDITION 1 ITEMS. IN ACCORDANCE WITH ESTABLISHED PROCEDURES, THE QUALITY HAS BEEN ASSURED AND I CERTIFY THAT THE ABOVE CALCULATION HAS BEEN ORIGINATED, CHECKED OR APPROVED AS NOTED BELOW:

ORIGINATED BY D V BENSON DATE 12-30-93

CHECKED BY [illegible] DATE 12-30-93

APPROVED BY TA [illegible] DATE 12-30-93

ISSUED TO DOCUMENT CONTROL DATE

RECEIVED BY DOCUMENT CONTROL DATE

MICROFICHE ATTACHMENT LIST: ☐ YES ☒ NO SEE FORM 101.4

REV. NO.	CALCULATION PAGES (VOL)			ATTACHMENTS (VOL)			VOLUMES		ORIG	CHKD	APPR	ISSUE DATE
	REVISED	DELETED	ADDED	REVISED	DELETED	ADDED	DELETED	ADDED	DATE	DATE	DATE	REC'D DATE

PURPOSE

The purpose of this evaluation is to determine if any unreviewed safety questions are involved with NSM ON-32911/0. The criteria of 10 CFR 50.59 a(2) will be used to make this determination. This evaluation is QA Condition 1 because it determines the presence or absence of a USQ.

MODIFICATION DESCRIPTION

NSM ON-32911 will revise the makeup control circuitry so that when the Upper Surge Tank (UST) level approaches the Tech Spec limit (6 feet) the hotwell level control signal will be overridden and the three makeup control valves, 3C-176, -187, and -192 will automatically close. This will prevent the USTs from draining if the hotwell should be lost. For more information, see Reference 1.

SAFETY REVIEW

The USTs provide the primary source of water for the emergency feedwater pumps. Makeup for the condenser hotwell also comes from the USTs. Water from the USTs flows to the condenser hotwell through three lines. Flow is regulated by air operated valves 3C-176, -187, and -192. If hotwell level should ever be lost with the existing makeup control circuitry, the three valves would open automatically, thus draining the USTs of their contents, and violating the Tech Spec (3.4.4) limit of 6 feet maintained in each tank. This problem is addressed in PIR 4-089-0111. [References 1, 2, 3, and 4]

Two new safety related differential pressure switches will be tapped into the existing Inadequate Core Cooling Monitor (ICCM) impulse lines. The new switches will have sufficient valving to isolate them from the impulse lines if necessary. This arrangement precludes having to adjust ICCM system operation (a Reg. Guide 1.97 requirement) or having to breach the UST shell, thereby reducing possible leakage paths and failure modes. The ICCM level transmitters are unaffected by this modification and will continue to perform their design functions. The reference leg of the new differential pressure transmitter tubing will be trace heated to prevent condensation. [References 1 and 6]

The new system will close the three hotwell makeup valves if the UST level decreases below a predetermined setpoint. A new electrical interlock and solenoid valve arrangement will cause the valves to close by bleeding off air from the valve actuators.

Adequate margin is provided to account for valve stroke time and instrument error to ensure that the Tech Spec required UST level will be maintained. Since portions of the hotwell system are not seismically reviewed, its availability in an accident scenario is not assured. [References 1, 2, 8, and 10]

New safety related cabling will run from the USTs (in the Turb. Building) to the Cable Room AT Cabinet, from the control valves (in the Turbine Building) to the cabinet, and from the Control Room to the cabinet. New trays and conduit will be necessary in some areas. All the new equipment installed by this modification will be safety related, seismically mounted, QA Condition 1 except for some non-safety components used only for testing. The new non-safety components will normally be de-energized and will not adversely affect any safety function. The areas affected have been checked for seismic interaction with existing equipment. [References 1 and 5]

The only change to the Control Board will be to label an existing unused annunciator window. It will read, "UST TO HW MAKEUP VALVES FAILED CLOSED." This change does not require seismic qualification review. [References 1 and 5]

It was recognized that a single failure of the new safety related interlock could cause the makeup valves to close spuriously, but this failure mode is already present, and could be caused by various non-safety components. Thus, it was determined that the new interlock will not significantly increase the probability of spurious valve closure. The system is designed such that a single failure in the circuitry cannot cause the makeup valves to remain open, which protects the UST inventory as required by Tech Specs. [References 1, 5, and 8]

The new electrical system has been evaluated for fault propagation and new failure modes. No new accidents or failure modes were identified. An Appendix R review has been initiated. The final design is complete. The new safety related components will be tested periodically according to procedure, but this evaluation does not address the adequacy of planned testing. The 10 CFR 50.59 evaluation for the procedure change will address the acceptability of the testing. [References 5 and 11]

FSAR Sections 10.4.1.2 10.4.6.2, and 10.4.7.1 will be updated to address this modification. References 8 and 10 will also need revision.

USQ EVALUATION

May the modification:

1) Increase the probability of an accident previously evaluated in the SAR?

No. The USTs and hotwell makeup valves do not initiate any design-basis accidents.

2) Increase the consequences of an accident previously evaluated in the SAR?

No. The new interlock system will ensure that sufficient water remains in the USTs for accident mitigation. Adequate margin will be provided in the setpoint to allow for instrument error and valve stroke time. All new equipment will be QA Condition 1, except for some equipment used only for testing. All the equipment will be mounted seismically (except for these non-safety components), and has been evaluated for new failure modes, system interactions, and fault propagation.

3) Create the possibility for an accident of a different type than any evaluated in the SAR?

No. No new accidents were identified.

4) Increase the probability of a malfunction of equipment important to safety evaluated in the SAR?

No. All new equipment will be QA Condition 1, and will be seismically qualified, except for the new components used only for testing. No single failure will be able to open the makeup valves spuriously. No new failure modes were identified. A single failure of the new safety related components could cause the makeup valves to close spuriously, but this failure mode already exists, and could be caused by any of several non-safety failures. Is was determined that the probability of this failure mode was not significantly increased by the new safety equipment. All new equipment has been evaluated for system interactions and fault propagation. The UST pressure boundary will not be degraded by the new level switches. The exising ICCM impulse lines will not be degraded, and will continue to perform its design functions.

5) Increase the consequences of a malfunction of equipment important to safety evaluated in the SAR?

No. The system will continue to respond as designed. The makeup control valves will fail closed on any single failure in the system, which will assure UST inventory even if the hotwell was lost. The system is not subject to any single failure which could open the valves. The Control Room operators will be alerted by a new annunciator that hotwell makeup from the USTs is stopped due to reading low UST levels.

6) Create the possibility for a malfunction of a different type than any evaluated in the SAR?

No. No new failure modes were identified.

7) Will the modification reduce the margin of safety as defined in the basis to any Technical Specification?

No. This modification will have no effect on plant safety limits, setpoints, or design parameters. Therefore, the margin of safety defined in the Technical Specifications will not be reduced.

CONCLUSION

Based on this discussion, no unreviewed safety questions are created by or involved with this modification. FSAR Sections 10.4.1.2, 10.4.6.2, and 10.4.7.1 should be updated to address this change. No changes to the Technical Specifications are required.

SUMMARY FOR 10 CFR 50.59 ANNUAL REPORT

NSM ON-32911 will provide a means to assure the Technical Specification Upper Surge Tank water inventory is available, even if the condenser hotwell is lost. The hotwell makeup control valves (3C-176, -187, and -192) will be modified to automatically close if surge tank inventory approaches the required level. The valves will also fail closed on any single failure. All new equipment required to perform this function will be QA Condition 1. No USQs are involved with this modification. FSAR Sections 10.4.1.2, 10.4.6.2, and 10.4.7.1 will be updated to include this new feature. No Technical Specification changes are required.

REFERENCES

1) Final Scope Document for NSM ON-32911, Rev. 1, dated 12-30-93.

2) Oconee Nuclear Station Final Safety Analysis Report, Sections 3.0, 7.4.3, 7.5.2.11, 9.2.3, 10.4, 15.0, and Figure 10-6, 1992 update

3) Oconee Nuclear Station Technical Specifications, Sections 3.4, 3.5.6, and 4.9, updated 9-16-93.

4) Problem Investigation Report 4-089-011, dated 6-30-89.

5) Personnal communications between John Boehme, Electrical Engineering, and Damon Bryson, Mechanical/Nuclear Engineering (MNE), on 12-28-93 and 12-30-93 stating that:
   1) the new system was evaluated for fault propagation,
   2) the new interlocks were evaluated for possible seismic system interaction, with no new interaction potential identified,
   3) all new components will be QA 1, safety related, except for the annunciator control cable and some solenoids and cable used exclusively for testing,
   4) all new safety components will be seismically mounted,
   5) the only change to the control boards will be to label an existing unused annunciator,
   6) all new cabling will be run according to seismic and fire protection specifications,
   7) the AT Cabinet is not required to be seismically reviewed even though safety related equipment is contained within,
   8) an Appendix R review has been initiated,
   9) the final design is complete,
   10) all the new electrical power sources used by the new safety components will be safety related, non-battery backed,
   11) all changes will be made in mild environments, so no environmental qualification will be necessary,
   12) the makeup control valves will all fail closed on a loss of power, or any of several other single failures,
   13) no new failure modes or accidents were identified,
   14) the new level switches will not provide a new leakage path out of the USTs,
   15) the new electrical interlock will be not be subject to a single failure causing the makeup valves to open, but it will be possible for a single failure to produce valve closure

6) Personal communications between Rounette Kellahan, MNE, and Damon Bryson, MNE, on 4-7-93 stating that the hotwell makeup lines are 20" and 3", but the valves which control them are 12" and 2". These valves are also the seismic class boundary. The UST side is seismically qualified, but the hotwell side is not. The trace heating mentioned in the Project Description is to prevent condensation problems with the UST level instrument lines, as the USTs are held at vaccuum.

7) Oconee Nuclear Station Selected Licensee Commitments, no sections apply, 11-15-93 update.

8) Oconee Nuclear Station Design Basis Specification OSS-0254.00-00-1000, Rev. 10, "Emergency Feedwater and Auxiliary Service Water Systems", 4-14-93.

9) Duke Drawings OFD-121A-3.7, Rev. 17, and
3.8, Rev. 7.

10) Oconee Nuclear Station Design Basis Specification OSS-0254.00-00-1027, Rev. 0, "Condensate System, Heater Drain System, and Heater Vent System", dated 5-11-93.

11) Personal conversations between Damon Bryson, MNE, and Steve Capps, Project Management, on 12-30-93 stating that the new instruments and interlock was evaluated for required periodic testing by the Instrument and Electrical Maintenance Supervisor responsible for the condensate system. It has been determined that the new system will be added to the refueling-frequency procedures for calibration.

Form 35372 (R10-84)

COMPLETE FORM BY PRINTING WITH BLACK BALL POINT PEN OR TYPE

DUKE POWER COMPANY
NUCLEAR STATION

Problem Investigation Report Serial No. 4-089-0111
Station OCONEE
Investigation Report No.

I. Problem Occurred-Time/Date: NSM Designs Completed: U1: 5/3/87 U2: 6/3/88 U3: 3/11/87 Discovered-Time/Date: 6-30-89
Units(s): 1-3 Unit Status At Time Problem Occurred/Discovered: U1 65% U2 S/D U3 100%
Description and Cause of Problem: FSAR violations at EFW seismic boundaries are being corrected by NSM's 1,2,32640. Although explicit FSAR requirements are satisfied by the NSM, the new design fails to account for automatic opening of the seismic boundary valves between the UST + hotwell. Loss of hotwell level will automatically drain the primary EFW supply (UST) into the hotwell. Valves affected: 1,2,3C-192, -187, -176

Other Duke Stations Affected ☐ Yes ☐ No ☒ Unknown Determined By/Date HA Hammond / RG 1-30-90
Location of Problem: TURBINE BUILDING
Method Used to Discover Problem: Review of valve functions during active valve study ONDS-239
Immediate Corrective Actions Taken/To Be Taken:

Work Stoppage Notification (Form QCK-2A) Serial No. N/A
Information Sources/References (Work Requests, Documents Violated, etc.): O-422 K-23 rev 10
I+C Detail - Hotwell Makeup Control
Originated By: [signature] Date: 6-30-89 Dept./Group/Section DE/NOSE/Mech

II. Compliance Evaluation-Item/System Operable ☒ Yes ☐ No ☐ Not Applicable
Evaluated By/Date: Hugh Hammond 24 July 89 Comments: Design Operability Eval
Reportable ☐ Yes ☒ No Reportable Per: ☒ 50.73 Section ☐ 50.72 Section
T.S./Lic Cond Section ☐ Part 21 ☐ Other ☐ Part 50.9
Evaluated By/Date: [signature] 5 July 89 Comments:

III. Telecon/ENS Report to NRC Time/Date:
NRC Contactee(s): DPC Contactor(s):
Telegraph/Mailgram/Facsimile Transmission to NRC-Date:
Date Notified: NRC Res. Inspector Station Manager
General Office Comments:

IV. Investigation Report Assigned To: NRC Report Due Date:
Date Due to Compliance after Evaluation
PIR Review (Compliance):
PIR Station Manager Approval Date:
Date:

V. Further Action/Evaluation Required ☒ Yes ☐ No (Explain Below):
Page 2 Assigned To: DESIGN
Comments:

Compliance Review Date QA Review CR Hansen Date 2-1-90
RBM 1-30-90

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PROPOSED RESOLUTION

Considering that the intent of the licensing commitment is already satisfied, the following is proposed as a permanent resolution:

1. Complete NSM 32640 as designed.

2. Of the three automatic hotwell make-up paths in question, revise the two larger lines to "normally closed" by closing valves 1,2,3C-175 & -186. Hotwell make-up through these lines would only be allowed during transient modes of operation such as Condensate and/or FDW clean-up. This will fully satisfy the seismic boundary licensing commitments for these two lines.

3. Retain the automatic function of the small (2") hotwell make-up valve. If this normal make-up path is disabled for any reason, an equivalent amount of flow would be allowed through one of the several manual bypasses. (Even if the scenario was again postulated which results in a seismic break, flow lost through this small line would be of little concern.)

4. If the hotwell and associated piping should become seismically qualified through response to GL 87-02 and other actions, valves 1,2,3C-175 & -186 could be returned to the "normally open" condition.

5. The Emergency Feedwater Design Basis Document will be updated appropriately, documenting this resolution to the seismic boundary issue.

NOTE: Corrective action in Design Engineering to preclude similar design deficiencies is considered already implemented. The TOPFORM program, with its emphasis on design verification and client review, provides added assurance that modification designs accomplish their intended purpose.

HHammond 7-21-89

WA Houston 7/21/89

Addendum #1 to Proposed Resolution
PIR 4-089-0111

The proposed resolution dated 7-21-89 must be revised since closure of valves 1,2,3C-175 and -186 would prevent required makeup flow from entering the hotwell for certain situations. As such, items 2 and 4 are re-written below and item 6 is added. The remainder of the proposed resolution is unchanged.

1. Complete NSM 32640 as designed.

2. Valves 1,2,3C-176 and -187 presently open to provide hotwell makeup upon receipt of a low hotwell level signal. A new signal will be provided to override this demand anytime the Upper Surge Tanks reach a minimally acceptable level. The UST level setpoint will be equal to the Tech Spec minimum level plus that required to compensate for volume potentially lost during valve closure.

If necessary, the controls will be safety grade and seismically qualified. They will not be required to meet single failure criteria. Valves 1,2,3C-176, -187 would be included in the active valve list with specific requirements for stroke time. Either valve 1,2,3C-175 or -186 will be normally closed by procedure, with the restriction that if one is open the other must be closed. This limits the impact of valve closure time on the level setpoint.

An SPR will be provided by Design Engineering to initiate this change.

3. Retain the automatic function of the small (2") hotwell make-up valve. If this normal make-up path is disabled for any reason, an equivalent amount of flow would be allowed through one of the several manual bypasses. (Even if the scenario was again postulated which results in a seismic break, flow lost through this small line would be of little concern.)

4. If the hotwell and associated piping should become seismically qualified through response to GL 87-02 and other actions, this new control function and the active classification of the valves would no longer be required.

5. The Emergency Feedwater Design Basis Document will be updated appropriately, documenting this resolution to the seismic boundary issue.

6. Regulatory Compliance should inform the NRC that more time is needed to upgrade the EFW boundaries to meet seismic design criteria imposed by Generic Letter 81-14. (The present commitment is to complete upgrade by the end of the current Unit 3 refueling outage.)

NOTE: Corrective action in Design Engineering to preclude similar design deficiencies is considered already implemented. The TOPFORM program, with its emphasis on design verification and client review, provides added assurance that modification designs accomplish their intended purpose.

HG Hammond 12-19-89
WA Houston 1/4/90

PIR A-089-0111
OPERABILITY EVALUATION

This PIR points out a licensing commitment regarding the seismic boundary of the EFW suction water supply which is not fully satisfied for Units 1 & 2. In the event the seismically non-qualified hotwell piping is postulated to fail during an earthquake, hotwell level would be lost and valves would automatically open and drain the UST. Since the valves are not remotely operable, credit is not allowed for operator action to stop the loss of UST volume.

NSM 12640 & 22640 were implemented and NSM 32640 is scheduled to correct seismic boundary deficiencies identified in 1986. The NSM moves the seismic boundary from normally open valves to valves 1,2,3C-176, -187, & -192. However, since these new boundary valves are not remotely operable, the NSM actually will not satisfy the licensing commitment.

Since the identification of boundary concerns in 1986, engineering analysis has determined the non-qualified portion of the EFW suction (the hotwell & associated piping) can withstand a seismic event. The analysis approach was based on the SQUG/seismic margin techniques presently being finalized for resolution of NRC Generic Letter 87-02 and seismic margin issues. As such, there is no realistic seismic/non-seismic boundary between the UST and the hotwell. Since there are no credible seismic failures which would cause hotwell low level, the EFW water supply would remain in the UST. No credible seismic threat to the EFW water supply exists for the Units 1 & 2 EFW pumps.

The same licensing issues affect Unit 3 as well. The schedule submitted to the NRC for implementing NSM 32640 does not impose the requirements until the end of the next Unit 3 RFO. Nevertheless, the same arguments would apply to Unit 3, since the Unit 3 hotwell and associated piping have also been analyzed for seismic capability.

Based on the above, the EFW system is considered operable for all three units.

[signature] 7-13-89

[signature] 7/21/89

STATION PROBLEM REPORT

(Side 1)

(1) Station OCONEE (2) Unit 1, 2, 3 (3) System(s) CONDENSATE STORAGE 4) SPR # 3/43

(5) Description of Problem SHOULD THE CONDENSER HOTWELL BE LOST THE HOTWELL MAKEUP VALVES WOULD AUTOMATICALLY OPEN AND THE UPPER SURGE TANK INVENTORY WOULD DRAIN OUT. THE UST IS THE PRIMARY SOURCE OF WATER FOR THE EMERGENCY FEEDWATER PUMPS

(6) Does this SPR: DISCUSSED SPR WITH TOM COOTU, WHO IS IN AGREEMENT)

Yes	No	
a. ✓	___	Require physical change to station?
b. ✓	___	Clear TM, PIR, Commitment item? TM, PIR, Commitment item # (RESOLUTION FOR PIR-4-O89-0111)
c. ✓	___	Commitment to regulatory agency (NRC, INPO, EPA, etc.). Originator attach written documentation of commitment. FINAL OUTSTANDING SSFI ACTION ITEM (# 2.1.7)

(7) Proposed Resolution PROVIDE LOW-LEVEL SWITCHES ON THE UPPER SURGE TANKS TO CLOSE VALVES 1,2,3C-176 AND 187 AUTOMATICALLY. SET THE SWITCHES FOR THE MINIMUM TECH SPEC LEVEL OF 6 FT (3.4.4) PLUS ADDITIONAL MARGIN TO COMPENSATE FOR WATER LOST DURING VALVE CLOSURE. HAVE THIS SIGNAL OVERRIDE THE LOW HOTWELL LEVEL SIGNAL. KEEP EITHER 1,2,3C-175 OR 186 CLOSED AT ALL TIMES. ADD VALVES 1,2,3C-176 AND 187 TO THE ACTIVE VALVE LIST WITH SPECIFIED STROKE-TIME REQUIREMENTS.

(8) Other Alternatives LEAVE BOTH 1,2,3C-175 AND 186 NORMALLY CLOSED AND OPEN THEM ONLY DURING CERTAIN OPERATING MODES. SEISMIC QUALIFICATION OF HOTWELL AND ASSOCIATED PIPING MIGHT BE POSSIBLE.

(9) Impact to Station if Problem is Not Resolved WOULD LEAVE OPEN THE QUESTION OF UST WATER INVENTORY BEING LOST FOLLOWING RUPTURE/FAILURE OF HOTWELL AND NON-SEISMIC PIPE.

(10) Benefits of Problem Resolution: Consider plant performance/reliability, outage duration, personnel safety, cost benefit.
CLOSE LAST OPEN SSFI ITEM, HELP ASSURE WATER SUPPLY FOR EFWS.

(11) If a Physical Change is Involved:
a. Approximate cost of equipment/material <$2000
b. Change involves (~~replacement of~~ / addition to) existing equipment.
circle one
c. Will implementation of change require outage? Yes / No If this affects existing instrumentation only, and not piping.
circle one

(12) Initiator [signature] ____ EC II 5th floor ____ Oconee Mech Section (DMS) Engineering ____ 3-14-90
Name ____ Work Location ____ Group (Abbrev.) ____ Date

(13) Approved: WRH / [signature] [signature] ____ Dated 6/1/90
K. [signature] 6-27-90

STATION PROBLEM REPORT

(Side 2)

(14) Station Oconee (15) Unit 1,2,3 (16) System(s) [illegible] (17) SPR # [illegible]

(18) Accountable Engineer Jim Graves
Name

(19) Resolution This SPR will be resolved per the NSM program and will be prioritized and placed on the NSM list.

[illegible signature]
[illegible date]

This SPR will be worked by NSMs 1,2,3 2911 [illegible signature]
1-11-92

(20) Approved: ____________________ Date ____________
Name

Oconee PRA Rev 2

A.6 EMERGENCY FEEDWATER SYSTEM

System Description

The Emergency Feedwater (EFW) System is designed to supply feedwater to the steam generators in the event Main Feedwater is lost. The EFW system provides the required flow rate to cool the Reactor Coolant System (RCS) down to the point at which the Decay Heat Removal system is designed to operate. The EFW system is also designed to cool the RCS following a small break Loss-of-Coolant-Accident.

Two alternate systems are also available to provide feedwater to the steam generators. The Auxiliary Service Water (ASW) System and the SSF Auxiliary Service Water System are raw water systems which can be manually aligned for steam generator cooling. The SSF ASW system is described in the SSF section, and the ASW System, sometimes known as the "Tornado Pump," is described later in this section.

Three EFW pumps are provided for each unit. Two motor-driven EFW pumps normally supply feedwater to one steam generator (SG) each. One turbine-driven pump, capable of supplying feedwater to both SGs simultaneously, is driven from steam contained in either SG. Each of the two motor-driven pumps is rated at a flow of 500 gpm. The turbine-driven pump has a rated capacity of 1080 gpm when feeding both SGs and 880 gpm when feeding only one SG. The reduction in flow while feeding one SG is due to line restrictions. Any single pump and SG combination provides adequate decay heat removal for safe shutdown. All three pumps have minimum-flow recirculation lines to the Upper Surge Tank (UST) for pump protection.

Redundant water supplies are available to the EFW System. The primary source of water for the EFW System is the UST: two 36,000 gallon tanks valved together to make one 72,000 gallon tank. The condenser hotwell can also serve as the source of suction to the EFW pumps and has a 142,000 gallon capacity when filled to a level of approximately 70 inches. The normal operating level of water in the hotwell is approximately 63 inches. In order to take suction from the hotwell, the condenser vacuum must be broken because a net positive suction head (NPSH) must be established. The turbine-driven pump is capable of taking suction from the hotwell under vacuum, however, total flow is limited by operating procedures to 500 gpm to protect the pump. This flow is considered inadequate for steam generator cooling when recirculation flow back to the UST is accounted for.

In addition to serving as the source of suction to the EFW pumps, the UST also provides makeup water to the hotwell by gravity feed. The UST can be made up from the following sources:

1. The Demineralized or Treated Water System
2. The condensate storage tank (CST)
3. The condenser hotwell

Each motor-driven EFW pump discharges through a single line to the SG it is dedicated to feed. The turbine-driven pump discharge line splits into two lines, one joining the discharge line of motor-driven pump '3A' to form a common header to SG '3A,' and the other performing a similar function for SG '3B.' Each pump discharge line contains a check valve and a normally open motor-operated valve to ensure an open flow path. The flow of EFW to the SGs is controlled by means of control valves 3FDW-315 and 3FDW-316.

The ASW System is shared among the three units at the station. It is designed to remove decay heat from all three units simultaneously upon loss of main feedwater (MFW) and EFW as a result of tornado wind or missile damage. The pump can supply adequate flow to one or both SGs of each unit assuming the atmospheric dump valves are open to depressurize the SGs.

The ASW pump is manually started at the ASW switchgear panel and the valves that align the pump discharge to the SGs are manually opened. Lake water is supplied to the suction of the ASW pump via the Unit 2 CCW pumps' discharge piping.

System Success Criteria

Success of the EFW System is accomplished by supplying flow to one of the two SGs from one of three EFW pumps.

Major Assumptions

1. It is assumed that the reactor coolant pumps are unavailable when EFW is demanded. Therefore, SG levels must achieve their natural circulation setpoint.

2. Manually cross-connecting the EFW Systems of other units is treated as a recovery and is applied after sequence cut sets are generated.

3. The hotwell pumps are the only source modeled for makeup to the UST.

4. The steam generator level control cabinets and instrumentation are not generally modeled. Operator action is normally relied upon to throttle EFW flow in the model.

5. A loadshed can fail power to the hotwell pumps and MOV 3C-391 if power is not reloaded.

6. Unavailability of the UST at the start will fail the pumps because there will be no suction source. Failure of makeup to UST will not fail the pumps unless hotwell switchover fails.

7. No credit is taken for the ASW System in the EFW model. Manual alignment of the ASW system is modeled only in the tornado model.

System Reliability Results

Turbine-driven pump run failure combined with a latent human error on the motor driven pump suction is the dominant failure mode for the EFW system. This failure could occur at any point during a 24 hour mission time. Common cause failure of the EFW flow control valves and unavailability of the UST also contribute significantly to overall system unavailability.

It should be noted that the EFW System was solved without assuming any particular initiator. If a main feed line break is assumed, the UST could be drained into the hotwell, thereby failing EFW's initial suction source. This is a potential failure mode which only appears during a main feed line (or condensate line) break. A steam line break in the TD pump supply line could render all secondary side cooling unavailable without operator action to isolate the break. This failure mode only appears with a main steam line break (T9) initiating event. Therefore, these failure modes do not show up in the system cut sets. The dominant minimal cut sets for the EFW System are shown in the table below.

Top Cut Sets For Gate F1: EFW System Fails

Cut Set Prob.	Event Name	Event Description	Event Prob.
1.15E-03	FCXMUSTLHE	UST Flow Line To MDEFWPs Fails Due To Latent Human Error	3.00E-03
	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01
1.94E-04	FEF1516COM	Common Cause Failure Of AOVs FDW-315 and 316 To Open	1.94E-04
1.00E-04	FEFWUSTLHE	Insufficient Inventory In UST For EFW Pump Suction	1.00E-04
9.72E-05	FCXC187AVT	Air-Operated Valve C-187 Transfers Open	9.72E-05
7.30E-05	FCX0572CVO	Check Valve 3C-572 Fails To Open On Demand	1.90E-04
	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01
4.26E-05	FEFMDPSCOM	Common Cause Failure Of Motor-Driven EFW Pumps To Start	1.11E-04
	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01
3.39E-05	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01
	FLS0527VVT	Manual Valve 3LPSW-527 Transfers Closed	8.83E-05
3.39E-05	FCX0180VVT	Locked Open Manual Valve 3C-180 Transfers Closed	8.83E-05
	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01
3.39E-05	FCX0166VVT	Manual Valve 3C-166 Transfers Closed	8.83E-05
	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01
2.52E-05	FEFMDPRCOM	Common Cause Failure Of Motor-Driven EFW Pumps To Run	6.56E-05
	FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	3.84E-01

Gate Probability: 1.97E-03

The three-train nature of the EFW System provides diversity and redundancy against system failure. Operator action is important for long-term operation of the system. Failure of the turbine-driven pump to run for the 24 hour mission time and failures of common suction lines are the weak point of the EFW System.

The most important system components are identified below and are ranked by contribution to system failure.

Importance Table For Gate F1: EFW System Fails

Event Name	Event Description	F - V	RAW
FEFTDFPTPR	Turbine-Driven EFW Pump Fails To Run For The Required Time	71.8%	2.15
FCXMUSTLHE	UST Flow Line To MDEFWPs Fails Due To Latent Human Error	62.4%	203
FEF1516COM	Common Cause Failure Of AOVs FDW-315 and 316 To Open	9.9%	509
FEFWUSTLHE	Insufficient Inventory In UST For EFW Pump Suction	5.1%	509
FCXC187AVT	Air-Operated Valve C-187 Transfers Open	4.9%	509
FCX0572CVO	Check Valve 3C-572 Fails To Open On Demand	3.7%	196
FEFMDPSCOM	Common Cause Failure Of Motor-Driven EFW Pumps To Start	2.2%	196
FCX0180VVT	Locked Open Manual Valve 3C-180 Transfers Closed	1.7%	196
FLS0527VVT	Manual Valve 3LPSW-527 Transfers Closed	1.7%	196
FCX0166VVT	Manual Valve 3C-166 Transfers Closed	1.7%	196
FEFTRNATRM	EFW System Train 3A Is In Maintenance	1.3%	3.62
FEFTRNBTRM	EFW System Train 3B Is In Maintenance	1.3%	3.62

10 CFR 50.59 USQ EVALUATION

(NSD 209.11.1)

(1) STATION (s):	(2) UNIT(s):	(3) TYPE OF ACTIVITY:	
[x] Oconee Nuclear Station	[x] Unit 1	[ ] Nuclear Station Modification	[ ] Operability Evaluation
[ ] McGuire Nuclear Station	[x] Unit 2	[ ] Minor Modification	[ ] Test or Experiment
[ ] Catawba Nuclear Station	[x] Unit 3	[ ] Procedure	[x] UFSAR Change
[ ] ______	[ ] ______	[ ] Other ______	[ ] Temporary Modification

(4) DOCUMENT NUMBER, REV. NUMBER, and DESCRIPTION: Revise UFSAR Section 10.4.7 to clarify EFW single failure design statements

(5) SCREENING FOR INCREASED MANAGEMENT INVOLVEMENT (NSD 209.11.2 & 213)

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 [x] No
2. Does this item involve infrequently performed tests or evolutions 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 the 10 CFR 50.59 regulation nor implemented without prior NRC approval. [ ] Yes [x] No
2. TECHNICAL SPECIFICATIONS AND ASSOCIATED BASES CONSULTED: 3.4, 4.9
3. UFSAR SECTIONS CONSULTED: 10.4.7, 7.4.3, 8.3.2.2.4, 15.0, 15.8, 15.14
4. OTHER SAR DOCUMENTS CONSULTED: Numerous other SAR documents consulted. See reference section of Attachment 1.
5. SAR DOCUMENT SECTIONS WHICH NEED REVISION: UFSAR 10.4.7

(7) SAFETY REVIEW (NSD 209.11.4)

Safety Review performed and documented as required per section 209.11.4 and 209.12? [x] 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 [x] No
2. Increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR? [ ] Yes [x] No
3. Increase the consequences of an accident previously evaluated in the SAR? [ ] Yes [x] No
4. Increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR? [ ] Yes [x] No
5. Create the possibility for an accident of a different type than any evaluated previously in the SAR? [ ] Yes [x] No
6. Create the possibility for a different type of malfunction of equipment important to safety than any evaluated previously in the SAR? [ ] Yes [x] No

Does the proposed activity:

7. Reduce the margin of safety as defined in the basis for any Technical Specification? [ ] Yes [x] No

* If the answer to any of the above seven (7) unreviewed safety questions in Part (8) is "Yes," the change cannot be performed under the 10 CFR 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. [x] 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? [x] Yes

(10) APPROVAL (NSD 209.11.7)

Preparer: SG Benesh Date: 11-18-98

Qualified Reviewer: KW Sandel Date: 11/18/98

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 11/18/98

Document UFSAR Change of 11/18/98

Safety Evaluation for FSAR 10.4.7 Change

Description of change

UFSAR 10.4.7 is being revised to clarify the EFW design basis. This evaluation process includes collecting information which is included in NRC/Duke correspondence and updating the UFSAR to include this information to clarify the licensing basis of EFW relative to single failure. The changes being evaluated are included in sections 10.4.7.1, 10.4.7.2, 10.4.7.3, and 10.4.8. Specifically, the following changes are proposed (Attachment 2):

Section 10.4.7.1 is being changed to clarify two statements. These first is:

> "Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

This statement is being revised to clarify that its original intent, which was to summarize the high energy line break analysis that was submitted to the NRC in 1973. In this analysis, feeding from another unit was credited. This statement will be changed to reflect this.

The second statement is:

> "In the event of a postulated break in the Main Steam or Main Feedwater system inside or outside containment coupled with a single active failure, the EFW system provides sufficient flow to ensure adequate core cooling."

This statement is also being revised to clarify that its' original intent. This was to summarize the design basis of the EFW system in the event of a pipe break that depressurizes a steam generator coupled with a single active failure of an EFW pump or control valve. This statement will be changed to reflect this.

Section 10.4.7.2 currently states that "Once automatically started, the EFW pumps will continue to operate until manually secured by the operator". This sentence is being revised to add "motor driven" to clarify that only the MDEFW pumps will always continue to run when automatically started until secured by the operator. The TDEFW pump is automatically secured by the Main Steam Line Break MSLB circuitry in the event of a MSLB. This needed UFSAR change was not made as a part of NSMs-x2873, which installed the MSLB circuitry. The modification installed circuitry that secured the TDEFW pump in the event of a MSLB. Therefore, after this modification, the TDEFW pump does not continue to run until manually secured by the operator in the event of a MSLB. This change was evaluated as a result of the modification NSM-x2873, therefore the evaluation below will focus on the changes related to single failure. (Ref. 34)

Section 10.4.7.3 is being revised to add clarification of the single failure design of the EFW system. A discussion of the High Energy Line Break (HELB) impact on the EFW system was included to document the Oconee reliance on the diverse sources of feedwater. Additionally, a clarification of the single failure analysis was provided. This docketed NRC/Duke correspondence reflected a review of the EFW pumps and EFW flow control valves for failures and the resultant impact on the EFW system.

Section 10.4.8 is being revised to add reference to Postulated Pipe Break Analysis, Duke letter about EFW conceptual design, and delete specific page reference to the April 3, 1981 letter.

SAFETY REVIEW

INTRODUCTION

The original Oconee feedwater system was designed with diversity so that in the event of a single failure, feedwater could be delivered to the steam generators. The diversity included the main feedwater system and EFW system for the affected unit, the EFW system from the other units, and the station ASW system. After the TMI accident, several enhancements were made to the emergency feedwater system to assure higher reliability. The major physical enhancement was the addition of the motor driven emergency feedwater pumps. As the result of the TMI accident, a reliability study was performed and a review of the EFW system to the standard review plan (SRP) was performed. These reviews and enhancements required major changes to the FSAR. Of these changes, two statements in particular were added that have created some confusion as to the true design basis requirements for the EFW system. The two specific UFSAR Chapter 10.4.7, "Emergency Feedwater System," statements which are in question are below.

> "Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

> "In the event of a postulated break in the Main Steam or Main Feedwater system inside or outside containment coupled with a single active failure, the EFW system provides sufficient flow to ensure adequate core cooling."

The following discussion reviews the development of the FSAR 10.4.7 for the EFW system and the licensing correspondence that supported the different changes.

EARLY OCONEE LICENSE BASIS INFORMATION

The early versions of the Final Safety Analysis Report (FSAR) Chapter 10, "Steam and Power Conversion System," stated that the feedwater supply to the steam generators following a reactor shutdown is assured by one of the following methods:

(a) either of the two feedwater pumps,
(b) the hotwell and condensate booster pump combination,
(c) the turbine driven emergency feedwater pump, or
(d) the turbine driven emergency feedwater pump from each of the other units.

The sources of feedwater were described as from the upper surge tank and hotwell. The original EFW system design consisted of a single turbine driven EFW pump for each unit and was not required to withstand a single failure. However, sufficient redundancy and diversity was designed into the feedwater system to ensure that the feedwater supply to the steam generators was maintained following a reactor shutdown (Ref. 1).

The early versions of the Oconee Technical Specifications for the Steam and Power Conversion system applied to the turbine cycle components for removal of reactor decay heat. In the Steam and Power Conversion section, the following operability requirements were included:

"The reactor shall not be heated above 250°F unless the following conditions are met:

3.4.1 Capability to remove a decay heat load of 5 percent full reactor power from at least one of the following means:

a. A hotwell pump, condensate booster pump, and a main feedwater pump.
b. The emergency feedwater pump.
c. A hotwell pump and condensate booster pump.

3.4.3 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell."

(Ref. 2)

Since only one of the decay heat removal methods in the early versions of the Oconee Technical Specifications was required, the early Oconee Technical Specifications allowed for reactor operation above 250°F without the turbine driven emergency feedwater pump being operable. The Technical Specifications outlined the requirements for the source of feedwater which consisted of the upper surge tank, condensate storage tank, and hotwell. The various methods of supplying feedwater could all provide water from the upper surge tank, condensate storage tank, and hotwell, as necessary.

Original Safety Evaluation Report

The original Unit 1 Safety Evaluation Report (SER), which is dated December 29, 1970, from the Atomic Energy Commission (AEC) reviewed Oconee's steam and power conversion system. In the SER, the AEC stated the following:

> "There are two principal intermediate heat removal routes: (1) by way of the steam and power conversion system (steam generators and main condensers), and (2) by way of the low pressure injection and low pressure service water systems. The heat removal capacity of the steam and power conversion system route is adequate to permit the loss of the low pressure injection route. Redundancy within the steam conversion system is such that the heat removal adequacy of this system is not impaired by single failures of components, equipment, or piping." (Ref. 3, Section 10.4)

The Oconee steam and power conversion system could adequately remove the decay heat without any reliance on the low pressure injection system. In addition, the redundancy of the steam and power conversion system (main feedwater pumps, hotwell pumps, condensate booster pumps, and turbine driven emergency feedwater pumps) ensured adequate decay heat removal capability following a single failure. Thus, the steam and power conversion system was designed with redundancy, however, each individual part of the steam and power conversion system (i.e., turbine driven emergency feedwater system) was not designed to single failure criteria.

HELB ANALYSIS

In a letter dated December 15, 1972, the AEC requested information on the effects of a piping system break outside of containment. As part of this request there was discussion about a requirement to assume a single active failure of a component needed to function to mitigate the event. (Ref. 5)

In a report dated April 25, 1973, supplemented on June 22, 1973, Duke submitted the analysis of effects resulting from postulated piping breaks outside containment for Oconee. The report identified that the main feedwater system and emergency feedwater system could be lost as the result of a feedwater line break, auxiliary steam line break, or condensate line break. In addition, the feedwater line break or auxiliary steam line break could result in the loss of the 4160 volt engineered safeguards switchgear (1TC, 1TD, and 1TE). Duke stated that the plant could mitigate the loss of the 4160 volt engineered safeguards switchgear following a postulated feedwater line break in the turbine building since it would result in a condition that was similar to the accident analysis in FSAR Chapter 14.1.2.8.3, "Results of a Complete Loss of All Station Power Analysis." The FSAR analysis stated that the loss of all station power did not require immediate operation of the emergency feedwater system. In addition, a method of decay heat removal was available from the Auxiliary Service Water system through manual operator action. Duke also stated that "Since the failure of the ASW system pump could leave the plant without adequate long-term cooling, design changes for a redundant Emergency Feedwater system [were planned]." After the planned modifications were completed, emergency feedwater would be available within 15 minutes by manual operation. These design changes included installation of an emergency feedwater bypass line and unit cross-connects which provided a means to deliver EFW to the affected unit from another unit's EFW system. (Refs. 1, 6, 7)

In the AEC's SER for Oconee Units 2 and 3 which was dated July 6, 1973, the AEC evaluated the high energy line rupture in piping systems external to the reactor building in Section 7.1.11. The AEC stated that "the staff has evaluated the analytical methods and assumptions used in the applicant's analyses and find them acceptable and concurs with the proposed plant modifications and the criteria to be used in their designs." (Ref. 4, Section 7.1.11 and Appendix E).

It should be noted that the basis for the AEC's acceptance of the Oconee design to mitigate the HELB, relied on the diversity of the EFW sources available at Oconee to deal with single failure criteria. This was consistent with the early licensing basis of Oconee Nuclear Station in that the acceptability of the EFW system was based on the diversity of feedwater sources across the site.

POST-TMI COMMITMENTS SPECIFIC TO SINGLE FAILURE

Shortly after the TMI event, IE bulletin 79-05A was issued specifying short term actions to enhance the reliability of the EFW system. The actions relevant to this discussion are:

1. Verify valves in the EFW flow path are in the open position
2. For manually operated valves which could defeat or compromise the flow of EFW to the steam generators, prepare and implement procedures that require valves to be locked open and maintain positive position controls.
3. Prepare procedures which assure that two independent steam generator EFW paths, each with 100% flow capacity, are operable.

(Ref. 8)

As a result of the TMI accident and IE bulletin, Duke took several actions to meet the short term actions specified in the IE Bulletin. Those include:

- All manual valves which could defeat or compromise EFW flow were locked open and controls of manually operated MOV's were tagged to denote the required position.
- Start TDEFW pumps on all three units on a demand from any unit and cross-connect the discharge piping between all three units.
- Make the TDEFW pump independent of ICS.
- Pursue the installation of MDEFW pumps.

(Ref. 9, 27, 28)

On May 7, 1979, the NRC issued an Order to Duke Power specifying the implementation of actions associated with EFW. The actions were relating to the assurance that flow paths were not blocked by closed valves and pumps were available. There were no specific requirements to review the impact of failures on systems. The design criteria listed for the MDEFW pumps were:

1. Two electrically operated pumps will provide a minimum of 100% design capacity for each unit.
2. The design of the pumps and the associated equipment will be seismic.
3. The controls for the electric pumps will be independent of the ICS.

4. The power supply for the pumps and controls will be from the 4160 volt safety related busses.
5. Piping and valves added will be Duke Class F (B31.1-SSE seismic design) which is like quality to the existing TDEFW feedwater pump system.
6. The existing feedwater control valves will be utilized thereby providing four functional flow paths to the steam generators on each unit.

(Ref. 9)

Installation of the MDEFW pumps

Following the issuance of the NRC Order, Duke submitted a system concept to the NRC which described the "new" EFW system with the MDEFW pumps installed. The following summary of the submittal is provided:

Design Bases

"The Emergency Feedwater (EFW) System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the Condensate/Feedwater System, to remove energy stored in the core and primary coolant. The EFW System is designed to provide sufficient secondary side steam generator heat sink to enable cooldown from reactor trip at power operation down to cold shutdown conditions. The EFW System may also be required in some other circumstances such as cooldown following a loss-of-coolant accident for a small break.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps or low feedwater header pressure. The EFW System will supply sufficient feedwater for approximately five hour cooldown at a flowrate of at least 720 GPM to enable the Reactor Coolant System to reach conditions at which the Decay Heat Removal System may be operated.

Three EFW pumps are provided, powered from diverse power sources. Two 65% capacity motor driven pumps are powered by the Emergency AC Power System, each supplying feedwater to one steam generator. One 150% capacity turbine driven pump, supplying feedwater to both steam generators, is driven from steam contained in either steam generator. Although the total rated capacity of all three EFW pumps is 2080 GPM, only 720 GPM is required as a minimum to enable safe and orderly cooldown of the Reactor Coolant System. Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS)."

(Ref. 10)

System Description (from this submittal)

- the control valves are designed to fail open on loss of air or AC power which would eliminate the possibility of valve failure from isolating the necessary feedwater.
- the EFW system control valves receive a control air signal for valve modulation in response to steam generator level which is independent of the ICS.
- the motor driven EFW pumps are powered from the emergency AC power system during the loss of offsite power.
- the turbine driven emergency feedwater pumps can be supplied by steam from either steam generator with either steam supply capable of being isolated if necessary.

- the feedwater sources which consist of the upper surge tanks, hotwell, condensate storage tanks, and makeup demineralizers are normally aligned to the EFW pump suctions.

(Ref. 10)

Svstem Function (from this submittal)

- each EFW pump is supplied with its own independent starting circuit which is powered from the 125 VDC station batteries
- the operator can manually start each EFW pump and sufficient indication is provided in the control room to allow the operator to monitor the plant parameters during a cooldown
- the control of the EFW control valves is by the steam generator level control signal and the level control setpoint is automatically raised following a loss of offsite power to promote natural circulation in the reactor coolant system
- the EFW system is provided with sufficient valving to allow isolation and cross connection as required to select and isolate water sources and assure system function in the event of various failures.
- During a shutdown following a blackout or loss of feedwater, no valve realignments or isolation are necessary since all the necessary valves are maintained in normal standby alignment to assure an open flow path for each EFW pump, and to assure piping separation and independence.

(Ref. 10)

Safetv Evaluation (from this submittal)

"Feedwater inventory is maintained in the steam generators following reactor shutdown by one of the following methods listed:

a) Either of the two main feedwater pumps is capable of supplying both steam generators at full secondary system pressure.
b) The two EFW motor driven pumps are capable of supplying both the steam generators at full secondary pressure.
c) The single EFW turbine driven pump is capable of supplying both steam generators at full secondary pressure.
d) Alternate EFW supplies may be available from the EFW systems of the other units, capable of supplying both steam generators at full secondary system pressure.
e) The hotwell and condensate booster pump combination has a discharge shutoff head of approximately 700 psia. Three pairs of pumps are provided. If required, the turbine bypass system of the ADV's can be used to reduce the secondary system pressure to the point where one hotwell and condensate booster pump combination can supply feedwater to both steam generators.
f) The Auxiliary Service Water system may be used to maintain steam generator water inventory following steam generator depressurization to remove decay heat in the long term.

A sufficient depth of backup measures is provided to allow steam generator water inventory to be maintained by any of the diverse methods listed above. Although redundancy and diversity is provided in the listed measures, the EFW system has been designed with special considerations to enable it to function when conventional means of feedwater may be unavailable.

Redundancy is provided with separate, full capacity, motor and turbine driven pump subsystems. Failure of either the motor driven pumps or the turbine driven pump will not reduce the EFW system below the minimum required capacity. Pump controls, instrumentation, and motive power are separate in design. Separate piping subsystems include redundant hotwell and upper surge tank condensate supply piping, aligned individually to the separate pump trains. Cross-connection is provided, however, to allow a subsystem to supply all pumps in the event of single failure of a suction piping subsystem. The same design philosophy is included in the discharge piping subsystems."

(Ref. 10)

There are several key items that should be noted from this May 17, 1979 Duke letter. The design basis for the EFW system was stated as being required following a loss of main feedwater (with and without AC power available) and a small break LOCA. This is a restatement of the requirements for EFW from the early Oconee licensing basis, modified to include the SBLOCA. As was the case for the original license, there is no specific mention of a main steam line break as EFW has no immediate role for some time after a MSLB since the break results in an overcooling event. The design bases did state that "Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power". This is a summary of the position stated in the HELB submittal of 1973, in which the EFW function was viewed as that provided by all the available site sources. This is supported by the Safety Evaluation section of this submittal which states that feedwater inventory is maintained in the steam generators by any of the six listed methods. Therefore, the basis for the statement relating to the secondary pipe break relies on the diversity of feedwater across the Oconee site for EFW system design acceptability.

Summary of EFW Related Post-TMI requirements

NUREG-0667, "Transient Response of B&W Designed Reactors", included many recommendations that resulted from the staff review of the TMI event. One of which was related to the design of the EFW system. The recommendation was paraphrased as follows:

> The EFW system on operating B&W plants should be classified as an Engineered Safety Feature system, and as such be upgraded as necessary to meet safety-grade requirements. As an alternative, assuming comparable reliability, consideration would be given to the addition of a dedicated EFW system (i.e., a separate train).
>
> (Ref. 13)

Duke responded to this recommendation stating that "The Oconee emergency feedwater system coupled with the dedicated Standby shutdown Facility, currently under construction, meet this recommendation and no additional modifications to the system are necessary." This position again expresses Duke's intent to credit the site diversity for the acceptability of the EFW system design (Ref. 13).

NUREG-0737 provided a summary of the post-TMI action items and current status for each plant. Included in those items were three that are relevant to EFW design requirements. The action items are:

1. Perform a simplified EFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for EFW system failure under various loss of main feedwater transients
2. Perform a deterministic review of the EFW system using the acceptance criteria of Standard Review Plan (SRP) 10.4.9 and Branch Technical Position (BTP) ASB 10-1 as principal guidance, and
3. Re-evaluate the EFW flowrate design bases and criteria.

(Ref. 41)

An NRC letter addressing TMI action items noted that Oconee had already performed the reliability evaluation and it was under staff review. The NRC said that when they finished their review of the evaluation, they would issue a letter including requests for information regarding items 2 and 3 and short and long term EFW upgrade requirements based on item 1. The reliability study included an assessment of the EFW system for three events; loss of main feedwater with a reactor trip, loss of main feedwater with loss of offsite power, and loss of main feedwater with loss of all AC power. The events analyzed in the reliability study were consistent with the events the EFW system was originally licensed to mitigate. There were no pipe breaks assumed during this evaluation. (Ref. 12, 30, 31, 41)

Since item 2 (above) required a review of the EFW system against the SRP and BTP criteria, an examination of those criteria provide an insight to the standards that the NRC was applying during their review of the ONS EFW system.

Standard Review Plan 10.4.9 (Ref. 31)

The requirements in the SRP 10.4.9 are based on the assumption that the EFW system is the only means to provide feedwater to the steam generators following a loss of main feedwater. This is evident in the Branch Technical Position ASB 10-1, which is included in the SRP 10.4.9, which states:

> "The auxiliary feedwater system functions as an engineering safety system because it is the only source of makeup water to the steam generators for decay heat removal when the main feedwater system becomes inoperable. It must, therefore, be designed to operate when needed, using principles of redundancy and diversity in order to assure that it can function under postulated accident conditions."

This statement does not apply to each unit's EFW system at Oconee, since there are many other sources of feedwater available in the event that main feedwater is lost. The NRC's recommendation was to classify the EFW system as ESF or add a dedicated EFW system such as a separate train. Duke in their letter implied that the EFW system would not be upgraded to meet ESF requirements because the SSF (along with the other feedwater sources available) would provide the diversity required. (Ref. 13)

Although not generally applicable to Oconee, the NRC staff appeared to review the EFW system to those requirements. This is evident from the questions that were asked in the NRC letter dated, November 14, 1980 (Ref. 14). That letter is discussed later. The specific requirements in the SRP relevant to this issue are:

1. The system satisfies the recommendations of the BTP ASB 3-1 with respect to the effects of pipe whip and jet impingement that may result from high or moderate energy pipe breaks or cracks.
2. The system is capable of withstanding a single active failure.
3. The system design possesses the capacity to automatically terminate auxiliary feedwater flow to a depressurized steam generator, and to automatically provide flow to the intact steam generator.

(Ref. 31)

The SRP requirement 1 above, which references BTP ASB 3-1 (Ref. 33), is directly related to the HELB analysis performed for Oconee in 1973. In fact, the BTP ASB 3-1 states that:

> "Designs of plants for which operating licenses are issued before July 1, 1975 are considered acceptable with regard to effects of piping failures outside containment on the basis of the analyses made and measures taken by applicants and licensees in response to the December 1972 letter from A. Giambusso, and the staff review and acceptance of these analyses and measures."

The December 1972 letter referenced is the letter that requested the HELB study be performed for Oconee (referenced earlier). This shows that the issue relating to item 1 above was not relevant to the post-TMI EFW upgrades or reviews. Any Duke statements made, relating to secondary pipe breaks outside containment, were based on the information and analysis from the HELB study. The only issue relating to piping failures in the post-TMI time frame were those, discussed in item 3 above, where the piping failures resulted in a depressurized steam generator (i.e. a break inside containment).

<u>NRC request for Information related to NUREG-0660</u> (Ref. 14)

In a letter to Duke, dated November 14, 1980, the NRC requested additional information concerning the reliability of the emergency feedwater system at Oconee. Additionally, questions relating to the NRC's review of the SRP requirements and EFW flow rate design bases were included. The following information, pertinent to this discussion was requested:

> Item 2 - "Emergency Procedures For Initiating Back-Up Water Supplies. Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
>
> (1) The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW pumps against self-damage before water flow is initiated."
>
> Item 14 - "In the event of a postulated break in the main steam or main feed system inside or outside containment coupled with a single active failure, discuss how the Oconee EFW design limits or terminates EFW system flow to the depressurzied steam generator and directs the minimum flow to the intact steam generator. If manual action is relied upon, verify that sufficient flow to the intact steam generator will occur in sufficient time to provide adequate core cooling."
>
> Enclosure 3 - EFW flow design basis information as applicable to various design basis transients and accident conditions. Question 3 of Enclosure 3 requires Duke to verify that the EFW pumps will supply the necessary flow to the steam generators during the various design basis transients and accident conditions considering a single failure. A listing of transients was included. As part of Question 3, the NRC requested the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Duke responded to the NRC's request for additional information in a letter dated April 3, 1981 (Ref. 16), as follows:

> Item 2 - "The primary source of water (the upper surge tanks) are normally available and assured by the locking open of all manual valves in the pump suction paths and by double verification of valve alignment which is performed following the monthly testing. Pump startup is automatic upon demand signal".
>
> Item 14 (postulated main steam and main feedwater line breaks) - "In order to provide sufficient EFW flow to the intact steam generator to ensure adequate core cooling, and under a main steam or main feedwater break in OTSG A with a single active failure of motor driven emergency feedwater pump B train, the operator must manually close the EMO isolation valve or the flow control valve FDW-315 on OTSG A. He is able to do this from the Control Room. The same is true for OTSG B and motor driven emergency feedwater pump A. The operator has sufficient Control Room indication of steam generator level and pressure and would immediately be aware of such a situation.

Concurrently, the operator would monitor the intact steam generator to assure adequate inventory and secondary heat removal via either main feedwater or emergency feedwater systems.

In the event of a postulated break in the main steam or main feed system, coupled with a single active failure of either one of the three emergency feedwater pumps, sufficient flow will occur to provide adequate core cooling.

With a postulated break associated with the 'A' OTSG and a failure of the 'B' motor driven emergency feedwater pump, the turbine driven emergency feedwater pump is available as is the normal feedwater system.

Similarly, if the active failure occurs with the flow control valve (FDW-316), emergency feedwater flow can be aligned through the main feedwater startup control valves to either the main or auxiliary nozzles. Additionally, in the unlikely event that FDW-315, 316 failed to open automatically, an operator could manually open either one of the valves as they are located in the Penetration Rooms which are adjacent to the Control Room."

From the above response to Item 14, it can be seen that the single active failure review only involved the impact on the EFW pumps and flow control valves. The focus of this NRC request and Duke review was on the ability to establish a path in the event of a failure in the discharge flow path. An integrated system review of the entire plant secondary side response was not performed. Additionally, Duke noted that manual operation would be required and that action outside of the control room may be necessary to re-establish flow. This being an overcooling event, adequate time exists for actions outside of the control room to be taken to direct feedwater to the intact steam generator.

In response to the information requested by the NRC in Enclosure 3 of their letter dated November 14, 1980, Duke provided an analysis of the different transients as Item 17 in Duke's April 3, 1981 letter (Ref. 16). The following response was provided by Duke:

"The Emergency Feedwater System (EFW) serves as a backup to the Feedwater/Condensate System for supplying feedwater to the steam generators when normal feedwater delivery is interrupted or unavailable, thereby maintaining the heat sink capabilities of the steam generators. The EFW system, as designed, is capable of delivering sufficient feedwater to remove decay heat and reactor coolant pump heat including the assumption of the worst single failure in the system.

The EFW system consists of one turbine driven pump capable of delivering to both steam generators (1080 gpm at 1065 psia total flow while feeding both SG's or 880 gpm at 1065 psia while feeding only one SG) and two motor driven pumps (450 gpm each at 1065 psia) each aligned to one steam generator. The EFW pumps will automatically start, following either a loss of both main feedwater pumps or a low feedwater header pressure signal, in addition, to manual actuation. Following pump start, the control valves will modulate to control steam generator level at the two foot minimum level, except in the event that all four reactor coolant pumps have tripped, in which case the level setpoint increased to 50% on the operating range to provide for natural circulation.

The EFW system is provided with sufficient feedwater sources to enable cooldown of the Reactor Coolant System to temperatures where a switch over to the Decay Heat Removal System (DHR) for long term decay heat removal is accomplished.

The plant transient which requires the highest Emergency Feedwater System flow, and as such constitutes the design basis transient, is the loss of main feedwater transient. This transient combines the highest heat load, decay heat plus reactor coolant pump heat, with the minimum heat sink due to the instantaneous loss of both main feedwater pumps. A discussion of the demand on the EFW system for each transient follows.

1) Loss of Main Feedwater - Those transients which result in losing feedwater delivery from the Feedwater/Condensate System are classified as a loss of main feedwater. This initiating event causes a turbine and reactor trip and automatically starts the EFW pumps. Since the reactor coolant pumps remain on, the control valves modulate to control steam generator level at two feet. The transient requires feedwater to be delivered at a rate sufficient to remove decay heat and reactor coolant pump heat. One motor driven emergency feedwater pump delivering 450 gpm at a steam generator pressure of 1065 psia will provide adequate heat removal capacity.

2) LMFW w/Loss of Offsite AC Power - This transient is the result of a station blackout condition. The loss of offsite AC power causes the reactor to trip, the turbine to trip, and the condensate booster pumps and hotwell pumps to trip and cause a loss of main feedwater. The emergency feedwater pumps are actuated on the main feedwater pump trip. Since the reactor coolant pumps have tripped, steam generator level control increases the level setpoint to 50% on the operating range to promote the natural circulation mode of heat removal. The emergency feedwater control valves open to allow full system flow until the controlling level is attained. Feedwater requirements are determined by core decay heat removal demand. One MDEFWP can deliver sufficient feedwater to meet the demand.

3) LMFW w/Loss of Onsite and Offsite AC Power - This transient is similar to Case 2 with the additional assumption that the onsite emergency AC power sources have been lost. This results in the loss of the motor driven emergency feedwater pumps. The transient requires the turbine driven emergency feedwater pump to deliver sufficient feedwater to move core decay heat. The TDEFWP has sufficient capacity to meet the heat removal demand.

4) Plant Cooldown - In addition to providing sufficient heat removal capacity immediately following a transient, the requirements for plant cooldown from full power operation to RCS temperatures where switchover to the Decay Heat Removal System can be accomplished has been determined. All heat sources have been included. The average hourly EFW flowrate to meet cooldown rates of 100°F/hr and 50°F/hr down to the switchover temperature of 250°F are given below.

	Cooldown Rate	
Time	100°F/hr.	50°F/hr.
0-1 hr	547 gpm	480 gpm
1-2 hr	464	390
2-3.3 hr	430	-
2-3 hr	-	354
3-4 hr	-	344
4-5 hr	-	331
5-6 hr	-	325
6-6.6 hr	-	320

Cooldown of the RCS is a manual function controlled by the operator such that the EFW flow is throttled to obtain the cooldown rate desired and within Technical Specification and administrative limits.

5) Turbine Trip - A turbine trip transient causes a reactor trip. The reactor trip initiates the ICS to control steam generator level at the minimum level so that the main feedwater pumps are

runback. With the main feedwater pumps in an untripped condition, there is no requirement for the EFW system to function.

6) Main Steam Isolation Valve Closure - This transient, similar to the turbine trip, does not trip the main feedwater pumps so that the EFW system is not required.

7) Main Feedwater Line Break - For a main feedwater line break upstream of the isolation check valve, the transient would have the same response as a loss of main feedwater. A break downstream of the check valve will cause the steam generator to blow down, but will be less severe than a steam line break transient due to less feedwater being delivered to the steam generators. The demand on the EFW system would be for decay heat and reactor coolant pump heat removal via the unaffected steam generator. One MDEFWP has sufficient capacity to perform this function.

8) Steam Line Break - A steam line break transient is primarily an overcooling transient. Only after the overcooling has been turned around and after isolation of the affected SG, the need for heat removal by the intact SG arises. Since the EFW system is capable of delivering to either steam generator, the heat removal demand on the EFW system can be met by one MDEFWP or the TDEFWP in the event the MFW system is unavailable.

9) Small Break LOCA - For small break loss of coolant accidents, feedwater is required to remove the decay heat and reactor coolant pump heat which is not relieved through the break. The analyses submitted in "Evaluation of Transient Behavior and Small Reactor Coolant System breaks in the 177-FA Plant", May 7, 1979, required a minimum flow rate of 300 gpm. One MDEFWP has the necessary capacity.

10) The above transients bound the EFW system performance requirements for all transients."

The focus of the NRC question, as was the focus of the Duke review, was associated with pumping capacity available during these events. As stated in the paragraph that preceded the discussion of each event, the demand on the EFW system for each transient was the focus. Additionally, the bottom line conclusion of each event discussion was that there was adequate pumping capacity. In addition, Duke's response to the postulation of a single failure coincident with the above transients to determine that the EFW pumps will supply the necessary flow to the steam generators is provided below.

> "The spectrum of transients which require EFW system performance for post trip heat removal have been evaluated assuming only one motor driven emergency feedwater pump is available to deliver the necessary feedwater. Any single failure in the three pump-two flowpath, EFW system design will not result in only one MDEFWP available, so that this assumption is overly conservative. A large margin of 10% reduction in pump flow was also included. These analyses verify the acceptability of the Oconee Emergency Feedwater System design."

Duke's response to the request to analyze the different transients above with the consideration of a single failure during the transients focused on the adequacy of the EFW system flow capacity to the steam generators following the transient. A review of the section which specifically addresses the single failure impact, shows again, that the single failures were limited to the EFW pumps and associated flow paths to the steam generators. There was no review of the plant response due to each of the events listed and the effect of failures resulting from plant interactions. They were simply requirements for sizing of EFW pumping capacity. It should also be noted that the questions relating to feedwater line break upstream of the isolation check valve were answered simply in terms of pumping capacity, because the requirements to perform an analysis of the HELB impact on overall plant equipment was previously addressed for Oconee as stated in the BTP ASB 3-1.

In an SER dated August 25, 1981 (Ref. 17), the NRC found the Oconee response to the various questions relating to the AFW reliability study and SRP requirements as follows:

> Item 2 - "... in the Oconee design there are parallel suction paths from the primary water source tanks to the motor driven and to the turbine driven emergency feedwater pumps.
>
> If a suction valve in one of these parallel paths were to block suction flow due to a mechanical failure, it is possible that either the motor driven or the turbine driven EFW pumps would be destroyed. However, this is a low probability occurrence. Additionally, one train of EFW cross-connects from the other units on the discharge side of the pumps, and by the Standby Shutdown Facility [are available]. Therefore, separate procedures for case 1 are not considered necessary. We find the Oconee design acceptable with respect with case 1 of this recommendation." [Case 1 was the case in which the primary water supply was not initially available].

Based on the NRC evaluation of a loss of primary water source tank, Oconee's design is found acceptable, in part because of the diversity of sources available to feed the steam generators. The NRC specifically credited the unit cross-connects and the SSF availability in the event of a loss of primary water source. This is evidence of the NRC's acceptance of the Duke position relative to Oconee's reliance on the diversity of systems capable of providing feedwater to the steam generators following a loss of main feedwater.

Item 14 (postulated main steam or main feedwater line break) - the NRC made the following statements:

> "By letter dated April 3, 1981, the licensee responded that in order to provide emergency feedwater flow to the intact steam generator and isolate the ruptured steam generator the operator must take manual action. The system is designed so that a single active failure of any of the emergency feedwater pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator. The operator has sufficient control room indication of steam generator level and pressure to take the actions necessary to provide sufficient flow to the intact steam generator in time to maintain adequate core cooling. We find the response to this request acceptable."

As can be inferred in the NRC's SER, the single failure, which was considered, consisted of a single active failure of any emergency feedwater pump or any valve in the EFW flow path to the steam generator. In addition, the NRC's SER states that the single active failure will not prevent the operator from directing sufficient flow to the intact steam generator which indicates credit for operator action to mitigate a main steam or main feedwater line break. Again, this review was only associated with the HELB that depressurized a steam generator, and therefore resulted in an overcooling event.

The completion of the NRC review of the EFW system flow requirements was documented in the SER that was dated April 8, 1982 (Ref. 19). The SER contains the following statement concerning the review of the EFW system flow requirements:

> "Duke's response evaluated various transient and accident conditions involving the use of the EFWS. The results of these evaluations showed that any one EFW pump (two electric motor driven and one steam turbine driven pumps are provided in each unit) could provide sufficient EFW flow to remove decay heat from the Reactor Coolant System. We have reviewed this information and have concluded that the flowrate design bases are acceptable at the ONS."

The NRC review of Duke's analysis of the various transients and accident conditions focused on the ability of the EFW system to adequately supply feedwater to the steam generators from any one EFW pump.

Thus, the focus of the NRC review relative to system flow requirements was on pumping capacity and the resultant capacity following the loss of one of the pumps.

FSAR Update as Required by 10CFR 50.71e

On July 19, 1982, the original FSAR was updated in accordance with the recently issued rule, 10 CFR 50.71e. In the updated FSAR (UFSAR), Duke added the EFW system discussion to incorporate the EFW modifications which included the installation of the motor driven EFW pumps (Ref. 1, 32, 42). The UFSAR was brought up to date as of December 31, 1981 by way of an FSAR update sent by letter dated July 19, 1982 (Ref. 32). The updated FSAR was stated to include the effects of: all changes made to Oconee or procedures as described in the original FSAR; all safety evaluations performed by Duke, either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or in behalf of Duke at Commission request. Duke did not intend for the FSAR updates to reflect any changes to the design basis of the plant, but to merely reflect the design basis at that time. It now appears that some of the statements were taken out of context and had the effect of making the design basis unclear on certain points.

A review of the EFW section in the UFSAR indicates that most of the information was obtained from the following three distinct sources:

1) EFW conceptual system information in Duke letter dated May 17, 1979. (Ref. 10)
2) Duke's Item 14 response in a letter dated April 3, 1981. (Ref. 16)
3) Duke's Item 17 response in a letter dated April 3, 1981. (Ref. 16)

As a result of the combination of the information from the above three sources, the UFSAR information was arranged in a manner which changed the intent and interpretation of the EFW system information. The first three paragraphs of Section 10.4.7.1 contain design bases information for the EFW system which was taken from the Duke's submittal that was dated May 17, 1979. One of the statements which is in question is contained in this part of the UFSAR and is as follows:

> "Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

As was indicated earlier, the above information was a summary of the HELB study results. The HELB analysis considered the EFW function to include multiple feedwater sources available to the site. In this study, Duke noted that a HELB of the main feedwater or auxiliary line could not only render EFW and feedwater inoperable, but also damage the 4160 V switchgear. The use of alternate units' EFW or the station ASW system was required to mitigate this event. So, when the statement is made that the EFW piping system can perform its safety related function in the event of a single failure coincident with a secondary pipe break, feedwater sources outside of the EFW system associated with the affected unit are credited. This was reviewed and accepted by the NRC in the 1973 Unit 2 & 3 SER (Ref. 4).

The other statement in question is contained in the next to last paragraph in Section 10.4.7.1. This statement consists of the following sentence:

> "In the event of a postulated break in the Main Steam or Main Feedwater system inside or outside containment coupled with a single active failure, the EFW system provides sufficient flow to ensure adequate core cooling."

This particular statement originated in the Item 14 response in Duke's letter dated April 3, 1981 and the NRC's SER dated August 25, 1981. Duke was responding to an NRC question regarding a line break that resulted in a depressurized steam generator. The following two quotes show the information that was contained in Duke's submittal and the NRC's associated SER:

Duke's April 3, 1981 statement

> "In the event of a postulated break in the main steam or main feed system, coupled with a single active failure of either one of the three emergency feedwater pumps, sufficient flow will occur to provide adequate core cooling."

NRC's August 25, 1981 statement

> "By letter dated April 3, 1981, the licensee responded that in order to provide emergency feedwater flow to the intact steam generator and isolate the ruptured steam generator the operator must take manual action. The system is designed so that a single active failure of any of the emergency feedwater pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator. The operator has sufficient control room indication of steam generator level and pressure to take the actions necessary to provide sufficient flow to the intact steam generator in time to maintain adequate core cooling. We find the response to this request acceptable."

The information from both Duke and the NRC show that the break that is being discussed is a line break that results in a depressurized steam generator. The original intent of the single failure requirement with the postulated main steam or main feedwater line break was focused on the capability of the EFW pumps to supply sufficient feedwater flow through the EFW flow paths to the steam generators. There was no integrated review of secondary system interaction. The FSAR has always included an analysis of the plant response to a main steam line break accident. In that analysis, continued EFW to the faulted steam generator is the only concern. The line break results in an overcooling of the plant, therefore there is no immediate need to establish EFW to the intact steam generator.

Generic Letter 81-14, "Seismic Qualification of Auxiliary Feedwater Systems"

As a result of the seismic design requirements of the SRP 10.4.9, the NRC issued Generic Letter 81-14, "Seismic Qualification of Auxiliary Feedwater Systems", in a letter dated February 10, 1981 (Ref. 15, 31). The generic letter requested licensees to define the extent to which the Auxiliary Feedwater systems are seismically qualified.

In a letter dated January 28, 1982 (Ref. 18), Duke responded to Generic Letter 81-14. The response indicated that the majority of the EFW system and necessary support systems were seismically qualified. In the response, Duke stated that the Oconee EFW system coupled with the dedicated SSF, currently under construction, meet the seismic requirements and no additional modifications to the system are necessary.

By a letter dated January 14, 1987 (Ref. 25), the NRC issued a safety evaluation for the review of the seismic qualification of the Oconee EFW system. In the safety evaluation, the NRC included the resolution of the potential backfit concerning the EFW system availability following a safe shutdown earthquake and concurrent single failure. Based on Duke's letters and the NRC's backfit analysis, the NRC concluded that the Oconee EFW system seismic qualification has been adequately addressed.

The NRC indicated that the SSF auxiliary service water system and HPI feed and bleed capability are important as alternate means of decay heat removal should the EFW system fail following a maximum hypothetical earthquake. The NRC stated that these additional means of decay heat removal are not only significant in the interim while the identified EFW system seismic deficiencies are corrected, but also serve as additional defense-in-depth protection against core melt in the long term given the seismically-induced flooding vulnerability of the EFW system. Based on the alternate means of decay heat removal, the NRC

Staff concluded that adequate post-seismic event shutdown decay heat removal capability is provided for assuring plant safety.

In the report section which discussed the EFW system and SSF single failure capability, the NRC outlined resolution of their potential backfit concern. The NRC reviewed the alternate means of decay heat removal and the modifications performed to address flooding in the turbine building. The NRC's review indicated that adequate core melt protection was provided and no further plant improvements were warranted. The NRC's determination was based on the flooding protection provided for the HPI and LPI pumps for use in the feed and bleed mode of operation along with the SSF auxiliary service pump. Both of the above alternate means of decay heat removal provide suitable redundancy to the EFW system since the EFW system itself is unprotected from flooding and, therefore, assumed unavailable following a maximum hypothetical earthquake. Thus, the NRC Staff closed the concern about decay heat removal capability and a concurrent single failure.

As can be seen in the review of the licensing basis of the EFW system relative to its seismic and single failure design, the NRC stated that the EFW system may not be available following a maximum hypothetical earthquake since the turbine building would be flooded and the EFW pumps would be considered unavailable. The NRC, again, accepted the Oconee EFW design based on the diversity of alternate means of decay heat removal. With the combination of HPI feed and bleed and the SSF auxiliary service water system, the NRC closed the issue for the EFW system seismic qualification concerning the concurrent single active failure.

Licensing Basis Conclusion

The licensing basis for the Oconee EFW system has evolved over the years with the TMI accident resulting in the greatest impact. The early versions of the FSAR were written with very little discussion on the EFW system, in fact it was included as part of the feedwater system section. The original design of Oconee consisted of a diverse and redundant steam and power conversion system for supplying feedwater to the steam generators. Each unit's emergency feedwater portion of the steam and power conversion system consisted of a single turbine driven emergency feedwater pump and was not designed to withstand a single failure.

The effects of a high energy line break on plant equipment and environment were considered during the postulated piping breaks analysis in 1973. The high energy line break analysis found that in addition to the main and emergency feedwater systems, the 4160 switchgear (TC, TD, and TE) could be lost following certain steam line or feedwater line breaks. Since the loss of the 4160 switchgear (TC, TD, and TE) resulted in a loss of all station power, similar to the FSAR Chapter 14 analysis, and the station had enough diverse systems available to cope with the event, the Oconee design was determined to be acceptable. To satisfy the single failure criteria, however, modifications were implemented (unit cross-connects and feedwater bypass lines) to make the other units' EFW systems available to the unit with the HELB. Therefore, diversity of EFW sources was credited to mitigate the HELB with a concurrent single failure.

Shortly after the TMI accident in 1979, but before the long term NRC plans for EFW upgrades were formalized and published, Duke committed to install two motor driven pumps in each unit's EFW system to enhance the system reliability. The initial concept for the enhanced system provided to the NRC, was a collection of the requirements that were applicable to the system at the time. The accidents that were evaluated for EFW, at the time of the initial system concept, were loss of main feedwater and SBLOCA. The system concept addressed a secondary line break, but based on the licensing history of the system, the break discussed was the break analyzed in the HELB study performed in 1973. Mitigation of the event, with a concurrent single failure, required the use of other units' EFW system or the station ASW system. Therefore, it is concluded that in making the statement "Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power", crediting other available sources of EFW at the site was required. This conclusion is supported by the EFW system Safety Evaluation described in the UFSAR which states that "A sufficient depth of backup measures is provided to allow steam generator water inventory to be maintained by any of the diverse methods listed above" These not only include, the affected unit's EFW system, but also the other units' EFW systems, the SSF ASW system, and the station ASW system. At the time of the original system concept, a line break that resulted in a depressurized steam generator was not considered for the EFW system since it resulted in an overcooling event, and therefore not a design basis event for EFW.

After the initial TMI-2 short term actions were complete, the NRC formalized their requirements in various NUREGS. The NRC requirements can be summarized into three basic areas, 1)EFW system was reviewed against requirements in the NRC's Standard Review Plan and associated Branch Technical Positions, 2) a reliability study was performed to identify any single point vulnerabilities during loss of feedwater events, and 3) EFW system flowrate design bases and criteria were re-evaluated.

Before the detailed reviews started, the NRC recommended that licensees upgrade their EFW systems to ESF grade or provide an alternate path of EFW. The Duke response, which was consistent with the original license of Oconee, was that the EFW system would not be upgraded to be an ESF system, because an alternate path was going to be available in the SSF.

As a result of the EFW system reviews, several key issues were discussed in correspondence between Duke and the NRC. Components associated with the feedwater water sources (hotwell, condensate storage tank, and upper surge tank) were not specifically considered as part of the EFW system review, however, the NRC did postulated the impact of the primary suction source not being immediately available and the operators ability to protect the EFW pumps during this event. The NRC credited the diversity of EFW sources (the unit cross-connects and the SSF) for the acceptability of the Oconee EFW system design. Additionally, the NRC accepted the EFW system design to limit or terminate EFW flow to the depressurized steam generators and direct the minimum flow to the intact steam generator in sufficient time to provide adequate core cooling following a main steam and feedwater line break. That review was very limited, in fact, the focus was on the EFW pumps and control valves, no review of system interaction was performed. It stated that manual operator action was required and that sufficient control room indication and time was available to direct sufficient flow to the steam generator. This conclusion does not negate the ability to use any unit's EFW system in the event of a failure of the affected unit's EFW system. Finally, as part of the NRC's review of the EFW system to the SRP requirements, they issued GL 81-14 "Seismic Qualification of Auxiliary Feedwater Systems". This issue was resolved taking credit for the diverse sources of feedwater at Oconee. It has been a consistent theme throughout the evolution of the design basis of the EFW system, acceptability because of the diversity of sources across the site.

The current UFSAR Chapter 10.4.7 includes information that was assembled from various submittals related to the EFW system design modifications following the TMI-2 accident and the NUREG-0660 EFW reliability study. The text in the UFSAR was taken from Duke submittals, however important background information was left out, making the statements misleading.

The first statement in question (below), was taken from the initial system concept when the motor driven pumps were installed. This submittal summarized the design bases of the EFW system as understood at that time.

> "Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

The statement was intended to reflect the high energy break evaluation performed for Oconee in 1973. When it states that the EFW piping system can perform its safety related function, it is crediting sources beyond that unit's EFW system. The FSAR safety evaluation also states that a sufficient depth of backup measures is provided to allow steam generator inventory to be maintained by any of the diverse measures listed. This is further justification that the EFW function is considered that provided across the site.

The second statement in question (below), was taken from Duke correspondence, in response to the NRC questions relating to limiting or terminating flow to the depressurized steam generator and directing flow to the intact steam generator.

> "In the event of a postulated break in the Main Steam or Main Feedwater system inside or outside containment coupled with a single active failure, the EFW system provides sufficient flow to ensure adequate core cooling."

This statement, without the question to which it was responding, is misleading. The NRC, in their SER stated the position much more accurately. They state:

> "... in order to provide emergency feedwater to the intact steam generator and isolate the ruptured steam generator the operator must take manual action. The system is designed so that a single failure of any of the emergency pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator. The operator has sufficient control room indication of steam generator level and pressure to take actions necessary to provide sufficient flow to the intact steam generator in time to maintain adequate core cooling."

As can be seen from the NRC statement the break is a secondary break that depressurizes the steam generator. In addition, this event is an overcooling event, and with a failure of the EFW system on the affected unit, other units' EFW could be aligned in time to provide adequate core cooling.

The licensing basis of Oconee has always relied on diverse and redundant methods of supplying feedwater to the steam generators to remove decay heat following various plant transients. The UFSAR is being revised by this change to more clearly reflect the position discussed in this evaluation.

Evaluation of USQ Questions

May the proposed activity:

1. Increase the probability of occurrence of an accident previously evaluated in the SAR?

No. The change does not increase the probability of an accident described in the SAR. The EFW system is used for mitigation of accidents. This change will clarify the FSAR bases for EFW single failure and to correct an omission that occurred with the installation of the MSLB circuitry. This change is not changing the physical plant or licensing basis. It is clarifying the design basis of EFW.

2. Increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

No. This change will clarify the UFSAR to describe the limits of the single failure evaluation specific to EFW as previously submitted to the NRC for review. There are no physical changes in the plant as a result of this change. As part of the NRC's review of the HELB, a loss of EFW was reviewed and based on the diversity of EFW sources, the design of the Oconee EFW system was found to be acceptable. As part of the post-TMI EFW review, the NRC reviewed a loss of the primary water source and again credited the diversity of the Oconee design and concluded that the design was acceptable. This change simply collects information provided in various submittals to clarify the basis for the EFW design. This change will also clarify the FSAR bases for EFW single failure and to correct an omission that occurred with the installation of the MSLB circuitry. This change does not increase the probability of malfunction of equipment important to safety evaluated in the SAR.

3. Increase the consequences of an accident previously evaluated in the SAR?

No. This change does not increase the consequences of an accident evaluated in the SAR. This change clarifies the UFSAR to better reflect the EFW system design that has been reviewed by the NRC. This change will also clarify the FSAR bases for EFW single failure and to correct an omission that occurred with the installation of the MSLB circuitry. The accidents/transients evaluated in the SAR which rely on EFW for mitigation are; Loss of Main Feedwater, Main Feedwater line break, Steam line break, Loss of Coolant Accident, and Anticipated Transient without SCRAM (ATWS). The consequences of the accidents will not be increased because of the clarification to the SAR.

4. Increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

No. This change does not increase the consequences of a malfunction of equipment important to safety evaluated in the SAR. This change clarifies the basis for the design of the Oconee EFW system relative to single failure. The statements in the current UFSAR addressed in the discussion above were taken from Duke/NRC correspondence and did not include enough background information to adequately describe the design of the EFW system. This change will add background information from the Duke/NRC correspondence to better describe the EFW design. This change will also clarify the FSAR bases for EFW single failure and to correct an omission that occurred with the installation of the MSLB circuitry.

5. Create the possibility for an accident of a different type than any evaluated previously in the SAR?

No. This change does not change the way EFW is operated, automatically or manually. It revises the UFSAR so that it includes a more accurate representation of the entire scope of events evaluated by the NRC and the response of the EFW system. The UFSAR, prior to this change, was not clear concerning information from the High Energy Line break analysis. This change includes the potential impact of the HELB on a particular unit's EFW system and the stations reliance on the diversity of the secondary side of the plant. The possibility of an accident of a different type than any evaluated in the SAR is not created as a result of this change.

6. Create the possibility for a different type of malfunction of equipment important to safety evaluated previously in the SAR?

No. This change does not alter the way EFW is operated, automatically or manually. There is no physical change to the plant as a result of this UFSAR change. This change is adding background information to the UFSAR to more accurately reflect the design basis. The possibility of a malfunction of a different type than any evaluated in the SAR is not created as a result of this change.

Does the proposed activity:

7. Reduce the margin of safety as defined in the bases for any Technical Specification?

No. The margin of safety as defined in the bases for any Technical Specification is not reduced as a result of this change. Currently T.S. 3.4 includes the EFW system operability requirements. This change does nothing to change the operability requirements or reduces the margin of safety as described in the T.S. These changes are clarifying the UFSAR relative to the single failure design of the EFW system.

Conclusion

These changes involve no Unreviewed Safety Questions. No Technical specification changes are required. Changes to UFSAR 10.4.7 are specified as included in Attachment 2.

Summary of the 50.59 evaluation for the annual report

UFSAR 10.4.7 is being revised to clarify the EFW design basis. This evaluation process includes collecting information which is included in NRC/Duke correspondence and updating the UFSAR to include this information to clarify the licensing basis of EFW relative to single failure. The changes being evaluated are included in sections 10.4.7.1, 10.4.7.2, 10.4.7.3, and 10.4.8. Specifically, the following changes are proposed:

Section 10.4.7.1 is being changed to clarify two statements. These first is:

"Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

This statement is being revised to clarify that its' original intent, which was to summarize the high energy line break analysis that was submitted to the NRC in 1973. In this analysis, feeding from another unit was credited. This statement will be changed to reflect this.

The second statement is:

"In the event of a postulated break in the Main Steam or Main Feedwater system inside or outside containment coupled with a single active failure, the EFW system provides sufficient flow to ensure adequate core cooling."

This statement is also being revised to clarify that its original intent. This was to summarize the design basis of the EFW system in the event of a pipe break that depressurizes a steam generator coupled with a single active failure of an EFW pump or control valve. This statement will be changed to reflect this.

Section 10.4.7.2 currently states that "Once automatically started, the EFW pumps will continue to operate until manually secured by the operator". This sentence is being revised to add "motor driven" to clarify that only the MDEFW pumps will always continue to run when automatically started until secured by the operator. The TDEFW pump is automatically secured by the Main Steam Line Break MSLB circuitry in the event of a MSLB. This needed UFSAR change was not made as a part of NSMs-x2873, which installed the MSLB circuitry. The modification installed circuitry that secured the TDEFW pump in the event of a MSLB. Therefore, after this modification, the TDEFW pump does not continue to run until manually secured by the operator in the event of a MSLB. This change was evaluated as a result of the modification NSM-x2873, therefore the evaluation below will focus on the changes related to single failure.

Section 10.4.7.3 is being revised to add clarification of the single failure design of the EFW system. A discussion of the High Energy Line Break (HELB) impact on the EFW system was included to document the Oconee reliance on the diverse sources of feedwater. Additionally, a clarification of the single failure analysis was provided. This docketed NRC/Duke correspondence reflected a review of the EFW pumps and EFW flow control valves for failures and the resultant impact on the EFW system.

Section 10.4.8 is being revised to add reference to Postulated Pipe Break Analysis, Duke letter about EFW conceptual design, and delete specific page reference to April 3, 1981 letter.

The licensing basis for the Oconee EFW system has evolved over the years with the TMI accident resulting in the greatest impact. The early versions of the FSAR were written with very little discussion on the EFW system, in fact it was included as part of the feedwater system section. The original design of Oconee consisted of a diverse and redundant steam and power conversion system for supplying feedwater to the steam generators. Each unit's emergency feedwater portion of the steam and power conversion system consisted of a single turbine driven emergency feedwater pump and was not designed to withstand a single failure.

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The first statement in question (below), was taken from the initial system concept when the motor driven pumps were installed. This submittal summarized the design bases of the EFW system as understood at that time.

"Sufficient redundancy and valving are provided in the design of EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary AC power."

The statement was intended to reflect the high energy break evaluation performed for Oconee in 1973. When it states that the EFW piping system can perform its safety related function, it is crediting sources beyond that unit's EFW system. The FSAR safety evaluation also states that a sufficient depth of backup measures is provided to allow steam generator inventory to be maintained by any of the diverse measures listed. This is further justification that the EFW function is considered that provided across the site.

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This statement, without the question to which it was responding, is misleading. The NRC, in their SER stated the position much more accurately. They state:

"... in order to provide emergency feedwater to the intact steam generator and isolate the ruptured steam generator the operator must take manual action. The system is designed so that a single failure of any of the emergency pumps or valves will not prevent the operator from directing sufficient flow to the intact steam generator. The operator has sufficient control room indication of steam generator level and pressure to take actions necessary to provide sufficient flow to the intact steam generator in time to maintain adequate core cooling."

As can be seen from the NRC statement the break is a secondary break that depressurizes the steam generator. In addition, this event is an overcooling event, and with a failure of the EFW system on the affected unit, other units' EFW could be aligned in time to provide adequate core cooling.

The licensing basis of Oconee has always relied on diverse and redundant methods of supplying feedwater to the steam generators to remove decay heat following various plant transients. The UFSAR is being revised by this change to more clearly reflect the position discussed in this evaluation.

These changes involve no Unreviewed Safety Questions. No Technical specification changes are required. Changes to UFSAR 10.4.7 are specified.

<u>References</u>

1. Early version of the FSAR for Oconee, amended as of 6/3/76, Sections 10.1, 14.1.2.8.3
2. Early version of the Technical Specifications for Oconee, as amended to 7/19/74, Section 3.4
3. Original SER for Oconee Unit 1 dated 12/29/70
4. Original SER for Oconee Units 2 & 3, dated 7/6/73
5. December 15, 1972 - NRC letter requesting high energy line break analysis.
6. June 22, 1973 - Duke letter regarding the high energy line break analysis.
7. April 25, 1973 - Duke letter transmitting the high energy line break analysis to the AEC.
8. April 1, 1979 and April 5, 1979 - Bulletin 79-05 and 79-05A issued by the NRC.
9. April 25, 1979 - Duke provided the design criteria for the motor driven EFW pumps.
10. May 17, 1979 - Duke supplied the conceptual design for the motor driven EFW pumps to the NRC.
11. August 24, 1979 - NRC meeting summary of the August 9, 1979, meeting.
12. December 21, 1979 - Duke provided the EFW reliability study to the NRC.
13. July 23, 1980 - Duke responded to NUREG-0667 recommendations.
14. November 14, 1980 - NRC request for additional information related to NUREG-0660.
15. February 10, 1981 - NRC issued Generic Letter 81-14.
16. April 3, 1981 - Duke response to NRC request for additional information related to NUREG-0660.
17. August 25, 1981- NRC SER on the EFW evaluation.
18. January 28, 1982 - Duke responded Generic Letter 81-14.
19. April 8, 1982 - NRC SER which discusses the EFW system flow requirements.
20. September 9, 1982 - NRC issued a Technical Evaluations Report for the EFW seismic evaluation.
21. October 13, 1982 - Duke responded to the NRC's Technical Evaluation Report.
22. December 26, 1984 - NRC provided a status of the EFW seismic qualification review which included a list of open items.
23. April 9, 1985 - Duke provided a response to the NRC's open items related to the EFW seismic qualification review.
24. April 28, 1986 - Duke provided an summary of modifications that addressed turbine building flooding.
25. January 14, 1987 - NRC SER for the EFW seismic qualification review.
26. PIP 0-O98-4208
27. April 13, 1979 - Duke response to IE Bulletin 05A.
28. April 26, 1979 - Duke response to TMI issues.
29. Specification OSS-0254.00-00-1000, "Design Basis Specification for the Emergency Feedwater and Auxiliary Service Water Systems", Revision 18.
30. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", Revised August 1980.
31. October 21, 1980 - NRC letter sending copy of Standard Review Plan Section 10.4.9, "Auxiliary Feedwater System (PWR)"..
32. July 19, 1982 - Duke submitted an update to the original FSAR Section 10, "Steam and Power Conversion", as amended to 12/31/81.

33. Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", Revisions 1 - July 1981.
34. CFR 50.59 Evaluations for NSM ON-12873 dated 7/18/96, ON-22873 dated 7/18/96, and ON-32873 dated 7/17/96 (Installation of MSLB Detection/Mitigation Circuitry).
35. Oconee Nuclear Station Technical Specifications, as amended to 9/24/98, Sections 3.4, 4.9.
36. Oconee Nuclear Station Selected Licensee Commitments, as amended to 10/13/98, Section 16.7.3.
37. Oconee Nuclear Station Updated Final Safety Analysis Report (UFSAR), as amended to 12/31/97, Sections 7.4.3, 8.3.2.2.4, 10.4.7, 15.0, 15.8, 15.14.
38. August 1, 1986 - NRC letter sending SSFI on EFW.
39. October 1, 1986 - Duke response to NRC's SSFI on EFW.
40. April 30, 1987 - NRC response on Duke response to NRC's SSFI on EFW.
41. October 31, 1980 - NRC letter incorporating all TMI related items approved for implementation.
42. Federal Register, Volume 45, Number 92, May 9, 1980, Pages 30614 through 30616, addressing new rule on periodic updating of the Final Safety Analysis Report.

10.4.6.6 Interactions with Reactor Coolant System

The effects of inadvertent steam relief or steam bypass are covered by the analysis of the steam line break given in Section 15.13, "Steam Line Break Accident." The effects of an inadvertent rapid throttle valve closure are covered by the loss of full load discussion in Section 15.8, "Loss of Electric Load Accidents."

Following a turbine trip, the reactor will trip automatically due to anticipatory trip logic. The safety valves will relieve excess steam until the output is reduced to the point at which the steam bypass to the condenser can handle all the steam generated.

In the event of failure of a main feedwater pump, there will be an automatic runback of the power demand. The one main feedwater pump remaining in service will carry approximately 60 percent of full load feedwater flow. If both main feedwater pumps fail, the turbine and reactor will be tripped, and the emergency feedwater pumps started.

1 On failure of a condensate booster pump, the spare condensate booster pump is automatically started.

10.4.7 EMERGENCY FEEDWATER SYSTEM

10.4.7.1 Design Bases

The Emergency Feedwater (EFW) System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the Condensate/Main Feedwater System, to remove energy stored in the core and primary coolant. The EFW System is designed to provide sufficient secondary side steam generator heat sink to enable cooldown from reactor trip at power operation down to cold shutdown conditions. The EFW System may also be required in some other circumstances such as cooldown following a loss-of-coolant accident for a small break. The EFW System is shown in Figure 10-8.

2 The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as
6 indicated by Main Feedwater Pump low hydraulic oil pressure. In addition, low water level in either
2 steam generator, after a 30 second delay to prevent spurious actuations, will start the Motor Driven
6 Emergency Feedwater Pumps. The EFW System will supply sufficient feedwater to enable the Reactor Coolant System to cool down to conditions at which the Decay Heat Removal System may be operated.

Three EFW pumps are provided, powered from diverse power sources. Two full capacity motor-driven pumps are powered by the emergency A.C. Power System, each supplying feedwater to one steam generator. One turbine-driven pump, supplying feedwater to both steam generators, may be driven by any of three separate steam sources; A Main Steam, B Main Steam, or plant start-up steam (also called the Auxiliary Steam System). Although the total rated capacity of all three EFW pumps is 1780 gal/min, the flow capacity of any one of the pumps is sufficient to enable safe and orderly cooldown of the Reactor Coolant System. ~~Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary A.C. power. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS)~~).

Insert 1 as new paragraph

The three units are provided with separate EFW Systems. The discharge header of each EFW System is cross connected making each system capable of supplying either unit.

Automatic initiation of the turbine-driven EFW pump is independent of AC power. Based on the required emergency feedwater flow, sufficient inventory of EFW is available for maintaining hot shutdown
6 for at least 75 minutes from both upper surge tanks. The inventory in the upper surge tanks is assured by

Insert 1
Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections (unit and train) allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary A.C. power. In the case of a secondary pipe break coincident with a single failure, the emergency feedwater function may be provided by another unit's EFW pumps, the SSF ASW pump or the station ASW pump. Manual action is required to align these other sources.

All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Insert 2
In the event of a postulated break in the Main Steam or Main Feedwater System inside or outside containment, that results in a depressurized steam generator, coupled with a single active failure of an EFW pump or control valve, the EFW System provides sufficient flow to ensure adequate core cooling.

4 auto closure of the hotwell makeup control valves on a low upper surge tank level signal. The upper
surge tanks and the associated piping from them to the EFW pump suctions are seismically qualified.
The condenser hotwell is also seismically qualified with a nominal capacity of 120,000 gallons. However,
the condenser hotwell is seismically qualified without any piping connected to it, and not all of the piping
from the hotwell to the EFW pump suctions has been seismically qualified.

Insert 2 as paragraph 2 of 6 (see p7)

~~In the event of a postulated break in the Main Steam or Main Feedwater System inside or outside containment coupled with a single active failure, the EFW System provides sufficient flow to ensure adequate core cooling.~~

The plant transient which requires the highest Emergency Feedwater System flow, and as such constitutes
2 the Emergency Feedwater design basis transient, is the loss of main feedwater transient. This transient
combines the highest heat load, decay heat plus reactor coolant pump heat, with the minimum heat sink
due to the instantaneous loss of both main feedwater pumps. A discussion of the demand on the EFW
5 system for each transient follows. The following, with the exception of Steam Line Break (Section
5 10.4.7.1.8, "Steam Line Break") and Small Break LOCA (Section 10.4.7.1.9, "Small Break LOCA"),
2 should not be considered Design Basis Transients for the entire plant, but for Emergency Feedwater only.

10.4.7.1.1 Loss of Main Feedwater (LMFW)

Those transients which result in losing feedwater delivery from the Main Feedwater/Condensate System
6 are classified as a loss of main feedwater. Since the reactor coolant pumps remain on, the control valves
0 modulate to control steam generator level at 30 inches. The transient requires feedwater to be delivered at
a rate sufficient to remove decay heat and reactor coolant pump heat. One motor driven emergency
6 feedwater pump delivering 400 gal/min. at a steam generator pressure of 1064.7 psia and an EFW
temperature of ≤ 130°F will provide adequate heat removal capacity.

3 10.4.7.1.2 LMFW with Loss of Offsite AC Power (LOOP)

3 The loss of offsite AC power causes the reactor to trip, the turbine to trip, and the condensate booster
pumps and hotwell pumps to trip causing a loss of main feedwater. The emergency feedwater pumps are
actuated on the main feedwater pump trip. Since the reactor coolant pumps have tripped, steam
3 generator level control increases the level setpoint to 240 inches on the extended startup range to promote
the natural circulation mode of heat removal. The emergency feedwater control valves open to allow full
system flow until the controlling level is attained. Feedwater requirements are determined by core decay
heat removal demand. One motor driven EFW pump can deliver sufficient feedwater to meet the
demand.

3 10.4.7.1.3 LMFW with Loss of Onsite and Offsite AC Power (Station Blackout)

6 This transient is the result of a station blackout condition. This transient is similar to the Section
6 10.4.7.1.2, "LMFW with Loss of Offsite AC Power (LOOP)" analysis with the additional assumption
that the onsite emergency AC power sources have been lost. This results in the loss of the motor driven
3 emergency feedwater pumps. This transient is not a design basis event. The turbine-driven emergency
3 feedwater pump should be available for this event because of its AC power independence; however, the
3 SSF ASW is required to remove the decay heat in this transient. The transient is described in Section
3 8.3.2.2.4, "Station Blackout Analysis."

10.4.7.1.4 Plant Cooldown

In addition to providing sufficient heat removal capacity immediately following a transient, the
requirements for plant cooldown from full power operation to RCS temperatures where switchover to the
Decay Heat Removal System can be accomplished has been analyzed. All heat sources have been

2 Automatic starting of the MDEFWP's is determined by the position of the control room selector switch
2 for each pump. The MDEFWP's are provided with a four position selector switch which allows the
2 operator to select between Off, Auto 1, Auto 2 and Run. When the selector switch is in the Auto 1
3 position, LOW STEAM GENERATOR WATER LEVEL in either steam generator (OTSG) will start
2 the pump after a 30 second time delay to prevent spurious actuations. When the selector switch is in the
2 Auto 2 position, LOW STEAM GENERATOR WATER LEVEL or LOSS OF BOTH MAIN
2 FEEDWATER PUMPS will start the pump. Loss of both main feedwater pumps is sensed by pressure
6 switches which monitor feedwater pump turbine control oil pressure. Loss of both Main Feedwater
6 Pumps actuation is by the control oil pressure switches sensing loss of feedwater pumps.

2 <u>Turbine Driven EFW Pump (TDEFWP)</u>:

2 Automatic starting of the TDEFWP is determined by the position of the control room selector switch for
2 the pump. The TDEFWP is provided with a three position-pull to lock selector switch which requires
2 that the control room operator manually take the switch to the OFF position through a deliberate action.
2 The operator can select between Off, Auto and Run. When the selector switch is in the Auto position,
2 LOSS OF BOTH MAIN FEEDWATER PUMPS will start the pump. Loss of both main feedwater
6 pumps is sensed by pressure switches which monitor feedwater pump turbine control oil pressure. If a
5 main steam line break signal is present and the selector switch is in AUTO, the TDEFWP will
5 automatically stop and prevent an auto start. The operator can manually start the TDEFWP by placing
7 the selector switch to RUN.

Once automatically started, the motor-driven EFW pumps will continue to operate until manually secured by the
operator. Each emergency feedwater discharge line to each steam generator is provided with a control
5 valve and check valve. The control valves are normally closed due to steam generator level > 30". The
valves are arranged to fail to the automatic control mode upon loss of DC control power to the
manual/auto select solenoid. If the selected train of automatic control fails, then the valve would fail
open. Also, upon loss of station air, the valves will maintain their position with N_2 backup. If N_2
backup fails then the valve would fail open. These modes of operation show that emergency feedwater
2 isolation is not possible with valve control circuitry or motive force failure. Open/Closed valve position
2 indication is provided for each control valve in the main control room at the valve manual loader.

5 In automatic, a solenoid valve on each control valve is de-energized, allowing the valve to receive a control air signal for valve modulation in response to steam generator level, independent from the ICS.

The EFW pumps normally discharge into separate lines feeding a separate steam generator through the auxiliary feedwater header.

A flow path is also provided to the upper surge tank dome (connected to the condenser) for minimum
0 recirculation flow and testing purposes. A continuous recirculation flow is provided for the turbine driven
pump, limited by fixed orifices. A self-contained automatic recirculation valve is provided for each motor
0 driven pump to assure individual pump minimum flow when needed during operation. A flow path is
0 provided from the discharge of each motor driven pump to the upper surge tank for full flow testing.
Power for the motor driven pumps is normally provided by the normal station auxiliary A.C. Power
3 System. During loss of offsite power operation, these pumps are aligned to the Emergency A.C. Power
System. Motive steam for the turbine driven pump is provided from either of the two steam generators by main steam lines upstream of the stop valves, and is exhausted to the atmosphere. Either steam supply will provide sufficient steam for turbine operation. Either steam supply may be isolated if necessary. A check valve is provided in each steam supply line to prevent uncontrolled blowdown of more than one steam generator.

A sufficient depth of backup measures is provided to allow steam generator water inventory to be maintained by any of the diverse methods listed above. Although redundancy and diversity is provided in the listed measures, the EFW System has been designed with special considerations to enable it to function when conventional means of feedwater makeup may be unavailable.

Redundancy is provided with separate, full capacity, motor and turbine driven pump subsystems. Failure of either the motor driven pumps or the turbine driven pump will not reduce the EFW System below minimum required capacity. Pump controls, instrumentation, and motive power are separate in design. Separate piping subsystems include redundant hotwell, upper surge tank, and condensate supply piping, aligned individually to the separate pump trains. Cross-connection is provided, however, to allow a subsystem to supply all pumps in the event of single failure of a suction piping subsystem. The same design philosophy is included in the discharge piping subsystems.

In order to provide sufficient EFW flow to the intact steam generator to ensure adequate core cooling, and under a main steam or main feedwater break in OTSG A with a single active failure of motor driven emergency feedwater pump B train, the operator must manually close the motor operated isolation valve
6 (FDW-372) or the flow control valve FDW 315 on OTSG A. This action can be done from the Control Room. The same is true for OTSG B and motor driven emergency feedwater Pump A. The operator has sufficient Control Room indication of steam generator level and pressure and would immediately be aware of such a situation.

Concurrently, the operator would monitor the intact steam generator to assure adequate inventory and secondary heat removal via either Main Feedwater or Emergency Feedwater Systems.

In the event of a postulated break in the Main Steam or Main Feed System, coupled with a single active failure of either one of the three emergency feed water pumps, sufficient flow will occur to provide adequate core cooling.

5 With a postulated break associated with the 'A' OTSG and a failure of the 'B' motor driven emergency
5 feedwater pump, the normal feedwater system will be isolated to both steam generators and the TDEFWP
5 will be inhibited from automatically starting. The TDEFWP can be manually started by placing its
7 control switch to RUN.

5 With a postulated break associated with the 'A' OTSG and an active failure occurs with the flow control
5 valve (FDW-316), the Main Steam Line Break Circuitry must be disabled by the operator to allow
5 emergency feedwater flow alignment through the main feedwater startup control valves to either the main
7 or auxiliary nozzles.

5 In the unlikely event that FDW-315, 316 fail open (on a loss of compressed air and nitrogen), an operator
5 could manually adjust either one of the valves as they are located in the Penetration Rooms which are
5 adjacent to the Control Room.

*TP → of the EFW pumps or EFW flow control valves,

The spectrum of transients which require EFW system performance for post trip heat removal have been evaluated assuming only one motor driven emergency feedwater pump is available to deliver the necessary feedwater. Any single failure in the three pump-two flowpath EFW system design will not result in only one motor driven EFW pump available, so that this assumption is overly conservative. These analyses verify the acceptability of the Emergency Feedwater System design.

TP In the unlikely event that a High Energy Line break, coincident with a single active failure, renders that unit's EFW System unavailable, feedwater can be re-established via the other units' EFW pumps through the unit cross-connects, the SSF Auxiliary Service Water pump, or the station Auxiliary Service Water pump.

10.4.8 REFERENCES

1. R. B. Davis (Babcock & Wilcox), letter to B&W 177 FA Owners Group TMI-2 Subcommittee, May 23, 1979.

2 2. W. O. Parker (Duke) letter to H. R. Denton (NRC), April 3, 1981~~, page 32~~.

7 3. ONOE-11376, changes to support multiple unit alignment to the ~~Auxiliary~~ Steam Header.

THIS IS THE LAST PAGE OF THE CHAPTER 10 TEXT PORTION.

4. Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, 1, 2, & 3, dated April 25, 1973 and Supplement 1, dated June 22, 1973.

5. Duke letter to NRC dated May 17, 1979; sending EFW conceptual design with addition of MDEFW pumps.