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SUBJECT: Forwards response to 980413 RAI re TS change 95-03, addressing newly installed MSLB protection circuitry.

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June 16, 1998

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Proposed Revision to Technical Specifications
Technical Specification Change # 95-03
Response to Request for Additional Information #3

In letters dated July 15, 1997, February 9, 1998, March 3, 1998, and April 13, 1998, Duke submitted a proposed amendment to the Oconee Nuclear Station (ONS) Technical Specifications to address the newly installed main steam line break protection circuitry.

In a Request for Additional Information dated April 13, 1998, the NRC staff requested additional information regarding certain aspects of the main steam line break protection circuitry. Please find the response to this Request for Additional Information in Attachment 1.

If there are any additional questions, please call David Nix at (864) 885-3634, or Ed Burchfield at (864) 885-3292.

Very truly yours,

W. R. McCollum, Jr.
Vice President
Oconee Nuclear Site

Attachment

A001/1

cc: Mr. L. A. Reyes
Regional Administrator, Region II

Mr. M. A. Scott
Senior Resident Inspector

Mr. D. E. LaBarge
ONRR, Project Manager

Mr. M. Batavia
DHEC

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Question #1:

Provide the results of a containment analysis (as requested by Bulletin 80-04) that reflects the actual design of the MFW isolation capability at Oconee. The analysis should assume that feedwater continues to be added to the faulted steam generator (via the condensate pumps) for ten minutes (assumed for operator action to secure flow) or for the time it takes to isolate MFW flow via the MFW block valves if they can close under the differential pressure associated with the condensate pumps. The analysis can take credit for all containment cooling features, since no other single failures need be assumed. The analysis should also assume that offsite power is available since the condensate pumps would not be running for a loss of offsite power.

Response #1:

An analysis was performed for a main steam line break (MSLB) with continued feedwater addition to the faulted steam generator via the condensate booster pumps for ten minutes due to the failure of the main feedwater (MFW) control valve to close. The MFW block valves are assumed to be unable to close under the differential pressure associated with the condensate pumps. It is assumed that the MSLB Detection and Feedwater Isolation System functions to trip the MFW pumps and the turbine-driven emergency feedwater (EFW) pump, and to close the MFW control valve to the unaffected steam generator. Since a failure of the MFW control valve to the affected steam generator is assumed, no other single failure is postulated. Thus, two trains of high pressure injection (HPI) are credited and the EFW flowrate is controlled to maintain steam generator level at the control setpoint. Credit is taken in the analysis for all containment cooling features. Offsite power remains available.

RETRAN-02 Steam Line Break Analysis

The methodology used to perform the RETRAN-02 steam line break analysis is detailed in topical report DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology." DPC-NE-3003-PA states that nominal full power heat addition by the MFW heaters is maintained until after MFW isolation. This conservative assumption has a minor impact on the containment response for cases with quick MFW isolation. However, for the case of continued MFW addition, this assumption sustains heat addition by the MFW heaters for a much longer time even though the steam source for feedwater heating is no longer available after the reactor trip occurs. Therefore, it is assumed that heat addition by the MFW heaters decreases linearly to zero over the 60 seconds following reactor trip. In addition, the reactor coolant pumps in the unaffected loop are assumed to be tripped at 235 seconds. This assumption was necessary to permit continued running of the computer model by avoiding a code job abort due to a code error in one of the unaffected loop volumes. Although this assumption stops the heat addition from these two pumps, the neglected energy addition is small compared to the total energy removed from the primary system.

The sustained MFW flow results in significant overcooling of the primary system. Both liquid accumulation in the steam generator tube bundle and liquid carryout into the steam line in the affected steam generator occur. The modeling described in DPC-NE-3003-PA conservatively compensates for liquid carryout by tracking the amount of liquid that is carried out and returning it to the steam generator as additional heated feedwater. This modeling approach results in artificially high compensating flowrates as the steam generator fills with liquid. To avoid unrealistic compensating flowrates, this modeling is stopped once the flowrate exceeds approximately 150% of the MFW flow rate. This modeling approach remains conservative. The break flow enthalpy prediction is still adjusted as described in DPC-NE-3003-PA such that all of the break flow prior to the steam line being flooded is released in the form of steam.

Even with continued feedwater addition to the affected steam generator, there is sufficient control rod worth and negative reactivity from the injected boron to prevent a return to power. Reactor Coolant System (RCS) pressure, pressurizer level, and RCS temperatures rapidly decrease due to the overcooling of the primary system.

Main feedwater flow is added to the affected steam generator via the condensate booster pumps and the D heater drain pump. The D heater drain pump trips at approximately 170 seconds due to depletion of the inventory in the D heater drain tank. The condensate booster pumps deliver a substantial amount of flow until main feedwater flow is terminated by the operator at ten minutes. Emergency feedwater is not delivered to the affected steam generator since level does not decrease below the control setpoint.

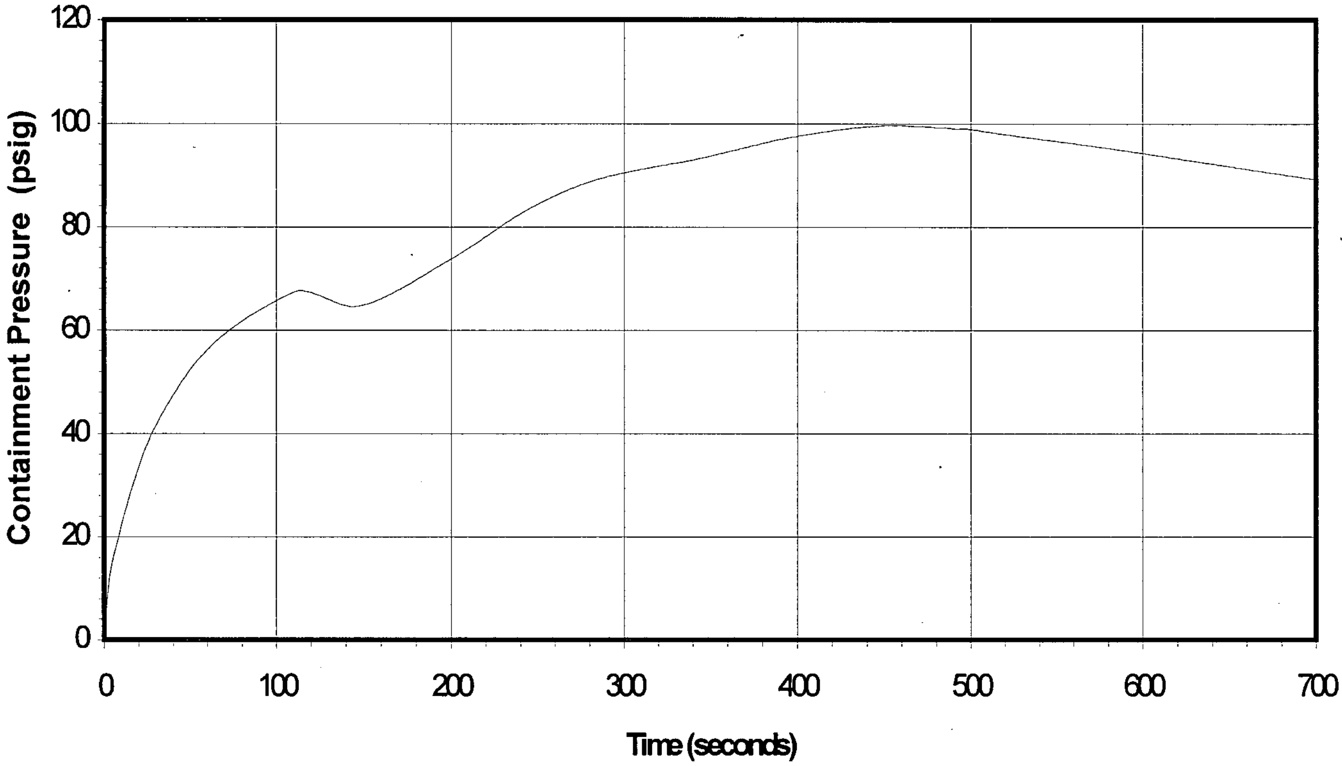
Break flow decreases steadily as the containment backpressure approaches the pressure in the affected steam generator. At 500 seconds, the hot leg temperature of the affected loop decreases below the saturation temperature corresponding to the pressure in the steam generator. Thus, no steam is released after this point and the break enthalpy decreases to that of a liquid release. By 510 seconds, the steam line between the affected steam generator and the break location is filled with liquid. Thus, a substantial amount of liquid break flow continues that is essentially equal to the rate of main feedwater addition by the condensate booster pumps. The transient simulation is terminated at 700 seconds.

FATHOMS Containment Analysis

The methodology used to perform the FATHOMS containment analysis is detailed in topical report DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology." Three reactor building cooling units (RBCUs) and two spray trains are assumed to be available. The RBCU actuation signal occurs at 2 seconds, and the units are running at 80 seconds. The containment spray actuation signal occurs at 17 seconds, and the spray flow starts at 109 seconds. Containment pressure increases steadily until the peak containment pressure (Figure 1) of 100 psig is reached at 445 seconds. The containment pressurization is terminated by the decreasing break flowrate as well as the change in the break flow from steam to liquid. The simulation is run for 700 seconds.

Figure 1

Oconee Steam Line Break With Continued MFW Addition: Containment Pressure



SUMMARY:

Duke performed a containment analysis to allow the NRC staff to assess the design of the MFW isolation capability at Oconee. The analysis assumed that feedwater continued to be added to the faulted steam generator, via the condensate pumps, for ten minutes. The analysis took credit for all containment cooling features since no other single failures were assumed. The analysis also assumed that offsite power was available. The analysis indicated that a peak reactor building pressure of 100 psig is reached at 445 seconds into the accident simulation. The containment temperature exceeded the equipment qualification limit during the 700 second transient simulation. However, this requested analysis is considered to be outside Oconee's current licensing basis. UFSAR Chapter 6 describes Oconee's current limiting MSLB analysis with respect to containment temperature and pressure.

Question #2:

Identify the stroke time of the MFW block valves that would exist with the worst case differential pressures following a MSLB with operation of the condensate pumps. If the stroke time is less than 10 minutes, provide further justification for not including these valves in the proposed technical specification.

Response #2:

In 1992, the MFW block valve closure capability was analyzed using methodology from the Generic Letter 89-10 program. The analysis results concluded that the MFW valves were not capable of closing under a differential pressure of 500 psig. The actual maximum discharge pressure of the Condensate Booster Pumps at shutoff conditions is greater than 600 psig. As a result, the MFW block valves were not credited for closure in the worst case main steam line break conditions.

Question #3:

In your response DEC stated that there is a high probability that the containment would not fail even if the design pressure was exceeded. Use the results of the analysis from Item 1 above to support the conclusion that the containment is not likely to fail following the worst case MSLB inside containment and a failure of the MFW control valve to close.

Response #3:

A calculation was performed to assess the ultimate capacity of the Unit 3 reactor building and associated penetrations. The intent of this calculation was to determine the effects of internal pressure, beyond the reactor building design pressure, on the structural integrity of the reactor building. This analysis used actual statistical material property values for liner plate steel, concrete and reinforcing steel, and tendons. The analysis was performed using a non-linear finite element computer analysis for the containment structure, and using existing documentation and hand calculations for the penetrations. The Unit 1 and 2 reactor buildings are of identical design and construction as the Unit 3 reactor building.

The overall containment mean ultimate capacity was determined to be 144.0 psi with a standard deviation of 1.95 psi. Initial yield of the tendons using mean material property values was found to occur at not less than 140.0 psi. The liner plate and reinforcing steel already had yielded in tension at 125 to 135 psi based on their mean material property values. Reactor building penetrations were determined to have a higher ultimate capacity than the reactor building. Therefore, the ultimate capacity of the penetrations is less limiting than the ultimate capacity of the reactor building. The results of Question #1 indicated that the peak reactor building pressure would be 100 psig. This peak pressure is less than 144 ± 1.95 psig. Therefore, the reactor building would not fail during the event analyzed in Question #1.

Question #4:

Identify all of the single active failures that could cause a MFW control valve not to go closed following an MSLB actuation signal; e.g., loss of power, loss of air, failure of the actuation circuitry, failures in the integrated control system, etc.

Response #4:

The MFW control valves are 16-inch Fisher model ENA, single port with cage guiding, balanced trim, and push down to close. The actuators are pneumatic cylinder, Fisher model 476L, which lock in the last controlled position upon loss of supply air.

For normal operation under the Integrated Control System (ICS), the MFW control valves receive an instrument signal (3-27 psig) from E/P # 1FDWEP0007 or 8. This signal goes to the Bailey positioner on the valve. The positioner interprets the signal to open, modulate, or close the valve via the actuator. If supply air is lost, the trip lock valves will lock the actuator in the last controlled position. Each actuator is equipped with a manual handwheel to close the valve if required. See the attached instrument detail.

The MSLB circuitry that closes the MFW control valve is comprised of two solenoid valves, one per each train of MSLB, in series between the electric/pneumatic (E/P) converter and the pneumatic MFW control valve positioner. Upon a MSLB actuation either or both of these solenoids will energize which vents the pneumatic signal going to the pneumatic MFW valve positioner. When the pneumatic signal is vented, the valve closes. The MSLB initiating components (solenoids) are downstream of the ICS component (E/P). A single failure in the ICS will not affect the operation of the MSLB circuitry. The circuitry and instrumentation of the MSLB is powered from two separate vital power panel boards. Therefore, no single active failure to the electrical/electronic controls would cause a MFW control valve not to go closed following a MSLB actuation.

The following failures, listed by type of failure, could cause a MFW control valve not to go closed:

I. Loss of Air Failures:

- A rupture of the supply air tubing to the positioner would result in the actuator not being able to close the valve.
- A rupture of the air output tubing from the positioner to the upper chamber of the actuator would result in the actuator not being able to close the valve.

II. Mechanical Failures:

- A failure of the valve positioner could cause the valve to fail as-is, open, or close.
- If the actuator stem to valve stem coupling fell off, then the actuator would not be capable of fully closing the valve.

FUNCTIONAL DESCRIPTION:
1FDWEP0007 (1FDWEP0008) RECEIVES SIGNAL FROM ICS TO CONTROL VALVE 1FDW-32 (1FDW-41). 1FDWSV1089 AND 1FDWSV1090 (1FDWSV1085 & 1FDWSV1086) RECEIVE SIGNAL FROM MSLB SYSTEM TO CLOSE VALVE 1FDW-32 (1FDW-41) UPON DETECTION OF MAIN STEAM LINE BREAK.

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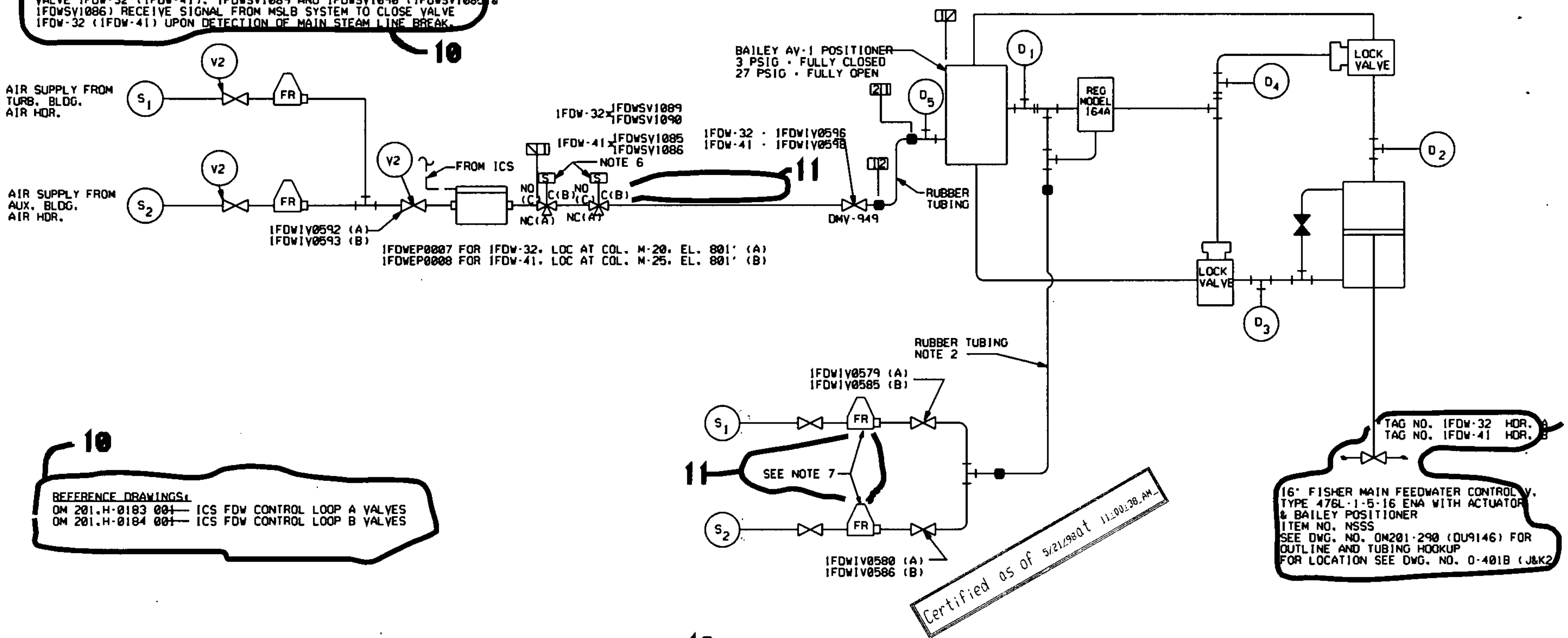

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REFERENCE DRAWINGS:
OM 201.H-0183 001 ICS FDW CONTROL LOOP A VALVES
OM 201.H-0184 001 ICS FDW CONTROL LOOP B VALVES

DESIGN PARAMETERS

LINE NO.	MATERIAL	PIPE SPEC.	CLASS
1	1/2" SS	PS-2701.2	F
2	1/2" RUBBER TUBING		CG*

* - COMMERCIAL GRADE

NOTES:
1. TWO (2) REQUIRED.
2. IMPERIAL EASTMAN TYPE C404 TUBING, OR SWAGELOK-PB SERIES HOSE.
3. D* - DIAGNOSTIC TEST PORT (QUICK CONNECTS)
4. ● - TUBING TO RUBBER TUBING ADAPTER.
5. ALL TUBING NOT DESIGNATED TO BE 1/4" COPPER, PS-162, CLASS G.
6. LETTERS IN ( ) INDICATE PORT ON SOLENOID VALVES.
7. AIR REGULATOR SET @ 90 PSIG.

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NUCLEAR SAFETY RELATED EXCLUSIONS:
1. FITTINGS, TUBING, PIPING & VALVES (NOT PART OF A SAFETY RELATED PIPING SYSTEM, ETC.)
   A) ALL EXCEPT LINE PARAMETERS 1 AND 2
2. INSTRUMENTS
   A) 1FDWEP0007
   B) 1FDWEP0008

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10	NSM ON-12989/00/AL1	DRL	10-13-97	JMB	10-13-97	DET	10-20-97	BHJ	BJS	DKH
ORIG.	ORIGINAL DRAWING RETIRED									

INSPECTED (CIVIL / ELEC. / MECH.)

DUKE POWER COMPANY
OCONEE NUCLEAR STATION UNIT 1

INSTRUMENT DETAILS
MAIN FDW VALVE CONTROL
1FDW-32 & 1FDW-4

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DESIGNER ____ DATE ____ INSP. ____ DATE ____
DRAWN RLB DATE 4-1-79 INSP. H.B. DATE 6-16-79
CHECKED BHS DATE 6-17-79 APPR. WHD DATE 6-17-79

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