NSIC

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

DECEMBER 21 1979

Docket No. 50-285

Mr. W. C. Jones
Division Manager, Production
Operations
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

SUBJECT: AUTOMATIC INITIATION OF AUXILIARY FEEDWATER SYSTEMS AT FORT CALHOUN STATION, UNIT NO. 1

In recent communications, your staff has indicated that a proposed design, using control grade components, which would automatically initiate the auxiliary feedwater systems at your facility upon the loss of main feedwater flow will be submitted in the near future. This submittal was in response to Short-Term Recommendation 2.1.7.a, "Auto Initiation of the Auxiliary Feedwater System", as clarified in our letter October 30, 1979 which was addressed to all operating nuclear power plants.

We will review your proposed design against each of the seven positions stipulated in Short-Term Recommendation 2.1.7.a. In response to this recommendation, some licensees have raised the issue of the applicability of current analysis of a main steam line break or main feedwater line break assuming early initiation of auxiliary feedwater flow with a failure to limit flow to the affected steam generator. In question is whether the change in assumptions would increase the calculated containment pressure or the likelihood of return to power. These questions are believed applicable for either manual or automatic initiation of the auxiliary feedwater system. You are requested to resolve this concern by submitting an analysis within twenty (20) days after receipt of this letter (telecopied on date signed). The enclosure to this letter provides a list of questions and information you should address as appropriate.

As a result of this concern and pursuant to our letter of October 30, 1979, you should not implement automatically initiated AFWS flow until we have completed our review and issued an approval. However, to resolve this matter as expeditiously as possible, you should continue with the procurement of equipment and proceed with the installation to the extent possible without activating the automatic-start system or adversely affecting the manual-start AFWS.

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You are requested to propose Technical Specifications for the AFWS modifications. Sample Technical Specifications are enclosed for your consideration. In addition, you will need to revise normal and emergency operating procedures as required by this modification and train the plant operations people as required by these procedures. Particular attention to the means of controlling the bypass capability of the automatic AFWS turbine start signal is recommended.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure:
Sample TS Pages

cc: w/enclosure
See next page

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Omaha Public Power District

cc w/enclosure(s):

Margaret R. A. Paradis
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D. C. 20036

Blair Public Library
1665 Lincoln Street
Blair, Nebraska 68008

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Enclosure

REQUEST FOR INFORMATION

AUTOMATIC INITIATION OF THE AFWS AFFECT ON

MAIN STEAM LINE BREAK ACCIDENT ANALYSIS

A. Return to Power

1. Provide the results of analyses of main steam line breaks that are the most limiting with respect to fuel failure resulting from return to power. Analyses should be presented covering:

   a. Break inside containment

   b. Break outside containment

   c. Availability or loss of offsite power

   Justify omitting an analysis for any of the above.

2. Provide the time sequence of all actions and events occurring during each of the postulated steam line break transients. These events and actions should include:

   a. Reactor scram

   b. Turbine trip

   c. Steam line isolation

   d. Feedwater isolation

   e. ECCS actuation

   f. Auxiliary feedwater actuation and control

   g. Safety/relief valve actuation (primary and secondary systems)

   h. Operator actions (define credit for operator action)

   i. Initiation of onsite power (if required).

3. For each of the above, identify the initiating signal, the protection system that initiates the action, and the extent of the action ending with the time the element (i.e., MSIV, turbine stop, turbine control, turbine bypass, etc.) reaches its new condition. The above events are to reflect the expected response of the plant and systems.

4. Identify and justify any equipment that does not meet Regulatory Guides and IEEE-279 requirements.

5. Provide a list of potential single failures that could affect each of the above actions and show how the analyses presented consider the worst single failures from a fuel failure standpoint. Note that normal control systems should not be considered to function if their action would be beneficial with respect to fuel failures.

6. Provide the following information as a function of time:

   a. Minimin DNBR

   b. Cladding temperature if DNBR limit is exceeded

   c. Feedwater flow into faulted and nonfaulted steam generators (main and auxiliary)

   d. Steam generator liquid mass, heat transfer area covered, heat transfer rate, and pressure

   e. Break flow rate

   f. Other steam release rates in secondary systems

   g. Primary system pressure

   h. Pressurizer level

   i. Hot channel flow rate

   j. Core inlet and outlet temperature

   k. Pressurizer safety/relief valve flow rate

   l. ECCS flow rate.

   The analysis should be carried out until the effects of delayed neutrons and moderator feedback have turned around and the subcriticality margin is increasing.

   Note the DNBR calculations must reflect the initial plant perturbations due to moderator and pressure decrease and loss of offsite power (if appropriate). Also discuss how the effects of a stuck rod are considered when calculating DNBRs after the rods have been inserted. If fuel damage occurs (i.e., violation of DNBR), provide fraction of fuel that failed and offsite dose calculations. Also provide and justify DNB correrlations used in the analyses.

B. Containment Pressure

Provide the following information to show that the containment pressure will be acceptable following a main steam line break.

1. Review your current analysis of this event, and provide NRC with the assupmtions used during this analysis. Particular emphasis should be placed on describing how AFS flow was accounted for in your original analysis. (Reference to previously submitted information is acceptable if identified as to page number and date.) Any changes in your design which would impact the conclusions of your original analysis should be discussed. We are particularly concerned with design changes that could lead to an underestimation of the containment pressure following a MSLB inside containment.

2. Provide the following information for the reanalyses performed to determine the maximum containment pressure for a spectrum of postulated main steam line breaks for various reactor power levels for the proposed AFS design.

   a. Specify the AFS flow rate that was used in your original containment pressurization analyses. Provide the basis for this assumed flow rate.

   b. Provide the rated flow rate, the run out flow rate, and the pump head capacity curve for your AFS design.

   c. Provide the time span over which it was assumed in your original analysis that AFS was added to the affected steam generator following a MSLB inside containment.

   d. Discuss the design provisions in the AFS used to terminate the AFS flow to the affected steam generator. If operator action is required to perform this function, discuss the information that will be available to the operator to alert him of the need to isolate the auxiliary feedwater to the affected steam generator, the time when this information would become available, and the time it would take the operator to complete this action. Define credit for operator action. If termination of AFS flow is dependent on automatic action, describe the basic operation of the auto-isolation system. Describe the failure modes of the system. Describe any annunciation devices associated with the system.

   e. Provide the single active failure analyses which specifically identifies those safety grade systems and components relied upon to limit the mass and energy release and the containment pressure response. The single failure analysis should include, but not necessarily be limited to: partial loss of containment cooling systems and failure of the AFS isolation valve to close.

   f. For the single active failure case which results in the maximum containment atmosphere pressure, provided a chronology of events. Graphically, show the containment atmosphere pressure as a function of time for at least 30 minutes following the accident. For this

case, assume the AFS flow to the broken loop steam generator to be at the pump run out flow (if a run out control system is not part of the current design) for the entire transient if no automatic isolation to auxiliary feedwater is part of the current design.

g. For the case identified in (f) above, provide the mass and energy release data in tabular form. Discuss and justify the assumptions made regarding the time at which active containment heat removal systems become effective.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. EMERGENCY FEEDWATER					
a. Manual	2 sets of 2 per FDW line	1 set of 2 per FDW line	2 sets of 2 per FDW line	1, 2, 3, 4	A
b. Steam Generator Level-Low	4/SG	2/SG	3/SG	1, 2, 3, 4	B*
c. Feedwater Flow-Low	4/FDW line	2/FDW line	3/FDW line	1, 2, 3, 4	B*
d. Steam Generator Pressure-Low	4/SG	2/SG	3/SG	1, 2, 3, 4	B*
e. Safety Injection	(See Safety Injection initiating functions and requirements)				

*The provisions of Specification 3.0.4 are not applicable.

3.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION statements.

ACTION STATEMENTS

ACTION A - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION B - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. All functional units receiving an input from the tripped channel are also placed in the tripped condition within 1 hour.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. EMERGENCY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Steam Generator Level-Low	≥ _____ %	≥ _____ %
c. Feedwater Flow -Low	≥ _____ gpm	≥ _____ gpm
d. Steam Generator Pressure-Low	≥ _____ psia	≥ _____ psia
e. Safety Injection	(see Safety Injection Setpoints)	

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
Emergency Feedwater System		Not Applicable
2. Steam Generator Pressure-Low		
Emergency Feedwater System	≤	_____*/_____**
3. Steam Generator Level-Low		
Emergency Feedwater System	≤	_____*/_____**
4. Feedwater Flow-Low		
Emergency Feedwater System	≤	_____*/_____**

NOTE: Response time for Motor-driven Emergency Feedwater Pumps on all Safety Injection signal starts ≤ _____

* Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays not included. Offsite power available.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. EMERGENCY FEEDWATER				
a. Manual Initiation	N.A.	N.A.	M(*)	1, 2, 3, 4
b. Steam Generator Level-Low	S	R	M	1, 2, 3, 4
c. Feedwater Flow-Low	S	R	M	1, 2, 3, 4
d. Steam Generator Pressure-Low	S	R	M	1, 2, 3, 4
e. Safety Injection	(See Safety Injection surveillance requirements)			

*Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

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