Omaha Public Power District

1623 HARNEY ■ OMAHA, NEBRASKA 68102 ■ TELEPHONE 536-4000 AREA CODE 402

January 13, 1982
LIC-82-020

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

Enclosure 2 to the Commission's letter dated October 22, 1979 requested the Omaha Public Power District provide information regarding flow requirements for the Fort Calhoun Station's auxiliary feedwater system. The District's response is attached.

Sincerely,

W. C. Jones

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/TLP/RWS:jmm

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae
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RESPONSE TO THE COMMISSION'S OCTOBER 22, 1979 LETTER REGARDING
AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

Introduction

Enclosure 2 to Reference (1) requested documentation of the design bases and criteria for the Fort Calhoun Station auxiliary feedwater system (AFWS) requirements for flow to the steam generators. Reference (2) provided the Commission's safety evaluation and approval of Fort Calhoun's control grade automatic initiation of AFWS. The control grade system utilized a steam generator low level initiation setpoint (using narrow range level instrumentation) and an initiation time delay of 180 seconds or more.

The control grade initiation system was replaced during the 1981 refueling outage with a safety grade system. The safety grade system uses wide range steam generator level instrumentation in conjunction with a steam generator pressure permissive signal to initiate AFWS flow. Figure 1 depicts the safety grade actuation logic and shows that, if a steam generator's pressure is greater than the low pressure setpoint and a low level condition exists, auxiliary feedwater will be provided. A broken steam generator is differentiated from an intact unit by a combination of low pressure and differential pressure setpoints. This differentiation prevents feeding a broken steam generator during a main steamline break event which would result in undesirable overcooling. The safety grade AFWS actuation design description was provided to the Commission in Reference (3).

Events Considered in Establishing Auxiliary Feedwater System Requirements

Each event required to be considered in Enclosure 2 of Reference (1) is addressed below with acceptance criteria and bases, assumptions and technical bases for the assumptions, and results of the analysis or justification is provided for not considering the event.

1. Loss of Main Feedwater (LMFW)

   The acceptance criteria for this event are:

   (1) The peak reactor coolant system (RCS) pressure will not exceed 2750 psia (110% of design), and

   (2) Neither steam generator dries out; i.e., there is no complete loss of primary heat sink. This must not occur because uncovering the tubesheet would create the potential for stress induced cracking when the relatively cold auxiliary feedwater enters the steam generator(s).

   The assumptions and their technical bases in this analysis are:

   (1) Maximum core power is 102%. This value includes a 2% power measurement uncertainty. The LMFW event is initiated from full power because this results in a higher initial core heat flux and decay heat, thus resulting in a quicker steam generator inventory depletion and a higher RCS pressure.

(2) The time delay from a LMFW event start to reactor trip due to a steam generator low level condition consists of a 0.9 second processing delay and a 0.5 second Control Element Assembly (CEA) clutch de-energization delay.

(3) The parameter initiating auxiliary feedwater is the steam generator low level setpoint of the Auxiliary Feedwater Actuation Signal (AFAS) circuitry. Only one of two auxiliary feedwater pumps is assumed to start, resulting in a time delay between AFAS and flow into both steam generators of 49.7 seconds.

(4) The LMFW event is initiated at a normal steam generkdor water level which corresponds to 62.95% narrow range, 84.64% wide range. The corresponding initial mass inventory of each steam generator is 81,656 lbm (at 102% power).

(5) The minimum steam generator level that initiates auxiliary feedwater flow is 30.3% of wide range (39,822 lbm for the assumed steam generator conditions). The average rate of inventory depletion just prior to AFW initiation is 44 lbm/sec. The American Nuclear Society (ANS) decay heat curve was used in a CESEC computer code analysis. After AFAS initiation and the stabilization of decay heat, the average net inventory increase rate is approximately 4 lbm/sec per steam generator, which results in a slow refilling. The minimum analyzed steam generator inventory during the event is 34,000 lbm, which corresponds to a wide range level of 28.4%. Steam slip, because of low steam flow through the Steam Dump and Bypass (SDB) system, causes the steam generator level indication to be higher than if a larger flow were to occur through the safety valves.

(6) The maximum steam generator pressure against which the auxiliary feedwater pump must develop a driving head is 900 psia. The pump capacities at this pressure are 388 gpm and 300 gpm for the electric motor driven and steam driven pumps, respectively. This exceeds the 260 gpm flow assumed in the analysis.

(7) The minimum number of steam generators which must receive AFW flow is 1 of 2 for this event. However, both steam generators are assumed to be fed in the analysis.

(8) Continued operation of the reactor coolant pumps (RCP's) is assumed for this case which results in faster RCS heat removal and subsequently, a faster depletion of secondary inventory. Therefore, a loss of main feedwater with offsite power available is more severe than that with a loss of offsite power. The steam generators will dryout approximately 210 seconds sooner with offsite power available and no auxiliary feedwater flow, than for the case with a loss of offsite power.

(9) The maximum AFW inlet temperature is assumed to be 120°F which is consistent with previous analyses.

(10) The AFW piping between the isolation valves and the steam generator is assumed to be empty. Before AFW can flow into the steam generators through the AFW nozzles, these lines need to be filled. The volume of each line is approximately 32 gallons. It should be noted that AFW flow through the AFW piping, rather than through the main feedwater piping, is the preferential path.

(11) The steam generator blowdown system isolates on a containment isolation actuation signal (CIAS), which does not occur for this event. However, to be conservative, blowdown system heat removal is not included in the analysis model.

(12) The reactor vessel's metal sensible heat is not included in the model because it becomes significant only after several hundred seconds into the event. During the time when minimum inventory occurs, its effect is negligible.

2. <u>Loss of Main Feedwater with a Loss of Offsite AC Power</u>

The acceptance criteria and assumptions used are the same as for the loss of main feedwater event. The only difference between the two events is that offsite power is lost which results in a natural circulation cooldown. As mentioned in Item (8) for the loss of main feedwater event, natural circulation results in a slower RCS cooldown and a slower depletion of the steam generator's inventory. Therefore, this case is less severe than the case with offsite power available.

3. <u>Loss of Main Feedwater with a Loss of Offsite and Onsite AC Power</u>

Reference (4) provided the District's response to the Commission's February 25, 1981 letter which requested all licensees to review current plant operations to determine their capability to mitigate a station blackout event. From an analytical standpoint, this event is essentially the same as the LMFW with loss of offsite power except the only available AFW pump is the steam driven unit. This pump is capable of exceeding the 260 gpm flow requirements, so no further analysis is considered necessary.

4. <u>Plant Cooldown</u>

Operation of the AFW system for plant cooldown is unchanged from the considerations detailed in the Fort Calhoun Station FSAR and will be performed with the system in the manual mode. The system has sufficient storage capacity to provide for the removal of heat produced during 10 seconds of full power operation, and the isothermal condition corresponding to a steam generator pressure of 1000 psia and the maximum decay heat produced during the eight hours following a reactor trip.

5. <u>Turbine Trip With and Without Steam Dump and Bypass</u>

The acceptance criterion for this event is that the AFW system will not actuate for these events; i.e., the safety system should not actuate for this expected transient. Actuation would occur if the steam generator level were to fall below the initiation setpoint. In analyzing this event, a best estimate simulation was run using a wide range level of 62% (8.3% narrow range) as the AFAS setpoint.

The assumptions and their technical bases in these analyses were:

(1) Maximum core power is 100%. The full power case results in the greatest decrease in steam generator level following trip.

(2) The standard trip time delays are assumed - a 0.9 second processing delay and a 0.5 second CEA clutch de-energization delay.

(3) The turbine trip events are initiated from the normal water level which corresponds to 62.95% of narrow range (84.65% of wide range).

(4) The ANS decay heat curve is used for analyzing these events. The rate of steam generator mass depletion is equal to the flow out of the SDB system or secondary safety valves minus the feedwater flow in. These values are positive. The decrease in steam generator level is due to void collapse upon closing the turbine admission valves.

(5) Continued operation of the reactor coolant pumps is assumed.

(6) The steam generator blowdown system is not credited by the analysis code.

AFW flow is not initiated for these cases because the minimum expected steam generator levels are 12% and 17% of the narrow range or 64% and 66% of the wide range for the steam dump and bypass (SDB) system available and unavailable, respectively. The AFAS level equipment setpoint is 8.3% on the narrow range or 62% on the wide range. The code used for these analyses predicts steam generator levels approximately 10-15% (of narrow range) lower than levels actually experienced. Thus, additional margin between the minimum steam generator level and the AFAS setpoint exists.

6. <u>Main Steam Isolation Valve (MSIV) Closure</u>

This event does not pose a challenge to the operation of the AFW system and, therefore, was not analyzed.

7. Main Feedwater Line Break

The acceptance criteria for this event are:

(1) The peak RCS pressure does not exceed 2750 psia (110% of design), and

(2) One steam generator must be maintained as a primary heat sink; i.e., dryout does not occur in the unaffected unit and the tubesheet does not uncover.

The assumptions and their technical bases for this event are:

(1) Maximum core power is 102%. This value includes a 2% power measurement uncertainty. This event is initiated from full power because it results in a higher initial core heat flux and decay heat, thus resulting in a more rapid steam generator inventory depletion and a higher RCS pressure.

(2) The standard trip time delays assumed for this event are a 0.9 second processing delay once trip conditions occur and a 0.5 second CEA clutch de-energization delay.

(3) The parameter responsible for initiating AFW is the steam generator low level AFAS setpoint. Figure 1 shows the initiation logic. The low pressure on the ruptured steam generator prevents it from being fed, so only the intact unit receives AFW.

(4) One AFW pump is assumed to function, so the delay between AFAS and flow into the intact steam generator is 42.3 seconds.

(5) The event is initiated at the normal steam generator water level for 102% power, which corresponds to 62.95% of the narrow range (84.64% of wide range). The initial mass inventory (including steam and liquid) of each steam generator is 81,656 lbm (at 102% power).

(6) The minimum steam generator level assumed to initiate AFW flow is 30.3% of wide range (38,000 lbm under these steam generator conditions). The average rate of inventory depletion just prior to flow into the steam generators is 154 lbm/sec. After AFAS and the stabilization of decay heat, the average net inventory increase rate is 10 lbm/sec which results in a slow refilling. The minimum steam generator inventory during the event is 28,080 lbm which corresponds to a wide range level of 13.2%. The ANS decay heat curve is used in the CESEC code for modeling the decay fission power.

(7) The maximum steam generator pressure against which the auxiliary feedwater pump must develop a driving head is 1000 psia. The pump capacities at this pressure are 315 gpm and 276 gpm for the electric motor driven and steam driven pumps, respectively. These values exceed the 260 gpm flow requirements assumed for the analysis.

(8) Continued operation of the RCP's is assumed for this case which results in more rapid RCS heat removal and subsequently a more rapid depletion of steam generator inventory. The feedwater line break was also analyzed, with a loss of offsite power (RCS natural circulation), and found to be less limiting, since the intact unit dries out 275 seconds later, assuming no AFW flow.

(9) The maximum AFW inlet temperature is 120°F.

(10) For this event, the break is assumed to occur at the feedwater nozzle and there is a concurrent loss of main feedwater to both steam generators. Thus, the feedwater line break is non-isolable. The low steam generator pressure also causes a main steam isolation signal (MSIS). There is a one second delay between the time when the low steam generator pressure condition occurs and the MSIV's begin to close. The closure time of the MSIV's is assumed to be 4.0 seconds, the maximum value allowed by Technical Specifications. The broken steam generator is not fed, because its pressure is below the AFAS low pressure limit, while the intact generator's pressure remains significantly above this value. The time delay to feed the intact unit is 42.3 seconds and consists of a 0.9 second sensor delay, a 0.6 second process delay, a 33.4 second delay for pump start and acceleration and isolation valve opening, and a 7.4 second delay to fill AFW lines. The pump acceleration time is estimated to be 10 seconds which is considerably less than the 33.4 second maximum isolation valve opening time. The 7.4 seconds allowed for AFW line fill is probably not needed because the isolation valves will be partially open and the pumps at full speed for at least 23.4 seconds, so this volume would already be displaced. It is, however, included for conservatism. The same assumptions, conditions, and feedwater line volumes apply for this event as in Item (10) of the loss of main feedwater, except only one AFW line needs to be filled.

(11) The steam generator blowdown system which removes primary system heat is not credited.

(12) Metal sensible heat is not modeled because it becomes significant only after several hundred seconds into the event. During the time of minimum inventory, its effects are negligible.

In summary, the feedwater line break is the most limiting of the loss of primary heat sink events and, consequently, results in the maximum loss of secondary inventory.

8. Main Steamline Break (MSLB)

The acceptance criteria for this event are:

(1) The return to power or return to critical following initiation of the event is bounded by the previous safety analysis for the control grade automatic AFW system, or

(2) The ruptured steam generator is not fed with AFW which will ensure that the requirements of Item (1), above, are fulfilled. The control grade system assumed runout flow from the AFW system and also a 5% main feedwater flow to the ruptured steam generator.

The assumptions and their technical bases in this analysis are:

(1) Maximum core power is 102% of full power which results in a higher initial core average heat flux and a greater amount of decay heat. The 102% power includes a 2% power measurement uncertainty. The zero power MSLB case was also examined but found to be bounded by the full power case.

(2) The standard time delays for reactor trip are used - a 0.9 second processing delay and a 0.5 second CEA clutch de-energization delay.

(3) This event is initiated from the normal steam generator water level which corresponds to 62.95% of the narrow range (84.64% of wide range). The mass inventory, including both steam and liquid, of each steam generator initially is 81,656 lbm (at 102% power).

(4) Continued operation of the RCP's is assumed.

(5) The steam generator blowdown system isolates on CIAS which occurs six seconds after trip. Therefore, its effects are negligible and need not be modeled.

(6) Metal sensible heat is included in the model although it has little effect until several minutes into the event and then does not provide a major contribution to the total heat generation.

AFAS does not occur for this event, because the unaffected steam generator maintains an inventory sufficient to prevent actuation and the ruptured steam generator's pressure falls below the low pressure setpoint prior to reaching the low level condition.

9. Small Break LOCA

This event was not analyzed with respect to the AFW system because primary system heat removal occurs through both the break and the steam generators. The smaller the break, the greater the percentage of total heat that must be removed through the steam generators. The limiting small break LOCA case with respect to operation of the AFW system is a zero-area break, which is bounded by the loss of main feedwater event.

10. Other Events Analyzed

The main feedwater line break event was analyzed with a loss of offsite power and this case found to be less limiting than the case with offsite power available.

The main steam line break event was also examined with a loss of offsite power and found to be less limiting than the case with offsite power available. Similar results were found in Reference (2).

AFW Pump Capabilities

Only one AFW pump was credited for the loss of primary heat sink events. For all events analyzed, sufficient capacity existed to ensure the events' acceptance criteria are met. Normally, additional AFW flow was available, but not credited or needed.

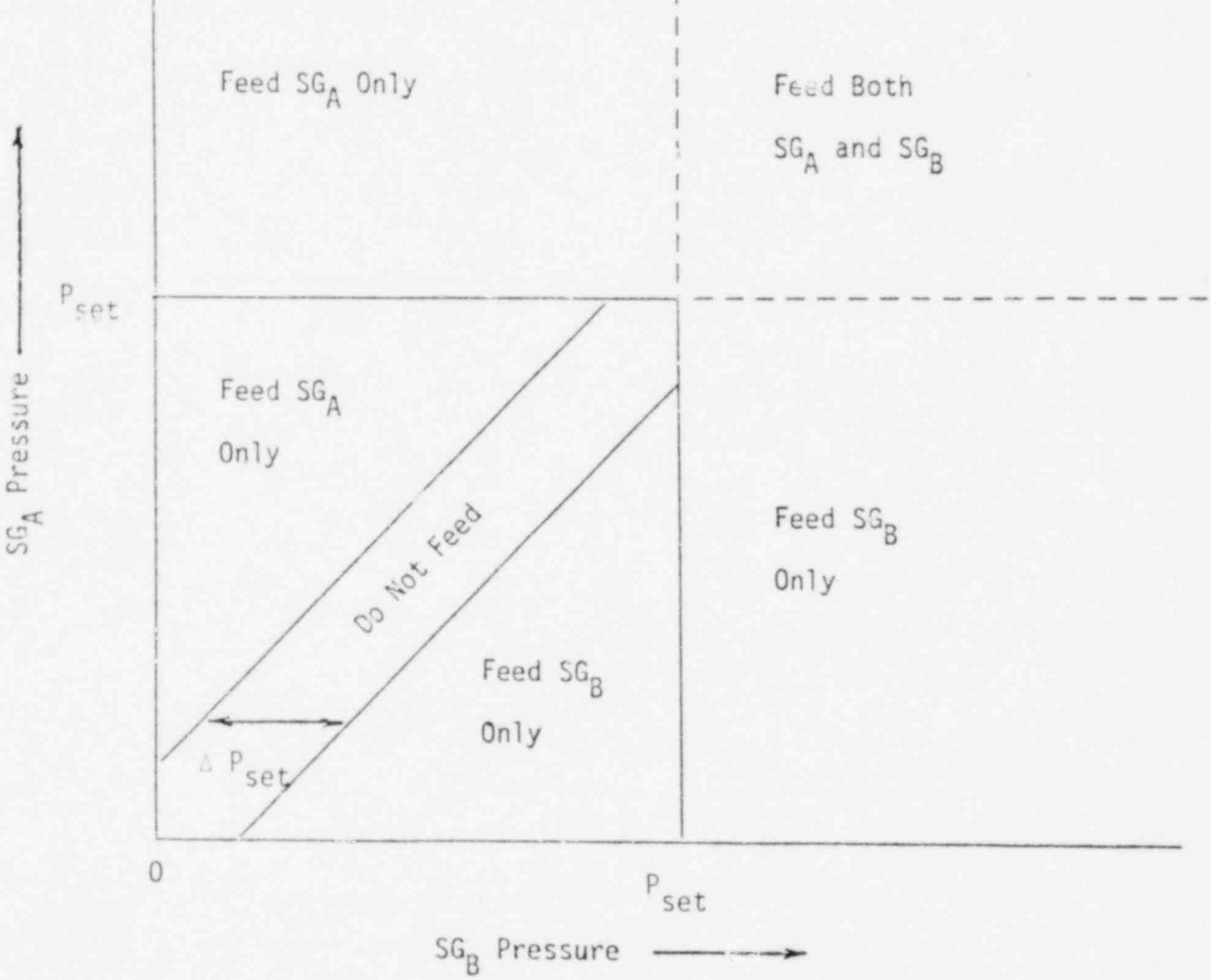
References

(1) "NRC Requirements for Auxiliary Feedwater Systems at Fort Calhoun Station, Unit No. 1", Letter from Darrell G. Eisenhut to W. C. Jones, October 22, 1979.

(2) "Fort Calhoun SER on Automatic Initiation of Auxiliary Feedwater", Letter from Robert A. Clark to W. C. Jones, February 20, 1981.

(3) Letter from W. C. Jones to Darrell G. Eisenhut, December 31, 1980.

(4) Letter from W. C. Jones to Darrell G. Eisenhut, June 5, 1981.



*Actuation only occurs if low SG level and pressure signals are concurrent

Figure 1

AFW Steam Generator Actuation* Logic