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Letter trans the following....... Letter Re. our letter of 11-27-75

ENCLOSURES:
Letter W/Attached responses concerning Re-Evaluation of ECCS Performance.......

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December 31, 1975

Director of Nuclear Reactor Regulation
ATTN: Dennis L. Ziemann, Chief
Operating Reactors Branch #2
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Power & Light Company
Arkansas Nuclear One-Unit 1
Docket No. 50-313
License No. DPR-51
Re-Evaluation of ECCS Performance

Gentlemen:

Attached find responses to those items contained in your letter of November 27, 1975 to Mr. J. D. Phillips concerning the above subject.

As stated in our initial submittal of July 9, 1975, your expeditious review, comments and/or approval is requested.

Very truly yours,

James H. Woodward

J. H. Woodward
Director, Power Production

JHW:ay

Attachment

142

Attachment to Letter
Woodward to Ziemann
December 31, 1975

NRC Question 1

It is the staff's position that Mode 1 should not be attempted as the primary method to control boron concentration in the core during long-term cooling. The possibility of gas or steam entrainment in the decay heat suction nozzle can result in severe damage to the decay heat removal pump. Long-term heat removal requirements can exist for long durations (days or months) after the accident and continuous operation of one train of the decay heat removal system is required. In the event of a equipment malfunction in this train, no method is available to remove the decay heat if the other train has been previously damaged. Therefore, implementation of Mode 1 should not be attempted since this action could result in the decrease of required safety equipment. To verify that gravity draining for boron dilution for Mode 2 is possible, provide the elevations of the piping and other components in the decay heat drop line from the hot leg nozzles through each of the trains to the reactor building sump.

Response

Mode 1, if attempted and successful, would have the following desirable features:

1. If the LOCA break were a cold leg break, Mode 1 would produce enough forced flow through the core to stop boiling in the core.

2. Mode 1 could determine whether the break elevation is high enough to shut off the other LPI string taking suction from the sump. This would stop coolant from spilling out of the break which should enhance reactor building entry, repair and cleanup. Of course, break location may be determined at some point in time by personnel entry into the reactor building.

Mode 1 will not be attempted until conditions (radiation level) in the auxiliary building permit a plant operator to be near the LPI pump to observe the pump for symptoms of cavitation and entrainment of vapor or gas. Also, operational changes (such as shutting down the other LPI pump) will not be attempted unless an operator can be near the LPI pump to observe the pump for signs of cavitation.

Mode 1 is a desirable method of operation and will only be considered the primary method of controlling boron concentration when the above conditions for operation are met.

In regard to the last sentence of NRC Question 1, the following elevations apply to ANO-1:

Elevation of Decay Heat Letdown Nozzle in Hot Leg -- 365'9"
Elevation of containment penetration of D. H. Letdown Line -- 362'0"
Elevation of suction piping on D. H. pumps -- 323'
Elevation of motor-operated valves in R. B. sump -- 333'

At no point in the piping system does the elevation of any component exceed the elevation of the decay heat letdown nozzle in the hot leg.

NRC Question 2

With regard to the single failure analysis, the analyses did not consider the failure of any one of the four valves in the decay heat drop line and the resulting effect to ECCS performance. Expand your single failure analyses to include spurious signals and the resulting consequences for all EMO valves in the ECCS. Confirm that post-LOCA long-term cooling requirements were considered (i.e., systems needed to limit boric acid concentration in the reactor vessel).

Response

The analysis concerning single failure criteria for manually-controlled electrically-operated valves did consider the failure of each of the three EMO valves in the decay heat drop line (there are only three valves in this line rather than four as stated in the question). All EMO valves in the ECCS were analyzed; however, valves with "ES" signals were not included in our transmittal as they are evaluated in the single failure analysis in Section 6 of the SAR.

The tabulation below addresses valves in systems needed to limit boron concentration in the reactor vessel.

Valve Description	Failure	Evaluation
DH drop line valves (CV1050, CV1410, or CV1404)	Closed	Would prevent Modes 1 & 2, Mode 3 unaffected.
Emergency sump outlet valve (CV1405 or CV1406)	Closed	Would prevent Mode 2, Modes 1 & 3 unaffected.
LPI flow control valve (CV1428 or CV1429)	Closed	Used for flow control in Mode 1, can use cooler bypass valve (CV1433 or CV1432) instead.
	Open	Would prevent Mode 1. For Modes 2 & 3, should be closed; CV1401 or CV1400 can be closed, which accomplishes intended function.
LPI cooler bypass valve (CV1433 or CV1432)	Open	Would prevent Mode 1. For Mode 2 should be closed; CV1401 or CV1400 can be closed, which accomplishes intended function.

LPI injection valve (CV1401 or CV1400)	Closed	Would prevent Mode 1. Modes 2 & 3 are not affected.
	Open	For Modes 2 & 3, should be closed. However, CV1428 or CV1429 can be closed, which accomplishes intended function.
Pressurizer main spray line block valve (CV1009)	Open	No effect. For Mode 3, CV1008 is closed, which accomplishes intended function.
Pressurizer main spray line control valve (CV1008)	Open	No effect. For Mode 3, CV1009 is closed, which accomplishes intended function.

NRC Question 3

For a core flooding tank (CFT) line break and an inadvertent closure of a valve in the unaffected low pressure injection line, the LPI-to-LPI crossover would be rendered ineffective. Station Technical Specifications must require that power be disconnected and breakers locked open to LPI motor-operated valves downstream of the LPI-to-LPI crossover (valves normally open) and that a periodic test be performed to warn of abnormal leakage of the check valves in the LPI injection lines inside the containment. These changes provide further assurance that abundant core cooling is available for a CFT line break and minimize the potential for a LOCA outside containment. Discuss the above concerns and submit the required Technical Specification change if necessary.

Response

As shown in Figure 6-1 of the Arkansas Nuclear One-Unit 1 FSAR and drawing M-230, Rev. 13 (transmitted to you by our letter of September 4, 1975), there are no motor-operated or power-operated valves located downstream of the LPI-to-LPI crossover at ANO-1. Also, the LPI-to-LPI crossover is located inside the reactor building. Therefore, this question is not applicable to ANO-1.

NRC Question 4

It is noted that motor-operated valves CV1407 and CV1408 on drawing M-232, Rev. 11, from the BWST are shown normally closed. It appears that, assuming sufficient static head were available, the potential for a water hammer when ECC is injected into a dry line would be reduced considerably if these valves were normally open. Discuss this concern and indicate your position in this matter.

Response

If CV1407 and CV1408 were normally open then a flow path would be opened from the BWST to the suction of makeup pumps P36A and C, respectively. We consider this to be an undesirable situation. Also, with these valves open, chere would be only one isolation valve between the BWST and the Reactor Building sump, another undesirable situation. It is our position that these lines will remain full of water and ensure this through periodic pump (HPI, LPI, and RBS) testing recirculating to the BWST, periodic valve cycling, and normal decay heat removal operations.

NRC Question 5

The operating methods to control boric acid concentration in the core during long-term cooling require operator action in areas where radiation dose levels may be excessive. Since operator actions may be delayed due to inaccessibility to manual valves, provide analyses of the boron concentration in the core as a function of time assuming the maximum possible time delay to implement the required systems.

Response

As stated in our April 21, 1975 letter to you we feel that the answer provided to Question 120 in BAW-10091, Rev. 1 adequately responds to the concern of analyses performed. Figure 10-4 of BAW-10103, Rev. 1 shows the boron concentration versus time for operator inaction and action situations.

Valve lineups may be accomplished for Mode 3 with no excessive radiation dose being received by the operators. The valves which will be necessary to be aligned are outside of containment in the piping penetration room at Elevation 360'.

Valve lineups for Modes 1 and 2 will be accomplished in the Decay Heat Pump Room at Elevation 317'. Dose rate calculations have been performed based on the assumptions stated in FSAR Section 14.2.2.5.6. The source term used for the 14" 10S decay heat removal pump pipe (the primary source from which an operator would receive radiation while operating valves in the decay heat pump room following LOCA) is the iodine fission product inventory given in Table 14-49 of the FSAR. The source term was diluted by the volume of the primary system and the refueling water storage tank. Assuming, conservatively, that an operator would be in the room for 15 minutes (no more than 10 minutes is actually expected) and that the surface dose rate is that which he will receive (valve stem length will ensure that he is not at the surface), at 24 hours following LOCA the dose which would be received is 27 Rem. We feel that this is not excessive in that it will be a one time dose, and the operator would perform no other functions in the plant following this action. To further ensure that the doses received in this area is not excessive, more than one operator could perform the operation, i.e. one person at a time would be in the room for a short duration and would be replaced by another operator, etc. to complete the operation of valve lineup. Also, we feel that this dose estimate is conservative and will not actually be received based on the assumptions stated above.

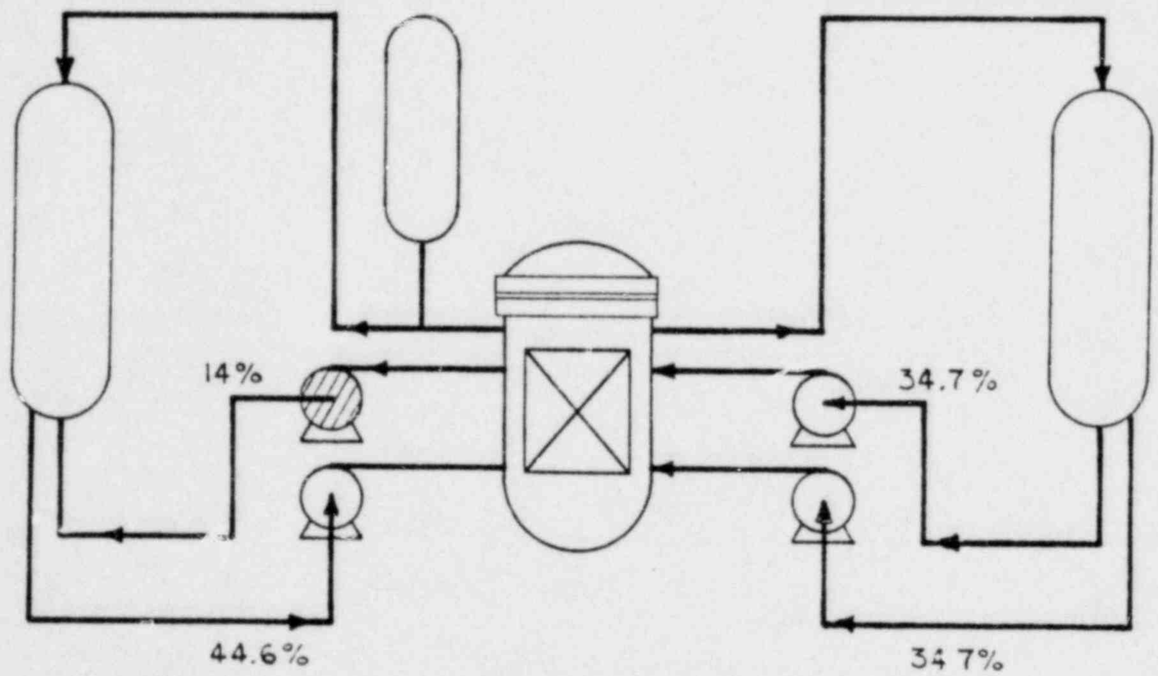
NRC Question 6a

With regard to partial loop analysis:

> Discuss the consequences of a break in an active cold leg of the fully active loop.

Response

During steady state, 3-pump operation, the pump in the active cold leg-inactive loop supplies 44.6% of system flow compared to the pump in the active cold leg-active loop which supplies 34% of system flow. Figure 1 shows the Reactor Coolant System flow for 3-pump operation. Therefore, placing the break at the discharge of the pump in the active cold leg-inactive loop instead of the active cold leg of the fully active loop, would yield the most degraded positive flow through the core during the first half of the blowdown. This is due to the loss of the largest source of positive flow and this would result in higher cladding temperature. Thus, analyzing this break location will ensure that the most conservative assumptions affecting core flow have been considered.



. FIGURE 1 Reactor Coolant System Flow for 3-pumps operating (percent of total system flow).

NRC Question 6b

Two pump operation will not be permitted unless an analysis is provided to support this mode of operation. Compare a break in the inactive cold leg to a break in the active cold leg.

Response

Based on the arguments presented below, conformance to the ECCS acceptance criteria given in 10CFR50.46 is assured for 2-pump operation. However, since no analysis has been performed, Technical Specifications for 2-pump operation will be revised to allow operation in the mode for a period not to exceed 5 days.

With 2-pumps operating, one in each loop, the maximum power level will be 51% of full power which includes 2% for uncertainty. The system flow rate is reduced to 50% of normal 4-pump operation at steady state conditions. The idle pumps in each loop are locked in position because flow is reversed in each of the inactive cold legs. About 18.8% of the RC flow from the downcomer plenum is directed back in each inactive cold leg. If the flow reverses to the positive direction during the transient the idle pump would act as a free spinning roter with no power.

The core flow for a break in an inactive leg with 2-pumps operating should be similar to a break in the active cold leg-inactive loop with 3-pumps operating. During the LOCA transient, the positive driving force for both breaks is with 2-pumps and therefore the core flow should be approximately the same. The reflooding rates for the 2-pump case should be greater than the 3-pump case because the core power is lower, 51% versus 77% of full power rating, thus a lower cladding temperature rise after the End of Blowdown (EOB) would be expected for 2-pumps operating.

A break at the pump discharge of either one of the active cold legs will cause a loss in positive flow during the first half of the transient compared to the above case. The transition from positive to negative flow should occur earlier. The negative flow would be substantially increased due to the decrease from 2 to 1 active pumps trying to force positive flow into the core region. The high negative flow rate through the core during the blowdown phase should provide good core cooling and remove a significant amount of stored energy in the fuel. Thus the cladding temperature during this phase of the LOCA should be maintained at a relatively low temperature. The reflooding phase should have the same improvement in clad temperature as described for the previous 2-pump case, i.e., a lower cladding temperature rise after the EOB would be expected.

Therefore, the maximum cladding temperature for 2-pump operation should be approximately equal to or less than that calculated for 3-pump operation. Since the calculated peak cladding temperature for a LOCA that occurs during 3-pump operation gives a large margin (434F) relative to the 2200F limit, 2-pump operation will easily comply to the acceptance criteria for the ECCS set forth in 10CFR50.46.

NRC Question 6c

Indicate and justify the worst-case pump status assumed at the time of the LOCA (tripped vs. powered).

Response

The partial loop analysis was performed assuming the worst case break (8.55 ft^2 DE, C_D = 1.0, pumps powered) reported in BAW-10103. Based on the results given in section 5.5 of BAW-10103, it was found that the pumps-powered case produced the highest peak cladding temperature (2114F vs. 2080F). The difference of 34F indicates that the LOCA analysis is relatively insensitive to assumptions regarding electrical power availability to the pump.

The effect of whether pumps tripped or pumps powered is the worst case has been essentially analyzed by the two cases that were analyzed in the partial loop analysis. Core flow for a break in the inactive loop-active cold leg and inactive loop-inactive cold leg is similar to the core flow with pumps tripped and powered, respectively, as can be seen by comparing Figures 2 and 4 of the partial loop analysis to Figure 5-7 of BAW-10103. Comparing results for the two partial loop cases illustrates that the results would be insensitive to a change in pump status. Since the maximum cladding temperature calculated for the partial loop analysis is only 1766F, which is 313F less than the same break at full power and flow conditions, the pump status would not adversely affect the results.

NRC Question 6d

Provide assurance that the PCT versus break size curve in BAW-10103 would not be significantly altered by either mode of partial loop operation.

Response

The partial loop analysis was performed assuming the worst case break (8.55 ft^2 DE, C_D = 1) reported in BAW-10103 at the maximum Kw/ft limits shown in Figure 2-2. Historically, the above break has resulted in the highest cladding temperature for LOCA analysis. In general as the break size decreases, the duration of the blowdown increases which results in decreased maximum cladding temperature. Table 6-1 of BAW-10103, Summary of Break Spectrum, verifies this statement, i.e., the maximum cladding temperature decreased 195F when the discharge coefficient for a 8.55 ft^2 DE break was changed from 1.0 to 0.6.

As mentioned in response to Question 6.c, the core flow for the partial loop cases are similar to those shown in BAW-10103. Therefore, core flow for smaller breaks during partial loop operation would be similar to that shown in Section 6 of BAW-10103. With similar flows, the peak clad temperature (PCT) versus break size curve should exhibit the same trend, i.e., decreasing PCT with break size. Since the PCT for the partial loop analysis is 313F less than that given for the worst break in BAW-10103, smaller breaks will exhibit large margins of safety relative to the 2200F criteria.