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May 27, 1980

1-050-18
2-050-21

Mr. K. V. Seyfrit, Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Comm.
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Subject: Arkansas Nuclear One - Units 1 & 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6
IE Bulletin 80-04
(File: 1510.1 and 2-1510.1)

Gentlemen:

The following information is provided in response to IE Bulletin 80-04. For each question you will find a separate response for ANO-1 and ANO-2.

QUESTION 1

Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

RESPONSE ANO-1

As a result of a main steam line break inside the reactor building of ANO-1, the steam pressure in both Once Through Steam Generators (OTSG) would decrease quite rapidly. The rate of depressurization would of course depend upon the break size. For a steam line break accident, the reactor power would increase with the decreasing average reactor coolant temperature as a result of a negative moderator coefficient. The ICS will cause insertion of control rods in an attempt to limit the reactor power to 102 percent. If the break were large, the reactor power increase could not be limited sufficiently by the ICS and a reactor trip would occur due to high neutron flux and/or low reactor coolant pressure.

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Following the reactor trip, the turbine will trip and the ICS will run back the feedwater flow. Due to the low OTSG pressure in the affected OTSG, the safety grade Steam Line Break Instrumentation and Control System (SLBIC) would actuate, isolating the affected OTSG by closing the respective feedwater isolation valve and both main steam block valves. A SLBIC signal also opens the steam supply to the turbine driven emergency feedwater pump. As the affected OTSG boils dry, the emergency feedwater actuation and control system will actuate the emergency feedwater system when it receives a OTSG level of less than 18 inches in either generator. This signal will actuate the motor driven emergency feedwater pump (the turbine driven pump has already been actuated by SLBIC) and align the emergency feedwater valves in both trains.

Upon realizing he has a steam line break accident, the operator, using Emergency Operating Procedure 1202.24, will determine the affected OTSG by observing the OTSG levels and pressures. Upon identifying the affected OTSG, the operator will close the affected OTSG's emergency feedwater system steam supply and feed valves, and open, if not presently open, the corresponding steam supply valve on the unaffected OTSG. The operator would then commence cooldown to cold shutdown utilizing the unaffected OTSG.

If for some unlikely reason the operator fails to isolate the emergency feedwater to the affected OTSG, it has been shown through analysis using the assumptions in Attachment A that the reactor building pressure would not reach the design pressure of 59 psig until approximately 3 hours and 45 minutes into the accident allowing more than sufficient time for the operators to take corrective action.

The ability of the emergency feedwater pumps to remain operable after extended operation at runout flow is at this time still being investigated. The findings of this study will be forwarded to you as soon as they are available. We expect this information to be in our possession within the next 30 days.

RESPONSE ANO-2

Following a main steam line break on ANO-2, the Engineered Safety Features Actuation System (ESFAS) will initiate a Main Steam Isolation Signal (MSIS). The MSIS, as described in the ANO-2 FSAR Section 10.3 and 7.3.1, closes both main steam block valves and both main feedwater isolation valves, stops both main feedwater pumps and stops all four condensate pumps. Upon receiving an Emergency Feedwater Actuation Signal (EFAS), as described in the ANO-2 FSAR Section 7.3.1, the emergency feedwater system will actuate. The safety grade EFAS will start both emergency feedwater pumps, determine which steam generator(s) are intact and open the appropriate emergency feedwater valves to the intact steam generator(s) (the emergency feedwater valves in this instance include the steam supply valves for the turbine driven steam water pump). The EFAS will then prevent a high level condition in the intact steam generator(s) by closing the emergency feedwater valves when the water level is reestablished above the low level trip set point.

Once the operator has verified proper MSIS and EFAS actuation by following ANO-2 Emergency Operating Procedure 2202.24, he will place the unit in cold shutdown utilizing the unaffected steam generator.

As indicated above, the emergency feedwater system's logic automatically isolates the affected steam generator even assuming a single failure. Should one of the emergency feedwater pump's discharge valves stick open in the line to the low pressure steam generator, the other valve in the series will remain closed as it is actuated by a different ESF bus. This redundancy will protect both emergency feedwater pumps from a runout flow condition under a postulated single failure.

QUESTION 2

Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

RESPONSE ANO-1

The steam line break accident has been analyzed in the Unit 1 FSAR in Section 14.2.2 considering no operator action. In this analysis, the affected OTSG is assumed to blow dry after the rupture at which time the minimum level control opens feedwater valves such that the OTSG maintains low-level. Assuming a minimum tripped rod worth with the maximum rod stuck out, the reactor will return to a maximum neutron power level of 2.6% at 44.5 seconds and return to subscriticality at 47.5 seconds. With the low level control valves maintaining a 30-inch minimum downcomer level in the affected OTSG, the average coolant temperature will remain below 475 degrees F until feedwater isolation on the affected OTSG is achieved.

RESPONSE ANO-2

Unit 2 has a safety grade feed only good generator system designed in accordance with BTP 10.1. Therefore, the affected steam generator would be isolated and there would be no additional water sources contributing to RCS cooling by blowdown.

QUESTION 3

If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed.

RESPONSE ANO-1

Although the potential exists on ANO-1 for reactor building overpressurization, this event will not take place until 3 hours and 45 minutes into the steam line break accident. It is our position that there is more than sufficient time for the operator to isolate the affected OTSG and terminate the event. Thus no corrective action is proposed.

RESPONSE ANO-2

The safety grade, redundant and single failure proof EFS is designed to preclude such an event. Therefore, no corrective action is proposed.

Very truly yours,

David C. Trimble

David C. Trimble
Manager, Licensing

DCT:DEJ:skm

Attachment

cc: Mr. Victor Stello, Jr., Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Comm.
Washington, D. C. 20555

ATTACHMENT A

The following conservative assumptions were used for the reactor building overpressurization analysis:

1. No operator action.

2. Two reactor building air coolers operable.

3. One 1000 GPM EFW pump feeding the affected steam generator.

4. EFW flow boiled in steam generator and dispersed into the reactor building through the break adding mass energy at h = 1200 BTU/LBM.

5. One reactor building spray train with an initiation setpoint of 44.7 psia.

6. FSAR Table 14-19 "Steam Line Failure Parameters" and "Mass and Energy Releases for Building Pressure Analysis".

7. No sump heat exchanger for spray recirculation.

8. Heat sink data as stated in FSAR Tables 14-41 and 14-42.

9. No safety injection into containment.