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U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Preliminary Accident Sequence Precursor Analysis

Gentlemen:

Arkansas Nuclear One Unit 2 submitted Licensee Event Report (LER) No. 50-368/95-001-00 to report discovery of the potential for a common mode failure of Emergency Feed Water (EFW) trains due to a failure of one DC electrical bus. The Nuclear Regulatory Commission (NRC) provided a preliminary Accident Sequence Precursor analysis of the condition described in the LER in a letter dated August 2, 1996. The following comments are provided concerning that analysis.

The "Event Description" is accurate in that it reflects the results of the simulator run. The LER stated that there is no conclusive evidence that actual plant response to the condition would have resulted in a generator coast down of sufficient duration to allow green train valves to close completely and block all EFW flow. Subsequent investigation has failed to establish a duration of valve motion. A detailed analysis of the voltage decay has not been performed due to the cost. If the EFW performance had been able to exceed the minimum requirement to preclude core uncovery, the event would not have proceeded to core damage via the event sequence originally postulated, and this condition would result in no net change in Core Damage Frequency (CDF). To preclude this from being the case, the valve would have had to travel at its normal speed (which would require its normal voltage) for 16 seconds after the generator tripped.

While flow blockage due to valve closure is uncertain, potential operator recoveries were examined in order to provide a complete evaluation of the significance of this condition. Two operator recovery actions were identified that would each be successful in restoring EFW flow to the steam generators. These recoveries would have been attempted in parallel to increase the EFW flow, and either would have been adequate if successful. Therefore, in order to have core damage, both recoveries would have to fail.

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These recoveries are:

1. Restore power to electrical buses 2A2/2A4 by manually aligning offsite power to 2A2. Reset Main Steam Isolation Signal (MSIS). Open Emergency Feed Water (EFW) discharge valve(s) from the control room.

2. Open EFW discharge valve(s) locally using the handwheels(s).

Only one valve must be opened because heat removal by one steam generator is adequate. The local manipulations for both recoveries are provided with specific lighting that is battery powered and is, therefore, unaffected by the loss of power situation. In addition, adequate power is available through 2A1 such that adequate lighting is available to permit ingress to the local position without impediment. Both recoveries are proceduralized in procedures 2202.001, "Standard Post Trip Actions," and 2203.037, "Loss of 125V DC," with the specific details of the MSIS reset in procedure 2202.010, "Standard Attachments," Attachment 14 "MSIS Reset." All these actions are a routine part of the training received by operators in completing their qualification cards.

Recovery #1 is partially accomplished in the control room and partially in a location that requires entry through a security door. Recovery #2 is accomplished in a location that requires entry into the radiologically controlled access area.

Recovery #1 requires manually opening one breaker and manually closing one breaker outside the control room and electrically opening one valve after resetting MSIS in the control room. In the calculation "ANO-1&2 Common Plant-Specific Data for PRA Use," no values for breakers failing to operate in the manual mode are provided, but the failure to operate in the remote electrical mode value of 6.45E-4 will be conservatively assumed for purposes of this evaluation. With two breakers to manipulate, the probability of one failing to operate is 1.29E-3. Adding to this the probability of the motor-operated valve (MOV) failing to operate of 9.23E-3 leads to a total equipment failure probability for recovery #1 of 1.05E-2. [The components involved in resetting MSIS are relay contacts, signal processors, bistables, transmitters, logic circuits and pushbutton contacts, at least two of which in separate channels must fail to prevent the reset. The individual failure probability for each of these components is in the range of 1E-6 or lower (based on the ANO IPE Generic Data Notebook). Even if two of these components in separate channels fail, one channel can be quickly bypassed and the reset successfully accomplished. Therefore, the equipment failure probability for this portion of the recovery is negligible.] Recovery #2 requires manually opening one valve. The failure to operate probability for the manual valve manipulation in recovery #2 is 7.93E-4 based on the calculation "ANO 1&2 Common Plant Specific Data for PRA Use."

The portion of recovery #1 requiring action outside the control room has been determined in the ANO-2 Human Reliability Analysis Work Package (HRAWP) to take 5 minutes and the default value of 4 minutes for the control room portion of the recovery will be conservatively used in

series with the portion of the time requirement for actions outside the control room. The time required to accomplish recovery #2 has been determined in the HRAWP to be 10 minutes.

ANO-2 analysis has determined that core uncovery would not begin for at least 40 minutes following steam generator dryout. Values established in ANO-2 analyses indicate that 38 minutes would elapse from the time of reactor trip to the time of steam generator dryout for this scenario with no EFW flow and four Reactor Coolant Pumps (RCPs) running (based on 4.5 Mw_{th} into the Reactor Coolant System from each RCP). Therefore, if the EFW valve receives adequate power to completely close, 78 minutes are available to accomplish the recovery. If the valve does not receive adequate power to close, the additional EFW flow that occurs during the post trip time frame will significantly lengthen the available recovery time even if there is not enough flow to prevent core uncovery without recovery action.

The following input parameters are, therefore, used in the determination of the failure probability of each recovery:

RECOVERY #1

Parameter	**Value**
Mean response time (min)	9
Additions to response time	0
Model error factor	4.3905 (default)
Adjustments to error factor	1 (security door entrance)
Model uncertainty error factor	1.68 (default)
Available time (min)	78
Failure probability of equipment used	1.05E-2

RECOVERY #2

Parameter	**Value**
Mean response time (min)	10
Additions to response time	0
Model error factor	4.3905 (default)
Adjustments to error factor	1 (radiologically controlled access entrance)
Model uncertainty error factor	1.68 (default)
Available time (min)	78
Failure probability of equipment used	7.93E-4

Using the Human Recovery Action numerical models developed in the Individual Plant Evaluation (IPE) model with these input parameters the failure probabilities for recovery are:

Recovery	**Failure Probability**
#1 with 78 min. available time	4.24E-2
#2 with 78 min. available time	3.98E-2
Both #1 & #2 with 78 min. available time	1.69E-3

For these recoveries, a combined failure probability of 1.69E-03 was determined. Since the failure of electrical bus 2D01 was already modeled in the IPE with the exception of this postulated EFW failure, the change in CDF due to the loss of 2D02 initiator (T11) is estimated to be essentially the T11 frequency times the operator failure to recover EFW Train B or 6.66E-07/rx-yr. This is a small increase in the ANO-2 CDF from its estimated value of 3.29E-5/rx-yr, as reported in the ANO-2 IPE/PRA. Note that none of these evaluations, either the original IPE or this re-evaluation, account for the availability of the Station Black Out diesel generator or the Auxiliary Feed Water train which were installed after the IPE freeze date. The availability of these systems for use in recovering from the T11 initiator could even further reduce the contribution of this new failure mode to CDF.

Considering the additional information presented above that is a result of a more detailed evaluation the section of the NRC letter concerning "Modeling Assumptions" should be reconsidered. The documents discussed above are available for review if desired.

Very truly yours,

Mark A. Smith

for Dwight C. Mims
Director, Nuclear Safety

DCM/tfs

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