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MILLER,C.L. Project Directorate I-2

SUBJECT: Responds to NRC 930610 request for comments on precursor anslyses,w/respect to LER 388/92-001 re three of five EDGs being unavailable for 11 h.

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JUN 30 1993

Director of Nuclear Reactor Regulation
Attention: Mr. C. L. Miller, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
REPLY TO REQUEST FOR COMMENTS ON
PRECURSOR ANALYSES
PLA-3994 FILE R41-2

Ref: NRC Letter from R.J. Clark to R.G. Byram, "Request for Comments on Precursor Analyses, Susquehanna Steam Electric Station, Unit 2, LER 388/92-001," dated June 10, 1993

Dear Mr. Miller:

PP&L has reviewed the preliminary precursor analysis of the Susquehanna Steam Electric Station (SSES) March 18, 1992 SCRAM forwarded by NRC under the referenced letter. We offer the following background summary and specific comments for NRC consideration regarding the inclusion of the SSES SCRAM event in the 1992 Accident Sequence Precursor (ASP) report.

BACKGROUND

The durations for diesel generator (D/G) unavailability cited in the NRC letter appear to be inaccurate. D/G "B" would have been immediately available had we manually switched to the intact generator field rectifier bridge 2 diode after bridge 1 diode failed. We elected to repair D/G "B" instead. Therefore, no D/G was available to cover the "B" bus from 0831 on March 18, 1992, when D/G "B" tripped, until 0750 on March 19, 1992 when D/G "E" was placed in service. (data from Unit Log - PCO Log). If it had become necessary, either D/G "B" or D/G "E" could have been made available for the B bus within 2 hours. The 72 hours referenced in the NRC letter refers only to the LCO time as stated in Technical Specifications. D/G "C" was never unavailable. D/G "C" was running in the emergency mode, ready and fully capable for use. Channel 2C bus only was unavailable. The 2C bus could have been energized immediately, however the decision was to proceed slowly and purposefully on the cause of the problem and bus/relay integrity.

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NRC estimates the conditional probability of core damage given the equipment failure combination that occurred on March 18, 1992 to be in the range of 3.6×10^{-6} to 1.7×10^{-7} per reactor year. The range in core damage frequency is a result of the values used to estimate the probability of AC power recovery used by the analysts who prepared the analysis. Based upon our understanding of the data and model used to analyze the SCRAM of 3/18/92, we believe that the conditional probability of core damage for SSES for this event falls orders of magnitude below the threshold value of 1.0×10^{-6}. In the report, the NRC is emphatic that "*the conditional probabilities determined for each precursor cannot be directly associated with the probability of severe core damage resulting from the specific event at the specific reactor plant at which it occurred.*" However, PP&L is sensitive to misinterpretations or misapplications of the "high" values of calculated core damage frequency ascribed to an event at SSES.

SPECIFIC COMMENTS

1. The title of the event, "Three of Five EDGs Unavailable for Eleven Hours", is not indicative of what occurred and does not represent SSES emergency D/G power requirements. Functionally, only one diesel failed. The D/G "C" was running and could have been loaded onto the bus after resetting the lockout. In addition, the spare D/G "E" was always available for tie in which requires less than 2 hours. D/G "B" was always available by manually switching which requires less than 2 hours. The SSES design is 3 of 4 emergency D/Gs with a spare that can be manually substituted for any of the 4 emergency D/Gs.

2. This event was modeled as a transient initiator, 'Loss of an AC bus' in the Susquehanna IPE. The AC bus loss precipitates a Main Steam Isolation Valve (MSIV) closure through the loss of containment instrument gas. The NRC instead chose to treat it as a Loss of Offsite Power (LOOP) probably due to the loss of D/G "B" and the C 4160V AC bus. (Note that MSIVs will also close on LOOP)

3. Given the NRC modeling of the event, we were unable to reproduce the calculated conditional core damage probabilities. When applying an AC non-recovery factor of 0.8 we calculate a conditional core damage probability of 1.8×10^{-6} instead of 3.8×10^{-6}. If we presume the LOOP exposure is about 2 to 3 hours, the NRC frequency can be reproduced. However, the actual event included a reactor trip so the choice of 2 to 3 hours is arbitrary on our part and does not reflect the risk reduction due to SCRAM and reduced decay heat at 3 hours.

4. The LOOP frequency used by the NRC is 1.6×10^{-5}/hr (0.14/year) compared with a Susquehanna specific value of 6.4×10^{-6}/hr (0.057/year). The NRC value is on the high side for Susquehanna which has not experienced a LOOP during 11 site years of operation, but probably reasonable when considering the entire reactor population.

5. The NRC presumes three D/Gs failed when in fact only one D/G failed. D/G "C" started and ran, but was not loaded onto the bus due to the bus lockout. The D/G "E" was in stand-by and was started and loaded onto the B ESS bus. This affects the treatment of subsequent diesel failures. With only D/G "B" diesel failed, the probability of failing the remaining diesels, A, D & E, using the emergency power (EP) branch model, should be;

$P_{EP} = 0.057 \times 0.190 \times 0.500 = 0.0054$

instead of;

$P_{EP} = 0.190 \times 0.500 = 0.095$

which was used in the precursor analysis. This adjustment results in the conditional core damage probability of 3.8×10^{-6} becoming 2.2×10^{-7}. This value is below the precursor cut off probability of 1.0×10^{-6}.

PP&L has included D/G "E" in the evaluation of onsite AC power recovery above because since this diesel can be connected into any of the four 4 kV busses. Without consideration of D/G "E" we recommend that a value of 0.01 (0.057 x 0.190), be used for the failure probability of "EP" when using the NRC diesel failure model. However, statistical analysis of Susquehanna diesel failure data indicates that multiple diesel failures occur at a rate consistent with what would be expected as the result of independent failures. Therefore, using Susquehanna specific data, a value of 0.0025 $(0.05)^2$ is recommended for the failure probability of EP.

6. It appears from the information provided, that long term station blackout is the dominant contributor to core damage for this event. Core damage occurs in this event because AC independent safety systems become unavailable when DC control power is lost due to battery discharge. Core damage is prevented by AC power recovery. It is presumed from the information in NUREG/CR-4674 that battery depletion occurs between 2 and 4 hours following station blackout. Therefore, AC power must be recovered within this time frame to avoid core damage. The precursor analysis accounts for AC power recovery. The NRC used three values for failure to recover AC power; 0.8, 0.34 and 0.04 with the 0.04 being considered the best choice for calculating core damage frequency. These values were compared with those representative of Susquehanna.

AC power recovery data from the Susquehanna IPE was obtained. Susquehanna is committed to cope with station blackout for at least 4 hours. Battery discharge calculations demonstrate that the limiting time is actually 6 hours. Susquehanna also has a standby diesel that can be tied into any of the four 4 kV busses. Additionally, we have placed a 100 kw diesel generator at

Susquehanna to provide AC power to the battery chargers. This diesel has a 24 hour fuel supply that can be replenished from the 1E diesel fuel source. Therefore, if this diesel is operable, DC control power will not be limiting at Susquehanna. Using this information, an AC power recovery table has been constructed for Susquehanna as shown below. This data was derived from NUREG-1032 and Pennsylvania New Jersey Maryland (PJM) power pool data for offsite power recovery, Susquehanna diesel maintenance records through 1989 for diesel repair and D/G "E"logs through 1989 for D/G "E" availability and tie-in time.

Susquehanna Specific non-Recovery Data

Recovery Time	Offsite Power	Offsite Power & Diesel Generator Repair	Offsite Power, Diesel Generator Repair & D/G "E" Tie-in.
3 hour battery discharge and 1 hour RPV water level boildown	0.087	0.046	0.015
4 hour battery depletion and 2 hour RPV water Level boil down	0.060	0.022	0.007
Credit for charger diesel, core damage postulated at 24 hours	0.0056	0.0012	0.0004

If D/G "E" diesel is included in the Event Tree Top Event "EP", then the third column should be used for assessing AC power recovery. If the D/G "E" is the Event Tree Top Event "LOOP REC (LONG)", then the forth column should be used for AC power recovery. The NRC recovery value of 0.04 seems to correspond to a 3 hour coping time. This implies that no credit was given for station blackout rule compliance or station blackout enhancements such as the charger diesel. We recommend that the NRC use a value of 0.0004 in conjunction with the EP failure probability given in comment 5 when assessing the probability of AC power recovery. This value accounts for those improvements made to comply with the station blackout rule and other plant enhancements installed by PP&L to reduce the risk

from station blackout.

7. The event tree model gives no credit for RCIC operation given a stuck open relief valve. PP&L calculations performed with the ORNL BWRSAR code demonstrate that the RCIC system will remain operable with a stuck open relief valve. The HPCI system on the other hand will trip on low steam pressure.

We hope this information will prove useful in helping NRC to decide whether to include the SSES March 18, 1992 SCRAM event in the 1992 ASP report.

Please contact Mr. J.B. Wesner at (215) 774-7911 with any questions you may have.

Very truly yours,

R. G. Byram

cc: NRC Document Control Desk (original)
Mr. G.S. Barber, NRC Sr. Resident Inspector
Mr. R.J. Clark, NRC Sr. Project Manager