PDR

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230-0355

Mr. Lou Anastasia
Commonwealth Edison Company
35th Floor FN West
Chicago, Illinois 60690

July 12, 1985
NS-RAT-PRRA-85-152

Re: Follow-up to NRC/BNL/CECo/W Meeting on Byron LCORP

50-454/455

Dear Lou:

Attached is a summary of our responses to the questions raised by Brookhaven at the review meeting. Action items have been identified which will be addressed in the next several days. Per our conversation on July 10, the asterisked items are best addressed by CECo due to the availability of information. Could you please send us a copy of everything that is sent to BNL to insure that our references are identical to those being used by the review team?

Feel free to call me if you need any further information. My new phone number is (412) 236-6470.

Very truly yours,

Jon F. Merz, Senior Engineer
Plant Risk Analysis

Approved: D. S. Sharp
D. S. Sharp, Manager
Product Risk Analysis

JFM/bbp

cc: Mr. Nam Cho - Brookhaven National Labs
Mr. Al Spano, Nuclear Regulatory Commission
Phillips Building

NS-RAT-PRRA-85-149
July 11, 1985

Response to Questions on Byron LCOR raised at the June 19 meeting

- **General Topics**
  - o **Methodology** - it is felt that the support state approach yields conservative results. It will be instructive to compare the approach to the SETS methodology.
  - o **Mean AOT Usage** - while components may be removed from service for maintenance for times up to (and possibly exceeding) the AOT, the frequency of such events should be much lower than the frequency of maintenance events of shorter duration. We feel that this inverse relationship is properly modeled by use of mean outage times.
  - o **Electric Power** - DC power has not been analyzed in this study, since no change in the LCO applicable to the batteries and inverters is being sought. It was considered to be beyond the scope of the study to verify proper design separation of Class 1 E equipment

    AC power has, for the most part, been modeled in the Support State model. 4160 V bus availability is explicitly modeled. 480 V bus availability is implicity included in the support state model. The failure rate of the 4160 - 480 V transformer and its breaker is small compared to the failure rates of both the offsite power source and the diesel generator associated with each bus. Thus, 480 V bus availability was assumed to be parallel with its associated 4160 V bus, thus a part of the support systems. Each component powered by 480 V AC includes the failure of its associated Motor Control Center, each of which is fed by that train's 480 V AC ESF bus. There are 5 MCC on each ESF bus.

- **Modeling**

1. See the discussion under Electric Power, above. The six support states modeled do not include states where one bus is powered and service water is available. the loss of one bus is clearly dominated by a loss of offsite power (to both buses) and the failure of one diesel generator. In this mode, if service water is lost, the diesel generator will fail, leading to support state 6. See, e.g., Figure 4.1.2-2 et seq.

   * **Action:** A copy of ESF 480 VAC Distribution Diagrams.

2. DC Power loss has not been considered as an initiating event. See the discussion under Electric Power, above.

   * **Action:** Copy of D.C. ESF Loading Tables.

3. The loss of ESW may be a more benign event than at Zion, since Byron loads the makeup and HHSI pumps on ESW, not CCW. A loss of ESW will

cause makeup pump trip, but will not cause an immediate loss of RCP thermal barrier cooling due to the heat capacity of the CCWS. Both the time to heat up the CCWS and the cooling of the RCS by Auxiliary Feedwater System operation will decrease the chance of seal damage.

Actions: A. List of CCW and ESW loads (& sizes).
B. CCWS heat capacity (following loss of ESW).
C. Recent Seal LOCA developments (See attachment).
D. Annual frequency of loss of ESW.

4. Average values applied to system operational lineups should obviate the need to analyze each normally operating system in each potential operating mode.

Action: None

5. The AFW pumps are tested monthly on a staggered schedule. Thus, the two trains are always separated by 1/2 month in their test period. The 3/4 λT unavailability was applied to the more unreliable of the two AFW trains, and is considered to be conservative. In addition, this time period is only applied to open valves, which do not comprise a significant contribution to AFW system unreliability. Refer to Tables 3.5-9 and 10.

Action: None

6. The support state approach conservatively models the simultaneous maintenance phenomenon between front-line systems and support systems. The unavailability of a given support system train includes that train's maintenance. Thus, in a given support state, a contribution to that support state is the maintenance of a support system component. For example, Support State 3, Bus 141 unavailable, includes the maintenance of Diesel Generator A. During quantification of front-line system unavailability, however, only Train B is included in, e.g., Low Pressure Injection. A contribution to LPI unavailability is the maintenance of the "B" RHR pump. This event, however, is precluded by the tech specs. Thus, the system unreliability is over-estimated by the product of the unavailabilities of the two systems due to maintenance.

Action: None

7. (1) RHR valves 8811A,B are not tested quarterly, they both are stroke tested and automatically actuated at 18 month intervals (refueling).

Action: Correct Table 3.7-1

(2) At present there is no flow test of the CCW valves at the RHR HX. There is a capability to test these valves by measuring design flow rates with the system properly configured. Also, during cold shutdown, flow conditions will be verified by proper reactor decay heat removal operation.

Action: None

(3) SI valves 8926, 8806, 8923A,B cannot be isolated during power operation, it is a violation of tech specs to do so. The unit would be shut down.

Action: None

(4) SI valves 8921A,B and CV8481A,B are not tested in the ISI program. The check valves will be tested at refueling (during vessel injection) and whenever the centrifugal charging pumps are utilized for makeup. These valves will not contribute significantly to the unavailability of the HHSI systems.
See Table 3.7-23.

Action: None

(5) SI8835 receives an S signal. Failure to restore after test should be ANDed with failure to open on demand. This error is conservative.

Action: Potentially correct fault trees

(6) High pressure recirculation path valves 8804A,B, 8807A,B and 8924 are stroke tested quarterly.

Action: None

(7) High pressure recirculation fault tree, figure 3.7-4 should have gate 181 input to gate 185.

Action: Verify and change fault tree

(8) Agree that our treatment of switchover operation was non-conservative in the model of high pressure recirculation. The operator failures on each of the several MOV movements should be interrelated.

Action: Possibly change fault tree

(9) It was assumed that the NPSH available to the RHR pumps from the containment sump upon opening the sump isolation values would be sufficient to close the RWST suction check valve. Under this assumption, manual closure of the RWST suction MOV would not be necessary to prevent the RHR pumps from draining the RSWT empty and cavitating. If the opposite were true, than the automatic actuation of recirculation operation by opening the sump valves would be defeated.

Action: Verify the NPSH (containment) exceeds NPSH (RWST) upon the actuation of recirculation of low RWST level.

(10) Accumulation discharge valves are electrically locked out in the open position, are alarmed for improper position above about 600 psig RCS pressure, and are verified open at regular surveillance intervals (once per 31 days) See T.S. 3.5.1.

Action: Correct the fault tree

(12) Operator failure to restore valves 8809, 8716, and 8812, which are alarmed on wrong position, is a conservative treatment. The inclusion of this failure mode does not impact the results.

Action: None

8. The mean times to repair components in this study were derived assuming: 3-day LCO - lognormal distribution with 5th and 95th percentiles of 2 hours and 60 hours, respectively; and 7-day LCO - lognormal dist. with 5th and 95th percentiles of 2 hours and 120 hours, respectively. The prior distribution was utilized throughout the study, since keeping the duration times consistent was the most comprehensible means for measuring the MTTR as a function of LCO. The updated values for different components at Zion were representative of 7-day LCO, and it was felt that it would not be meaningful to try to derive 3-day LCO durations for comparison to the updated values. No data existed (at the time of the study) to substantiate MTTR on 3-day LCO plants, nor on the Byron facility.

Action: Possible sensitivity analysis

9. Diesel generator failure was modeled in two modes: demand failure to start, for which the data comprising the failure probabilitiy inherently includes a standard test period of about 1 month; and hourly failure to run, which is a failure rate derived from operating experience.

Action: None

10. Common cause failures in a normally operating system were modeled as follows:
    a. One pump running and one standby - common cause failure to run;
    b. Running pump trips and both must start (loss of offsite AC) - common cause failure to start and failure to run.

Action: None

11. Common cause linking of the two dissimilar AFW pumps was included since the two pmps are identical, although the drivers are different. For this model, Atwood's beta factor for diesel driven pumps was assumed to be entirely embodied in the pump, and thus used to represent the failure of both pumps.

Action: None

12. Typical quantification cutoff probabilities were as follows:
    WAMBAM - 10^{-11} to 10^{-15}
    WAMCUT - roughly 3 orders of magnitude below the WAMBAM result.

Action: None

13. Agree that coefficients in common cause equations are wrong, see pages 3.8-3,5. These coefficients are applied to higher order failures, and should not impact the results.

    Action: Check that proper equations were used, correct report tables

14. AFW diesel driven pump maintenance frequency was taken from the Torrey Pines 0611 Report.

    ✳ **Action:** Look into additional data on Zion CS pump and Byron AFW pump

15. The startup feedwater system was not credited in the LCOR analysis.

    Action: None

16. The provision for crosstying the A diesel generators has not been credited in the analysis. Thus, the study assumes that 2A DG is available when 1A is taken out of service. In actuality, if 2A were unavailable, the units would be shutdown within two hours.

    Action: None

17. Sequence information is available and will be provided.

    Action: Attach Adam output

18. Maintenance procedures are extremely voluminous and probably of little value.

    ✳ **Action:** CECo to identify system engineers at Byron who may help with review questions as they arise

Additional Questions

1. Reasons for the requested changes is AOTs include:
   A. fewer unnecessary plant shutdowns due to T.S.;
   B. more time to diagnose failures, seek out the root causes;
   C. less chance of human error since rush situations will be avoided;
   D. allow procurement of spare/replacement parts; and
   E. there are limits on the number of hours any one worker can work on safety-related equipment, which impacts the actual time available for the performance of the repair effort.

2. There are not substantive differences in the Zion and Byron maintenance policies, since the Byron staff has largely been derived from the Zion staff. The major constraint on the Byron staff is the 3-day LCO, which restricts the time available for preparation and trouble shooting.

3. Refer to Question 3 under Modeling, above.

4. Unit 2 Diesel Generator A is available.

5. There are no room coolers for the AFW pumps. Auxiliary building

ventilation is available for general area cooling.

6. Emergency operating procedures for feed and bleed are used. #1ZFR-H.1, which follows the Westinghouse ERGs, calls for feed and bleed when AFW flow falls below 485 gpm.

Other Materials

- Will RHR discharge valves (cont. Isol.) open on S when system in test and valves closed?
- Need copy of latest revisions of system P&ID.
- Do ESW valves to HX O open on S signal?
- Fan cooler operability tech spec modeled incorrectly. Assess the impact of the current LCO.

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REACTOR COOLANT PUMP SEAL PERFORMANCE

The issue is concerned with Westinghouse reactor coolant pump (RCP) seal performance in a postulated loss of all seal cooling scenario. The postulated occurrence of loss-of-all-AC power (station blackout) was evaluated generically by Westinghouse and found to be the dominant contributor leading to loss of all seal cooling. This scenario involves loss of all offsite power and failure of all the diesel generators to start on demand. The NRC has designated this postulated event as USI A-44. This condition is beyond the current licensing basis and beyond the Westinghouse design basis for nuclear power plants. Westinghouse Owners Group sponsored analysis has shown that the seals will continue their sealing function and limit the reactor coolant system leakage to minimal levels if a station blackout occurs. When the probabilities of a loss-of-all-AC event are factored into the low estimated leakage, the frequency of core uncovery from these events is lower than the current preliminary NRC safety goal. Therefore, Westinghouse does not consider plant modifications necessary to address the issue.

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The analysis of RCP seal performance was submitted to the NRC for comment in April, 1984 as WCAP-10541 (Westinghouse proprietary), "Westinghouse Owners Group Report, Reactor Coolant Pump Seal Performance Following A Loss Of All AC Power." The NRC contracted Energy Technology Engineering Center (ETEC) to perform a review of the WCAP and to perform independant audit calculations. The ETEC review, which was completed in December, 1984, concluded that the Westinghouse methods were conservative while the audit calculations found leakage rates that were 7 percent to 20 percent lower than the Westinghouse results. NRC comments have been forwarded to the Westinghouse Owners Group in a letter dated April 17, 1985. The NRC letter concluded that substantial additional testing, including a full scale demonstration test, was required due to the complexity of a failed seal flow path. Many of the NRC comments are already being addressed by the most current Westinghouse Owners Group program.

Secondary sealing materials testing at the Chalk River National Laboratory of Atomic Energy of Canada, Ltd. (AECL) indicated that the currently used O-ring material would probably not survive the expected loss of seal cooling re--

onditions for the required duration of the loss of all AC ower event. An alternate O-ring material subsequently tested y AECL did demonstrate a much higher resistance to failure nd maintained it's integrity at conditions which are more xtreme than expected during the event. Based upon successful ormal operation testing, the Westinghouse Owners Group and estinghouse concluded that it is appropriate to change out he O-ring material to the alternate O-ring material during ormally scheduled pump maintainence outages. Westinghouse ill be able to begin supplying the alternate O-ring material n April, 1986.

The Westinghouse Owners Group also funded tests at AECL of the teflon based channel seal material subject to the extremes of pressure and temperature identified in the analysis. Tests of unirradiated channel seals were successful, showing very little or no extrusion. However, tests of highly irradiated channel seals indicated that extrusion may result, particularly if the channel seals are exposed to oxygen after extended irradiation. The NRC has expressed concern that extrusion of the channel seal material may result in the RCP seals being forced open due to thermal growth of the RCP shaft and housing during the loss of all seal cooling event.

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Additional testing will be required to demonstrate acceptable extrusion limits for better estimate irradiation and oxygen exposure, or an alternate channel seal material will require qualification.

The Westinghouse Owners Group participated, through a Westinghouse three-party agreement, in a loss of all seal cooling test on a 7-inch RCP seal system in a static RCP mockup at the Montereau Power station in France on May 29, 1985. The test was conducted by Electricite de France (EDF) with joint participation by Framatome and Jeaumont-Schneider and demonstrated acceptable leakage rates during the loss of all seal cooling event. The Westinghouse Owners Group participation made the test more representative of the loss of all AC power event.

The response of the 7-inch seal system to the loss of all seal cooling was unknown prior to the test since there are significant dimensional design differences between the 8-inch RCP seal systems which were analyzed and the 7-inch RCP seal system. A less detaed evaluation of the design indicated

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that higher leakage rates were expected in the 7-inch seal system design. However, the 7-inch RCP seal preliminary test results indicate that the leakage rates were approximately 40 percent lower than the analysis calculations for the 8-inch RCP seal system. The 7-inch RCP seal system was well behaved and stable throughout the 20 hour duration of the test which simulated the expected reactor coolant system conditions at the inlet of the RCP seal system during a loss of all AC power event.

The test results indicate that resolution of the channel seal extrusion concern and substitution of the alternate O-ring material should be sufficient to close out the issue. However, the NRC may require analysis of the 7-inch RCP seal system as well as a full scale demonstration test of the 8-inch RCP seal system to resolve the issue. The Westinghouse Owners Group will be meeting with the NRC in July, 1985 to discuss the remaining NRC concerns and methods for resolving the issue. Westinghouse and the Westinghouse Owners Group will petition the NRC to review the EdF 7-inch RCP seal full scale test results for applicability to the Westinghouse desigbefore embarking on a full scale demonstration test of the 8-inch RCP seal.

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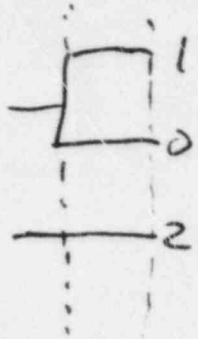
iditionally, the Atomic Industrial Forum has proposed a
oluntary utility program that would supply the NRC with
implified deterministic plant-specific analyses in order to
stablish the tolerable duration times for station blackout.
omments on this program have only recently been solicited.

EVENT TREE # SUPPORT STATE PATH #

LIST OF DOMINANT ACCIDENT SEQUENCES

INIT EVENT			CM STAT	CM FREQ.	EVENT SEQ.	FAILED SYSTEMS
03.01	SLO	2	SLFC	1.6250E-05	011111211011	R-2
11.02	TT	17	S2/SEC	6.4000E-06	0022222222	
07.02	LMFW	44	S2/SEC	4.8000E-06	002222222222	
14.02	RT	44	S2/SEC	4.8000E-06	002222222222	
12.05	LOSP	56	SE	3.7750E-06	010020222222	
12.05	LOSP	53	SEC	9.7509E-07	010021021011	
11.01	TT	3	SLFC	9.5772E-07	0111010111	
07.01	LMFW	3	SLFC	7.1929E-07	011112010111	
14.01	RT	3	SLFC	7.1829E-07	011112010111	
03.02	SLO	59	SEC	6.9259E-07	011101201011	
04.02	SGTR	48	V2E	6.4335E-07	001110002	
01.01	LLO	19	AEFC	6.2789E-07	0111011122	
02.01	MLO	37	AEFC	6.2789E-07	0111011122	
09.02	LRCS	44	S2/SEC	5.7500E-07	002222222222	
11.01	TT	9	TEFC	4.1909E-07	0111002111	
08.02	MSIV	45	S2/SEC	4.0300E-07	002222222222	
12.05	LOSP	29	TE	3.4686E-07	010020222222	
14.01	RT	9	TEFC	3.1432E-07	011112002111	
07.01	LMFW	9	TEFC	3.1432E-07	011112002111	
01.01	LLO	2	ALFC	2.3413E-07	0111111110	
02.01	MLO	2	ALFC	2.3413E-07	0111111110	
11.02	TT	12	TEC	2.1715E-07	0111002011	
15.01	ISL	2	V	2.1300E-07	00	
12.03	LOSP	3	SLFC	2.0489E-07	0111010111	
11.01	TT	17	S2/SLFC	1.9000E-07	0022222222	R-2
04.01	SGTR	29	V2L	1.8273E-07	012001102	
14.02	RT	12	TEC	1.6286E-07	011112002011	
07.02	LMFW	12	TEC	1.6286E-07	011112002011	
07.01	LMFW	44	S2/SLFC	1.5000E-07	002222222222	R-2
14.01	RT	44	S2/SLFC	1.5000E-07	002222222222	R-2
13.02	SSI	58	S2/SEC	1.2600E-07	002222222222	
13.02	SSI	6	SLC	1.2539E-07	011112100011	
12.05	LOSP	23	TEFC	1.2163E-07	011001102111	
	COREMELT FREQ TOTAL			4.66 E-05		
04.01	SGTR	4	V2L	7.1835E-08	012111020	
04.01	SGTR	16	V2L	1.3346E-08	012100020	
04.01	SGTR	48	V2E	1.3313E-08	001110002	
04.01	SGTR	8	V2L	1.2461E-08	012110020	
04.02	SGTR	60	V2E	7.1484E-08	001100002	
04.02	SGTR	61	V2E	2.5160E-08	001022222	
04.05	SGTR	48	V2E	1.2259E-08	001110002	
03.05	SLO	51	SE	1.3041E-08	011101201002	
07.05	LMFW	44	S2/SE	3.3000E-08	002222222222	
14.05	RT	44	S2/SE	3.3000E-08	002222222222	
12.05	LOSP	56	SE	1.0212E-08	010020222222	
12.05	LOSP	29	TE	9.4350E-09	010020222222	
11.05	TT	17	S2/SE	4.4000E-09	0022222222	
03.02	SM	77	SEC	5.3933E-02	2.4025E-08	011100201011
03.01	SM	83	SEC	2.3915E-07	1.0235E-08	011021221111
08.01	C	45	S2/SLFC		1.2000E-08	002222222222 R-2
08.01	C	3	SLFC	2.3943E-07	6.0336E-09	011121201111
08.01	C	9	TEFC	1.0477E-07	2.6403E-09	011121002111

REFER TO THE SPECIFIC EVENT TREE FOR DETERMINATION OF ACTUAL SEQUENCE PATH



08.02	CL	12	TEC	1.5955E-02	1.3674E-08	011121202011	
					1.7000E-08	002222222222	R-2
09.01	LO	44	S2/SLFC				
09.01	LO	3	SLFC	2.3943E-07	8.5716E-08	011112010111	
09.01	LO	9	TEFC	1.0477E-07	3.7508E-08	011112002111	
09.01	LO	12	TEC	1.5955E-02	1.9510E-06	011112002011	
09.02	LO	3	SLFC	2.3943E-07	1.3504E-08	011101010111	
12.01	LO	17	S2/SEC	5.0030E-01	5.7500E-08	002222222222	
12.02	LO	9	TEFC	5.1824E-05	2.9850E-08	011100002111	
12.03	LO	26	TEC	4.4113E-03	3.3085E-08	011001002011	
12.05	LO	10	SLFC	2.5937E-07	1.9150E-08	011112010111	
13.01	SP	16	TEFC	2.2451E-07	1.7961E-08	011112002111	
13.01	SP	3	SLFC	2.1542E-07	1.7234E-08	011112100111	
16.01	ATW	44	SEFC	5.9352E-04	1.3623E-08	010222202222111	

1 —

2 SW

3 1DG

4 1DG

5 2DG

6 2DG + SW