The Light company
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498; STN 50-499
Review of Preliminary Accident Sequence Precursor
Analysis of Event at South Texas Project, Unit 1

Reference: Correspondence from Thomas W. Alexion, U. S. Nuclear Regulatory Commission to William T. Cottle, HL&P, dated April 7, 1995 (ST-AE-HL-94175)

The referenced correspondence transmitted a preliminary Accident Sequence Precursor analysis of an operational event which was discovered at South Texas Project, Unit 1, on March 11, 1994, and was reported in Licensee Event Report No. 94-012. As requested, Houston Lighting & Power has completed a review of this analysis and provides a response in Attachments 1 through 7.

A review of sequences as analyzed as part of the Nuclear Regulatory Commission's Accident Sequence Precursor Program was performed by Houston Lighting & Power using insights gained from the South Texas Project Individual Plant Examination model and actual conditions existing during the event. The details of the review, provided in Attachments 1 through 7 to this letter. indicate that the event described in Licensee Event Report No. 94-012 should not be considered an accident sequence precursor.

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If you should have any questions, please contact Mr. K. J. Taplett at (512) 972-8416 or me at (512) 972-8787.

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Attachments:
1) Review of Accident Sequence Precursor Analysis for LER 94-012
2) Unit 1 Daily Summary from 09/10/92 to 04/15/94
3) Diesel Generator Starts Log
4) Generic Implications of Event
5) 0POP05-EO-EC00, Loss of All AC Power
6) 0POP04-AE-0001, Loss of any 13.8KV or 4.16KV Bus
7) Design and Operating Parameters of the Westinghouse Model E Steam Generators for the South Texas Project

c:

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* Above copies distributed without Attachments 2-7 except as noted by asterisk (*).

ATTACHMENT 1

REVIEW OF ACCIDENT

SEQUENCE PRECURSOR ANALYSIS

FOR LER 94-012

Review of Accident Sequence Precursor Analysis for LER 94-012

Reference: 1) Correspondence from Thomas W. Alexion, U. S. Nuclear Regulatory Commission to William T. Cottle, HL&P, dated April 7, 1995 (ST-AE-HL-94175)

The preliminary Accident Sequence Precursor (ASP) analysis was reviewed in accordance with Attachment A of Reference 1. The ASP analysis was found to present an accurate event description based on the information given in Licensee Event Report (LER) 94-012, however, more details required to accurately reflect the risk significance of this event are provided in this review. Based on the review of the ASP documentation and additional information not included in LER 94-012, HL&P believes that some of the assumptions are overly conservative, and the event as it occurred is not an accident precursor.

EVENT DESCRIPTION REVIEW:

The overall event description provided in the ASP package provides an accurate description of the event being considered based on the information provided in LER 94-012.

Two points brought out in the ASP event description are discussed further in the Modeling Assumptions Review section below. These are, that the Standby Diesel Generator (SDG) 11 had 6 successful starts between the two K1 failures, and that Unit 1 was shutdown for much of the time this event occurred. Neither LER 94-012 or the ASP event description mention that the shutdown for approximately the first three weeks of this event was on the end of an extended shutdown of approximately one year. This is probably the most significant item unaccounted for in the ASP analysis.

One other point that is not brought out in the event description is that the turbine driven auxiliary feedwater pump (TDAFWP) was fully functional for much of the period it was administratively inoperable. The modeling of the TDAFWP as guaranteed to fail and unrecoverable for the entire period of 121.5 hours is extremely conservative. LER 94-012 specifically states (page 5 of 6) that the TDAFWP was functional during the majority of the time the TDAFWP was inoperable. LER 94-012 does not give the actual time that the TDAFWP was fully functional. However, based on the evaluation of the decay heat level at the time the TDAFWP was technically inoperable, the functionality of the TDAFWP is irrelevant to the analysis.

ADDITIONAL EVENT RELATED INFORMATION REVIEW:

This section was reviewed, and found to accurately reflect the redundancy provided by the STP design. Some additional design features which affect the ASP analysis are given in the Modeling Assumptions Review section of this review.

MODELING ASSUMPTIONS REVIEW:

The modeling assumptions were reviewed to determine if plant/event specific assumptions would have a significant impact on the results.

Probably the most significant aspect of this event related to its safety significance is the decay heat level of the core. LER 94-012 only mentions that decay heat was minimal, however, it needs to be recognized that Unit 1 had been in an extended shutdown since February 4, 1993, and had only operated approximately 31 days since the beginning of a refueling outage on September 19, 1992. The decay heat was therefore extremely low, and the time available to recover offsite power based on drying out the steam generators was significantly longer than if this event would have occurred with the reactor operating. The daily plant summaries showing reactor power level are included in Attachment 2.

An estimate of the decay heat level in the core was made for 2 conditions: 1 hour after shutdown and 1 year after shutdown (the approximate time in the extended outage). Based on the decay heat equations in NUREG-0800 ASB 9-2, the decay heat level 1 hour after shutdown is approximately 2.1E8 BTU/hr, and 1 year after shutdown is 4.4E6 BTU/hr. Therefore, the decay heat level for much of this event was almost 2 orders of magnitude lower than if the reactor had been operating continuously. These calculations were done using a conservative operating history for the core, and the calculated steam generator dryout time with no Auxiliary Feedwater (AFW) would be well over 24 hours. In addition, with the low decay heat load and steam generators available, the pressurizer PORVs would not likely be challenged. This calculation is included at the end of this Attachment.

After the TDAFWP outage, the reactor did operate for approximately 11 days with the highest attained power level approximately 28 percent of full power. Given the short duration of operation and the reduced power level, the time available for electric power recovery before core damage could occur would have been significantly longer than if the reactor would have been at full power for a long duration. To accurately characterize this event, the recovery analysis would have to be revised to account for the longer time available to recover electric power.

An assumption is made in the ASP analysis that Standby Diesel Generator (SDG) 11 was guaranteed to fail for the entire 633 hour duration of this event. This was not the case, and, as stated in LER 94-012, the failure was intermittent during this period. The classification as an intermittent failure mechanism is justified, because as stated in LER 94-012, SDG 11 had 6 successful starts between the two related failures giving the diesel a failure to start probability of 2/8 for the overall period. From a PSA modeling standpoint based on the actual events, SDG 11 should not have been guaranteed to fail before the completion of the last successful test. Between the first failed test and the last successful test, the SDG was shown to be functional, and therefore no increase in the failure rate is justified. The dates of these tests were not provided in LER 94-012, but are given in the start logs for SDG 11 which are included in Attachment 3. The last successful test of SDG 11 before the March 3 failure was performed on February 23, 1994 at 06:57, so the actual time that should be considered as a **guaranteed** failure of SDG 11 is 148 hours between February 23 and March 1, 1994. The failure mechanism of the K1 relay during this period would not have caused a failure during operation after a successful start. The 148 hour period does not overlap the TDAFWP outage, but does exceed the Allowed Outage Time (AOT) for a single diesel in Modes 1-4.

The common cause failure probability was adjusted in the ASP analysis due to the observed failure of one of the SDGs. It is unclear from the ASP documentation how the probability was changed. It appears reasonable to assume that this failure could result in a common cause failure of two or more SDGs, however, the generic implications of this event were reviewed during the event investigation and the other SDGs were found to have their relays in a satisfactory condition. Therefore, no increase in a common cause failure of SDGs 12 and 13 should be modeled. The generic implications section of the investigation of this event is included in Attachment 4 as part of South Texas Project Station Problem Report 940250.

LER 94-012 states that the TDAFWP was inoperable for 121.5 hours during this event, and further states that during a majority of this time it was fully capable of performing its safety function. Without more specific information, the ASP analysis assumes that the TDAFWP is guaranteed to fail and is unrecoverable for the entire 121.5 hour period. For this specific event, the TDAFWP is actually irrelevant. As discussed previously, the decay heat level at the time of the TDAFWP outage was such that the time to boil the steam generators dry was significantly longer than the mission time used in the STP PSA for at power events (much greater than 24 hours).

The ASP analysis assumes that given a loss of reactor coolant pump (RCP) seal cooling, there is a 0.32 probability that an RCP seal LOCA will occur. STP has a method of alternate seal injection with a positive displacement pump (PDP) backed by the separate and diverse Technical Support Center diesel generator (TSCDG). This method of alternate seal injection is included in Emergency Operating Procedure 0POP05-EO-EC00 (step 6) which is included in Attachment 5. The failure rate of the PDP (when backed by the TSCDG) is defined by split fraction PDJ in the STP IPE, and equals 0.28. In the ASP analysis, the probability of a seal LOCA would then become 0.32 times 0.28 or 9E-2. The STP IPE assumes the probability of seal failure to be 1.0 after one hour without injection or cooling, so the probability of a seal LOCA after one hour is equal to the failure probability of the PDP when backed by the TSCDG (0.28).

Sequences 38 and 39 of the ASP analysis appear to evaluate the failure to recover a SDG or the TDAFWP, but neglect a recovery failure probability for offsite power. It appears that the failure to recover emergency power, EPS-XHE-NOREC, (assumed in this review to mean the SDGs) is used with the failure rate of the SDGs, and the failure to recover the TDAFWP, AFW-XHE-NOREC-EP, is used with the failure rate of the TDAFWP. A failure to recover offsite power does not appear to be included in sequences 38 and 39, but would be appropriate since offsite power recovery is considered in Off Normal Operating Procedure 0POP04-AE-0001, Attachment 6, which is entered from 0POP05-EO-EC00, step 9.

In order to determine the degree of conservatism used in the ASP models, a comparison of conditional CDFs from the ASP analysis and the STP average CDF model was made for the configuration of one SDG out of service. For the 511.5 hour period that the ASP analysis considers with SDG 11 unavailable, the core damage probability is 5.1E-5 (Case 1 A). The conditional CDF calculated in the ASP analysis is then:

CCDF = (5.1E-5 /511.5hr) * (8760 hrs/yr) = 8.73E-4/yr

The estimated conditional core damage frequency at power with one SDG out of service is estimated to be 2.4E-5/reactor yr using the current STP specific risk model which includes plant specific failure data, as well as, a site specific LOOP frequency and electric power recovery model. Therefore, the ASP analysis is very conservative. The CCDF based on the actual conditions for this event would be even lower if the recovery analysis were revised to account for the low decay heat level.

CONCLUSIONS:

This event, as it occurred, had essentially no safety significance. Due to the extended outage with the extremely low decay heat load, the plant was in a relatively safe condition with regard to station blackout scenarios. Based on a mission time of 24 hours, decay heat removal would have been available the entire time through the steam generators. Also, the pressurizer PORVs would not have lifted except possibly for the brief reduced power operation. The available recovery times based on the decay heat level would have virtually guaranteed the recovery of electric power before core damage occurred from a loss of decay heat removal. Given the additional detail not contained in the LER but provided by this review, the event described in LER 94-012 should not be considered an accident sequence precursor.

Some other factors found by this review relevant to the safety significance of this event are:

- SDG 11 was guaranteed to fail for only 148 hours,
- There is no increased common cause contribution to SDGs 12 and 13,
- The TDAFWP was out for a one time extended period in order to perform extensive startup tests. For much of the time that the TDAFWP was administratively inoperable, it was functional and capable of performing its intended safety function. In addition, if this event would have occurred at any other time during power operation, the outage duration for planned maintenance would have been significantly less and limited by the 72 hour allowed outage time,
- Other STP design features were available such as the PDP/TSCDG that are not accounted for in the ASP analysis and would provide additional capabilities to mitigate the impact of a LOOP.

STEAM GENERATOR DRY OUT CALCULATION:

The following calculation estimates the time to dry out the steam generators using the conditions that existed for Unit 1 before startup in February 1994.

Decay Heat Assumptions:

The power before shutdown was 3876 MW thermal. This is 2 percent higher than the 3800 MW rating of the units, and corresponds to a maximum uncertainty on power level used in the UFSAR Chapter 15 accident analyses.

All 193 assemblies in the core had an average burnup of 40,000 effective full power hours (EFPH). This is conservative because this core was relatively fresh having operated only approximately 31 days since the last refueling outage.

The time since shutdown was 1 year. Unit 1 entered Mode 1 on February 22, 1994 and had been shutdown since February 4, 1993.

The decay heat equations identified in NUREG 0800 ASB 9-2 were used to calculate decay heat levels.

The results of the decay heat analysis are shown on the following page.

Steam Generator Assumptions:

The mass of water in each steam generator was estimated to be 200,000 lbm at zero percent power. This value is based on the table included in Attachment 7 which gives the secondary mass in a steam generator as a function of power level. The zero percent power secondary mass from Attachment 7 is 209,000 lbm, but 200,000 lbm was assumed because some steam would be present.

All heat addition is latent heat, the sensible heat addition to pressurize the steam generators up to the safety valve pressure is conservatively neglected.

The steam generator safety valves lift at 1300 PSIA, the latent heat of vaporization at this pressure is 594.6 BTU/lbm.

Calculation

Decay heat 1 yr after shutdown = 4.41E6 BTU/hr
Mass of water/steam generator = 200,000 lbm
Total mass of water in steam generators = 4 * 200000lbm = 800000 lbm
Latent heat of vaporization at 1300 PSIA = 594.6 BTU/lbm

Energy to vaporize total mass of water in the steam generators assuming all energy input is latent heat at the steam generator safety valve setpoint:

E = 800000 lbm * 594.6 BTU/lbm = 4.76E8 BTU

Time to boil steam generators dry at decay heat level of 4.41E6 BTU/hr:

T = (4.76E8 BTU) / (4.41E6 BTU/hr) = **108 hours**

ATTACHMENT 2

UNIT 1 DAILY SUMMARY

From 09/10/92 To 04/15/94

(20 pages)

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
09/10/92	1,035.4	
09/11/92	1,032.1	
09/12/92	1,027.9	
09/13/92	1,017.5	
09/14/92	1,016.3	
09/15/92	1,0[illegible].3	
09/16/92	1,018.8	
09/17/92	1,016.3	
09/18/92	908.3	The unit commenced shutdown for 1RE04 at 1800.
09/19/92	12.1	The first coastdown mode of operation was successfully completed. The unit commenced shutdown operations on August 13, 1992 at an initial reactor power if 99.5%. Due to burnup of excess reactivity during the fuel cycle, the reactor could no longer sustain 100% operation, therefore, reactor power was slowly decreased to 78.5% prior to the scheduled shutdown for refueling. The unit was removed from service at 0202. Forth refueling outage (1RE04) begins at 0202. Conclusion of OPERATING CYCLE 4. Start of REFUELING CYCLE 4. (Refueling cycle includes unit down time for refueling; Breaker open to allow refueling through breaker open to allow the following refueling. Operating cycle includes Breaker closed following refueling through Breaker open to allow refueling.) The unit entered Mode 2 at 0203 and Mode 3 at 0326.
09/20/92	0.0	The unit entered Mode 4 at 1509.
09/21/92	0.0	Unit entered Mode 5 at 0338.
09/22/92	0.0	
09/23/92	0.0	
09/24/92	0.0	
09/25/92	0.0	Unit entered Mode 6 at 1045.
09/26/92	0.0	
09/27/92	0.0	
09/28/92	0.0	
09/29/92	0.0	
09/30/92	0.0	
10/01/92	0.0	
10/02/92	0.0	
10/03/92	0.0	
10/04/92	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
10/05/92	0.0	Core off-load complete entered no mode at 1648.
10/06/92	0.0	
10/07/92	0.0	
10/08/92	0.0	
10/09/92	0.0	
10/10/92	0.0	
10/11/92	0.0	
10/12/92	0.0	
10/13/92	0.0	
10/14/92	0.0	A failed fuel oil system check valve caused a valid failure of Standby Diesel Generator 12 during testing . The disc plate of a 3/4-inch swing check valve installed between the fuel oil filter and fuel oil supply header had become detatched from the hinge. The disc plate is held on to the hinge by the disc stud. The nut on the stud was not locked in place. The plate became lodged in the fuel oil flow path, cutting off the fuel oil from the fuel injection pumps. As a result, the diesel engine lost power and was unable to maintain generator electrical load. Inspection of two other similar, but larger check valves on the same engine revealed the potential for the nuts on the disc stud to loosen. These check valves will be staked in place as determined by engineering. Additionally, the one 3/4-inch check valve in each of the six Station Standby Diesels will have the internals removed.
10/15/92	0.0	
10/16/92	0.0	
10/17/92	0.0	
10/18/92	0.0	
10/19/92	0.0	
10/20/92	0.0	
10/21/92	0.0	
10/22/92	0.0	
10/23/92	0.0	
10/24/92	0.0	
10/25/92	0.0	
10/26/92	0.0	
10/27/92	0.0	
10/28/92	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
10/29/92	0.0	
10/30/92	0.0	
10/31/92	0.0	
11/01/92	0.0	
11/02/92	0.0	
11/03/92	0.0	
11/04/92	0.0	
11/05/92	0.0	
11/06/92	0.0	
11/07/92	0.0	
11/08/92	0.0	
11/09/92	0.0	
11/10/92	0.0	
11/11/92	0.0	
11/12/92	0.0	
11/13/92	0.0	
11/14/92	0.0	
11/15/92	0.0	
11/16/92	0.0	
11/17/92	0.0	
11/18/92	0.0	
11/19/92	0.0	
11/20/92	0.0	Due to corrective maintenance work on equipment required to allow plant start-up, the refueling outage was extended beyond the original scheduled start-up date of November 20. November 21 marked the beginning of unplanned shutdown time accrued to the unit.
11/21/92	0.0	
11/22/92	0.0	Unit entered Mode 5 at 0305.
11/23/92	0.0	
11/24/92	0.0	
11/25/92	0.0	
11/26/92	0.0	
11/27/92	0.0	
11/28/92	0.0	
11/29/92	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
11/30/92	0.0	
12/01/92	0.0	
12/02/92	0.0	
12/03/92	0.0	
12/04/92	0.0	
12/05/92	0.0	
12/06/92	0.0	
12/07/92	0.0	
12/08/92	0.0	Unit entered Mode 3 at 1253.
12/09/92	0.0	
12/10/92	0.0	Unit entered Mode 4 at 0053 and Mode 5 at 0606.
12/11/92	0.0	
12/12/92	0.0	
12/13/92	0.0	
12/14/92	0.0	
12/15/92	0.0	
12/16/92	0.0	
12/17/92	0.0	
12/18/92	0.0	
12/19/92	0.0	
12/20/92	0.0	
12/21/92	0.0	
12/22/92	0.0	
12/23/92	0.0	
12/24/92	0.0	
12/25/92	0.0	Unit entered Mode 4 at 1459.
12/26/92	0.0	Unit entered Mode 3 at 0202.
12/27/92	0.0	
12/28/92	0.0	Unit entered Mode 2 at 0042 and went critical at 0953.
12/29/92	0.0	
12/30/92	0.0	Unit entered Mode 1 at 0451.
12/31/92	76.7	Generator on line at 0549. The Turbine Overspeed Trip Test was performed at 1931 and the unit returned to service at 204[illegible]. Conclusion of 1RE04. Start of OPERATING CYCLE 5. (Refueling cycle includes unit down time for refueling; Breaker open to allow refueling through breaker open to allow

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
12/31/92	76.7	the following refueling. Operating cycle includes breaker closed following refueling through breaker open to allow refueling.)
01/01/93	296.3	
01/02/93	504.6	
01/03/93	561.3	
01/04/93	846.3	
01/05/93	995.4	Reactor power stabalized at 77 percent for Xenon stabalization.
01/06/93	998.8	
01/07/93	983.3	Reactor power reduced to 73 percent to facilitate work activities on the Incore and Excore nuclear detectors.
01/08/93	967.9	
01/09/93	984.6	Reactor power ascension to 90 percent commenced at 1735.
01/10/93	1,111.3	
01/11/93	1,141.3	Power ascension to 90 percent achieved at 0157. At 2128 reactor power reduced to 74.5 percent to allow calibration and troubleshooting of the nuclear instruments.
01/12/93	695.4	The unit was removed from service at 1859 and the reactor shutdown to comply with technical specification requirements. While reviewing surveillance procedure data for a steam pressure loop calibration, it was determined that some instrument setpoints had been improperly set due to deficiencies in the governing procedures. When the setpoints could not be corrected in the required time, a plant shutdown was initiated. Mode 2 was entered at 2136 and Mode 3 at 2146. (An Unusual Event was declared at 1551.)
01/13/93	0.0	The Unusual Event was terminated at 0130.
01/14/93	0.0	
01/15/93	0.0	
01/16/93	0.0	
01/17/93	0.0	The reactor was made critical at 1131 and the unit entered Mode 1 at 1446. Generator Breaker closure was attempted at 2105 but was delayed to allow corrective maintenance on the main generator voltage regulator.
01/18/93	0.0	
01/19/93	646.7	Generator synchronized to the grid at 0416 and power ascension to 90 percent commenced.
01/20/93	1,238.3	
01/21/93	1,240.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
01/22/93	1,241.7	
01/23/93	1,243.3	
01/24/93	1,261.3	
01/25/93	1,266.7	
01/26/93	1,272.1	Ascension to 98 percent reactor power achieved at 2204.
01/27/93	1,077.5	Reactor power reduction to 65 percent initiated due to oscillations on #13 Steam Generator Feedwater Pump Turbine. 65 percent reached at 1519 and corrective maintenance performed.
01/28/93	1,131.3	Reactor power ascension to 98 percent commenced at 0629 and reached at 1250. Reactor being maintained at 98 percent to allow performance of a flux map.
01/29/93	1,288.3	
01/30/93	1,287.5	
01/31/93	585.0	At 0001 reactor power was reduced to 28 percent to safely transfer the power panel that supplies 120 VAC to the steam generator feedwater pump control circuitry back to its normal supply. At 0700 power was at 28 percent and the panel was transferred back to its inverter at 0810. Reactor power increase to 98 percent was commenced at 1200.
02/01/93	1,206.7	During the performance of a surveillance test, Turbine-Driven Auxiliary Feedwater Pump (TDAFWP) 14 tripped on overspeed. This resulted in the pump being declared inoperable and the unit entering a 72 hour technical specification action statement.
02/02/93	1,292.5	
02/03/93	1,290.8	
02/04/93	639.6	The unit was removed from service and shutdown due to the failure to return Turbine-Driven Auxiliary Feedwater Pump (TDAFWP) 14 to operable status within the technical specification allowed time. On February 1, TDAFWP 14 tripped on overspeed. The cause of this overspeed trip and recent overspeed events was water intrusion into the TDAFWP adversely affecting performance. Corrective actions included extensive testing, analysis and component examination to determine the cause of overspeed trips on TDAFWP 14 and unit 2 TDAFWP 24. The unit was removed from service at 1422, entered Mode 2 at 1449, Mode 3 at 1514 and Mode 4 at 2132.
02/05/93	0.0	
02/06/93	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
02/07/93	0.0	
02/08/93	0.0	
02/09/93	0.0	
02/10/93	0.0	
02/11/93	0.0	Unit entered Mode 5 at 0143.
02/12/93	0.0	
02/13/93	0.0	
02/14/93	0.0	
02/15/93	0.0	
02/16/93	0.0	
02/17/93	0.0	
02/18/93	0.0	
02/19/93	0.0	
02/20/93	0.0	
02/21/93	0.0	
02/22/93	0.0	
02/23/93	0.0	
02/24/93	0.0	
02/25/93	0.0	
02/26/93	0.0	
02/27/93	0.0	
02/28/93	0.0	
03/01/93	0.0	
03/02/93	0.0	
03/03/93	0.0	
03/04/93	0.0	
03/05/93	0.0	The Reactor Coolant System was drained to mid-loop level to support steam generator primary manway leak repairs.
03/06/93	0.0	
03/07/93	0.0	
03/08/93	0.0	
03/09/93	0.0	
03/10/93	0.0	
03/11/93	0.0	
03/12/93	0.0	The primary manway leak repairs were completed and the RCS was

Date	Avg MWe-Gross	DAILY SUMMARY
03/12/93	0.0	filled to above the mid-loop level.
03/13/93	0.0	
03/14/93	0.0	
03/15/93	0.0	
03/16/93	0.0	
03/17/93	0.0	
03/18/93	0.0	
03/19/93	0.0	RCS fill and vent activities were completed and the system was repressurized.
03/20/93	0.0	
03/21/93	0.0	
03/22/93	0.0	During post-maintenance testing, one group of the reactor control rods, which includes four rods, did not indicate motion. Troubleshooting verified that the rod control system and the drive rod position indication system were operating properly. Investigation revealed that 36 of the 57 reactor control rod drive mechanisms (CRDMs) were stuck on the bottom of the core. If the control rods are driven to the bottom position and the Rod Control System is de-energized then the stationary grippers of the CRDM are engaged. If the plant is then cooled down, the core barrel contracts and raises the fuel assemblies and drive rods to where the drive rods contact the bottom of the stationary grippers which prevents the grippers from opening. This condition does not cause damage to the components or the CRDM and can be corrected by returning to normal operating temperature. Through additional attempts to move the control rods, 32 of the 36 rods were successfully freed. All of the CRDMs will be retested in Mode 3, followed by rod drop testing.
03/23/93	0.0	
03/24/93	0.0	RCS was depressurized to facilitate maintenance on the reactor coolant pump seal injection motor operated valves. A concern was identified regarding use of a load washer (strain gauge) manufactured by T-Hydronics for testing small motor operated valves where the normal size strain gauge would not fit. This instrument was used during 1RE04 on six valves and also used on seven Unit 2 valves during 2RE03. A controlled test was performed using the load washers which determined these devices to be installation sensitive, hence, less adequate for MOV testing. This required retesting of the MOVs that used this type of device. Unit 1 MOVs were retested using a stem mounted strain gauge and

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
03/24/93	0.0	Unit 2 MOVs will be tested during 2RE03.
03/25/93	0.0	
03/26/93	0.0	
03/27/93	0.0	
03/28/93	0.0	
03/29/93	0.0	
03/30/93	0.0	
03/31/93	0.0	
04/01/93	0.0	
04/02/93	0.0	
04/03/93	0.0	
04/04/93	0.0	
04/05/93	0.0	
04/06/93	0.0	
04/07/93	0.0	High Head Safety Injection Pumps 1A and 1C were removed from service for corrective maintenance. Lubricating oil sample analysis of the 1A pump motor indicated the presence of elevated levels of bearing wear particles in the lower bearing sump. The pump was removed from service to allow for bearing replacement. Lubricating oil sample analysis of the 1C HHSI pump motor indicated the presence of dirt particles in the upper bearing sump. The oil was drained to rid the sump of the contamination. Following the required post maintenance test run, the subsequent oil sample was unsatisfactory due to the continued presence of foreign material. Maintenance was performed to clean and inspect the oil reservoir. During the maintenance, the upper bearing oil sump gauge glass was removed for cleaning. The gauge was reinstalled upside down. The upper bearing was filled to reach the desired level on the sight glass with approximately 20 quarts of oil instead of the specified 11 quarts. This resulted in the overflowing of the oil into the motor windings.
04/08/93	0.0	
04/09/93	0.0	
04/10/93	0.0	
04/11/93	0.0	
04/12/93	0.0	
04/13/93	0.0	
04/14/93	0.0	

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UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
04/15/93	0.0	
04/16/93	0.0	
04/17/93	0.0	
04/18/93	0.0	
04/19/93	0.0	
04/20/93	0.0	
04/21/93	0.0	
04/22/93	0.0	
04/23/93	0.0	
04/24/93	0.0	
04/25/93	0.0	
04/26/93	0.0	
04/27/93	0.0	
04/28/93	0.0	
04/29/93	0.0	
04/30/93	0.0	
05/01/93	0.0	
05/02/93	0.0	
05/03/93	0.0	
05/04/93	0.0	
05/05/93	0.0	
05/06/93	0.0	
05/07/93	0.0	
05/08/93	0.0	
05/09/93	0.0	
05/10/93	0.0	
05/11/93	0.0	
05/12/93	0.0	
05/13/93	0.0	
05/14/93	0.0	
05/15/93	0.0	
05/16/93	0.0	
05/17/93	0.0	
05/18/93	0.0	
05/19/93	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
05/20/93	0.0	
05/21/93	0.0	
05/22/93	0.0	
05/23/93	0.0	
05/24/93	0.0	
05/25/93	0.0	
05/26/93	0.0	
05/27/93	0.0	
05/28/93	0.0	
05/29/93	0.0	
05/30/93	0.0	
05/31/93	0.0	
06/01/93	0.0	
06/02/93	0.0	
06/03/93	0.0	
06/04/93	0.0	
06/05/93	0.0	
06/06/93	0.0	
06/07/93	0.0	
06/08/93	0.0	
06/09/93	0.0	
06/10/93	0.0	
06/11/93	0.0	
06/12/93	0.0	
06/13/93	0.0	
06/14/93	0.0	
06/15/93	0.0	
06/16/93	0.0	
06/17/93	0.0	
06/18/93	0.0	
06/19/93	0.0	
06/20/93	0.0	
06/21/93	0.0	
06/22/93	0.0	
06/23/93	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
06/24/93	0.0	
06/25/93	0.0	
06/26/93	0.0	
06/27/93	0.0	
06/28/93	0.0	
06/29/93	0.0	
06/30/93	0.0	
07/01/93	0.0	
07/02/93	0.0	
07/03/93	0.0	
07/04/93	0.0	
07/05/93	0.0	
07/06/93	0.0	
07/07/93	0.0	
07/08/93	0.0	
07/09/93	0.0	
07/10/93	0.0	
07/11/93	0.0	
07/12/93	0.0	
07/13/93	0.0	
07/14/93	0.0	
07/15/93	0.0	
07/16/93	0.0	
07/17/93	0.0	
07/18/93	0.0	
07/19/93	0.0	
07/20/93	0.0	
07/21/93	0.0	
07/22/93	0.0	
07/23/93	0.0	
07/24/93	0.0	
07/25/93	0.0	
07/26/93	0.0	
07/27/93	0.0	
07/28/93	0.0	

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Date	Avg MWe-Gross	DAILY SUMMARY
07/29/93	0.0	
07/30/93	0.0	
07/31/93	0.0	
08/01/93	0.0	
08/02/93	0.0	
08/03/93	0.0	
08/04/93	0.0	
08/05/93	0.0	Unit entered Mode 4 at 1854.
08/06/93	0.0	
08/07/93	0.0	
08/08/93	0.0	
08/09/93	0.0	
08/10/93	0.0	
08/11/93	0.0	
08/12/93	0.0	Unit entered Mode 3 at 2043.
08/13/93	0.0	Unit entered Mode 4 at 2215.
08/14/93	0.0	Unit entered Mode 5 at 0500.
08/15/93	0.0	
08/16/93	0.0	
08/17/93	0.0	
08/18/93	0.0	
08/19/93	0.0	
08/20/93	0.0	
08/21/93	0.0	
08/22/93	0.0	
08/23/93	0.0	
08/24/93	0.0	
08/25/93	0.0	
08/26/93	0.0	
08/27/93	0.0	
08/28/93	0.0	
08/29/93	0.0	
08/30/93	0.0	
08/31/93	0.0	
09/01/93	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
09/02/93	0.0	
09/03/93	0.0	
09/04/93	0.0	
09/05/93	0.0	
09/06/93	0.0	
09/07/93	0.0	
09/08/93	0.0	
09/09/93	0.0	
09/10/93	0.0	
09/11/93	0.0	
09/12/93	0.0	
09/13/93	0.0	
09/14/93	0.0	
09/15/93	0.0	
09/16/93	0.0	
09/17/93	0.0	
09/18/93	0.0	
09/19/93	0.0	
09/20/93	0.0	
09/21/93	0.0	
09/22/93	0.0	
09/23/93	0.0	
09/24/93	0.0	
09/25/93	0.0	
09/26/93	0.0	
09/27/93	0.0	
09/28/93	0.0	
09/29/93	0.0	
09/30/93	0.0	
10/01/93	0.0	
10/02/93	0.0	
10/03/93	0.0	
10/04/93	0.0	
10/05/93	0.0	
10/06/93	0.0	

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UNIT 1

Date	Avg MW Gross	DAILY SUMMARY
10/07/93	0.0	
10/08/93	0.0	
10/09/93	0.0	
10/10/93	0.0	
10/11/93	0.0	
10/12/93	0.0	
10/13/93	0.0	
10/14/93	0.0	
10/15/93	0.0	
10/16/93	0.0	
10/17/93	0.0	
10/18/93	0.0	
10/19/93	0.0	
10/20/93	0.0	
10/21/93	0.0	
10/22/93	0.0	
10/23/93	0.0	
10/24/93	0.0	
10/25/93	0.0	
10/26/93	0.0	
10/27/93	0.0	
10/28/93	0.0	
10/29/93	0.0	
10/30/93	0.0	
10/31/93	0.0	
11/01/93	0.0	
11/02/93	0.0	
11/03/93	0.0	
11/04/93	0.0	
11/05/93	0.0	
11/06/93	0.0	
11/07/93	0.0	
11/08/93	0.0	
11/09/93	0.0	
11/10/93	0.0	

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Date	Avg MWe-Gross	DAILY SUMMARY
11/11/93	0.0	
11/12/93	0.0	
11/13/93	0.0	
11/14/93	0.0	
11/15/93	0.0	
11/16/93	0.0	
11/17/93	0.0	
11/18/93	0.0	The unit entered Mode 6 at 1302.
11/19/93	0.0	
11/20/93	0.0	
11/21/93	0.0	
11/22/93	0.0	
11/23/93	0.0	
11/24/93	0.0	
11/25/93	0.0	
11/26/93	0.0	The unit entered Mode 5 at 1218.
11/27/93	0.0	
11/28/93	0.0	
11/29/93	0.0	
11/30/93	0.0	
12/01/93	0.0	
12/02/93	0.0	
12/03/93	0.0	
12/04/93	0.0	
12/05/93	0.0	
12/06/93	0.0	
12/07/93	0.0	
12/08/93	0.0	
12/09/93	0.0	
12/10/93	0.0	
12/11/93	0.0	
12/12/93	0.0	
12/13/93	0.0	
12/14/93	0.0	
12/15/93	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
12/16/93	0.0	
12/17/93	0.0	
12/18/93	0.0	
12/19/93	0.0	
12/20/93	0.0	
12/21/93	0.0	
12/22/93	0.0	
12/23/93	0.0	
12/24/93	0.0	
12/25/93	0.0	
12/26/93	0.0	
12/27/93	0.0	
12/28/93	0.0	
12/29/93	0.0	
12/30/93	0.0	
12/31/93	0.0	
01/01/94	0.0	
01/02/94	0.0	
01/03/94	0.0	
01/04/94	0.0	
01/05/94	0.0	
01/06/94	0.0	
01/07/94	0.0	
01/08/94	0.0	
01/09/94	0.0	
01/10/94	0.0	
01/11/94	0.0	
01/12/94	0.0	
01/13/94	0.0	
01/14/94	0.0	
01/15/94	0.0	
01/16/94	0.0	
01/17/94	0.0	
01/18/94	0.0	
01/19/94	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
01/20/94	0.0	
01/21/94	0.0	
01/22/94	0.0	
01/23/94	0.0	
01/24/94	0.0	
01/25/94	0.0	
01/26/94	0.0	
01/27/94	0.0	
01/28/94	0.0	
01/29/94	0.0	
01/30/94	0.0	
01/31/94	0.0	
02/01/94	0.0	
02/02/94	0.0	
02/03/94	0.0	
02/04/94	0.0	
02/05/94	0.0	
02/06/94	0.0	The unit entered Mode 4 at 0533.
02/07/94	0.0	
02/08/94	0.0	The unit entered Mode 3 at 1800.
02/09/94	0.0	
02/10/94	0.0	
02/11/94	0.0	
02/12/94	0.0	
02/13/94	0.0	
02/14/94	0.0	
02/15/94	0.0	The Nuclear Regulatory Commission acknowledged completion of the Confirmatory Action Letter issues associated with Unit 1 and authorized the restart of the unit in accordance with the startup and power ascension plan.
02/16/94	0.0	
02/17/94	0.0	The unit entered Mode 2 at 2252.
02/18/94	0.0	The reactor was made critical at 0023.
02/19/94	0.0	
02/20/94	0.0	
02/21/94	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
02/22/94	0.0	The unit entered Mode 1 at 0039.
02/23/94	0.0	
02/24/94	0.0	
02/25/94	78.8	The unit was returned to service at 0121. The main turbine over-speed trip test was performed and the main generator was tripped at 1109. The unit was returned to service at 1700.
02/26/94	150.4	
02/27/94	138.3	
02/28/94	144.2	The unit was in the power ascension testing program, operating at 28% reactor power, when a manual reactor trip was initiated due to dropping water level in the 1D Steam Generator (S/G). The event was caused by a malfunctioning 1D Main Feedwater Regulating Valve (MFRV), which prevented the secondary reactor operator from taking remote manual control of the level in the S/G. The malfunctioning 1D MFRV was caused by a failure of the torque motor in the electro-pneumatic (I/P) converter which resulted in a closure failure of the 1D MFRV and subsequent loss of feedwater flow to 1D S/G. The failed I/P converter was replaced. Following the unit's removal from service, a small leak was discovered from the primary to the secondary side of the 1C S/G. The unit was taken to Mode 5 to facilitate the location and repair of the leak. A pressure test of the secondary side was performed and verified only one tube leak. Testing of the surrounding tubes identified no other leaks. The leaking tube was bored out and a remote plug was inserted.
03/01/94	0.0	
03/02/94	0.0	
03/03/94	0.0	Unit entered Mode 4 at 0113.
03/04/94	0.0	Unit entered Mode 5 at 0018.
03/05/94	0.0	
03/06/94	0.0	
03/07/94	0.0	
03/08/94	0.0	
03/09/94	0.0	
03/10/94	0.0	
03/11/94	0.0	
03/12/94	0.0	
03/13/94	0.0	

UNIT 1

Date	Avg MWe-Gross	DAILY SUMMARY
03/14/94	0.0	
03/15/94	0.0	
03/16/94	0.0	
03/17/94	0.0	
03/18/94	0.0	
03/19/94	0.0	Unit entered Mode 4 at 0257 and Mode 3 at 1517.
03/20/94	0.0	Unit entered Mode 2 at 2305.
03/21/94	0.0	Reactor made critical at 0114 and entered Mode 1 at 1917.
03/22/94	43.3	Generator on line at 1731
03/23/94	316.7	
03/24/94	553.3	Unit deration classified as unplanned until 29% reactor power achieved and surpassed, (Power level prior to the manual trip/forced outage) in which the unit continues the scheduled power ascension testing program.
03/25/94	571.7	
03/26/94	569.2	
03/27/94	574.2	
03/28/94	647.1	
03/29/94	907.9	
03/30/94	1,001.7	
03/31/94	1,014.2	
04/01/94	1,011.3	
04/02/94	1,163.3	
04/03/94	1,196.1	
04/04/94	1,095.0	
04/05/94	1,102.5	
04/06/94	1,201.3	
04/07/94	1,270.4	
04/08/94	1,322.1	Unit achieved 100% reactor power at 0048.
04/09/94	1,325.0	
04/10/94	1,322.9	
04/11/94	1,323.3	
04/12/94	1,322.9	
04/13/94	1,322.9	
04/14/94	1,323.8	
04/15/94	1,322.9	

ATTACHMENT 3

DIESEL GENERATOR

STARTS LOG

(2 pages)

ATTACHMENT 3
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DIESEL GENERATOR STARTS LOG

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Diesel Generator #11

START NUMBER	START DATE	START TIME	SIGNAL TYPE: AUTO	SIGNAL TYPE: TEST	SIGNAL TYPE: NONE	SPEED, VOLTAGE & FREQUENCY OBTAINED IN ≤ 10 SECONDS: YES	NO	TIME OF LOAD	LOAD IN KW	END DATE	END TIME	RESULT: NO TEST	RESULT: VALID TEST	RESULT: VALID FAILURE	RESULT: NON VALID FAILURE	POST SAMPLE T.S.4.8.1.1.2.B	REMARKS: Use as many lines as necessary to show how the determination of result was arrived at. Refer to the flow chart steps by location letter, in order to trace the decision. List problem report and service request numbers.	S. S. REVIEW	PED REVIEW	DATE	TEST FREQUENCY
389	1-5-94	0511		✓		✓		0656	5600	1-5-94	0545		✓			✓	A, E, G, H, J, K, L, M	[illegible]	[illegible]	[illegible]	M
—								—	—								Started D/G #11 per OPSP03-DG-0001, Run ≥ 2750 for 1 Hour No Problems occured				
—																					
—																					
(*) 390	2-3-94	0204		✓			✓	—	—	2 -94	0210			✓ (*)		NA	A, E, G, V, W, X, Y, Z STARTED DG #11 PER OPSP03-DG-0001. NO VOLT OR FREQ LOCALLY or in CR. SR# 305619 GENERATED, SPR INITIATED.	[illegible]	[illegible]	[illegible]	W
—																					
—																					
—																					
(*) 391	2-4-94	0112		✓		✓		0136	1400	2-4-94	0208	✓				NA	A, B, C, D PMT RUN				
(*) 392	2-4-94	0229		✓		✓		—	—	2-4-94	0230	✓				NA	A, B, C, D PMT RUN	[illegible]	[illegible]	[illegible]	W
(*) 393	2-4-94	0310		✓		✓		0327	1400	2-4-94	0750		✓			✓	A, E, G, H, J, K, L, M	[illegible]	[illegible]	[illegible]	W
~~394~~	~~2-6-94~~	~~0650~~		✓		✓		~~1136~~	~~5600~~	~~2-6-94~~	~~1426~~	MAN	~ 2-6-94				Performed OPSP03-DG-0001 (monthly)	[illegible]	[illegible]	[illegible]	W
(*) 394	2-6-94	0950		✓		✓		1135	5600	2-6-94	1426		✓			✓	AEGHJKLM				
(*) 395	2-16-94	0556		✓		✓		0748	5600	2-16-94	1040		✓			✓	A, B, E, G, H, J, K, L, M	[illegible]	[illegible]	[illegible]	W
																	INCREASE Freq OPSP03-DG-0001	[illegible]	[illegible]	[illegible]	W
2/23/94 ~~396~~ 397	2-23-94	0657		✓		✓		0706	5600	2-23-94	0858		✓			✓	A, E, G, H, J, K, L, M. / Performed SEMI-ANNUAL OPSP03-DG-0001	[illegible]	[illegible]	[illegible]	W

This form, when completed, SHALL be retained for the life of the plant.

(*) See note on next page GAH 4/4/94

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DIESEL GENERATOR STARTS LOG

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Diesel Generator #11

START NUMBER	START DATE	START TIME	SIGNAL TYPE: AUTO	SIGNAL TYPE: TEST	SIGNAL TYPE: NONE	SPEED, VOLTAGE & FREQUENCY OBTAINED IN ≤ 10 SECONDS: YES	SPEED, VOLTAGE & FREQUENCY OBTAINED IN ≤ 10 SECONDS: NO	TIME OF LOAD	LOAD IN KW	END DATE	END TIME	RESULT: NO TEST	RESULT: VALID TEST	RESULT: VALID FAILURE	RESULT: NON VALID FAILURE	FOST SAMPLE T.S.4.8.1.1.2.B	REMARKS Use as many lines as necessary to show how the determination of result was arrived at. Refer to the flow chart steps by location letter, in order to trace the decision. List problem report and service request numbers.	S. S. REVIEW	PED REVIEW	DATE	TEST FREQUENCY
(*) 397	3-1-94	1106		✓			✓	—	—	3-1-94	1115			✓ (*)		(1) N/A	A, E, G, V, W, X, Y, Z - Started DG per OPSP03-DG-0001 - Came up to speed but Volts + Freq failed to generate - Locally or in C.R. SR#305641 written + SPR Generated - Also see start # 390 on 2-3-94 for same problem	[illegible]	[illegible]	[illegible]	W
		(1) N/A For FOST Sample Due to failure - FOST Sample will be performed During ESF D/G #11 return to service + subsequent surveillance																[illegible]			
398	3-3-94	1221		✓		✓		—	—	3-3-94	1237	✓					A, B, C, D STARTED D/G #11 FOR MM TESTING K-1 RELAY - SAT	[illegible]	[illegible]	[illegible]	W
399	3-3-94	1315		✓		✓		1331	1375	3-3-94	1408	✓					A, B, C, D STARTED D/G #11 FOR MM TESTING IN EMERG FOR K-1 RELAY - SAT	[illegible]	[illegible]	[illegible]	W
400	3-4-94	0613		✓		✓		0630	1400	3-4-94	0724	✓				✓	A, B, C, D; ~~For Retake~~ 2386	[illegible]	[illegible]	[illegible]	W

This form, when completed, SHALL be retained for the life of the plant.

3-4-94
~~920250~~ 940250

(*) Start group 390 - 397 Considered a single failure. See SPR ~~921455 and notification Letter to NRC ST-HL-AE-4735~~. All starts in this group are counted as one failure.

[illegible] 4/14/94

[illegible] 4/14/94

ATTACHMENT 4

GENERIC IMPLICATIONS

OF EVENT

ATTACHMENT 4
ST-HL-AE-5084
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GENERIC IMPLICATIONS

These Potter Brumfield voltage release relays are used in all six diesel generators on site. Historically these relays have been very reliable. Other sites that have this type of voltage release relays have been contacted, and report that they have had no problems with these relays.

The degree of discoloration of the moving contact arm on the VR1 relay indicates that this condition has developed over some period of time. This can be correlated to the number of operations that the relay has seen, since each operation would weaken the relay's spring thereby decreasing contact pressure. The discoloration of the moving contact arm can be easily seen through the clear case of the relay. All other diesel's VR relays have been closely inspected for this same condition. All were found satisfactory, with no indications of overheated or damaged contacts. Additionally, the weaker stainless steel springs can be identified through the clear plastic case. All diesels have been inspected for these stainless steel springs. The only diesels that have these springs are DG-22 and DG-23. Service Request DG-313366 and DG-313367 have been written to replace these relays at the next available opportunity. All VR relays installed, including those in DG-22 and DG-23, were visually inspected, and found to have no indication of damaged contacts. Since the causal factor is also dependent on relay operations, the start counters for each of the diesels were surveyed. DG-11 had the greatest number of circuit operations by far at 2830 successful operations. DG-22 had 823 and DG-23 had 508 starts tallied.

Since its initial adjustment in startup the K1 contactor has performed reliably. Six problems involving the K1 has occurred on site since 1988. Only three of these can be attributed to the device itself (see K1 relay History attached). Additional sites were checked and found to have even fewer problems, with Palo Verde reporting that they have had no problems since they were set up during startup.

VI CORRECTIVE ACTIONS

R.1 Service Requests have been initiated for near term replacement of all remaining VR relays in the diesel generators at the next opportunity.