UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 1994

Docket No. 50-293

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Boulette:

SUBJECT: PRELIMINARY ACCIDENT SEQUENCE PRECURSOR (ASP) ANALYSIS OF PILGRIM EVENT FOR LICENSEE PEER REVIEW

Enclosed is a copy of the preliminary ASP Program analysis of an operational event which occurred at the Pilgrim plant on March 13, 1993. The preliminary results of the analysis of this event indicate that it may be a precursor event for 1993 (Enclosure 1).

In recent years, licensees of U.S. nuclear power plants have added safety equipment, and have improved plant and emergency operating procedures. Some of these changes, particularly those involving use of alternate equipment or recovery actions in response to specific accident scenarios, are not currently incorporated in the basic ASP models. Consequently, the ASP estimates of core damage probabilities could be conservative for certain accident sequences. To address this issue, we are providing each preliminary ASP analysis to the pertinent plant licensee for peer review. The licensee is requested to review and comment on the technical adequacy of the analyses, including the depiction of their plant equipment and equipment capabilities. We will then evaluate the comments received during this peer review for reasonableness and pertinence to the ASP analysis in an attempt to use best estimate values. Upon completion of this evaluation, we will revise the conditional core damage probability calculations where necessary to consider information provided by the licensee during the review. The object of the peer review process is to provide as realistic an analysis of the significance of the event as possible.

This year, we are sending the preliminary analyses out for peer review as they are completed, rather than in a batch mode, as was done with the 1992 events reviewed last year. The analysis of the Pilgrim event represents the first completed preliminary ASP analysis of a 1993 event. In order to maintain our schedule for issuance of the 1993 Precursor Report, the licensee is requested to complete their review and provide their comments within 60 days from the date that they receive your letter. In order to facilitate the licensee's review, we are providing the following additional enclosures: Guidance for Licensee Peer Review of Preliminary ASP Analysis (Enclosure 2), Licensee Event Report (LER) No. 293/93004 (Enclosure 3), Accident Sequence Precursor Identification and Quantification (Enclosure 4), and Appendix A ASP Models (Enclosure 5).

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We appreciate your timely review and comments. No new OMB clearance is needed for the ASP peer review process, as it is covered by the existing OMB clearance addressing staff follow up review of events documented in LERs. If you have any questions about the ASP Program Peer Review process or any of the enclosures, please contact either Dr. Dale Rasmuson or Dr. Patrick O'Reilly directly. Dr. Rasmuson can be reached on (301) 492-4490, and Dr. O'Reilly can be reached on (301) 492-8858.

Sincerely,

original signed by
Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Preliminary ASP Analysis of LER 293/93-004
2. Guidance for Licensee Peer Review of Preliminary ASP Analysis
3. Licensee Event Report 293/93-004
4. Accident Sequence Precursor Identification and Quantification
5. ASP Models, Appendix A to NUREG/CR-4674, Volume 17

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ENCLOSURE 1

0.1 LER Number 293/93-004

Event Description: Weather-Induced LOOP, Vessel P/T Limits Violated

Date of Event: March 13, 1993

Plant: Pilgrim

0.1.1 Summary

Pilgrim was operating at 100% power when a severe coastal storm caused a loss-of-load scram and loss of normal power supply to the plant. Failures in the 120-Vac control power system prevented automatic operation of plant service and cooling water systems. Difficulties were experienced during cooldown, during which time the reactor repressurized to at least 820 psig, with vessel bottom head temperature declining to around 110°F. The conditional core damage probability estimated for this event is 2.5×10^{-5}. The relative significance of this event compared to other postulated events at Pilgrim is shown in Fig. X1.

0.1.2 Event Description

Pilgrim experienced a load rejection and scram from 100% power when wind-driven snow and ice accumulated on switchyard insulators, causing a fault. Circuit breakers opened to isolate the main and unit auxiliary transformers. Most loads fed from the unit auxiliary transformer (UAT) fast-transferred to the alternate source, the startup transformer (SUT). However, a breaker control failure prevented 4160-Vac bus A3 from transferring. Loads fed from bus A3 including the "A" recirculation pump motor-generator set, the "A" circulating water pump, the "A" main turbine auxiliary oil pump, and 480-Vac bus B3, were deenergized. Deenergization of 480-Vac bus B3, in turn, removed power from reactor protection system (RPS) bus "A". Operators closed breaker 304 to reenergize bus A3 from the SUT.

Protective breakers for 120-Vac safeguard buses "A" and "B" tripped due to improper trip settings. As a result, the auto-starts for the salt service water (SSW) system and reactor building closed cooling water (RBCCW) system pumps were disabled. Manual operation of these pumps was not affected.

Approximately 12 min after the scram, a fault on one of two feeders to the SUT occurred, and the feeder was isolated. As a precaution, EDGs "1" and "2" were started and connected to their respective buses. RPS bus A was energized via the standby RPS transformer, fed from EDG 1. At 1710 hours (42 min after the trip), the remaining off-site supply to the SUT isolated, and nonsafety buses A1, A2, A3, and A4 deenergized.

Approximately 5 h after the start of the event, one off-site line to the SUT was reenergized, and operators began returning the plant to a normal shutdown lineup.

Reactor vessel pressure began to rise at about 10 psi/min, and vessel level rose to the high-level trip setpoint for HPCI, RCIC, and the MSIVs. When reactor pressure reached 572 psig, another scram resulted. Reactor vessel bottom head temperature was noted at 110°F. Suppression pool inventory was routed to radwaste via the RHR suppression pool cooling system. With reactor pressure approximately 900 psig, relief valves were opened to reduce pressure to about 450 psig. HPCI and RCIC were placed in service for vessel pressure and level control, respectively. Approximately 2 h after actuation of the relief valves, the vessel P/T limit was no longer exceeded, with pressure at 320 psig and bottom head temperature at 92°F. The plant then proceeded to cold shutdown.

0.1.3 Additional Event-Related Information

Pilgrim has four nonsafety-related 4160 Vac busses (See Fig. X2). Each of these nonsafety-related busses can be powered from (1) the Unit Auxiliary Transformer (UAT) which is energized by the main generator output or (2) the Startup Transformer (SUT; which is connected to two off-site 345 kV lines. Upon loss of the UAT, the nonsafety-related busses are automatically fast transferred to the SUT. The two safety-related 4160 Vac busses can also receive power from the UAT and SUT. In addition, they can be powered from (1) a 23 kV off-site line or (2) from the blackout diesel generator (BODG). The BODG is a nonsafety-related supply which is not dependent upon any other on-site systems for its operation. It can be started manually from the control room and is capable of providing power to one of the two safeguards busses and associated loads for blackout events without a concurrent LOCA event. Upon loss of the UAT, the safety-related busses are fast transferred to the SUT. If the SUT is lost, the busses automatically load onto the safeguards EDGs. The BODG is manually aligned to one of the safeguards busses and loaded as required. Each of the 4160 Vac busses supply a number of 480 Vac busses.

0.1.4 Modeling Assumptions

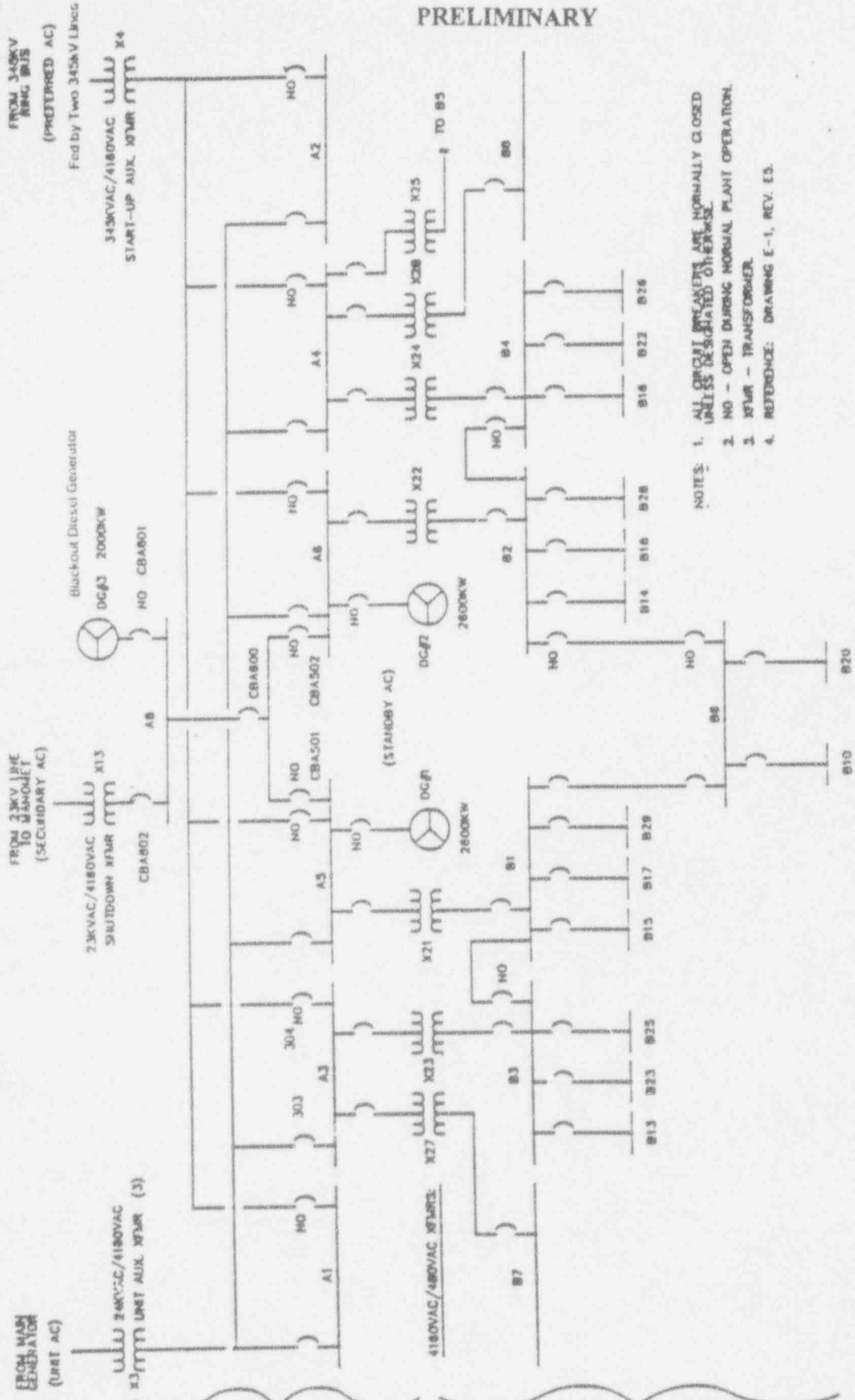
This event was modeled as a severe weather induced loss of off-site power since it was caused by a widespread ice storm. This is consistent with the categorization of LOOPs in NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*. The short-term LOOP nonrecovery probability and long-term nonrecovery of emergency power values were both modified using the models described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89-11, August 1989. These models are based on the results of the data distributions contained in NUREG-1032. The results of this yield a short-term nonrecovery (within the first 30 min) of 0.9 and a long-term nonrecovery of 5.5E-2. The BODG described in the previous section was included in the modeling. According to the Pilgrim IPE, the operator failure to align the blackout diesel is 0.05 (page A1-27). The probability that power is not available from the BODG is 7.5E-2 (Table B.10-1). This results in a total failure rate of the BODG of 0.125 (.05 + .075). The end states and sequences following loss of emergency power, and successful reactor trip are identical for recovery of off-site power and successful loading of the BODG. Therefore, the long-term nonrecovery value was multiplied by the failure probability of the BODG. This results in a long-term nonrecovery of 6.9E-3 (5.5E-2 from the ORNL Model × 0.125 for the BODG).

Because the loss of 120 volt ac safeguards buses would have prevented auto-start of the SSW and RBCCW systems, these systems were assumed to require operator recovery when demanded. Since this recovery could be performed from the control room and was procedurally based, a nonrecovery value of 0.04 was assigned in accordance with the methods discussed in Section A.3.2 of NUREG/CR-4674, Volume 17 (*Precursors to Potential Core Damage Accidents: 1992 A Status Report*). Cooling water system design at Pilgrim differs from some other BWRs in that the residual heat removal (RHR) heat exchangers are cooled by RBCCW instead of by raw water, and cooling for the RBCCW system is provided by SSW. Therefore, the RHR service water function incorporated in the ASP model for this class of plant represents the SSW/RBCCW function for Pilgrim.

No analytical evaluation was made of potential consequences of the reactor vessel repressurization which occurred during this event.

0.1.5 Analysis Results

The conditional core damage probability estimated for this event is 2.5×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Fig. 1, involves failure of emergency power and failure to recover emergency power before battery depletion.



Simplified Diagram of the Pilgrim Station 4160V and 480V AC Power Distribution System

FIG. X2

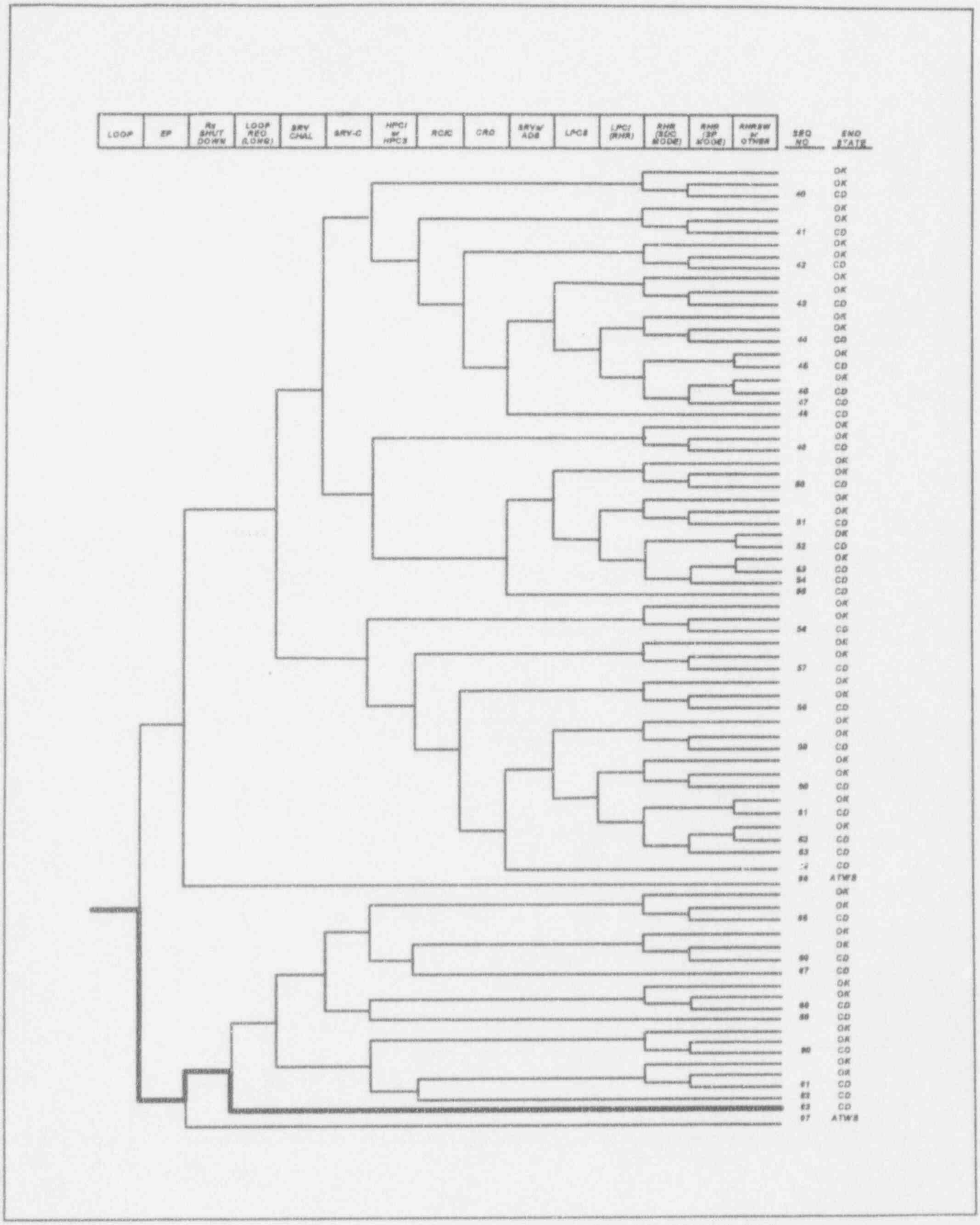


Fig. 1. Dominant core damage sequence for LER 293/93-004.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 293/93-004
Event Description: Weather Induced LOOP
Event Date: March 13, 1993
Plant: Pilgrim 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	9.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	2.5E-05
Total	2.5E-05
ATWS	
LOOP	2.7E-05
Total	2.7E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
83	LOOP emerg.power -rx.shutdown/ep EP.REC	CD	1.4E-05	9.0E-02
40	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	4.8E-06	1.0E-01
55	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close hpci srv.ads	CD	3.0E-06	4.5E-01
67	LOOP emerg.power -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	1.7E-06	3.5E-01
69	LOOP emerg.power -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	5.5E-07	5.0E-01
98	LOOP -emerg.power rx.shutdown	ATWS	2.7E-05	9.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
40	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	4.8E-06	1.0E-01
55	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close hpci srv.ads	CD	3.0E-06	4.5E-01
98	LOOP -emerg.power rx.shutdown	ATWS	2.7E-05	9.0E-01
67	LOOP emerg.power -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	1.7E-06	3.5E-01
69	LOOP emerg.power -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	5.5E-07	5.0E-01
83	LOOP emerg.power -rx.shutdown/ep EP.REC	CD	1.4E-05	9.0E-02

** non-recovery credit for edited case

Event Identifier: 293/93-004

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL: s:\asp\prog\models\pilgrim.sll
PROBABILITY FILE: s:\asp\prog\models\bwr_csll.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	5.5E-04	1.0E+00	
LOOP	2.0E-05 > 2.0E-05	4.3E-01 > 9.0E-01	
Branch Model: INITOR			
Initiator Freq:	2.0E-05		
loca	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	1.7E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	1.3E-02	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
EP.REC	3.1E-02 > 5.5E-02	1.0E+00 > 1.3E-01[1]	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	3.1E-02 > 5.5E-02		
fw/pcs.trans	2.9E-01	3.4E-01	
fw/pcs.loca	4.0E-02	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	3.0E-03	3.4E-01	
lpci(rhr)/lpcs	1.0E-03	7.1E-01	
rhr(sdc)	2.1E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
RHRSW	2.0E-02 > 2.0E-02	3.4E-01	2.0E-03 > 4.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	2.0E-02		

* branch model file
** forced

NOTES

1 This accounts for the BODG.

Event Identifier: 293/93-004

ENCLOSURE 2

GUIDANCE FOR LICENSEE PEER REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event which occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP rogram uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include loss of off-site power (LOOP), Loss-of-Coolant Accident (LOCA), degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences. This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and the licensee event report (LER) for this event. These sources are identified in the write-up documenting the analysis. The analysis methodology followed the process described in Section 2.1 and Appendix A of Volume 17 of NUREG/CR-4674, copies of which have been provided in this package for your use in this review.

Guidance for Peer Review and Criteria for Recovery Credit

The review of the preliminary analysis should use Section 2.1 and Appendix A of NUREG/CR-4674 for guidance. Comments regarding the analysis should address:

- Characterization of possible plant response,
- Representation of expected plant response used in the analytical models,
- Representation of plant safety equipment configuration and capabilities at the time of the event, and
- Assumptions regarding equipment recovery probabilities.

If you desire credit for plant features or recovery measures that were not considered in our preliminary analysis of this event (e.g., the use of additional systems, equipment, or specific actions), your request must be supported by appropriate, documented information that will allow us to reanalyze the event in the light of the information you provide. The identified plant features or recovery measures must have existed at the time of the event, and should include:

- Normal or emergency operating procedures,
- Piping and instrumentation diagrams (P&IDs),
- Electrical one-line diagrams,
- Results of thermal-hydraulic analysis,
- Operator training (both procedures and simulator), etc.
- Plant-specific system reliability - supporting information should include the basis for the stated reliability value (method of determining the system's reliability, available data used in determination, etc.)

Also, the documentation should address the impact of the use of the specific recovery measure on:

- The sequence of events,
- The timing of events,
- The probability of operator error in using the system or equipment, and
- Other systems/processes already modeled in the analysis.

For example, Plant A (a PWR) experiences a reactor trip and, during the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW train modeling would be patterned after information gathered either from the plant PSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be mitigated by the use of the standby feedwater system. The mitigation effect for the standby feedwater system would be credited in the analysis provided that the standby feedwater system characteristics are documented in the FSAR, accounted for in the IPE, procedures for using the system during recovery existed at the time of the event, the plant operators had been trained in the use of the system prior to the event, a clear diagram (one-line diagram or better) of the system is available, previous analyses have indicated that there would be sufficient time available to implement the procedure successfully, and results of an assessment that evaluates the effect that use of the standby feedwater system has on already existing processes of procedures that would normally be used to deal with the event are available.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the operational event:

- The specific licensee event report (LER), augmented inspection team AIT) report, or other pertinent reports as appropriate (separate enclosure).

- A calculation summary sheet indicating the dominant sequences and pertinent aspects of the modeling details (contained in the analysis writeup).

- An event tree with the dominant sequence(s) highlighted (contained in the analysis writeup).

- A copy of Section 2.1 and Appendix A of NUREG/CR-4674, Volume 17 (separate enclosures).

ENCLOSURE 3