Docket File

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

Docket No. 50-331

May 12, 1994

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

SUBJECT: DUANE ARNOLD ENERGY CENTER - TRANSMITTAL OF PRELIMINARY ASP ANALYSIS FOR LICENSEE PEER REVIEW

Enclosed is a copy of the preliminary Accident Sequence Precursor (ASP) Program analysis of an operational event which occurred at the Duane Arnold Energy Center on October 26, 1993. The preliminary results of our contractor's (ORNL) analysis of this event indicate that it may be a precursor event for 1993. The purpose of this letter is to request your review and comment on the technical adequacy of the analyses, including the depiction of the plant equipment and equipment capabilities.

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 and to the NRC staff for Peer Review. 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.

In order to maintain our schedule for issuance of the 1993 Precursor Report, we request that you complete your review and provide your comments within 30 days from the date you receive this letter. In order to facilitate your review, we have also enclosed several items for guidance. Enclosure 2: (1) contains specific guidance for the Peer Review, (2) identifies the criteria which we will apply to determine whether any credit should be given in the analysis for the use of licensee-identified additional equipment or specific actions in recovering from the event, and (3) describe the specific information that should be provided by you to support such a claim. Enclosure 3 is the licensee event report (LER) documenting the subject event.

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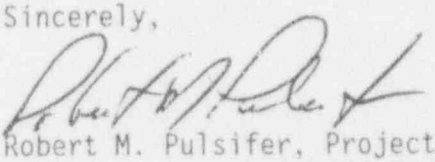
Enclosures 4 and 5 contain background information regarding the ASP methodology which may be useful to you in reviewing the analysis. Enclosure 4, which is Section 2.0 from the 1992 ASP Annual Precursor Report, describes the precursor event identification and quantification process. Enclosure 5, which is Appendix A from the same report, describes the ASP models used in precursor analyses.

We appreciate your assistance in the review of this preliminary report. No new OMB clearance is needed for the ASP Peer Review process, since the process is already covered by the existing OMB clearance addressing staff followup 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. Dr. Rasmuson can be reached on (301) 492-4490, and Dr. O'Reilly can be reached on (301) 492-8858.

Sincerely,



Robert M. Pulsifer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Preliminary Accident Sequence Precursor Program
2. Guidance for Licensee Peer Review of Preliminary ASP Analysis
3. Licensee Event Report
4. Background Information (Section 20. 1992) ASP Annual Precursor Report
5. Appendix A. ASP Models

cc w/enclosures:
See next page

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Original Signed By:

Robert M. Pulsifer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
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2. Guidance for Licensee Peer Review of Preliminary ASP Analysis
3. Licensee Event Report
4. Background Information (Section 20. 1992) ASP Annual Precursor Report
5. Appendix A. ASP Models

cc w/enclosures:
See next page

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DRoss, AEOD	LCollins, AEOD	

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MRushbrook	RPulsifer:sw	JHannon
5/12/94	5/12/94	5/12/94

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Mr. Stephen N. Brown
Utilities Division
Iowa Department of Commerce
Lucas Office Building, 5th floor
Des Moines, Iowa 50319

ENCLOSURE 1

0.1 LER Number 331/93-010

Event Description: Reactor Scram with LPCI Unavailable

Date of Event: October 26, 1993

Plant: Duane Arnold

0.1.1 Summary

Duane Arnold was operating at 100% power when a reactor scram occurred from turbine control valve fast closure and average power range monitor high flux. One loop of low-pressure coolant injection (LPCI) was out of service for planned maintenance. The conditional core damage probability estimated for this event is 1.1×10^{-6}. The relative significance of this event compared to other postulated events at Duane Arnold is shown in Fig. 1 below.

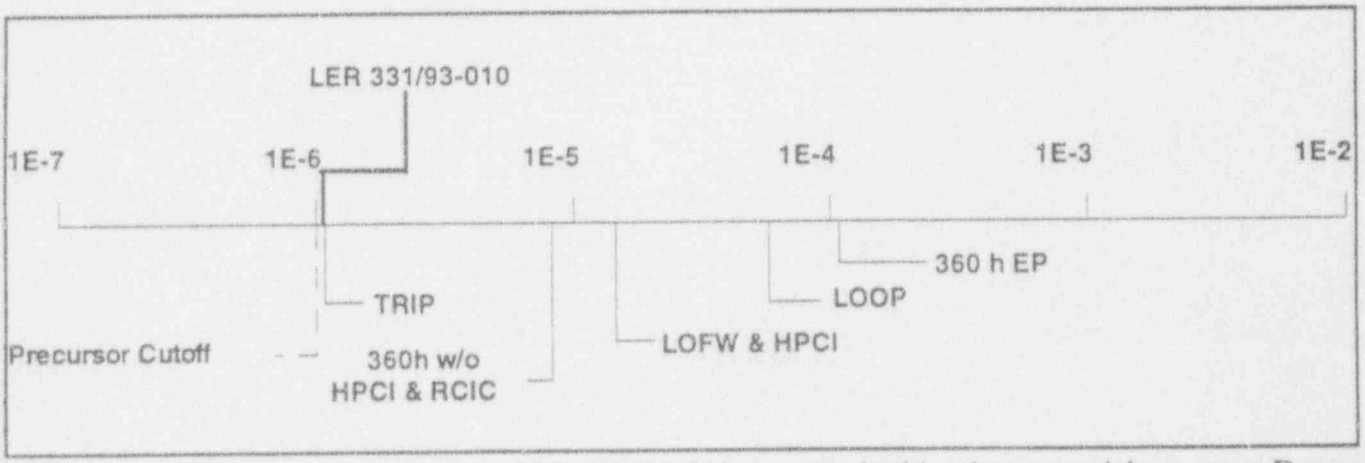

Fig. 1. Relative event significance for LER 331/93-010 compared with other potential events at Duane Arnold.

0.1.2 Event Description

On October 26, 1993, the plant was operating at 100% power. The B control building chiller was out of service, and the LPCI system was undergoing planned maintenance. A technician was replacing a cooling fan in a turbine electro-hydraulic control (EHC) cabinet in the control room when he inadvertently caused a short circuit. The short caused a loss of EHC hydraulic pressure, resulting in a turbine control valve (TCV) fast closure, and a subsequent reactor scram. There were no emergency core cooling system actuations and no safety relief valve openings following the scram.

0.1.3 Additional Event-Related Information

LPCI is an operating mode of the residual heat removal (RHR) system. Four pumps deliver water from the suppression pool to the selected recirculation loop. This provides vessel inventory makeup following large pipe breaks, or small pipe breaks when it is used in conjunction with the automatic depressurization system.

0.1.4 Modeling Assumptions

Since LPCI was in a planned maintenance outage, this event was modeled as a reactor trip with one loop of LPCI unavailable and nonrecoverable.

0.1.5 Analysis Results

The conditional probability of severe core damage estimated for this event is 1.1×10^{-6}. The dominant sequence associated with the event, shown on the event tree in Fig. 2, involves successful scram and high-pressure core cooling with failure to remove heat from the suppression pool in the long term.

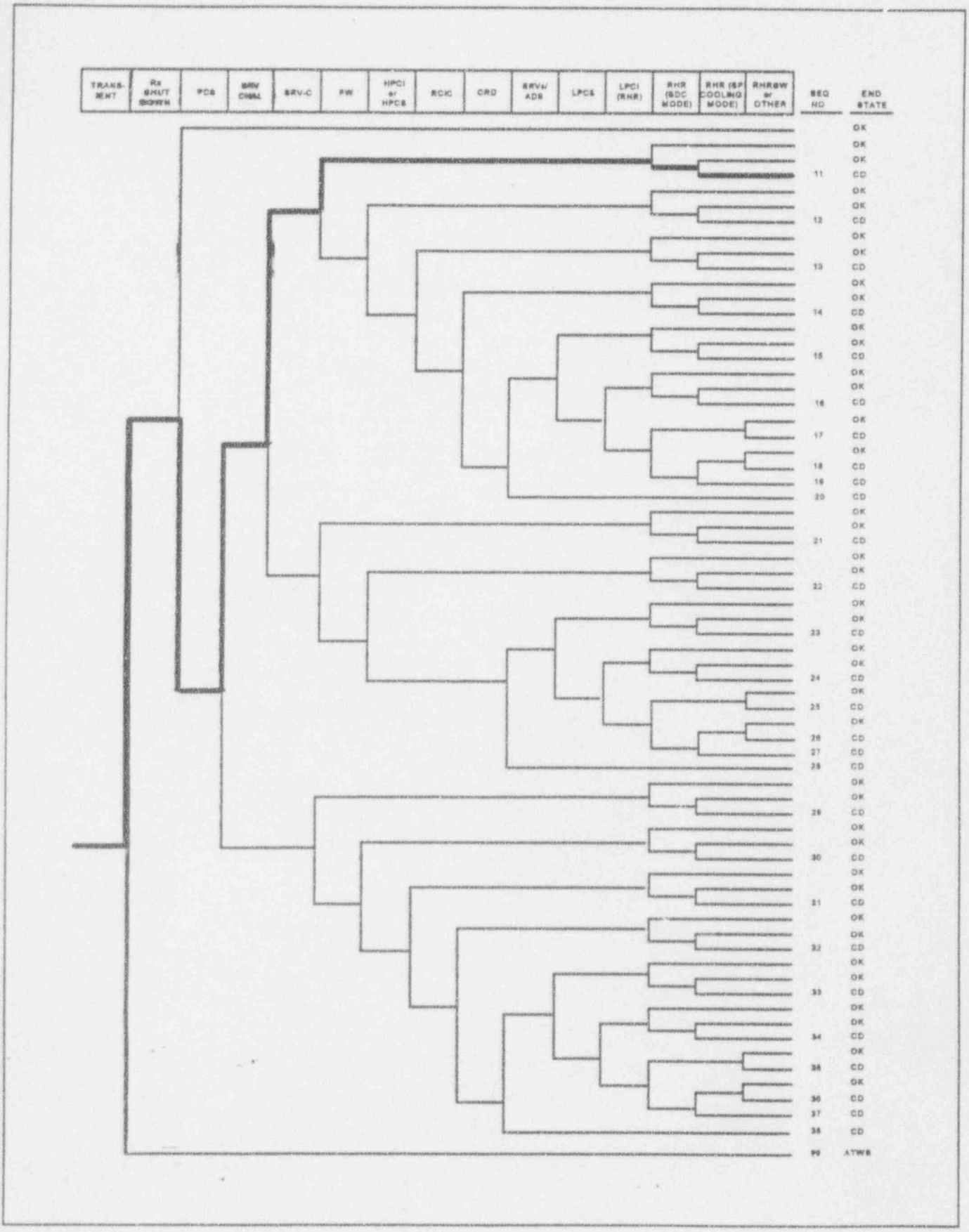

Fig. 2. Dominant core damage sequence for LER 331/93-010.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 331/93-010
Case: Scram w/LPCI O.O.S.
Plant: Duane Arnold

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.1E-06
Total	1.1E-06
ATWS	
TRANS	3.0E-05
Total	3.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close -fw/pcs.trans rhr(sdc) rhr(spcool)/rhr(sdc)	CD	8.2E-07	1.1E-01
28	trans -rx.shutdown pcs/trans srv.chall/trans.-scram srv.close fw/pcs.trans hpci srv.ads	CD	1.1E-07	1.7E-01
12	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close fw/pcs.trans -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	8.8E-08	3.9E-02
99	trans rx.shutdown	ATWS	3.0E-05	1.0E+00

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close -fw/pcs.trans rhr(sdc) rhr(spcool)/rhr(sdc)	CD	8.2E-07	1.1E-01
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28	trans -rx.shutdown pcs/trans srv.chall/trans.-scram srv.close fw/pcs.trans hpci srv.ads	CD	1.1E-07	1.7E-01
99	trans rx.shutdown	ATWS	3.0E-05	1.0E+00

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL: s:\asp\prog\models\duarnold.sll

Event Identifier: 331/93-010

PROBABILITY FILE: s:\asp\prog\models\bwr_cs11.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	8.2E-05	1.0E+00	
loop	1.6E-05	3.6E-01	
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	2.6E-02	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
ep.rec	6.6E-02	1.0E+00	
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 > 1.0E-02	7.1E-01	
Branch Model: 1.0f.2			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
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	3.4E-01	2.0E-03

* branch model file
** forced

Event Identifier: 331/93-010

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 Program 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.

Any claims for credit for the use of additional systems, equipment, or specific actions in the recovery process must be supported by appropriate documentation in your response. The identified 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.

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