50-313

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

April 23, 1996

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF LICENSEE EVENT REPORT NO. 95-005-00 AT ARKANSAS NUCLEAR ONE, UNIT 1

Dear Mr. Yelverton:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which occurred at Arkansas Nuclear One, Unit-1 (ANO-1) on April 20, 1996, (Enclosure 1), and was reported in Licensee Event Report (LER) No. 95-005-00. This analysis was prepared by our contractor at the Oak Ridge National Laboratory (ORNL). The results of this preliminary analysis indicate that this event may be a precursor for 1995. In assessing operational events, an effort was made to make the ASP models as realistic as possible regarding the specific features and response of a given plant to various accident sequence initiators. We realize that licensees may have additional systems and emergency procedures, or other features at their plants that might affect the analysis. Therefore, we are providing you an opportunity to review and comment on the technical adequacy of the preliminary ASP analysis, including the depiction of plant equipment and equipment capabilities. Upon receipt and evaluation of your comments, we will revise the conditional core damage probability calculations where necessary to consider the specific information you have provided. The object of the review process is to provide as realistic an analysis of the significance of the event as possible.

In order for us to incorporate your comments, perform any required reanalysis, and prepare the final report of our analysis of this event in a timely manner, you are requested to complete your review and to provide any comments within 30 days of receipt of this letter. We have streamlined the ASP Program with the objective of significantly improving the time after an event in which the final precursor analysis of the event is made publicly available. As soon as our final analysis of the event has been completed, we will provide for your information the final precursor analysis of the event and the resolution of your comments. In previous years, licensees have had to wait until publication of the Annual Precursor Report (in some cases, up to 23 months after an event) for the final precursor analysis of an event and the resolution of their comments.

We have also enclosed several items to facilitate your review. Enclosure 2 contains specific guidance for performing the requested review, 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 describes the specific information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 95-005-00, which documented the event.

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If you have any questions regarding this request, please contact me at (301) 415-1308. This request is covered by the existing OMB clearance number (3150-0104) for NRC staff followup review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

Original signed by
George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Accident Sequence Precursor
2. Guidance for Licensee Review
3. LER No. 95-005-00

cc w/encls: See next page

DISTRIBUTION:

Docket File	PUBLIC	PD4-1 r/f
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E. Adensam (EGA1)		

Document Name: AR195005.LTR

OFC	LA/PD4-1	PM/PD4-1
NAME	PNoonan	GKalman/vw
DATE	4/23/96	4/23/96
COPY	YES/NO	YES/NO

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Sincerely,

George Kalman

George Kalman, Senior Project Manager
Project Directorate IV-1
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Docket No. 50-313

Enclosures 1. Accident Sequence Precursor
2. Guidance for Licensee Review
3. LER No. 95-005-00

cc w/encls: See next page

Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-199

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Manager, Rockville Nuclear Licensing
Framatone Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

LER No. 313/95-005

Event Description: Trip with one EFW train unavailable

Date of Event: April 20, 1995

Plant: Arkansas Nuclear One, Unit 1

Event Summary

Arkansas 1 was operating at 100 percent power when a spurious trip of the main generator resulted in a main turbine trip, thereby causing an automatic trip of the reactor. Multiple equipment malfunctions were experienced, including failure of both flow control valves associated with the motor-driven emergency feedwater pump (MDEFWP) train. The conditional core damage probability estimated for this event is 2.0×10^{-5}.

Event Description

Arkansas 1 was operating at full power when a ground fault on the "B" phase of the current transformer lead to the negative sequence relay (NSR) initiated a generator lockout and subsequent turbine and reactor trip. (The NSR protects the main generator from thermal damage due to negative sequence current caused by system faults or an open phase condition.) During the post trip response, one main steam safety valve, PSV-2684 (see Fig. 1), remained open longer than operators expected. To reduce the pressure in the "B" Once-Through-Steam-Generator (OTSG), operators opened the "B" turbine bypass valve to approximately 50 percent. As pressure in the "B" steam generator (SG) dropped, PSV-2684 seated and the "B" turbine bypass valve closed. PSV-2684 reopened and operators again opened the B turbine bypass valve, thereby allowing PSV-2684 to reclose.

Both main feedwater pumps (MFPs) were used to maintain SG levels and ran back to minimum speed after the reactor trip, as expected. After SG levels stabilized, the MFPs should have automatically returned to automatic level control. The "A" MFP returned to automatic control as designed, but the "B"

MFP did not. Operators manually adjusted the "B" MFP flow and returned it to automatic control. The "B" MFP failed to shift back to automatic control because foreign material (a calibration sticker) on a module connector prevented a proper electrical connection to a relay coil.

During the first hour after the trip, condenser vacuum gradually decreased to about 20 inches Hg. This was attributed to excessive air in-leakage, coupled with a failure of the "B" vacuum pump to automatically shift into hogging mode (higher flow rate at reduced vacuum). Operators determined that the excessive air in-leakage was entering through the moisture separator reheater relief (MSR) valves. By increasing the MSR steam seal pressure and switching the "B" vacuum pump to hogging mode, the vacuum in the condenser was recovered.

About an hour after the trip, a +5 volt dc power supply for train "A" of the emergency feedwater initiation and control (EFIC) system failed. This failure, believed to be caused by component failure in the voltage regulating circuit for the power supply, resulted in a half-trip of the EFIC system. Train "A" SG level indication was lost, as was control of atmospheric dump valve (ADV) CV-2668 and emergency feedwater valves CV-2646 and CV-2648 (see Fig. 2).

Additional Event-Related Information

To adequately remove heat from the reactor core after a scram or a trip, only one of two EFW pump trains needs to be available to deliver water to at least one of the two OSTGs. The failure of the +5 volt power supply resulted in the loss of EFW flow control valves in the MDEFW train (CV-2646 and CV-2648) and ADV CV-2668 control in either automatic or manual control.

Modeling Assumptions

The licensee event report (LER) for this event is not specific regarding the as-failed position of the motor-driven emergency feedwater pump (MDEFWP) flow control valves and the impact of the failure on system performance. If the valves failed closed, the auxiliary feedwater supply from the MDEFWP would be unavailable. If the valves failed full-open, they would not be capable of regulating flow. This latter condition could eventually require the operators to trip the MDEFWP to prevent steam generator overfill. In this case, tripping the MDEFWP would be modeled as a recoverable system failure. Either of the above cases (failed open or failed closed) leads to the unavailability of the MDEFWP; therefore, this event was modeled as a reactor trip with flow from the MDEFWP made unavailable by failure of its EFW flow control valves. Even though the EFW control valves were not declared unavailable until about 1 hour after the trip, the event was modeled as a simple trip with MDEFW unavailable. Consistent with other presursor analyses, the probability of not recovering the failed MDEFW train was not revised in the models because failures were not observed in the TDEFW train.

Control of EFW flow control valves CV-2646 and CV-2648 was lost when a +5 vdc power supply in EFIC train "A" failed. This was apparently due to a random failure of a voltage regulator within the power supply. No information was provided which specifically indicated an increased potential for common-cause failure of the TDEFW train valves, so no increase in common-cause failure probability was modeled.

To implement the assumed failure of the MDEFWP flow control valves, the valves associated with the MDEFWP (Basic Event EFW-MOV-CF-DISM) were set to TRUE (i.e., the valves were failed). This caused the motor driven train of the EFW to be failed in the model. The turbine driven train was still available and not subject to the common cause failure which rendered the MDEFW valves inoperable. The 4-valve common-cause failure event (EFW-MOV-DF-DISAL) was therefore "removed" from the model by setting it as FALSE (i.e., p = 0.0) because the cause of the failure of the MDEFWP valves would not affect the TDEFWP valves.

Analysis Results

The conditional core damage probability estimated for this event is 2.0×10^{-5}. The dominant sequence, highlighted on the event tree in Fig. 3 involves the observed trip and loss of MDEFW. The assumed inoperability of MDEFWP valves increased the failure probability for the MDEFW. In addition, with

MDEFW degraded, the feed and bleed capabilities become more important with respect to removing decay heat.

Reference

1. LER 313/95-005, "Reactor Trip Initiated by Main Turbine Generator Protective Circuitry as a Result of a Logic Circuit Ground Caused by Vibration Induced Insulation Wear," May 19, 1995.

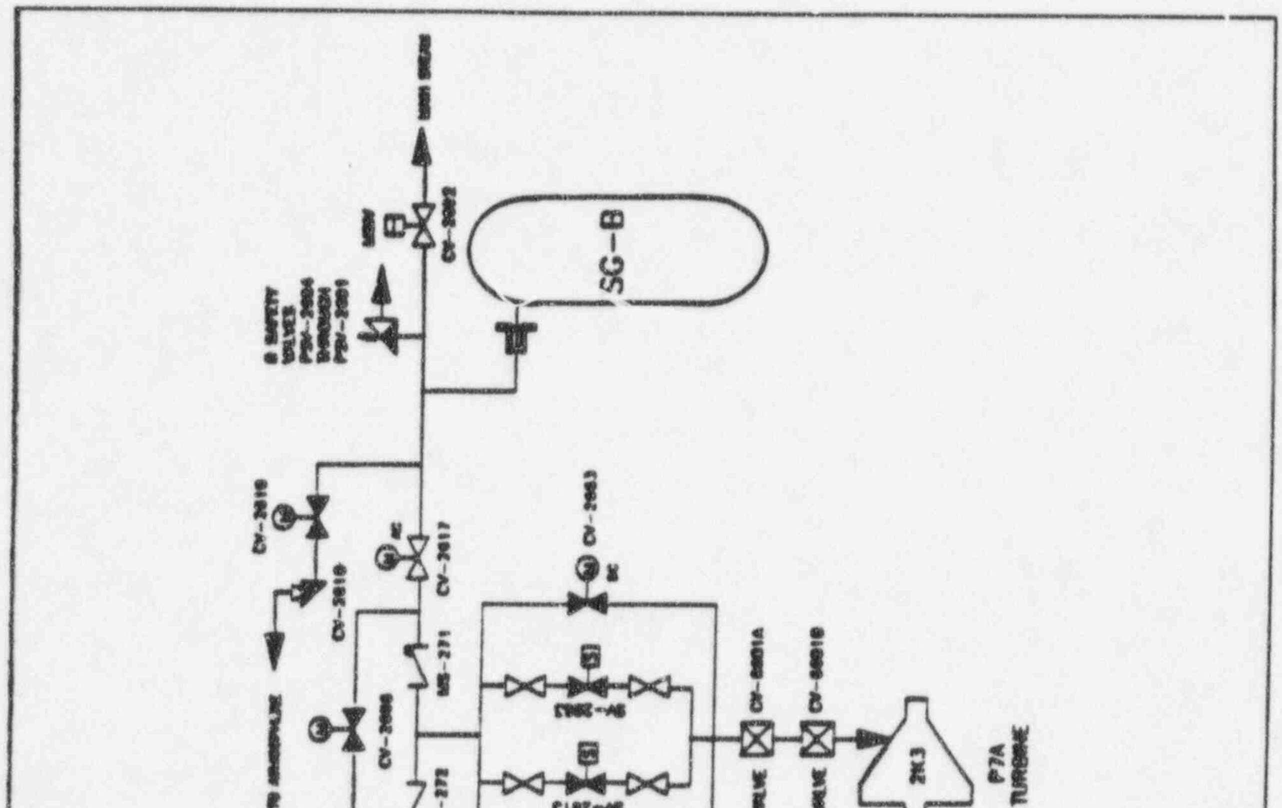

Figure 1 ANO 1 Emergency Feedwater System

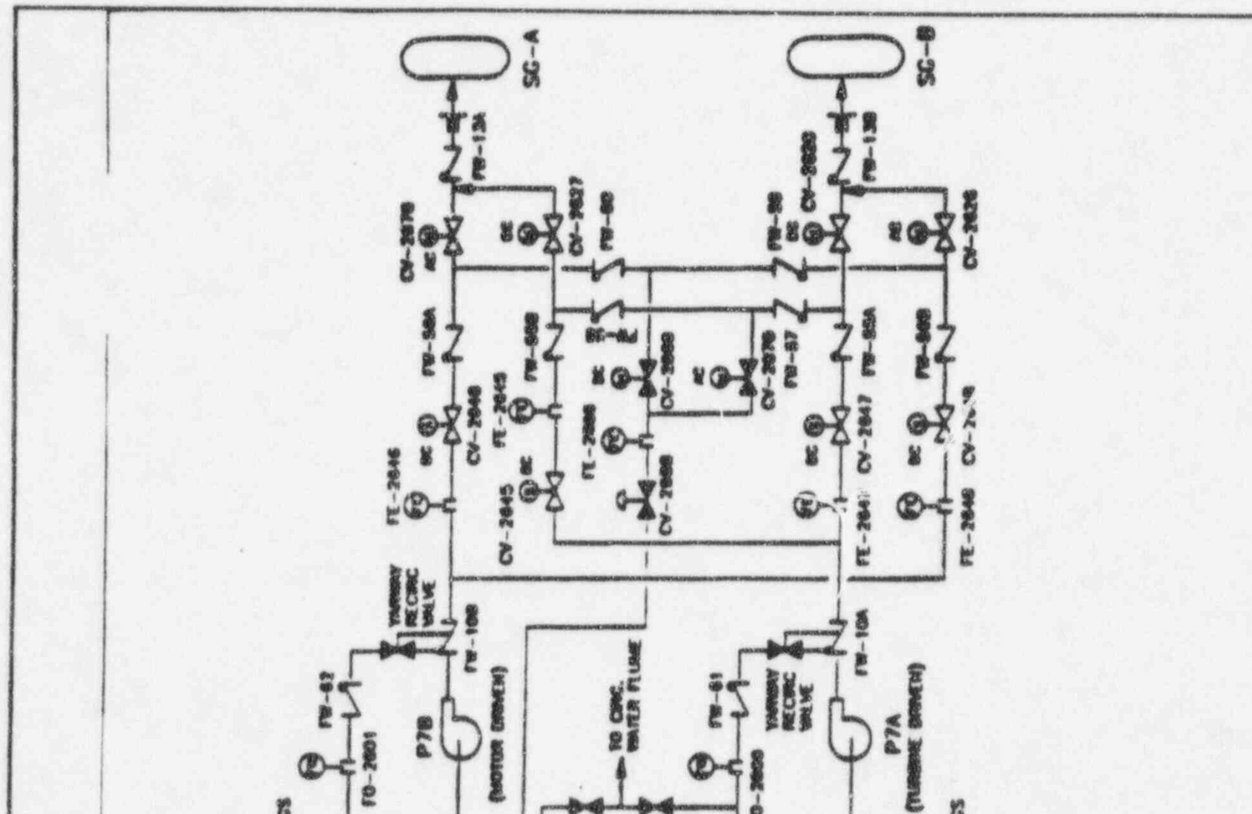

Figure 2 ANO 1 Emergency Feedwater System.

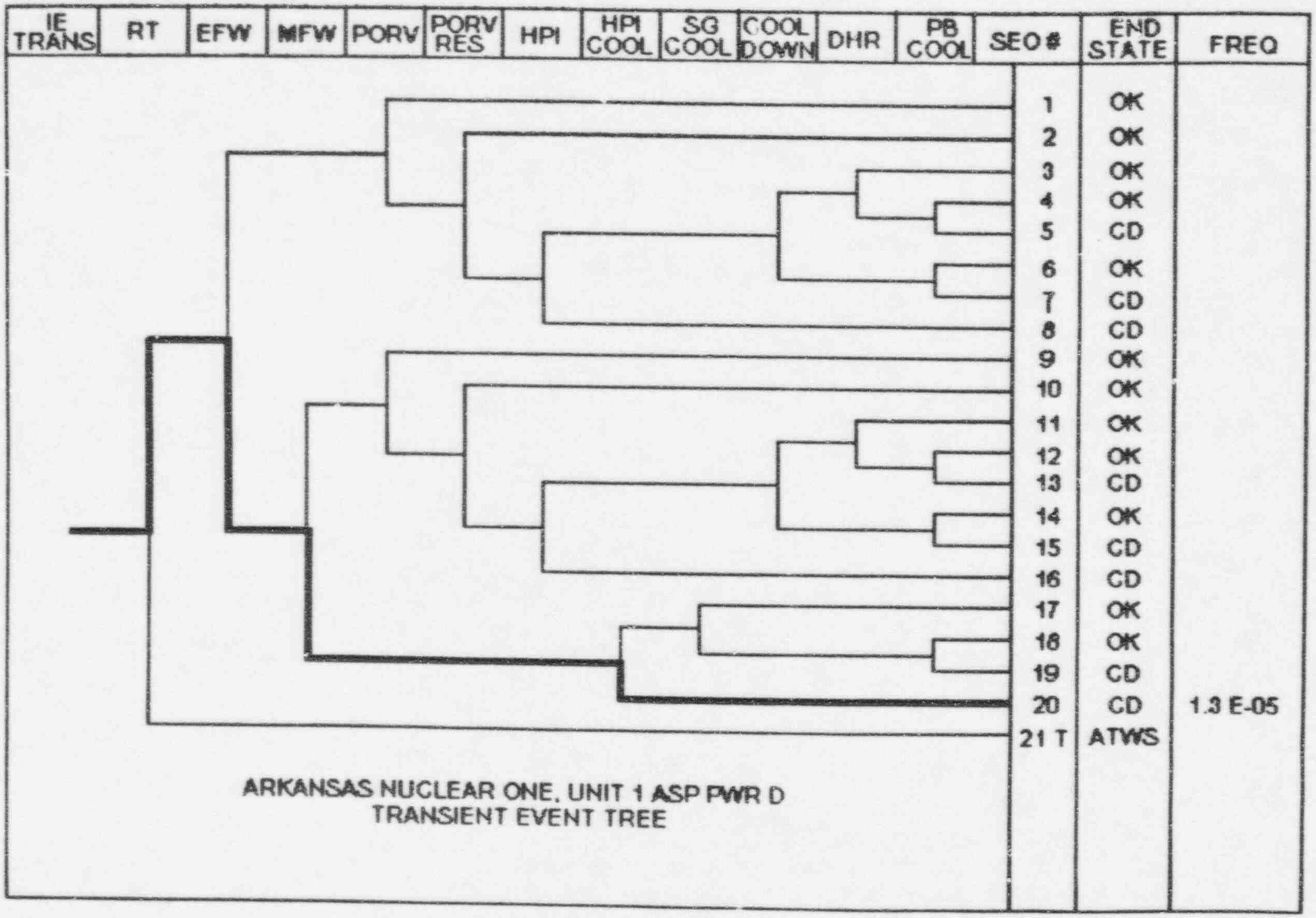

Figure 3 Dominant core damage sequence for LER 313/95-005.

Table 1. Definitions and probabilities for selected basic events for LER 313/95-005

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss of Offsite Power Initiating Event	8.5 E-006	0.0 E+000	IGNORE	Yes
IE-STGR	Steam Generator Tube Rupture Initiating Event	1.6 E-006	0.0 E+000	IGNORE	Yes
IE-SLOCA	Small Loss of Coolant Accident Intiiating Event	1.6 E-006	0.0 E+000	IGNORE	Yes
IE-TRANS	Transient Initiating Event	1.3 E-004	1.0 E+000		Yes
EFW-MOV-CF-DISAL	EFW Discharge Valves Fail from Common Cause	5.5 E-005	0.0 E+000	FALSE	Yes
EFW-MOV-CF-DISM	MDP Discharge Valves Fail From Common Cause	2.6 E-004	1.0 E+000	TRUE	Yes
EFW-TDP-FC-1B	Failure of EFW Turbine Driven Pump	3.2 E-002	3.2 E-002		No
EFW-XHE-NOREC	Operator Fails to Recover EFW System	2.6 E-001	2.6 E-001		No
EFW-XHE-NOTHROT	Operator Fails to Throttle EFW Flow	5.0 E-003	5.0 E-003		No
EFW-XHE-XA-CST	Operator Fails to Align a Backup Water Supply	1.0 E-003	1.0 E-003		No
HPI-CKV-00-MST	MST Suction Isolation MOV Common Cause Failures	3.0 E-003	3.0 E-003		No
HPI-MDP-CF-ABC	HPI MDP Common Cause Failures	1.1 E-004	1.1 E-004		No
HPI-MOV-CF-SUCT	HPI Suction Isolation MOV Common Cause Failures	2.6 E-004	2.6 E-004		No
HPI-XHE-NOREC	Operator Fails to Recover the HPI System	8.4 E-001	8.4 E-001		No
HPI-XHE-XM-HPIC	Operator Fails to Initiate HPI Cooling	1.0 E-002	1.0 E-002		No
MFW-SYS-TRIP	Main Feedwater System Trips	2.0 E-001	2.0 E-001		No

Event name	Description	Base probability	Current probability	Type	Modified for this event
MFW-XHE-NOREC	Operator Fails to Recover Main Feedwater	3.4 E-001	3.4 E-001		No
PCS-ICC-FA-TT	Failure of the Main Turbine to Trip	1.0 E-003	1.0 E-003		No
PPR-MOV-OO-BLK	PORV Block Valve Fails to Close	4.0 E-003	4.0 E-003		No
PPR-SRV-CC-PORV	PORV Fails to Open on Demand	6.3 E-003	6.3 E-003		No
PPR-SRV-CC-RCS	Relief Valves Fail to Limit RCS Pressure	4.4 E-004	4.4 E-004		No
PPR-SRV-OO-PORV	PORV Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No
PPR-XHE-NOREC	Operator Fails to Close the Block Valve	1.1 E-002	1.1 E-002		No
RCS-PHN-MODPOOR	Moderator Temperature Coefficient is not Negative Enough	1.4 E-002	1.4 E-002		No
RPS-SYS-FC-ELECT	Control Rod Drives Remain Energized	6.0 E-005	6.0 E-005		No
PRX-XHE-XM-SCRAM	Operator Fails to Manually Trip the Reactor	3.4 E-001	3.4 E-001		No

Table 2. Sequence conditional probabilities for LER 313/95-005

Event tree name	Sequence name	Conditional core damage probability (CCDP)	% Contribution
TRANS	20	1.3 E-005	64.6
TRANS	21-8	5.3 E-006	26.2
TRANS	08	1.2 E-006	6.3
TRANS	21-9	3.1 E-007	1.5
Total (all sequences)		2.0 E-005	

Table 3. Sequence logic for dominant sequences for LER 313/95-005

Event tree name	Sequence name	Logic
TRANS	20	EFW HPI-COOL
TRANS	21-8	/RCSPRESS
TRANS	08	/EFW PORV-RES
TRANS	21-9	RCSPRESS

Table 4. System names for LER 313/95-005

System name	Logic
EFW	No or Insufficient EFW System Flow
EFW-ATWS	No or Insufficient EFW System Flow
HPI	No or Insufficient Flow from the HPI System
HPI-COOL	Failure to Provide HPI Cooling
MFW	Failure of the Main Feedwater System
PORV	PORV Opens During Transient
PORV-RES	PORV Fails to Reseat
RCSPRESS	Failure to Limit RCS Pressure
RT	Reactor Fails to Trip During Transient

Table 5. Conditional cut sets for higher probability sequences for LER 313/95-005

Cut set No.	% Contribution	Frequency	Cut sets
TRANS Sequence 20		1.3 E-005	
1	43.2	5.6 E-006	EFW-XHE-NOREC, HPI-XHE-XM-HPIC, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-TDP-FC-1B
2	27.2	3.5 E-006	PPR-SRV-CC-PORV, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-TDP-FC-1B
3	10.8	1.4 E-006	EFW-XHE-NOREC, HPI-XHE-NOREC, HPI-CKV-00-MST, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-TDP-FC-1B
4	6.7	8.8 E-007	EFW-XHE-NOREC, HPI-XHE-XM-HPIC, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-XHE-NOTHROT
5	4.2	5.5 E-007	PPR-SRV-CC-PORV, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-XHE-NOTHROT
6	1.7	2.2 E-007	EFW-XHE-NOREC, HPI-XHE-NOREC, HPI-CKV-00-MST, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-XHE-NOTHROT
7	1.3	1.7 E-007	EFW-XHE-NOREC, HPI-XHE-SM-HPIC, MFW-SYS-TRIP, MFW-XHE-NOREC, EFW-XHE-XA-CST
TRANS Sequence 21-8		5.3 E-006	
1	99.9	5.3 E-006	EFW-XHE-NOREC, RPS-XHE-SM-SCRAM, RPS-SYS-FC-ELECT
TRANS Sequence 08		1.2 E-006	
1	64.6	8.3 E-007	HPI-CKV-OO-MST, HPI-CKV-OO-MST, PPR-SRV-PORV, PPR-XHE-NOREC
2	23.5	3.0 E-007	HPI-XHE-NOREC, HPI-CHV-OO-MST, PPR-SRV-OO-PORV, PPR-MOV-OO-BLK
3	5.6	7.3 E-008	HPI-XHE-NOREC, HPI-MOV-CF-SUCT, PPR-SRV-OO-PORV, PPR-XHE-NOREC
4	2.3	3.0 E-008	HPI-XHE-NOREC, HPI-FMDP-CF-ABC, PPR-SRV-OO-PORV, PPR-XHE-NOREC
5	2.0	2.6 E-008	HPI-XHE-NOREC, HPI-[illegible]V-CF-SUCT, PPR-SRV-OO-PORV, PPR-MOV-OO-BLK
TRANS Sequence 20-9		3.1 E-007	
1	90.6	2.8 E-007	RCS-PHN-MODPOOR, RPS-XHE-SM-SCRAM, RPS-SYS-FC-ELECT
2	6.4	2.0 E-008	PCS-ICC-FA-TT, RPS-XHE-XM-SCRAM, RPS-SYS-FC-ELECT
3	2.8	8.9 E-009	PPR-SRV-CC-RCS, RPS-XHE-XM-SCRAM, RPS-SYS-FC-ELECT
Total (all sequences)		2.0 E-006	

GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event that 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 actual initiating events such as a loss of off-site power (LOOP) or 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.

Modeling Techniques

The models used for the analysis of 1995 events were developed by the Idaho National Engineering Laboratory (INEL). The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The models are based on linked fault trees. Four initiating events are considered: (1) transients, (2) loss-of-coolant accidents (LOCAs), (3) loss of offsite power (LOOPs), and (4) Steam Generator Tube Ruptures (PWR only). Fault trees were developed for each top event on the event trees to a supercomponent level of detail. The only support system currently modeled is the electric power system.

The models may be modified to include additional detail for the systems/components of interest for a particular event. This may include additional equipment or mitigation strategies as outlined in the FSAR or IPE. Probabilities are modified to reflect the particular circumstances of the event being analyzed.

Guidance of Peer Review

Comments regarding the analysis should address:

- Does the "Event Description" section accurately describe the event as it occurred?
- Does the "Additional Event-Related Information" section provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?
- Does the "Modeling Assumptions" section accurately describe the modeling done for the event? Is the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions? This also includes assumptions regarding the likelihood of equipment recovery.

Appendix H of Reference 1 provides examples of comments and responses for previous ASP analyses.

Criteria for Evaluating Comments

Modifications to the event analysis may be made based on the comments that you provide. Specific documentation will be required to consider modifications to the event analysis. References should be made to portions of the LER, AIT, or other event documentation concerning the sequence of events. System and component capabilities should be supported by references to the FSAR, IPE, plant procedures, or analyses. Comments related to operator response times and capabilities should reference plant procedures, the FSAR, the IPE, or applicable operator response models. Assumptions used in determining failure probabilities should be clearly stated.

Criteria for Evaluating Additional Recovery Measures

Additional systems, equipment, or specific recovery actions may be considered for incorporation into the analysis. However, to assess the viability and effectiveness of the components and methods, the appropriate documentation must be included in your response. This includes:

- normal or emergency operating procedures.*
- piping and instrumentation diagrams (P&IDs),*
- electrical one-line diagrams,*
- results of thermal-hydraulic analyses, and
- operator training (both procedures and simulator),* etc.

Systems, equipment, or specific recovery actions that were not in place at the time of the event will not be considered. Also, the documentation should address the impact (both positive and negative) 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 (including operator actions).

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 modeling would be patterned after information gathered either from the plant FSAR 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

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* Revision or practices at the time the event occurred.

effect for the standby feedwater system would be credited in the analysis provided that the following material was available:

- standby feedwater system characteristics are documented in the FSAR or 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 of the system is available (either in the FSAR, IPE, or supplied by the licensee),
- previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis,
- the effects of using the standby feedwater system have on the operation and recovery of systems or procedures that are already included in the event modeling. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

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 LER, augmented inspection team (AIT) report, or other pertinent reports.

- A summary of the calculation results. An event tree with the dominant sequence(s) highlighted. Four tables in the analysis indicate (1) a summary of the relevant basic events including modifications to the probabilities reflect the circumstances of the event, (2) the dominant core damage sequences, (3) the system names for the systems cited in the dominant core damage sequences, and (4) cut sets for the dominant core damage sequences.

Schedule

Please refer to the transmittal letter for schedules and procedures for submitting your comments.

References

1. L. N. Vanden Heuvel et al., *Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Volumes 21 and 22), Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory and Science Applications International Corp., December 1995.

ENTERGY

Entergy Operations, Inc.

May 19, 1995

1CAN059504

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/95-005-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), enclosed is the subject report concerning a reactor trip.

Very truly yours,

Rick J. King

for Dwight C. Mims
Director, Licensing

DCM/rhs

enclosure

Enclosure 3

cc: Mr. Leonard J. Callan
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, GA 30339-5957

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Arkansas Nuclear One - Unit 1	05000313	1 OF 5

TITLE (4)
Reactor Trip Initiated by Main Turbine Generator Protective Circuitry as a Result of a Logic Circuit Ground Caused by Vibration Induced Insulation Wear

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
04	20	95	95	-- 005 --	00	05	19	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		70.71(b)
POWER LEVEL (10)	100		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		70.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		Specify in
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		Abstract Below
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		and in Text

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard H. Scheide, Nuclear Safety and Licensing Specialist	501-858-5000

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 20, 1995, at approximately 0313, an automatic reactor trip was initiated by the Reactor Protection System as a result of a main turbine trip. The turbine trip was initiated by a generator lockout which was caused by a negative sequence relay (NSR) actuation. All control rods inserted into the core, as designed, and immediate operator actions were accomplished with no complications. The plant was safely taken to Hot Shutdown although some minor abnormalities occurred post-trip. Investigation into the cause of the trip identified a ground in the NSR circuitry which caused a current imbalance to the NSR which resulted in actuation of the relay. The most probable cause of the ground was vibration induced wear of wiring insulation inside an electrical junction box. The grounded wire was repaired and a rubber insulation mat was installed in the junction box to prevent vibration induced insulation wear. Other similar junction boxes were examined and no additional discrepancies were identified

A. Plant Status

At the time of this event, Arkansas Nuclear One, Unit One (ANO-1) was operating at approximately 100 percent power. Reactor Coolant System (RCS) [AB] temperature was 579 degrees and RCS pressure was 2155 psig.

B. Event Description

On April 20, 1995, at approximately 0313, an automatic reactor trip was initiated by the Reactor Protection System (RPS)[JE] as a result of a main turbine trip. The turbine trip was initiated by a generator lockout which was caused by a negative sequence relay (NSR) actuation. All control rods inserted into the core, as designed, and immediate post trip operator actions were accomplished with no significant complications.

The NSR is intended to protect the main generator from thermal damage due to negative sequence current caused by system faults or an open phase condition. The NSR is set to coordinate with system protective relays and will operate to lockout the main generator if a fault or open phase condition occurs. A generator lockout initiates a turbine trip which will, in turn, initiate a RPS trip of the reactor if power is above 43 percent.

During the post trip response, the Main Steam Safety Valves (MSSVs) opened as expected. However, one valve (PSV-2684) appeared to remain open longer than normal. Operators initiated action to reduce the "B" Once Through Steam Generator (OTSG) pressure to assist the MSSV in closing. The "B" Turbine Bypass Valve (TBV) was rapidly opened to approximately 50 percent, at which time PSV-2684 seated. The "B" TBV was then returned to automatic control and the "B" OTSG pressure began to increase slowly resulting in PSV-2684 opening again. The TBV was again placed in manual and OTSG pressure was lowered until the MSSV seated. The TBV was left in manual control until the plant was stabilized at Hot Shutdown.

Several abnormal system responses were observed after the trip:

- Following the reactor trip, both main feedwater pumps (MFPs) ran back to minimum speed, as designed. Upon reaching appropriate OTSG levels, the MFPs should be automatically released from Rapid Feedwater Reduction (RFR) to maintain OTSG levels. Operators observed that the "B" MFP transferred to manual instead of returning to automatic control, as required. The "B" Hand/Auto (H/A) station signal was matched to the "A" MFP H/A station signal, and returned to automatic. This condition did not present a significant challenge to the operators and there were no further problems with MFP control.

- Over a period of approximately one hour following the trip, condenser vacuum degraded to approximately 20.4 inches Hg. Condenser vacuum pump C5B was found to be running in the "Holding Mode" while pump C5A was running in the "Hogging Mode". At the time of the trip, C5A was the running pump and should have been capable of maintaining condenser vacuum in the absence of significant air in-leakage. However, C5B should have automatically shifted to the "Hogging Mode" when vacuum decreased to 24 inches Hg. The operators manually shifted C5B to the "Hogging Mode" and increased Moisture Separator Reheater (MSR) seal pressure to approximately 7 psig. Condenser vacuum then returned to normal.

- Approximately one hour after the trip, channel "A" of the Emergency Feedwater Initiation and Control (EFIC) system received a half trip as a result of the failure of a +5 VDC power supply. This condition resulted in the loss of Train "A" OTSG level indication, a low level initiate to Train "A" EFIC, and the loss of control function for Emergency Feedwater control valves CV-2646 and CV-2648 and remote control of Atmospheric Dump Valve (ADV) CV-2668. As a contingency in the event that the ADV might be needed, operators opened the valve locally. The ADV block valves remained closed and no steam release occurred through the ADVs. No EFW actuation occurred as a result of the power supply failure.

C. Root Cause

Investigation into the cause of the negative sequence relay trip identified a ground on the "B" phase current transformer (CT) lead from the transformer to the relay. Further investigation revealed brittle and cracked insulation on the "B" phase wires inside a junction box at the generator. Evidence of arcing to ground was found at that location. Indications of wear resulting from the CT wiring rubbing against the cover plate was identified on one of the CT leads. No other brittle or cracked wiring was identified.

The most probable root cause of this event was determined to be vibration induced wear of the CT wiring that resulted in a ground, causing a current imbalance to the negative sequence relay which resulted in relay actuation and ultimately, the reactor trip.

PSV 2684 was lift pressure tested as a conservative measure to determine the potential blowdown range of the valve based on actual setpoint since it was the last valve to reseat and was the cause of the perception by the operators that a valve was open too long. The as-found setpoint was 1037 psig. This would correlate to an acceptable blowdown range of 943 to 1006 psig. A review of all the Safety Parameter Display System data showed that PSV-2684 responded normally on blowdown and reseat through several valve strokes as the valve reseated within the acceptable blowdown range.

The cause of the failure of the "B" MFP to shift back to automatic control was determined to be foreign material on a module connector which provides power to a RFR logic relay coil. A 1/2 inch diameter calibration sticker was found covering a module connector which prevented proper connection to the cabinet back plane.

The probable cause for the degraded condenser vacuum over a one hour period was significant condenser air in-leakage. A primary indicator of this leakage was a corresponding increase in vacuum when steam seal pressure was increased on the MSR relief valves. It was noted that MSR pressure decreased in parallel with the condenser vacuum drop during the evolution. Vacuum loss cannot be attributed to operation of only one pump in the hogging mode The pump performance curve indicates one pump has the ability to remove nearly 800 CFM of non-condensables. This suggests one hogging pump was not able to keep up with the "air in-leakage" rate. It appears the "A" pump switched to hogging mode at the 23" Hg setpoint while the "B" pump remained in holding mode for nearly 90 minutes. Subsequent trouble-shooting determined that the setpoint for the pressure switch which controls the "B" pump was three inches low (21" Hg versus 24" Hg).

The failure of the +5 VDC Power Supply was apparently due to a failure of the voltage regulating circuit within the supply which was unrelated to the reactor trip. The loss of the ADV remote control is also directly related to the loss of the +5 VDC power supply. This supply provides integrated circuit logic power to the compensation module and control module portions of the EFIC Channel "A" ADV control circuit. The compensation modules provide density compensation for the EFIC OTSG Level inputs based on OTSG pressure. The control module provides level and pressure control for the OTSGs by modulating the Emergency Feedwater (EFW) flow control valves (CV-2646 and CV-2648) and the ADV (CV-2668). The failure of the power supply resulted in the loss of both EFW Flow Control Valves and CV-2668 control in either automatic or manual.

D. Corrective Actions

Immediate:

- The grounded wire from the "B" phase CT was repaired and a rubber insulation mat installed in the junction box to prevent vibration induced insulation wear.

- An inspection and megger check of the leads from the CTs of all three phases was performed. No additional wear or unsatisfactory megger readings were identified.

- PSV-2684 was lift pressure tested and verified to be operable.

- The foreign material was removed from the MFP control contacts and the circuit was proven operable.

- Condenser vacuum pump C-5B pressure switch was reset to 25" Hg increasing. In addition, MSR relief valve seal steam pressure has been increased and set in accordance with plant operating procedures.

E. Safety Significance

The turbine and reactor protective circuitry performed as designed during this event and the plant was safely taken to Hot Shutdown conditions. The operators expeditiously and properly compensated for all identified post trip abnormalities. Therefore, this event is considered to be of minimal safety significance.

F. Basis for Reportability

A reactor trip is a reportable event in accordance with 10CFR50.73(a)(2)(iv). This event was also reported to the NRC Operations Center at 0453 CST on April 20, 1995, pursuant to 10CFR50.72(b)(2)(ii).

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G. Additional Information

LER 50-313/93-001-00 reported a reactor trip which resulted from two grounds on the 125 VDC system. One of the grounds was caused by vibration induced wear of wiring that passed through an ungrommeted hole in the wall of the main turbine front standard. The corrective actions associated with this LER were focused on the main turbine front standard, MFP control circuitry, and wiring passing through panel walls and could not reasonably be expected to identify the condition that caused the trip discussed in this report.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].