October 29, 1996

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF EVENT AT ARKNASAS NUCLEAR ONE - UNIT 1

Dear Mr. Hutchinson:

Enclosed for your information is a copy of the final Accident Sequence Precursor analysis of the operational event at Arkansas Nuclear One - Unit 1, reported in Licensee Event Report (LER) No. 313/95-005. Enclosure 1 contains the final analysis prepared by the Oak Ridge National Laboratory, our contractor, based on review and evaluation of your comments on the preliminary analysis and comments received from the Nuclear Regulatory Commission (NRC) staff and Sandia National Laboratories, our contractor. Enclosure 2 contains our responses to your specific comments. Our review of your comments used the criteria contained in the material which accompanied the preliminary analysis. The results of the final analysis indicate that this event is a precursor for 1995.

Please contact me at (301) 415-1367 if you have any questions regarding the enclosures. We recognize and appreciate the effort expended by you and your staff in reviewing and providing comments on the preliminary analysis.

Sincerely,

Original signed by
Kombiz Salehi, Acting Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Final Accident Sequence Precursor Analysis
2. Responses to your comments

cc: See next page

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

October 29, 1996

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cc w/encls: See next page

Mr. C. Randy Hutchinson
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Arkansas Nuclear One, Unit 1

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# LER No. 313/95-005

Event Description: Trip with one Emergency Feedwater (EFW) train unavailable

Date of Event: April 20, 1995

Plant: Arkansas Nuclear One, Unit 1

## Event Summary

Arkansas Nuclear One, Unit 1 (ANO 1) was operating at 100% 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 (CCDP) estimated for this event is $6.4 \times 10^{-6}$.

## Event Description

Arkansas 1 was operating at full power when a ground fault on the B phase of the current transformer supplying the negative sequence relay (NSR) caused a generator lockout followed by turbine and reactor trips. (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), appeared to remain 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%. 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. Subsequent review verified that valve PSV-2684 responded normally on blowdown and reseat.

Both main feedwater pumps (MFPs) were used to maintain SG levels, running back to minimum speed after the reactor trip, as expected. After SG levels stabilized, the MFPs should have 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 in. Hg. The decrease 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 occurring through the moisture separator reheater (MSR) relief 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 remote 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 must be



ENCLOSURE 1

available to deliver water to at least one of the two OTSGs. The failure of the +5 Volt power supply resulted in the loss of EFW flow control valves in the MDEFWP train (CV-2646 and CV-2648) and ADV CV-2668 control in either automatic or manual control (local control of the ADV was still possible).

## Modeling Assumptions

About 1 hour after the trip, EFIC Train A failed, resulting in a loss of automatic and manual control of EFW flow control valves CV-2646 and -2648. The licensee event report (LER) for this event is not specific regarding the as-failed position of the MDEFWP flow control valves and the impact of the failure on system performance. If the valves failed closed, then the auxiliary feedwater supply from the MDEFWP would be unavailable. If the valves failed full-open, then 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. Note that failure of the flow control valves in the open position in conjunction with operator failure to control SG level by tripping the MDEFWP could result in failure of the turbine-driven EFW pump (TDEFWP). This potential failure mode was not explored.

Control of EFW flow control valves CV-2646 and CV-2648 was lost when a +5 Volt dc power supply in EFIC Train A failed. This failure was apparently caused by a random failure of a voltage regulator within the power supply. No information was provided that specifically indicated an increased potential for common cause failure of the flow control valves in the TDEFWP train, so no increase in common cause failure probability was modeled.

To implement the assumed failure of the MDEFWP flow control valves, the set of valves associated with the MDEFWP (Basic Event EFW-MOV-CF-DISM) were set to TRUE (i.e., the valves were failed). This setting caused the motor-driven train of the EFW to be failed in the model. The turbine-driven train was still available and was not subject to the common cause failure (i.e., loss of the +5 Volt dc power supply in EFIC Train A) that rendered the MDEFWP flow control valves inoperable. Basic event probability changes are noted in Table 1.

## Analysis Results

The CCDP estimated for this event is $6.4 \times 10^{-6}$. The dominant sequence, highlighted on the event trees in Figs. 3 and 4, involves

- the observed trip demand with a failure to trip, and
- failure of EFW to provide sufficient flow for ATWS mitigation.

The assumed inoperability of MDEFWP valves increased the failure probability for the MDEFWP train.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences are shown in Table 2. Table 3 lists the sequence logic associated with the sequences listed in Table 2. Table 4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table 5.

## Acronyms

| | |
|---|---|
| ADV | atmospheric dump valve |
| ANO 1 | Arkansas Nuclear One, Unit 1 |

| | |
|---|---|
| ATWS | anticipated transient without scram |
| CCDP | conditional core damage probability |
| EFIC | emergency feedwater isolation control |
| EFW | emergency feedwater |
| FSAR | final safety analysis report |
| LER | licensee event report |
| MDEFWP | motor-driven emergency feedwater pump |
| MFP | main feedwater pump |
| MSR | moisture separator reheater |
| NSR | negative sequence relay |
| RCS | reactor coolant system |
| OTSG | once-through steam generator |
| SG | steam generator |
| TDEFWP | turbine-driven emergency feedwater pump |

## References

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.

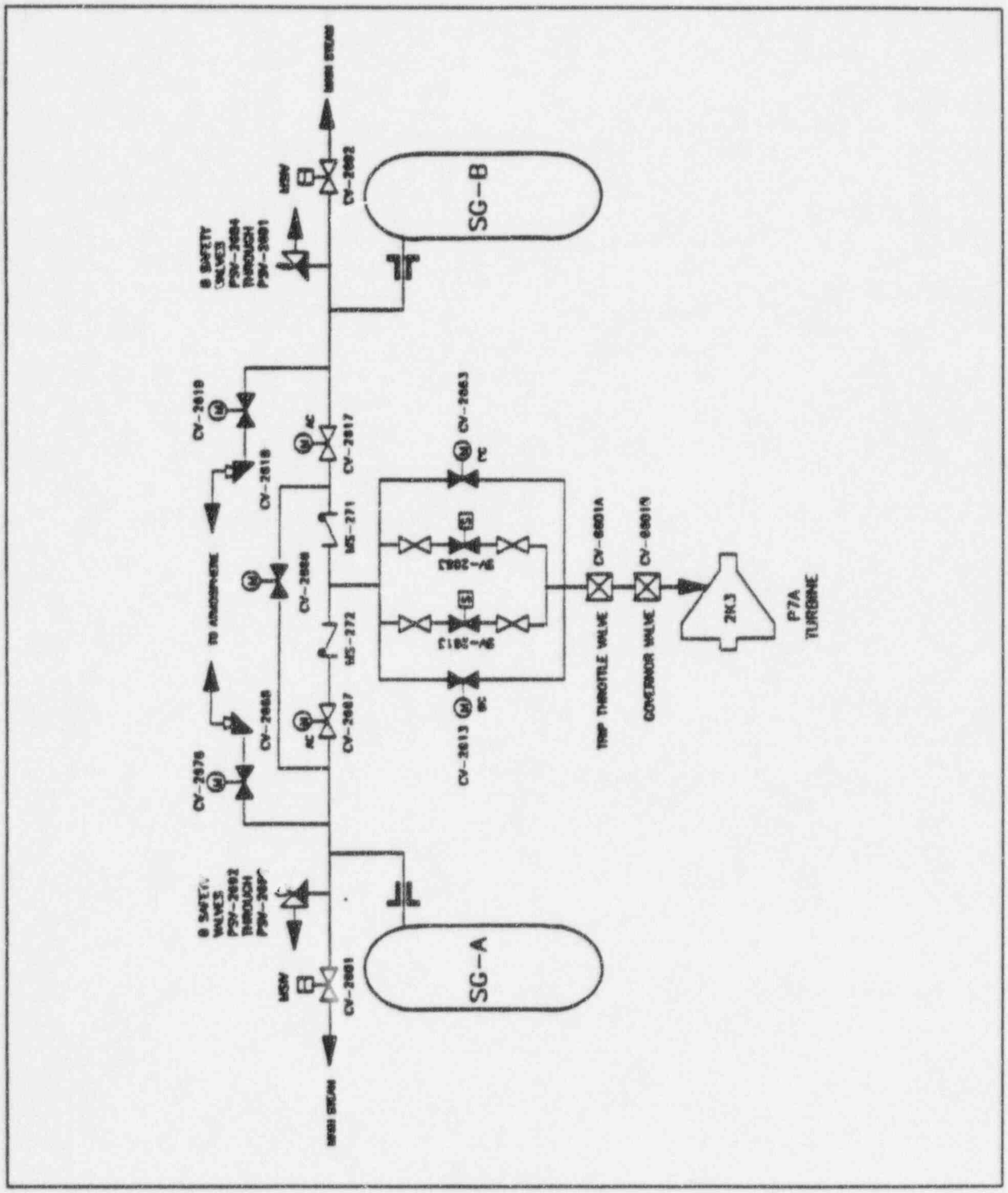

Fig. 1 ANO 1 Emergency Feedwater System.

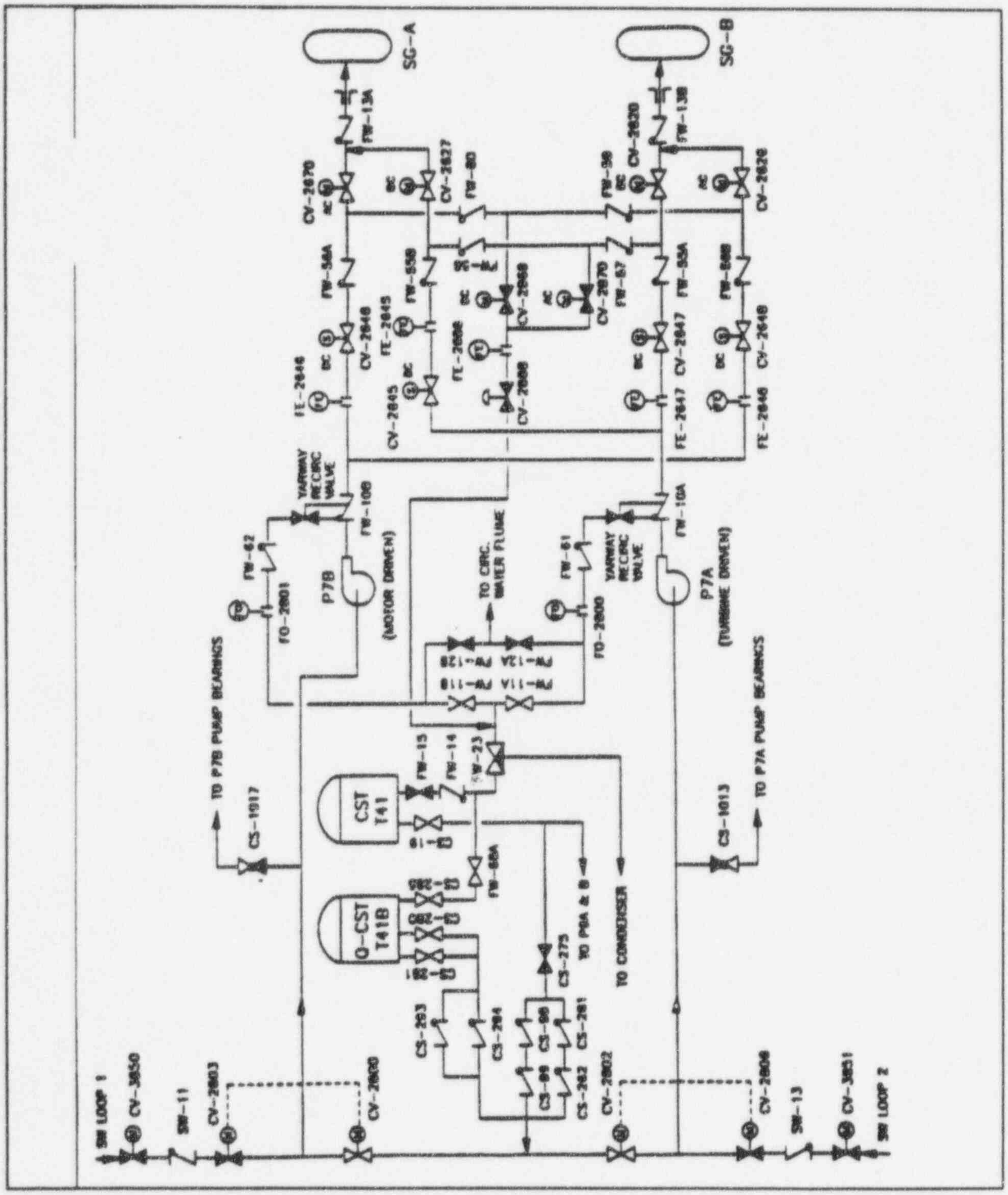


Fig. 2 ANO 1 Emergency Feedwater System.

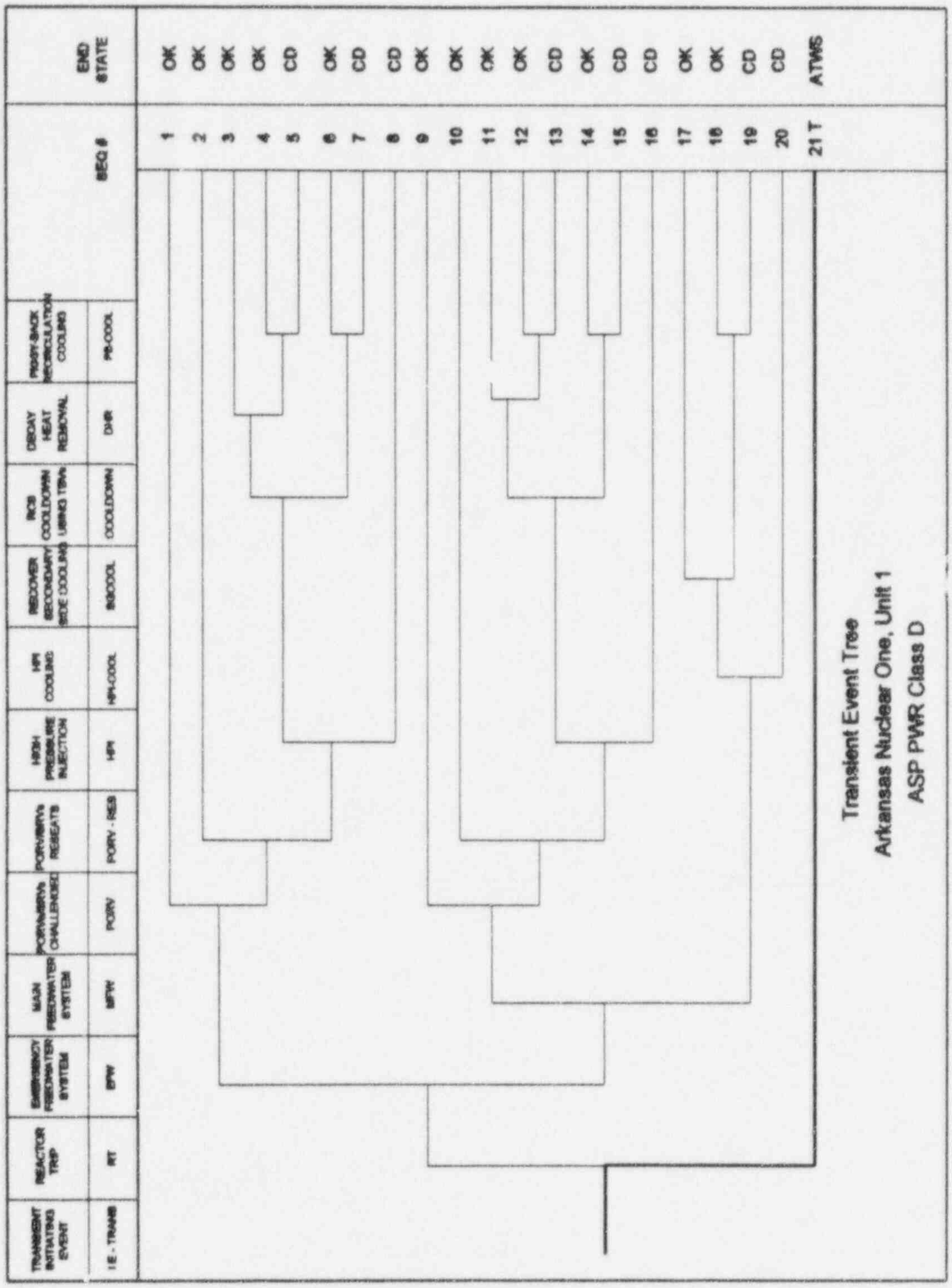


Fig. 3. Dominant core damage sequence for LER 313/95-005.

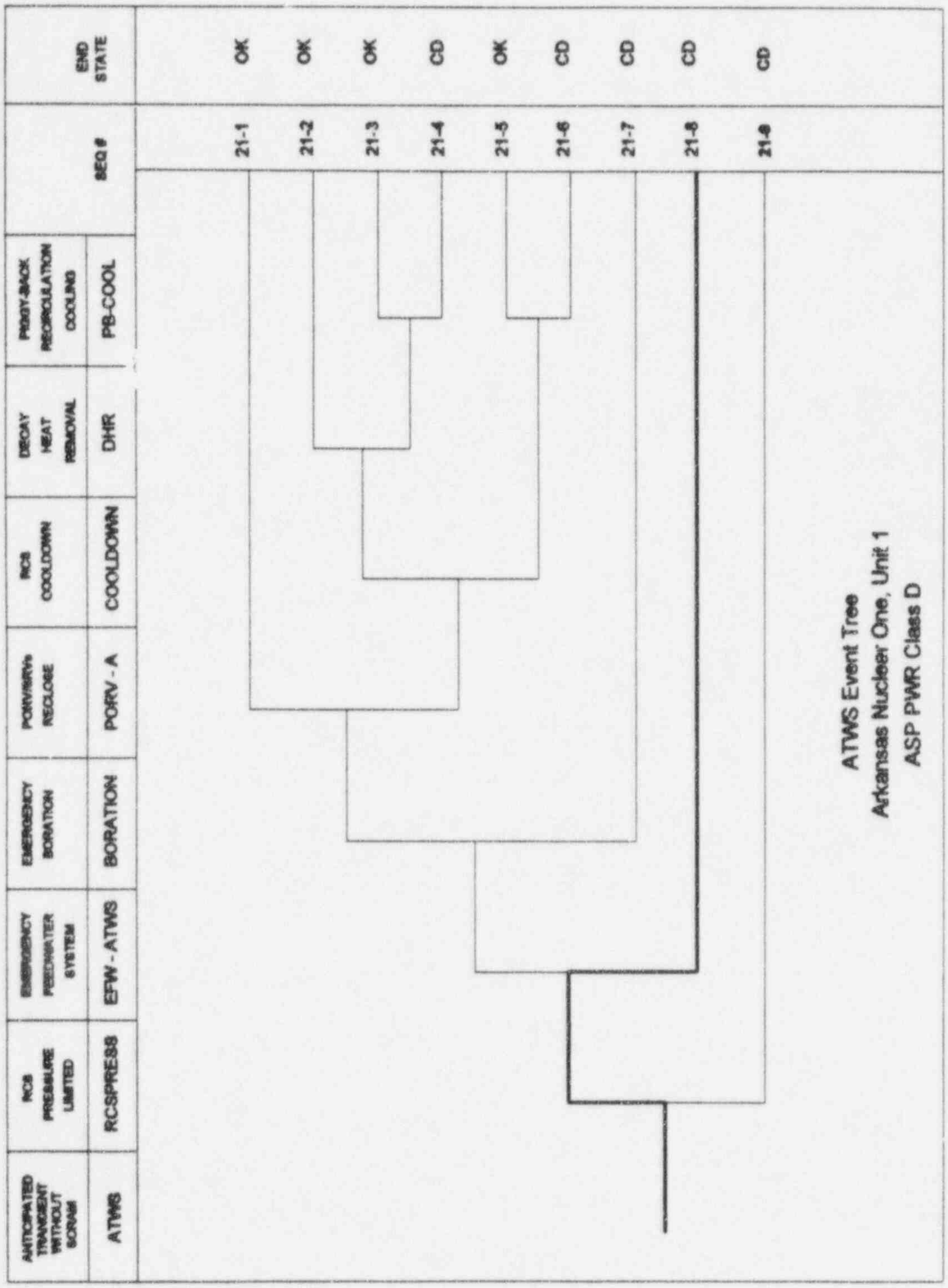


Fig. 4. Anticipated transient without scram (ATWS) event tree for Arkansas Nuclear One, Unit 1.

**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 Initiating Event | 1.0 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-DISM | Emergency Feedwater (EFW) Motor-Driven Pump (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-OO-MST | Makeup Storage Tank Suction Isolation Motor-Operated Valve (MOV) Common Cause Failures | 3.0 E-003 | 3.0 E-003 | | No |
| HPI-MDP-CF-ABC | High Pressure Injection (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 | 1.0 E+000 | | Yes |
| 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 |
| MFW-XHE-NOREC | Operator Fails to Recover Main Feedwater | 1.6 E-002 | 1.6 E-002 | | 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 | Power-Operated Relief Valve (PORV) Block Valve Fails to Close | 4.0 E-003 | 4.0 E-003 | | No |

**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 |
|---|---|---|---|---|---|
| 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 Reactor Coolant System Pressure | 4.4 E-004 | 4.4 E-004 | | No |
| PPR-SRV-CO-TRAN | PORV Opens During a Transient | 8.0 E-002 | 8.0 E-002 | | 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-NONREC | Nonrecoverable Reactor Protection System (RPS) Failures | 2.0 E-005 | 2.0 E-005 | | No |
| RPS-REC | Recoverable RPS Failures | 4.0 E-005 | 4.0 E-005 | | No |
| RPS-XHE-XM-SCRAM | Operator Fails to Manually Trip the Reactor | 1.0 E-002 | 1.0 E-002 | | No |

**Table 2. Sequence Conditional Probabilities for LER 313/95-005**

| Event tree name | Sequence name | Conditional core damage probability (CCDP) | Percent Contribution |
|---|---|---|---|
| TRANS | 21-8 | 5.3 E-006 | 82.7 |
| TRANS | 20 | 6.3 E-007 | 9.9 |
| TRANS | 21-9 | 3.1 E-007 | 4.9 |
| Total (all sequences) | | 6.4 E-006 | |

**Table 3. Sequence Logic for Dominant Sequences for LER 313/95-005**

| Event tree name | Sequence name | Logic |
|---|---|---|
| TRANS | 21-8 | RT, /RCSPRESS, EFW-ATWS |
| TRANS | 20 | /RT, EFW, MFW, HPI-COOL |
| TRANS | 21-9 | RT, 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 During an ATWS Event |
| 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. | Percent contribution | Conditional probability[a] | Cut sets[b] |
|---|---|---|---|
| **TRANS Sequence 21-8** | | 5.3 E-006 | |
| 1 | 98.0 | 5.2 E-006 | RPS-NONREC, EFW-XHE-NOREC |
| 2 | 1.9 | 1.0 E-[illegible] | RPS-XHE-XM-SCRAM, RPS-REC, EFW-XHE-NOREC |
| **TRANS Sequence 20** | | 6.3 E-007 | |
| 1 | 41.9 | 2.6 E-007 | EFW-MOV-CF-DISM, EFW-TDP-FC-1B, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC, HPI-XHE-NOREC |
| 2 | 26.4 | 1.6 E-007 | EFW-MOV-CF-DISM, EFW-TDP-FC-1B, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-PORV |
| 3 | 12.5 | 7.9 E-008 | EFW-MOV-CF-DISM, EFW-TDP-FC-1B, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-CKV-OO-MST, HPI-XHE-NOREC |
| 4 | 6.5 | 4.1 E-008 | EFW-XHE-NOTHROT, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC, HPI-XHE-NOREC |
| 5 | 4.1 | 2.6 E-008 | EFW-XHE-NOTHROT, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-PORV |
| 6 | 1.9 | 1.2 E-008 | EFW-XHE-NOTHROT, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-CKV-OO-MST, HPI-XHE-NOREC |
| 7 | 1.3 | 8.3 E-009 | EFW-XHE-XA-CST, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-HPIC, HPI-XHE-NOREC |
| 8 | 1.1 | 7.0 E-009 | EFW-MOV-CF-DISM, EFW-TDP-FC-1B, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-MOV-CF-SUCT, HPI-XHE-NOREC |
| **TRANS Sequence 21-9** | | 3.1 E-007 | |
| 1 | 88.9 | 2.8 E-007 | RPS-NONREC, RCS-PHN-MODPOOR |
| 2 | 6.3 | 2.0 E-008 | RPS-NONREC, PCS-ICC-FA-TT |
| 3 | 2.7 | 8.8 E-009 | RPS-NONREC, PPR-SRV-CC-RCS |
| 4 | 1.7 | 5.6 E-009 | RPS-XHE-XM-SCRAM, RPS-REC, RCS-PHN-MODPOOR |
| **Total (all sequences)** | | **6.4 E-006** | |

[a] The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probability of the initiating events are given in Table 1 and begin with the designator "IE". The probabilities for the basic events are also given in Table 1.

[b] Basic event EFW-MOV-CF-DISM is a type TRUE event and these type of events are normally not included in the output of fault tree reduction programs. This event has been added to aid in understanding the sequences to potential core damage associated with the event.

## 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

### Licensee Comments

**Reference:** Letter from D. C. Mims, Director, Nuclear Safety, Entergy Operations, Inc. to the U.S. Nuclear Regulatory Commission, *Review of Preliminary Accident Sequence Precursor Analysis*, ICAN059606, May 31, 1996.

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**Comment 1:** The event description incorrectly states that PSV-2684 "remained open longer than operators expected." The root cause section of the LER states that subsequent review verified that the valve responded normally on blowdown and reseat.

**Response 1:** As the comment itself notes, the ASP analysis event description did not report that PSV-2684 behaved abnormally, only that the operators believed that it was behaving abnormally. This is supported by the statement in the licensee event report (LER), "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 [main steam safety valve] in closing." To reflect that the valve operated properly during subsequent tests, the first paragraph in the **Event Description** section was changed to (changes noted in italics):

Arkansas 1 was operating at full power when a ground fault on the B phase of the current transformer supplying the negative sequence relay (NSR) caused a generator lockout followed by turbine and reactor trips. (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), *appeared to remain* 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%. 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. *Subsequent review verified that valve PSV-2684 responded normally on blowdown and reseat.*

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**Comment 2:** The event description states that control of the atmospheric dump valve was lost. Adding the word "remote" to the description clarifies that local control was still available. The same clarification may be added in the Additional Event-Related Information section.

**Response 2:** This clarification has been made. The sentence in the **Event Description** has been changed to (change noted in italics) "Train A SG level indication was lost, as was *remote* control of atmospheric dump valve (ADV) CV-2668 and emergency feedwater valves CV-2646 and CV-2648 (see Fig. 2)." The sentence in the **Additional Event-Related Information** section has been changed to "The failure of the +5 Volt power supply resulted in the loss of EFW flow control valves in the MDEFWP train (CV-

2646 and CV-2648) and ADV CV-2668 control in either automatic or manual control *(local control of the ADV was still possible).*"

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**Comment 3:** The Modeling Assumptions section states that the EFW control valves were not declared unavailable until about one hour after the trip, leaving the impression that they were unavailable previous to that time. A clarification that the valves actually became unavailable at that time would seem appropriate.

**Response 3:** This has been clarified by deleting the sentence in question.

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**Comment 4:** Upon evaluating the event description, and reviewing the Event Tree and resulting cut sets, it was identified that credit was not given for use of ANO-1's Auxiliary (Startup) Feedwater Pump (P-75). The Auxiliary Feedwater pump is an electric motor-driven centrifugal pump that is normally used during startup and shutdown conditions when there is insufficient steam available to run the main feed pumps. This pump is credited in the ANO-1 PSA as a recovery. . . .

The auxiliary feedwater pump can be credited in all transient sequences except for Loss of Offsite Power and Loss of Power Conversion System. The probability of failure for this recovery includes a mechanical failure probability as well as an operator failure. ANO-1's success criteria regarding the transient sequences involving loss of all feedwater defines an available time of 36 minutes for Loss of Power Conversion System sequences with reactor coolant pumps still running. . . .

**Response 4:** The event has been reanalyzed, giving credit for recovering the main feedwater supply using the motor-driven startup feedwater pump. The nonrecovery probability for the main feedwater system (MFW-XHE-NOREC) was changed to reflect this. The procedures supplied with the comment letter indicate that two valves must close, at least one of these valves must re-open, and the motor-driven pump must start and run for the recovery to be successful. In addition, adequate suction supply must be provided and operators must manually (remotely) align and start the system. The system failure probability given in Attachment 1 to the comment letter (1.6E-02) was used in the reanalysis.

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**Comment 5:** TRANS Sequence 20: The probability of failure for the Aux. Feed pump as a recovery can be applied to all cut sets in this sequence. . . .

TRANS Sequence 21-8: The cut set for this sequence includes a failure of the EFW system. Therefore, the Aux. Feedwater recovery can be applied. . . .

TRANS Sequence 08: The common cause failure probability assumed for the HPI-CKV-OO-MST is conservatively high based upon the ANO-1 Common Cause failure probability for these valves. The ANO-1 Common Cause probability for these valves is 7.38E-04 based upon a beta factor of .08 . . . and a probability of failure for the MOV of 9.23E-03.

For HPI-MOV-CF-SUCT, the CCF probability is also higher than is representative for ANO-1. The HPI suction valves are stop check valves. . . the beta factor for check valves is .06. Using ANO-1's

failure to open probability for these valves of 4.93E-04, yields a common cause failure probability of 2.96E-05 when the beta factor is applied. . . .

**Response 5:** Credit for the MDSFWP is accounted for in TRANS Sequence 20 through the recovery of the MFW system (MFW-XHE-NOREC). The nonrecovery probability provided by ANO 1 (0.016—see Comment 4) was used rather than a typical recovery class R2 nonrecovery probability (0.34), as given in Appendix A in NUREG/CR-4674, Vol. 21.

ATWS sequences, such as TRANS Sequence 21-8, require emergency feedwater in a very short amount of time. The procedures supplied with the comment letter indicate that two valves must close, at least one of these valves must re-open, and the motor-driven pump must start and run for the recovery to be successful. In addition, adequate suction supply must be provided and operators must manually (remotely) align and start the system. Because of the time available for recovery actions, credit for the MDSFWP was given in the recovery of MFW, but not in the recovery of EFW.

With respect to the common cause failure probabilities (CCF) for basic events HPI-CKV-OO-MST and HPI-MOV-CF-SUCT, data from many plants must be combined to estimate the probability of low- and moderate-frequency events because of the sparseness of data. Because of this, the model values will tend toward an average response for a group of plants. Regardless, because basic events HPI-CKV-OO-MST and HPI-MOV-CF-SUCT do not appear in TRANS Sequences 21-8 and 21-9 (which contribute to almost 90% of the total CCDP), any change in the CCF for these basic events will not significantly affect the overall CCDP. In fact, based on TRANS Sequence 21-8 alone, this event qualifies as an ASP event (i.e., CCDP $\geq 10^{-6}$).