50-368

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

August 2, 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 POTENTIAL FOR COMMON MODE FAILURE OF EMERGENCY FEED WATER TRAINS AT ARKANSAS NUCLEAR ONE, UNIT 2

Dear Mr. Yelverton:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of a design condition which was discovered at Arkansas Nuclear One, Unit 2 on July 19, 1995 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 50-368/95-001-00. This analysis was prepared by our contractor at the Oak Ridge National Laboratory (ORNL). The results of this preliminary analysis indicate that this condition 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 design condition as possible. We are aware that you have already changed the plant design to eliminate the common mode failure potential that is being analyzed.

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. 50-369/95-001-00, which documented the event.

Please contact me at (301) 415-1308 if you have any questions regarding this request. 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. 50-368

Enclosures: 1. ASP
2. Guidance for Licensee Review
3. LER

cc w/encls: See next page

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Sincerely,

George Kalman

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

Docket No. 50-368

Enclosures: 1. ASP
2. Guidance for Licensee Review
3. LER

cc w/encls: See next page

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Arkansas Nuclear One, Unit 2

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LER No. 368/95-001

Event Description: Loss of DC bus could fail both EFW trains

Date of Event: July 19, 1995

Plant: Arkansas Nuclear One, Unit 2

Event Summary

During a simulator procedure validation exercise, personnel discovered that both trains of emergency feedwater (EFW) could be failed by the loss of a single train of dc power at Arkansas Nuclear One, Unit 2 (ANO-2). The conditional core damage probability (CCDP) estimated for this event is 3.9×10^{-5}.

Event Description

While validating Abnormal Operating Procedures (AOP) on the plant simulator, a loss of "green" train direct current (dc) power during power operations was simulated. Approximately 3 seconds into the scenario, the main turbine tripped due to loss of dc power to the electrohydraulic control system. The turbine trip resulted in the trip of the main generator output breaker but, due to the loss of dc control power, the generator field breaker did not trip and the generator remained tied to alternating current (ac) bus 2A2 via the Unit Auxiliary Transformer. Generator voltage decayed over the next 30 seconds.

The loss of green-train dc power rendered multiple dependent systems and sub-systems inoperable, including ac buses 2A2 and 2A4, emergency diesel generator (EDG) B, and the A-train turbine-driven EFW pump. In addition, an unexpected interaction rendered the B-train ("red-train") motor-driven EFW pump unavailable. The discharge of EFW pump B can be routed to either steam generator via lines which each contain two isolation valves. The inboard (closest to the pump) valves are normally closed and are supplied by "red train" power. The outboard valves are normally open and are supplied by green-train ac power. These valves have a normally energized green-train dc relay which signals the valves to close on loss of dc control power. When this configuration was designed, it was assumed that any loss of green-train dc power would be accompanied by a simultaneous loss of green-train ac power. During the simulator run, the ac power remained available for approximately 30 seconds. This allowed the B-train EFW isolation valves to close, contrary to design intent. Once closed, the valves could not be reopened until ac power was restored and an open command was given.

Additional Event-Related Information

The Licensee Event Report (LER) for this event indicates that there is no conclusive evidence that the actual plant response would have resulted in complete closure of the affected EFW valves. Based upon a review of plant documentation, the LER indicates that sufficient voltage to operate the EFW isolation valves might only have existed for about 10 s after a trip. In this case, the valves would have only closed partially. In that event, some EFW flow but less than that amount required by Technical Specifications, might have been maintained.

The ANO-2 Individual Plant Examination (IPE) indicates that the expected frequency for the loss of one dc bus is 3.94×10^{-4} per year. The IPE also provides information about the potential impacts of a loss of dc power. Feed and Bleed (F&B) cooling requires that either the high-point vent line or one of the low temperature overpressure (LTOP) paths be opened. The loss of green-train power would render all of these pathways unavailable, hence high pressure recirculation would be unavailable.

In addition, the IPE provides a dependency table which indicates that the following systems, in addition to those already mentioned, are dependent upon green-train dc power: high pressure safety injection (HPSI) train B, shutdown cooling (SDC) train A, and the power conversion system (PCS).

Modeling Assumptions

The wiring logic error apparently existed from the time of a plant modification made in 1984 until it was discovered in 1995. In this analysis, it was assumed that the plant performance would be similar to that of its simulator. For a period of one operating year (1 year at 70% availability, or 6132 h), both trains of EFW were assumed to be initially inoperable given the loss of the green-train dc power. The frequency of this initiator, 3.9×10^{-4} per year, was taken from the ANO-2 IPE. Other systems dependent upon the green-train of dc power were also assumed to be failed: PCS/MFW, SDC A, HPSI B, HPR, and EDG B.

Because it was assumed that manual recovery of EFW train B was possible, a probability of EFW non-recovery of 0.1 was estimated using the methodology detailed in Reference 2.

This event was modeled as the unavailability for one operating year of systems dependent upon green dc power, and the reactor trip frequency was set equal to the loss of dc bus frequency, 3.9×10^{-4} per year. Frequencies for other initiators were set to zero. The calculation thus estimated the core damage frequency associated with a potential transient resulting from loss of "green" dc power during a one-year period.

Analysis Results

The CCDP estimated for this event is 3.9×10^{-5}. The dominant sequence involves:

- a postulated loss of green-bus DC power and a reactor trip,
- the unavailability and non-recovery of EFW,
- the unavailability of main feedwater,
- Safety Relief Valve (SRV) operation and successful SRV reseat, and
- unavailability of the condensate system to remove heat.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences for the condition assessment 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 for the condition assessment. Minimal cut sets associated with the dominant sequences for the condition assessment are shown in Table 5.

Acronyms

ac	alternating current
ANO-2	Arkansas Nuclear One, Unit 2
AOP	Abnormal Operating Procedures
CCDP	Conditional Core Damage Probability
dc	direct current
EFW	Emergency Feedwater
EDG	Emergency Diesel Generator
F&B	Feed and Bleed
HPSI	High Pressure Safety Injection
IPE	Individual Plant Examination
LER	Licensee Event Report
LTOP	Low Temperature Overpressure
PCS	Power Conversion System
SDC	Shutdown Cooling
SRV	Safety Relief Valve

References

1. LER 368/95-001, Rev. 0, "Unanticipated effect of analyzed failure of DC electrical bus upon train of EFW system containing AC motor-driven pump," July 19, 1995.

2. NUREG/CP-0140, Proceedings of the USNRC Twenty-Second Water Reactor Safety Information Meeting, "Methods Improvements Incorporated into the Saphire ASP Models," Sattison et. al., October, 1994.

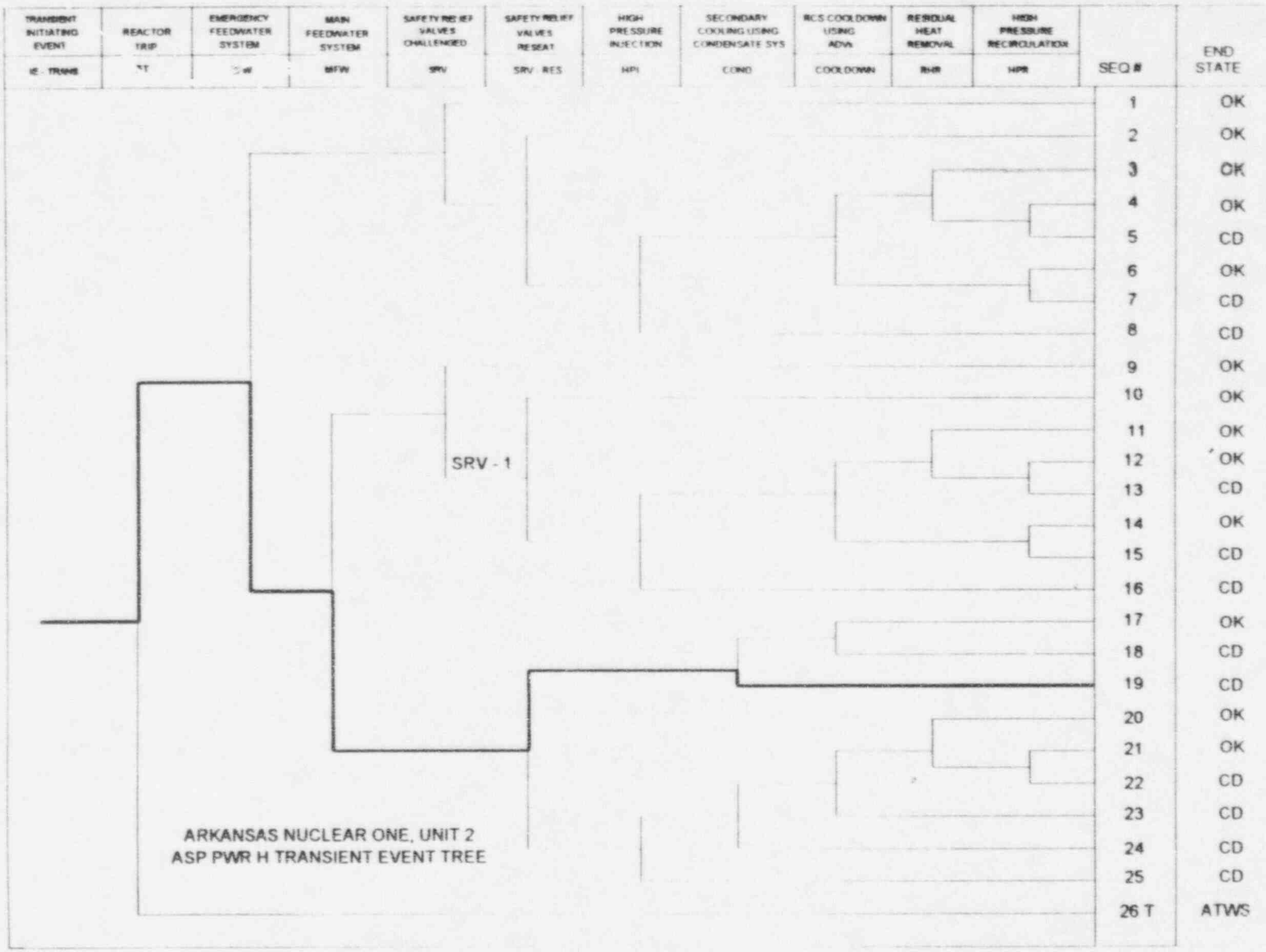

Figure 1. Dominant core damage sequence for LER 368/95-001.

Table 1. Definitions and probabilities for selected basic events for LER No. 369/95-001

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-TRANS	Initiating Event - Transient	0.0 E+000	6.4 E-008		Yes
EFW-XHE-NOREC	Operator Fails to Recover EFW System	2.6 E-001	1.0 E-001		Yes

Table 2. Sequence conditional probabilities for LER No. 368/95-001

Event tree name	Sequence name	Conditional core damage probability (CCDF)	Core damage probability (CDP)	Importance (CCDP-CDP)	Percent Contribution[a]
TRANS	19	3.9 E-005	0.0 E+000	3.9 E-005	100.0

[a] Percent Contribution to total Importance

Table 3. Sequence logic for dominant sequences for LER No. 368/95-001

Event tree name	Sequence name	Logic
TRANS	19	/RT, EFW, MFW, /SRV-RES, COND

Table 4. System names for LER No. 368/95-001

System name	Logic
COND	Secondary Heat Removal Using Condensate System Fails
EFW	No or Insufficient EFW Flow
HPI	No or Insufficient Flow from HPI System
MFW	Failure of the Main Feedwater System
RT	Reactor Fails to Trip During Transient
SRV-RES	SRVs Fail to Reseat

Table 5. Conditional cut sets for higher probability sequences for LER No. 368/95-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets
TRANS Sequence 19		3.9 E-005	
1	100.0	3.9 E-005	EFW-XHE-NOREC
Total (all sequences)		**3.9 E-005**	

[a] The conditional probability for each cut set is determined by multiplying the probability that the portion of the sequence that makes the precursor visible (e.g., the system with a failure is demanded) will occur during the duration of the event by the probabilities of the remaining basic events in the minimal cut set. This can be approximated by $1 - e^{-p}$, where p is determined by multiplying the expected number of initiators that occur during the duration of the event by the probabilities of the basic events in that minimal cut set. The expected number of initiators is given by λt, where λ is the frequency of the initiating event (given on a per hour basis), and t is the duration time of the event (in this case, 6132 h). This approximation is conservative for precursors made visible by the initiating event. The frequency of interest for this event is $\lambda_{TRANS} = 6.4 \times 10^{-4}$/h.

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 and 1996 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 types of initiating events are considered: (1) transients, (2) loss-of-coolant accidents (LOCAs), (3) losses 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 for 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 equipment 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

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* Revision or practices at the time the event occurred.

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 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 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 to 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.

August 18, 1995

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

Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Licensee Event Report 50-368/95-001-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B), 10CFR50.73(a)(2)(ii)(B), and 10CFR50.73(a)(2)(v), enclosed is the subject report concerning the unanticipated effect of an analyzed failure of a DC electrical bus upon the train of the Emergency Feedwater system containing the AC motor-driven pump.

Very truly yours,

Dwight C. Mims

Dwight C. Mims
Director, Licensing

DCM/tfs

enclosure

ENCLOSURE 3

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Arkansas Nuclear One, Unit Two	05000368	1 OF 6

TITLE (4) HUMAN ERROR IN THE DESIGN OF A PLANT MODIFICATION CREATED THE POTENTIAL FOR FAILURE OF ONE DC ELECTRICAL BUS TO HAVE CAUSED CONSEQUENTIAL FAILURE OF THE OPPOSITE TRAIN OF EMERGENCY FEEDWATER

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
07	19	95	95	001	00	08	18	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	98		20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		Specify in
			20.405(a)(1)(iv)	X	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)
Thomas F. Scott, Nuclear Safety and Licensing Specialist	501-858-4623

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 (If yes, complete EXPECTED SUBMISSION DATE.)	X	NO				

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

During validation of procedures on the plant simulator, a condition was discovered where failure of one DC electrical bus could potentially render the opposite train of the Emergency Feedwater (EFW) system inoperable. Consistent with system design, a failure of the green train DC bus would cause a loss of control power to the normally closed green train EFW injection valves and trip of the main turbine generator. This same loss of power could cause the two normally open green powered injection valves in series with the two red powered valves for the motor-driven EFW pump to close enough to restrict flow, contrary to intended system design, during a series of events involving loss of control power to the main turbine generator with its subsequent coast down. The green powered injection valves that closed during the event would not re-open until AC power was manually transferred to Startup Transformer #3 and an open command was present. Upon confirmation of the validity of the condition, the motor-driven EFW pump was declared to be inoperable and a 72 hour Technical Specification action statement was entered until the bus providing power to the normally open green powered valves could be transferred to Startup Transformer #3. The root cause of this condition was determined to be human error during the design of a plant modification installed in the mid-1980s to replace the electro-hydraulic EFW injection valves with motor-operated valves. A modification was completed on July 27, 1995, to correct the condition.

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A. Plant Status

At the time this condition was discovered, Arkansas Nuclear One Unit 2 (ANO-2) was operating at approximately 98 percent power in normal steady-state conditions.

B. Event Description

On July 19, 1995, ANO-2 discovered a condition in which failure of one DC electrical bus [EJ] could render the opposite train of the Emergency Feedwater system (EFW) [BA] inoperable.

At ANO-2, the EFW system has two trains. One train (green) contains a turbine-driven pump (2P-7A) and the other train (red) contains a motor-driven pump (2P-7B). Both pumps are capable of feeding both steam generators. Pump 2P-7A feeds both steam generators through four DC motor-operated valves, two valves for each generator, with two normally closed green-powered valves closer to the pump and two normally open red-powered valves closer to the steam generators. The 2P-7A governor and the common steam line isolation valves receive green DC power. Pump 2P-7B feeds both steam generators through four AC motor-operated valves, two for each generator, with two normally closed red-powered valves closer to the pump and two normally open green-powered valves closer to the steam generators. The normally open valves in the red train have a normally energized green-powered DC control relay. The main function of the relay is for normal open and close operation of the valves. It also provides functions associated with closing the valves for a Main Steam Isolation Signal (MSIS) [JE] and for Engineered Safety Features Actuation System (ESFAS) [JE] override capability. AC buses 2A2 and 2A4, Emergency Diesel Generator (EDG) "B", main turbine Electro-Hydraulic Control (EHC) [TG] controls, and main generator excitation field breaker are dependent upon green DC for control power.

During a validation of Abnormal Operating Procedures on the simulator, a loss of green train DC voltage was initiated from a normal operating configuration. The main turbine tripped in approximately three seconds due to a loss of DC power to the EHC system. Closing the turbine valves tripped the generator and its output breakers, but the generator field breaker did not trip since control power was not available to the trip coil. The main generator remained tied to AC bus 2A2 via the Unit Auxiliary Transformer (UAT). Generator voltage decayed in approximately 30 seconds. The DC control relays for green train valves in the motor-driven EFW pump (2P-7B) discharge were de-energized which initiated closure of those valves. Green AC needed to close those valves was available from the coasting main generator. The simulator model resulted in a coast-down of sufficient duration for these two valves to close. Approximately 30 seconds after initiation of the event, steam generator levels dropped due to shrink and boil-off causing an Emergency Feedwater Actuation Signal (EFAS). EFAS applied open signals to all eight EFW discharge valves and start signals to both EFW pumps. Depending upon the particular scenario in which the DC power loss was initiated in different tests, 2P-7A either did not start or started and tripped. 2P-7B received a start signal. The two AC green-powered valves in the red train (2P-7B discharge) had closed (or partially closed) and could not open because the generator had coasted down and AC power was not available (no fast transfer to Startup Transformer #3 occurred and EDG "B" was inoperable). These valves would not re-open until power was restored to the AC bus and an open command was present. This was an unanticipated effect of green DC bus failure on operability of the red EFW train.

There is no conclusive evidence that actual plant response to this condition would have resulted in a generator coast down of sufficient duration to allow the green train valves to close completely and block all EFW flow. A review of plant design documentation indicated that it is more likely that the generator voltage would decay in approximately 10 seconds and result in a throttling condition allowing some EFW flow, but less than required for operability as defined in Technical Specifications. After confirming that the simulator response reflected possible plant response, the EFW red train was declared to be inoperable at 2010 hours on July 19, 1995, and a 72-hour action statement of Technical Specification 3.7.1.2 was entered. The condition, operability of the EFW red train potentially affected by a green train DC bus failure, was determined not to cause the green EFW train to be declared inoperable since the green train vulnerability to this failure is a basis for designing redundant trains. At 1210 hours on July 20, 1995, AC electrical bus 2A2 that provides power to the green train AC valves was transferred from being supplied by the Unit Auxiliary Transformer to Startup Transformer #3. In this configuration, the green-powered AC valves in the red train would still close following the postulated failure of the green train DC bus, but they would re-open with actuation of an EFAS. The Technical Specification action statement was exited at 1224 hours on July 20, 1995.

C. Root Cause

The root cause of this condition was determined to have been human error in the design of a plant modification installed in 1984 that replaced electro-hydraulic discharge valves in the EFW system with motor-operated valves. During this modification, red and green power was mixed within each EFW flow path to provide single failure protection for the conflicting fail states associated with the dual functions of opening to supply EFW and closing to isolate the steam generators for MSIS. The human error was an assumption by the design engineer that the normally open green-powered valves in the red train would fail "as-is" upon loss of power. This obscure error was not detected during the review process for that modification or during the numerous subsequent reviews of the EFW system that have occurred since then.

The design process attempts to determine the effects of various failure modes; however, the depth of the analysis of each potential fault event is a matter of engineering judgment. At some point in the design phase, assumptions are made which terminate each postulated fault branch. While the specific depth of analysis is not documented for this modification, it was concluded that loss of green DC power in the red EFW train was acceptable based on the assumptions that the valves would normally be open and that there would be no AC power available to close the green AC valves after the turbine tripped. This specific design error was determined to be unique, non-recurring, and not indicative of a programmatic flaw with the modification design process. The modification process has also undergone significant changes since this particular design change was implemented that serve as additional barriers to prevent a similar condition from occurring. A comprehensive program was implemented to improve the quality, depth, and documentation of reviews for plant design changes. The relocation of Design Engineering to the site in 1990 allows for increased involvement during the construction, testing and close-out of design change packages. For these reasons, no corrective actions associated with the modifications program are required for this condition.

D. Corrective Actions

An evaluation of the potential generic implications of this condition was conducted. The unique combination and depth of red and green power interrelations in the EFW system appear to be isolated to this modification. The other train of EFW, other ESFAS actuations, and other ANO-2 systems were examined and found not to have similar problems. The condition was not applicable to ANO-1 which uses modulating EFW discharge valves due to the different design of their steam generators.

On July 27, 1995, a modification to the control relays of the green-powered valves in the red EFW train was completed to correct the potential consequential failure of the red EFW train. Following successful post-modification testing, AC electrical bus 2A2 was transferred from Startup Transformer #3 to the Unit Auxiliary Transformer at 1800 hours.

A "lessons learned" module concerning this condition will be provided to applicable ANO Engineering departments by October 20, 1995.

E. Safety Significance

The EFW system performs the safety function of removing residual heat from the Reactor Coolant System (RCS) [AB] following a reactor trip. It was designed to meet the requirements of General Design Criterion (GDC) 34, which requires that the system have sufficient redundancy to assure that its function can be accomplished assuming a single failure. The potential inability of the red train to supply EFW with a failure of the green train DC bus violates GDC 34 since the safety system function may not be accomplished assuming a single failure. Therefore, this condition represents a potentially significant safety issue.

As described above, a review of design documents indicated that the more likely plant response to the green train DC bus failure would have resulted in the valves in the red train not reaching their fully closed position. Having some amount of EFW flow, even less than required by Technical Specifications for operability, would mitigate the consequences of this accident. Other factors acting to reduce the safety significance are availability of the Station Blackout diesel generator as a backup power supply and the Auxiliary Feedwater system capable of adding feedwater to the steam generators, both of which require operator action. Since these systems were not available for the full period over which this condition existed, credit was not taken for them in evaluating Core Damage Frequency (CDF).

The safety significance of this condition is also mitigated by the expected rapid response of Operations personnel to the postulated inability to supply EFW. Operator response is directed by a "Loss of Feedwater" procedure that verifies the discharge valves are open or directs manually opening them. This recovery is anticipated to occur well before the steam generators dry out and core damage is postulated to occur. Restoration of EFW from the red train to either steam generator will restore the required safety function. Other operator actions to restore electrical power to AC and DC buses that were de-energized will also assist in terminating the event.

From a Probabilistic Safety Analysis (PSA) perspective, this condition is safety significant. When the dependence of the red EFW train on the green DC bus is accounted for in the ANO-2 PSA model, the CDF is estimated to become 6.49E-5/rx-yr. This is a significant increase in the ANO-2 CDF from its estimated value of 3.29E-5/rx-yr, as reported in the ANO-2 Individual Plant Evaluation (IPE). However, the revised CDF value is still well below the NRC safety goal of 1E-4/rx-yr (SECY-91-270). Therefore, this condition is not considered to have represented an undue risk to the public health and safety.