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Precursors to Potential Severe Core Damage Accidents: 1980–1981

A Status Report

W. B. Cottrell J. W. Minarick

P. N. Austin
E. W. Hagen
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Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

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Engineering Technology Division

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A STATUS REPORT

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NOTE

This document is bound in two volumes: Vol. 1 contains the main report and Appendixes A, C, D, E, and F; Vol. 2 contains Appendix B.

LIST OF ACRONYMS AND INITIALISMS

ADS	automatic depressurization system
AFW	auxiliary feedwater
ATWS	anticipated transient without scram
AUO	assistant unit operator
BIT	boron injection tank
BWR	boiling-water reactor
BWST	borated water storage tank
CCW	component cooling water
CFT	core flood tank
CRD	control rod drive
CST	condensate storage tank
CVCS	chemical and volume control system
D	demand
DG	diesel generator
DH	decay heat
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFW	emergency feedwater
EHC	electrohydraulic control
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESW	emergency service water
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPI	high-pressure injection
HPSW	high-pressure service water
HX	heat exchanger
ICS	integrated control system
ID	identification
IE	inspection and enforcement
LCO	limiting conditions for operation
LER	Licensee Event Report
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPR	low-pressure recirculation
LPSI	low-pressure safety injection
MFW	main feedwater
MG	motor generator
MSIV	main steam isolation valve
MSLB	main steam line break
NNI	nonnuclear instrumentation
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
NSST	normal station service transformer
PORV	pilot-operated relief valve

PSRV	pressure safety relief valve
PT	potential transformer
PWR	pressurized-water reactor
RBCCWS	reactor building closed cooling water system
RBEDT	reactor building equipment drain tank
RC	reactor coolant
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RSST	reserve station service transformer
RWST	refueling water storage tank
RWT	refueling water tank
SCS	shutdown cooling system
SDV	scram discharge volume
SFAS	safety features actuation system
SI	safety injection
SIAS	safety injection actuation system
SIV	scram instrument volume
SLBIC	steam line break instrumentation and control
SU	startup
SW	service water
SWC	saltwater cooling
TVA	Tennessee Valley Authority
ΔP	pressure differential

Appendix B

PRECURSOR SUMMARY SHEETS AND EVENT TREES

Fifty-eight PWR and BWR events for the years 1980 and 1981 were selected as precursors to potential severe core damage, based on the selection methods and criteria described in Vol. 1 of this report. These precursors are documented herein. For each precursor, four items are included.

1. A precursor description and data sheet lists the report date, briefly describes the precursor event, identifies what corrective action was taken after the event, and describes the purpose of the failed system or component.

2. An actual-occurrence event tree describes the major functions associated with the reported event which maintained the plant in a safe condition. Failed items on these trees are identified by hatch marks. The actual sequence of the event is indicated by arrows on the appropriate tree branches.

3. A sequence-of-interest event tree describes safety-related functions required to prevent severe core damage for a postulated initiating event. The choice of postulated events is discussed in Vol. 1 of this report. Four degraded or failed-function states are indicated when applicable on the sequence-of-interest tree:

- degraded (D) — when a safety-related function is degraded;
- consequently degraded (CD) — when a safety-related function is degraded as a consequence of a degraded or failed associated function, for example, when an auxiliary feedwater function is degraded during a loss of offsite power because of a diesel generator failure;
- failed (F) — when a safety-related function fails;
- consequently failed (CF) — when a safety-related function fails as a consequence of a failed or degraded associated function, for example, when an auxiliary feedwater function fails because of the failure of both plant emergency diesels during a loss of offsite power (if all auxiliary feedwater pumps are motor driven).

In addition to the four states discussed above, arrows are used in the sequence-of-interest event trees to indicate an inherent unavailability of a particular function at a plant, for example, turbine runback capability following a load rejection from 100% power (which does not exist on all PWRs).

4. A categorization of accident sequence precursors sheet lists characteristics identified for each precursor. These characteristics have been used in trends analyses. In this listing, the entry for DURATION includes "estimated" if an assumed value (based on testing interval) was used because an event-specific value was unavailable.

The 58 precursor descriptions are arranged in numerical order of the associated NSIC accession number. Table B.1 lists the precursor events sorted by accession number. The last column provides page numbers for this volume.

Table B.1. Index of potential precursor descriptions

Table B.1 (continued)

NSIC accession number	Description of actual occurrence	Plant name	Page number
166072	RHR heat exchangers damaged	Brunswick 1	B-163
166082	Loss of RCIC and HPCI systems	Brunswick 2	B-168
166384	Loss of 4.16-kV bus network	Monticello	B-172
166650	SI water supply valve unlocked and closed	Beaver Valley 1	B-176
166745	Two shutdown sequences and one DG unavailable for short time	Palisades	B-180
167117	Low switchyard voltages	Rancho Seco	B-184
167611	Inadvertent loss of RHR system and RCS blowdown occurs	Sequoyah 1	B-188
167624	LOOP and failure of one diesel to start	Crystal River 3	B-193
168548	Switchyard voltage drops below low limit	Rancho Seco	B-197
168829	Safety injection valves fail in an LOFW event	San Onofre 1	B-201
169042	LOOP and failure of HPI valve to open	Arkansas 1	B-206
169587	BIT flow path to RCS obstructed	Turkey Point 4	B-210
170098	Two BIT inlet valves fail to open	Salem 1	B-214
170199	Unavailability of both CCW trains	Kewaunee	B-219
171202	Both diesel generators unavailable	Turkey Point 3	B-223
171667	Loss of vital bus	Davis-Besse 1	B-227
171700	Emergency power unavailable	San Onofre 1	B-232
171733	Auxiliary feedwater pumps fail to auto-start	Zion 2	B-236
171842	Insufficient cooling to DG heat exchanger	Dresden 3	B-240
171939	All steam dump valves opened after a scram	North Anna 2	B-244
172198	MSIV closure and safety valve lift	St. Lucie 1	B-248
174073	Stuck open relief valve	Duane Arnold	B-252

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 154451

Date: February 12, 1980

Title: Inadvertent Opening of PORV at Haddam Neck

The failure sequence was:

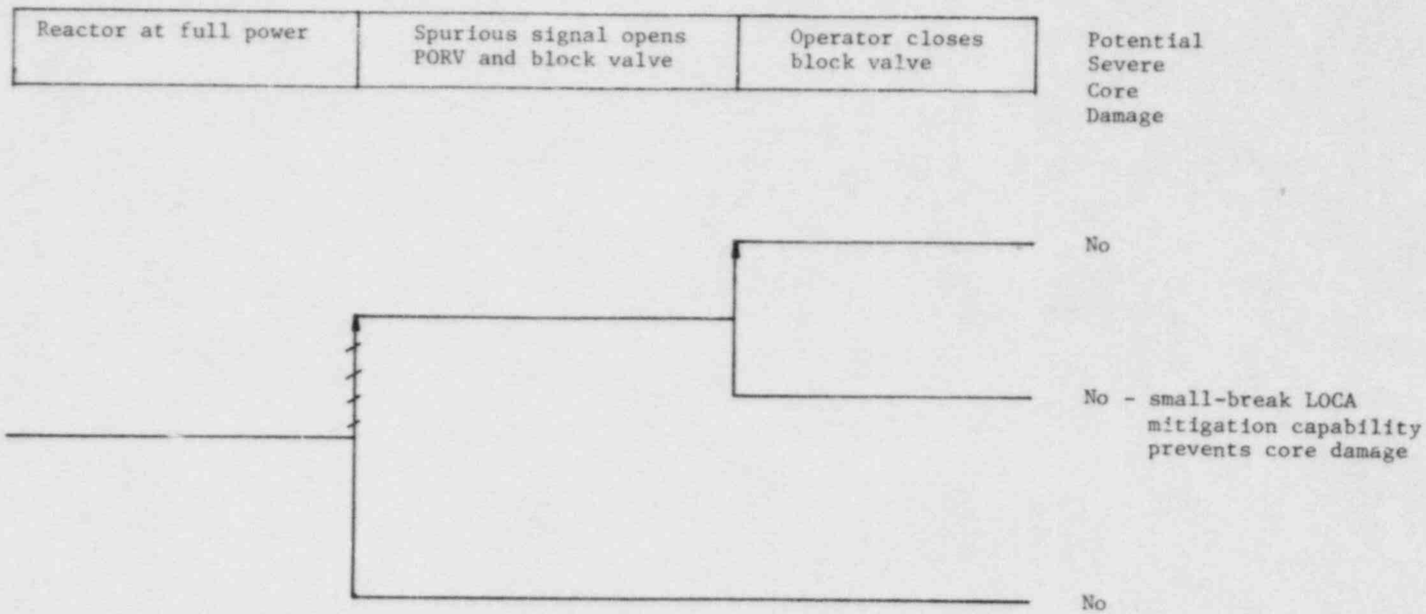
1. With the reactor at full power, spurious signal to the PORV and its isolation block valve was initiated in Pressurizer Pressure Control Channel 1.
2. Both valves opened and the RCS depressurized to 1992 psig.
3. The PORV and its associated block valve were remote-manually closed within approximately 10 seconds of the start of the transient. RCS pressure was restored to greater than 2000 psig within 2 minutes, terminating the transient.

Corrective action:

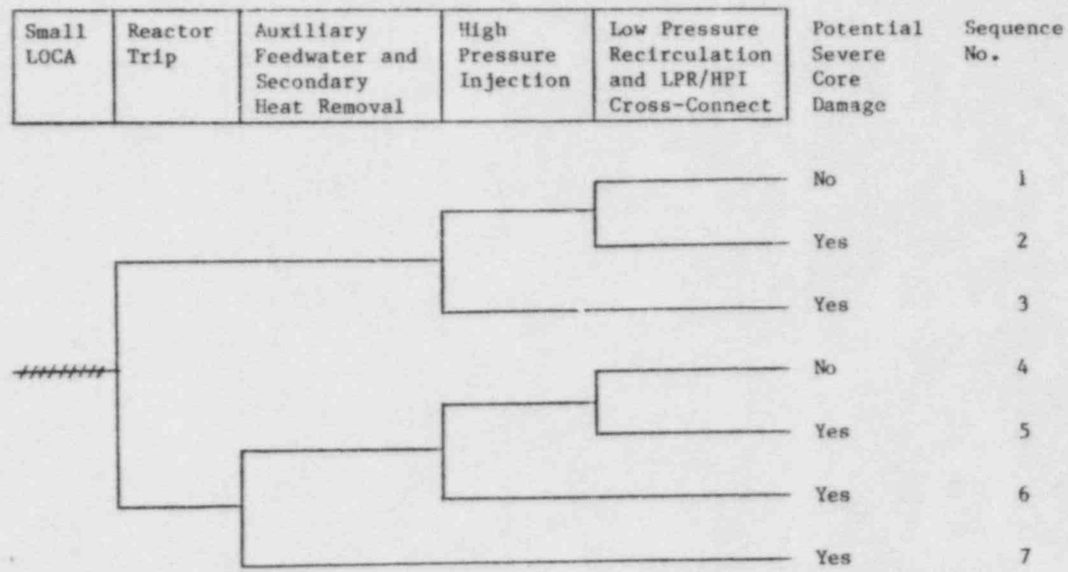
Investigation and testing did not reveal the source of the problem. The channel and valve controls were placed in the normal operating mode.

Design purpose of failed system or component:

PORVs are designed to prevent challenges to the main safety valves and to assist in control of RCS pressure.



NSIC 154451 - Actual Occurrence for Inadvertent Opening of PORV at Haddam Neck



NSIC 154451 - Sequence of Interest for Inadvertent Opening of PORV at Haddam Neck

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 154451

LER NO.: 80-04/3L

DATE OF LER: February 12, 1980

DATE OF EVENT: February 4, 1980

SYSTEM INVOLVED: Pressurizer relief system

COMPONENT INVOLVED: PORV and block valve

CAUSE: Spurious actuation signal

SEQUENCE OF INTEREST: Small break LOCA

ACTUAL OCCURRENCE: Inadvertent opening of a PORV and block valve

REACTOR NAME: Haddam Neck

DOCKET NUMBER: 50-213

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 580 MWe

REACTOR AGE: 12.5 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Connecticut Yankee Atomic Power Co.

LOCATION: 13 miles east of Meriden, Connecticut

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Inadequate performance;
inadvertent operation

DISCOVERY METHOD: Operational event

COMMENT: Actuation logic for the PORV and block valve has been modified from a one channel pressure signal to a 2 out of 3 channels to prevent spurious actuations.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 154674

Date: February 8, 1980

Title: Emergency Power System Unavailable at Calvert Cliffs 2

The failure sequence was:

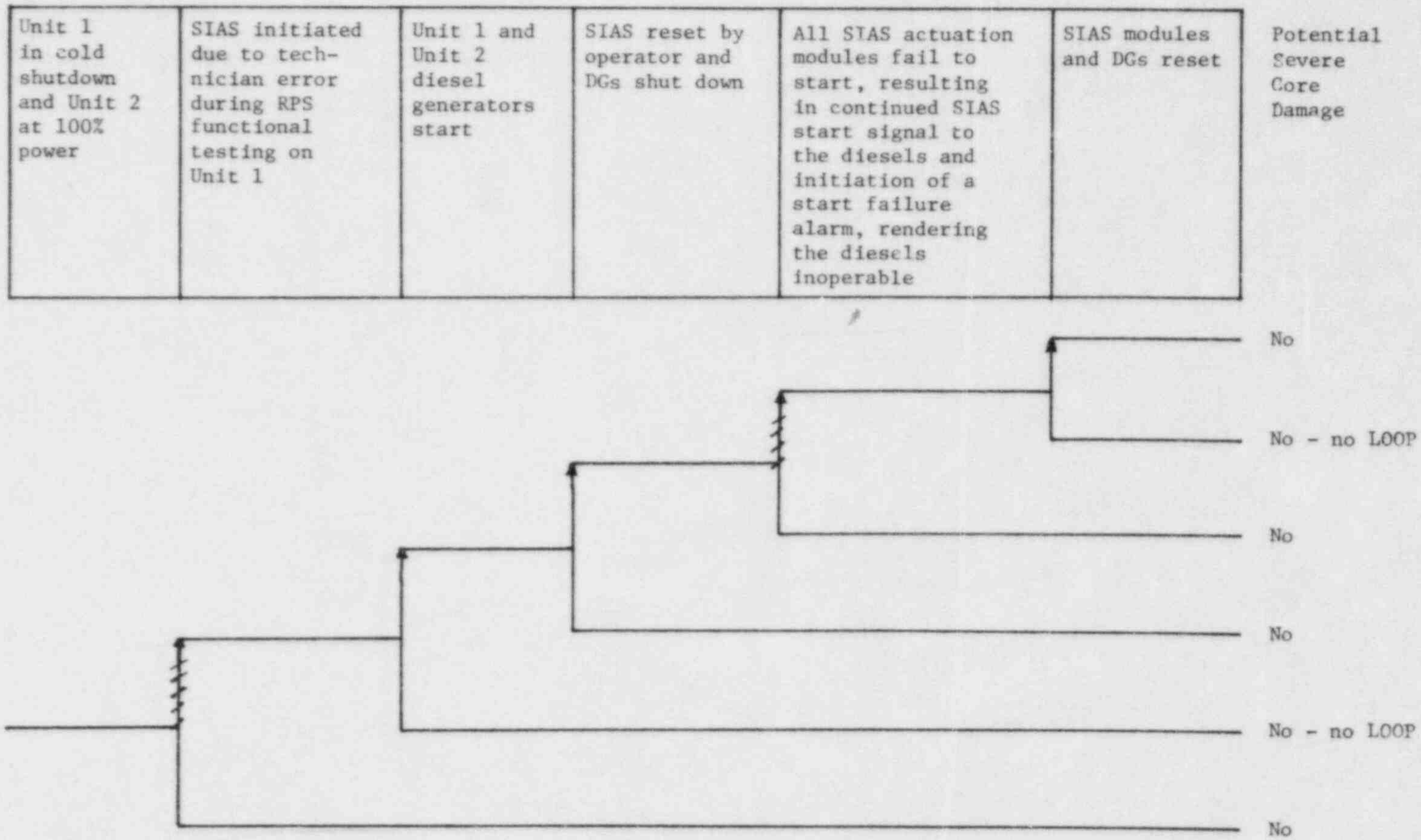
1. With Unit 1 in cold shutdown and Unit 2 at 100% power, safety injection was inadvertently initiated on Unit 1 during surveillance testing, a result of a missed step in the surveillance procedure.
2. Unit 1 SI components actuated, including all three diesel generators.
3. SIAS was reset and the diesel generators shut down.
4. All SIAS actuation modules did not reset, because either individual modules did not reset or because a technician at the ESFAS cabinets interfered with the control room operator when he reset SIAS.
5. Both unit DGs continued to receive a false start signal which caused a start failure alarm.
6. This rendered the DGs inoperable in the event of another SIAS.

Corrective action:

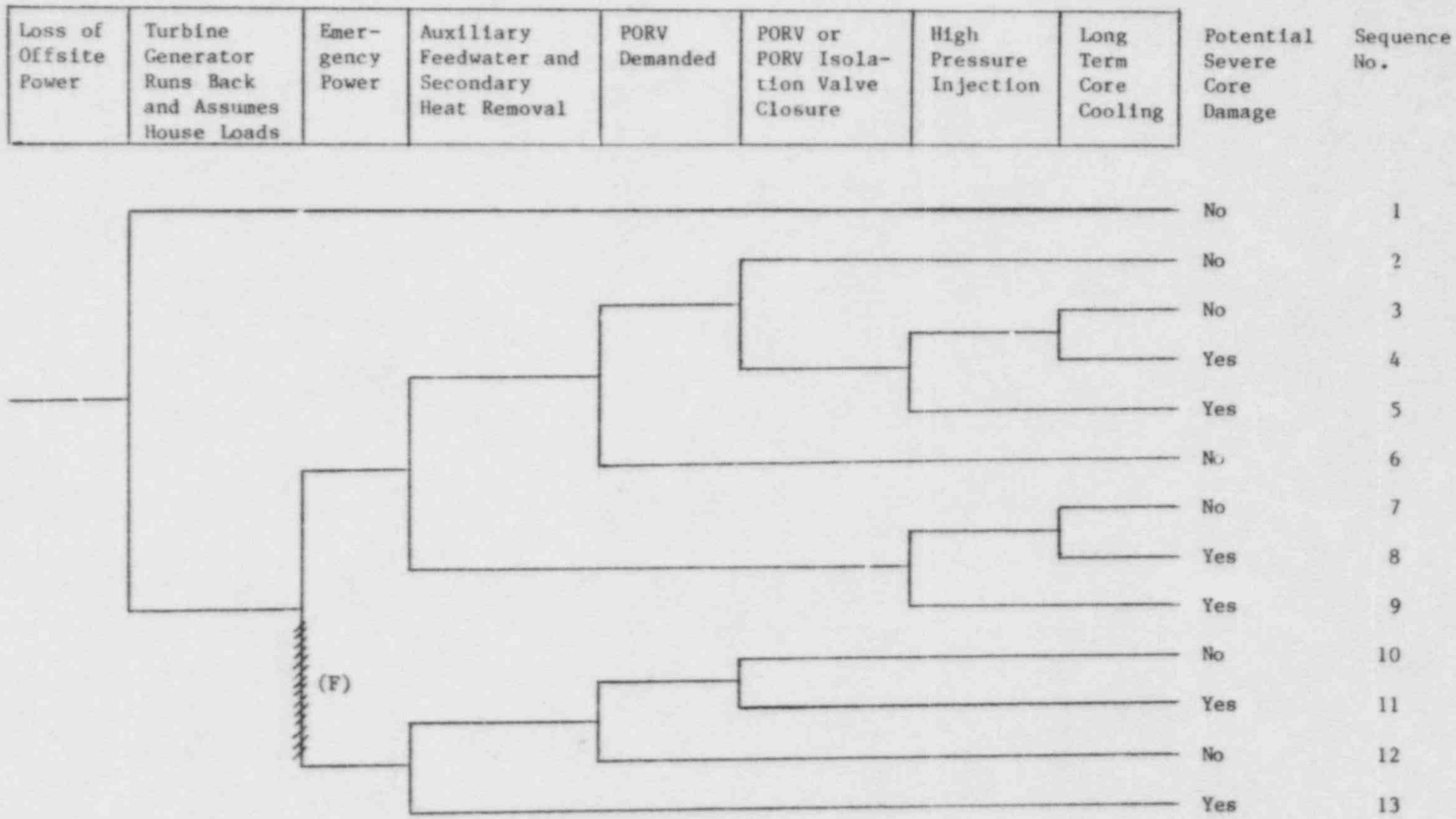
The operators manually reset the ESFAS actuation modules and diesel generator start failure relays at the respective local panels and restored the DGs to service.

Design purpose of failed system or component:

The emergency power system is designed to provide emergency power (three diesel generators) to both Units 1 and 2 in the event of a loss of offsite power.



NSIC 154674 - Actual Occurrence for Emergency Power System Unavailable at Calvert Cliffs 2



NSIC 154674 - Sequence of Interest for Emergency Power System Unavailability at Calvert Cliffs 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 154674

LER NO.: 80-10/3L

DATE OF LER: February 8, 1980

DATE OF EVENT: February 2, 1980

SYSTEM INVOLVED: Emergency power system

COMPONENT INVOLVED: Three diesel generators

CAUSE: SIAS modules improperly reset

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Emergency power system unavailable

REACTOR NAME: Calvert Cliffs Unit 2

DOCKET NUMBER: 50-318

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 845 MWe

REACTOR AGE: 3.2 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Baltimore Gas and Electric

LOCATION: 40 miles south of Annapolis, Maryland

DURATION: 15 min

PLANT OPERATING CONDITION: Unit 2 at 100% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Alarm/annunciator in control room

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Acces ion Number: 155475

Date: March 24, 1980

Title: Total Loss of Saltwater Cooling System at San Onofre 1

The failure sequence was:

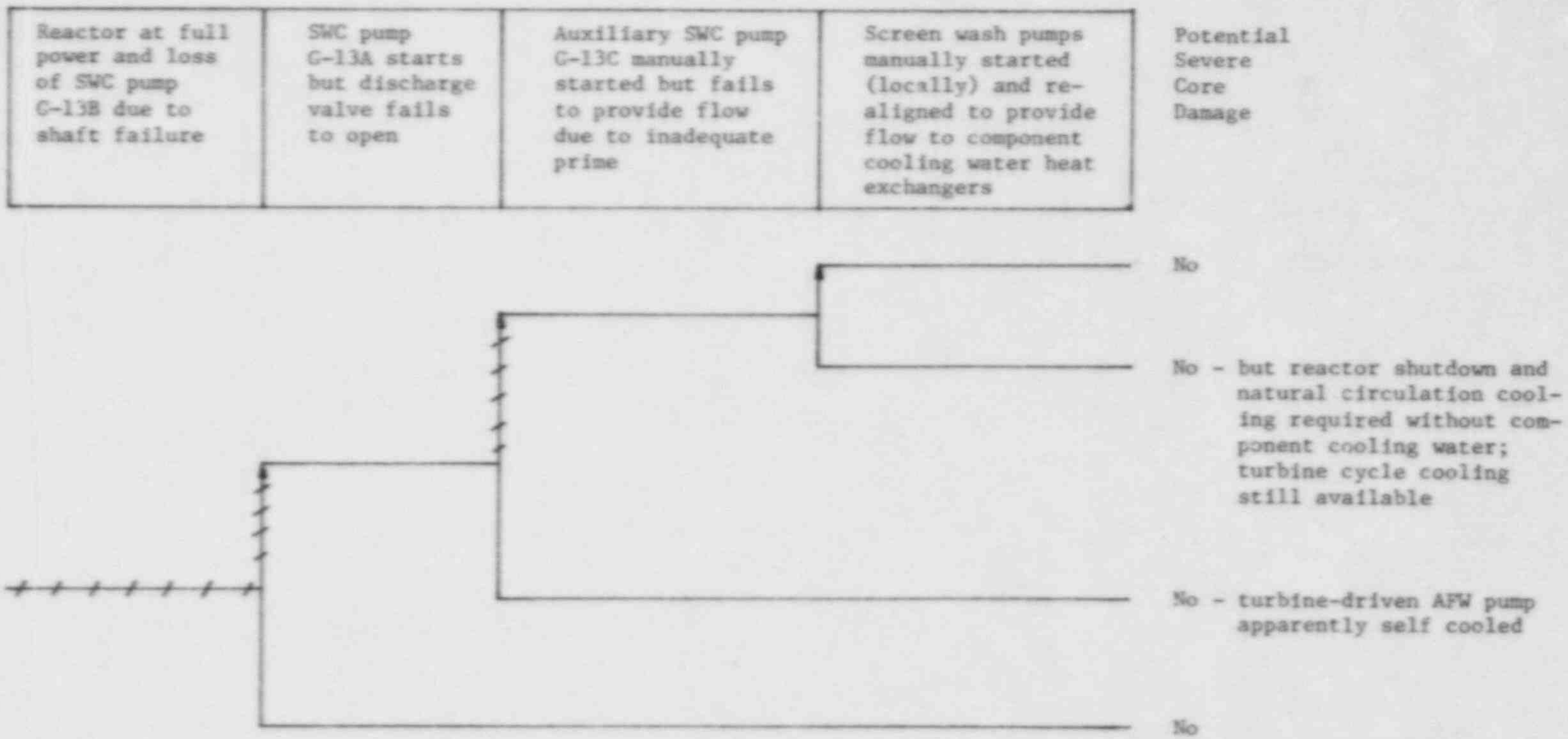
1. With the reactor at full power and south saltwater cooling (SWC) pump G-13B in operation, G-13B shaft failed due to excessive vibration from worn bearings.
2. This resulted in low flow and low discharge pressure alarms in the control room and auto start of north SWC pumps G-13A.
3. North pump G-13A discharge valve failed to open due to an O-ring Failure caused by desiccant in the instrument air lines. This resulted in no flow from either pump.
4. Auxiliary saltwater cooling pump G-13C was manually started from the control room but failed to develop sufficient flow because of insufficient prime, and the pump was shut down.
5. The screen wash pumps were manually started from the local panel and valves manually alig ed to discharge to the bottom component cooling water (CCW) heat exchanger, which reestablished CCW cooling.
6. Adequate prime was restored to SWC pump G13-C and the pump was placed in service.

Corrective action:

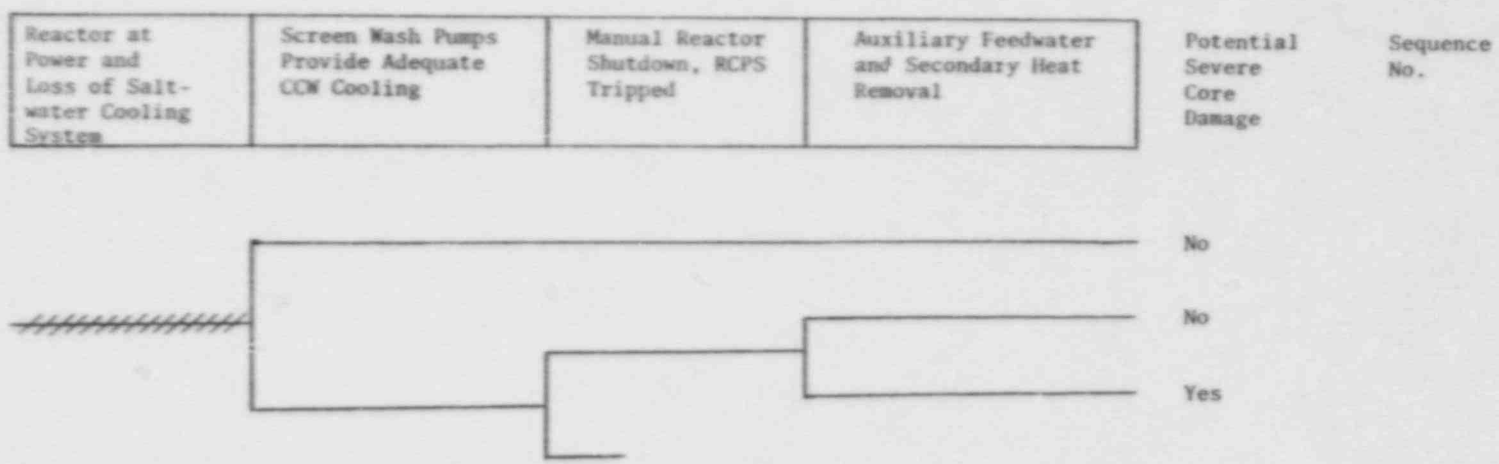
North SWC pump valve POV-5 was opened and the north SWC pump placed in service. The auxiliary SWC pump surveillance testing was increased from once per week to daily during low tide conditions.

Design purpose of failed system or component:

The salt water cooling system provides essential cooling flow to the component cooling water system (RC pumps, RHR pumps, charging pumps, recirculation heat exchanger).



NSIC 155475 - Actual Occurrence for Loss of Saltwater Cooling System at San Onofre 1



NSIC 155475 - Sequence of Interest for Loss of Saltwater Cooling System at San Onofre 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 155475

LER NO.: 80-006

DATE OF LER: March 24, 1980

DATE OF EVENT: March 10, 1980

SYSTEM INVOLVED: Saltwater cooling system

COMPONENT INVOLVED: South pump, north pump discharge valve, auxiliary pump

CAUSE: Pump and valve failures

SEQUENCE OF INTEREST: Loss of service water system

ACTUAL OCCURRENCE: Loss of service water system

REACTOR NAME: San Onofre 1

DOCKET NUMBER: 206

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 436 MWe

REACTOR AGE: 12.7 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Southern California Edison

LOCATION: 5 miles south of San Clemente, California

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Inadequate performance; failed to start

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 156204

Date: August 29, 1980

Title: Failure of MSIVs to Close on Demand at Trojan

The failure sequence was:

1. With the reactor in hot standby, three of four MSIVs failed to close on demand during the annual surveillance test due to sticking valve stems.
2. The plant was taken to cold shutdown.

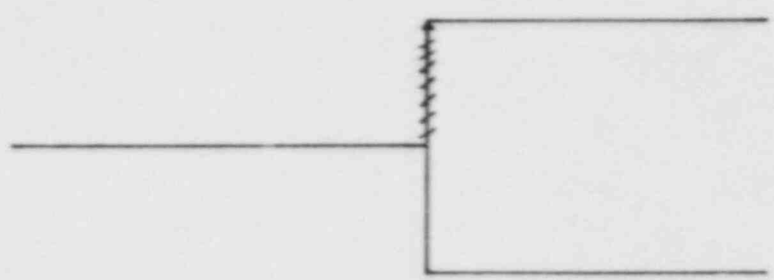
Corrective action:

The short-term corrective action was to loosen the valve stem packings and cycle the valves. Permanent corrective action was still being investigated at the time the report was written.

Design purpose of failed system or component:

The MSIVs are designed to isolate the steam generators during a steam line break to prevent excessive RCS cooldown.

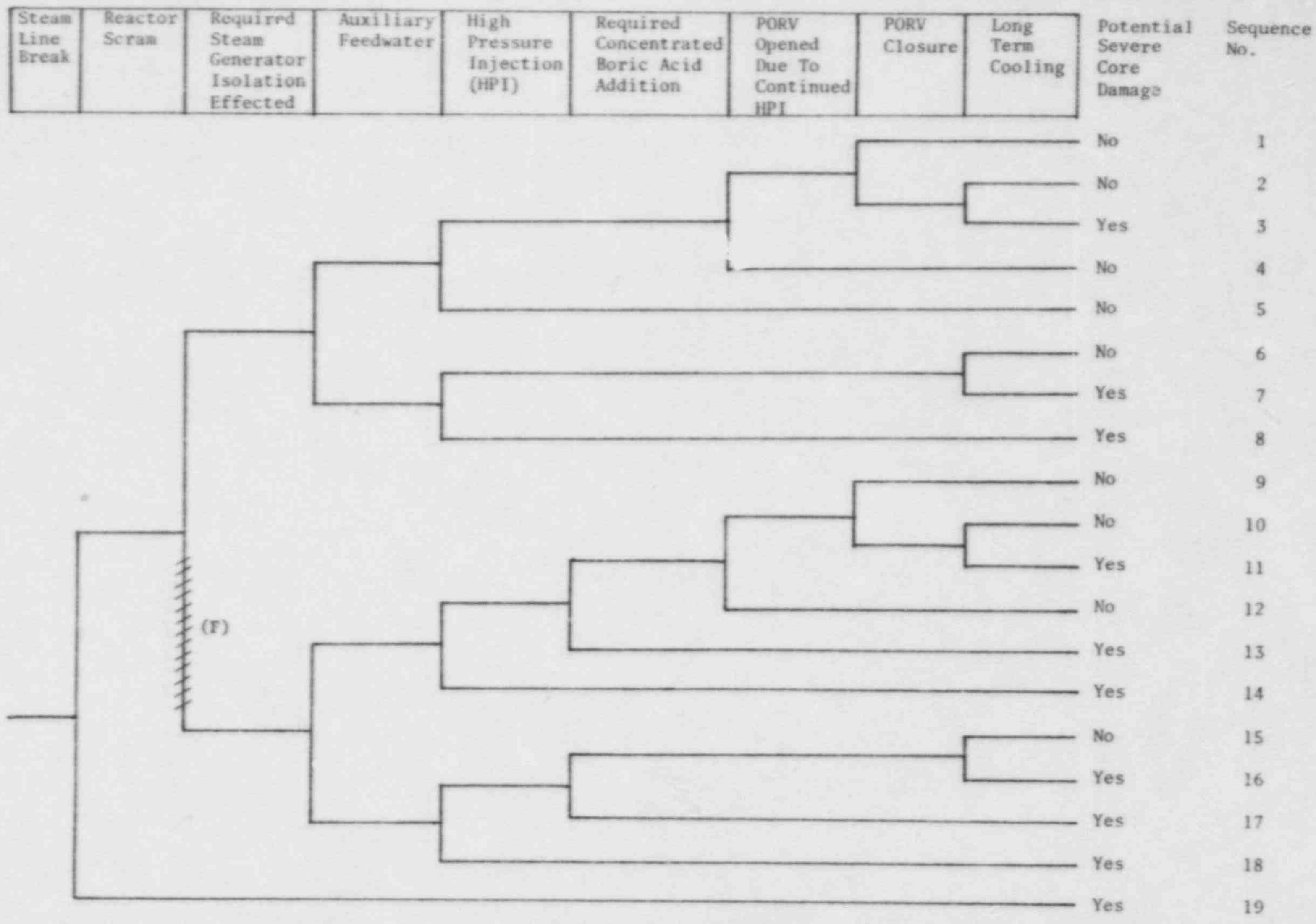
Reactor in hot standby and MSIV surveillance testing	Three of four MSIVs fail to close due to sticking valve stems	Potential Severe Core Damage



No - no demand for valve closure

No

NSIC 156204 - Actual Occurrence for Failure of MSI7s to Close on Demand at Trojan



NSIC 156204 - Sequence of Interest for Failure of MSIVs to Close on Demand at Trojan

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 156204

LER NO.: 80-005 Rev. 1

DATE OF LER: August 29, 1980

DATE OF EVENT: April 11, 1980

SYSTEM INVOLVED: Containment isolation

COMPONENT INVOLVED: Main steam isolation valves (three of four)

CAUSE: Failure to close due to sticking valve stems

SEQUENCE OF INTEREST: Main steam line break

ACTUAL OCCURRENCE: Failure of three MSIVs to close during testing

REACTOR NAME: Trojan

DOCKET NUMBER: 50-344

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1130 MWe

REACTOR AGE: 4.3 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Portland General Electric

LOCATION: 30 miles NW of Portland, Oregon

DURATION: 46 days (estimated)

PLANT OPERATING CONDITION: Hot shutdown

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Surveillance testing

A

COMMENT: Surveillance testing is conducted on these valves with no steam flow through the valves. The valves utilize a swinging disk which is assisted in seating by steam flowing through the valves. The utility felt that these valves would have closed with steam passing through the valves, although the utility was not able to confirm that that would have been the case. Main steam line check valves are provided to permit backflow into the affected steam generator for certain steam line breaks.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158228

Date: July 29, 1980

Title: Loss of Offsite Power at Prairie Island 2

The failure sequence was:

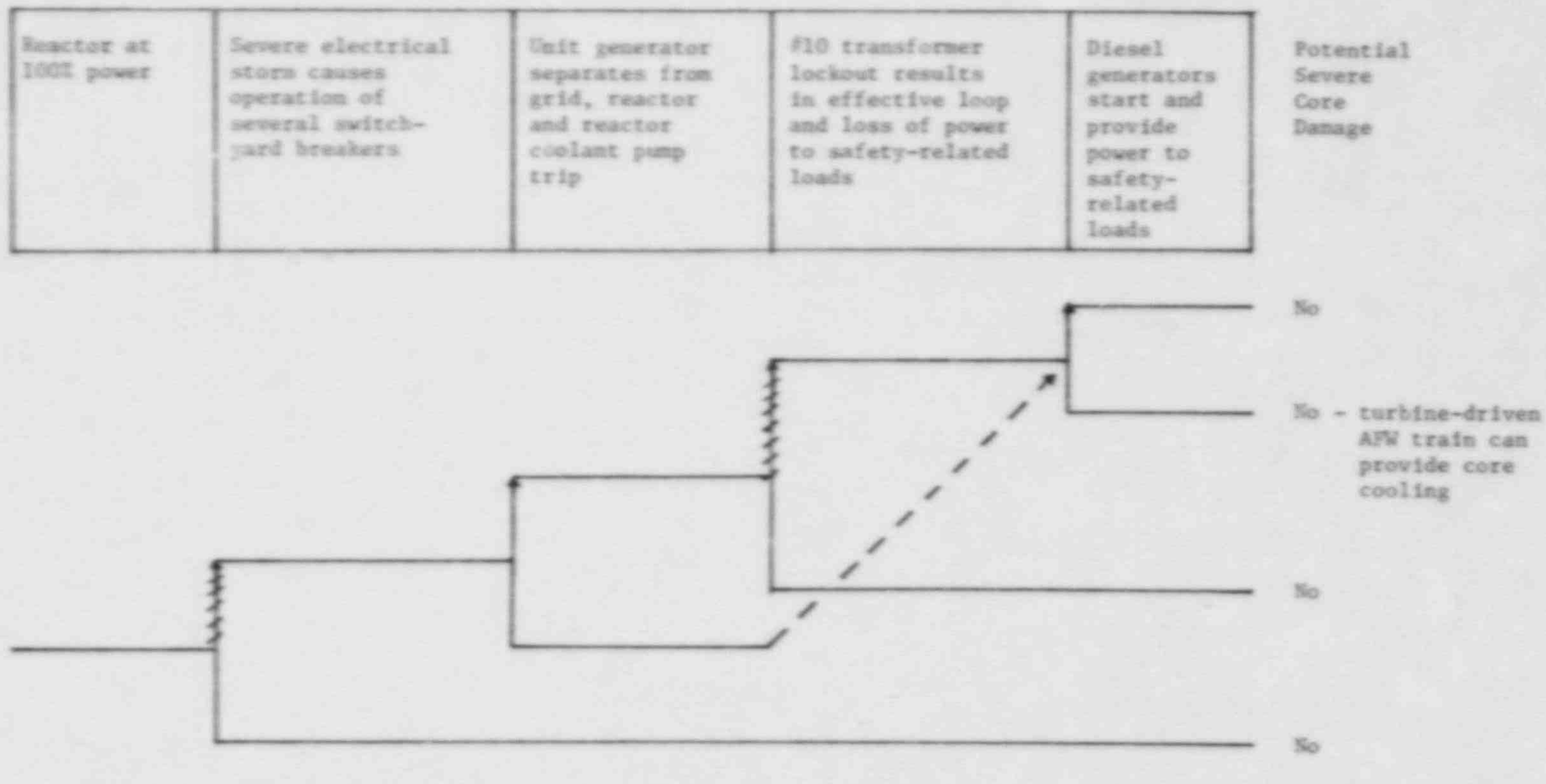
1. With Unit 2 at 100% power a severe electrical storm caused operation of several breakers in the substation.
2. The unit generator separated from the grid and the reactor and reactor coolant pumps tripped.
3. About 8 minutes later, the 345/161/13.8 kV No. 10 transformer locked out.
4. The diesel generators started and provided power to safety-related loads.

Corrective action:

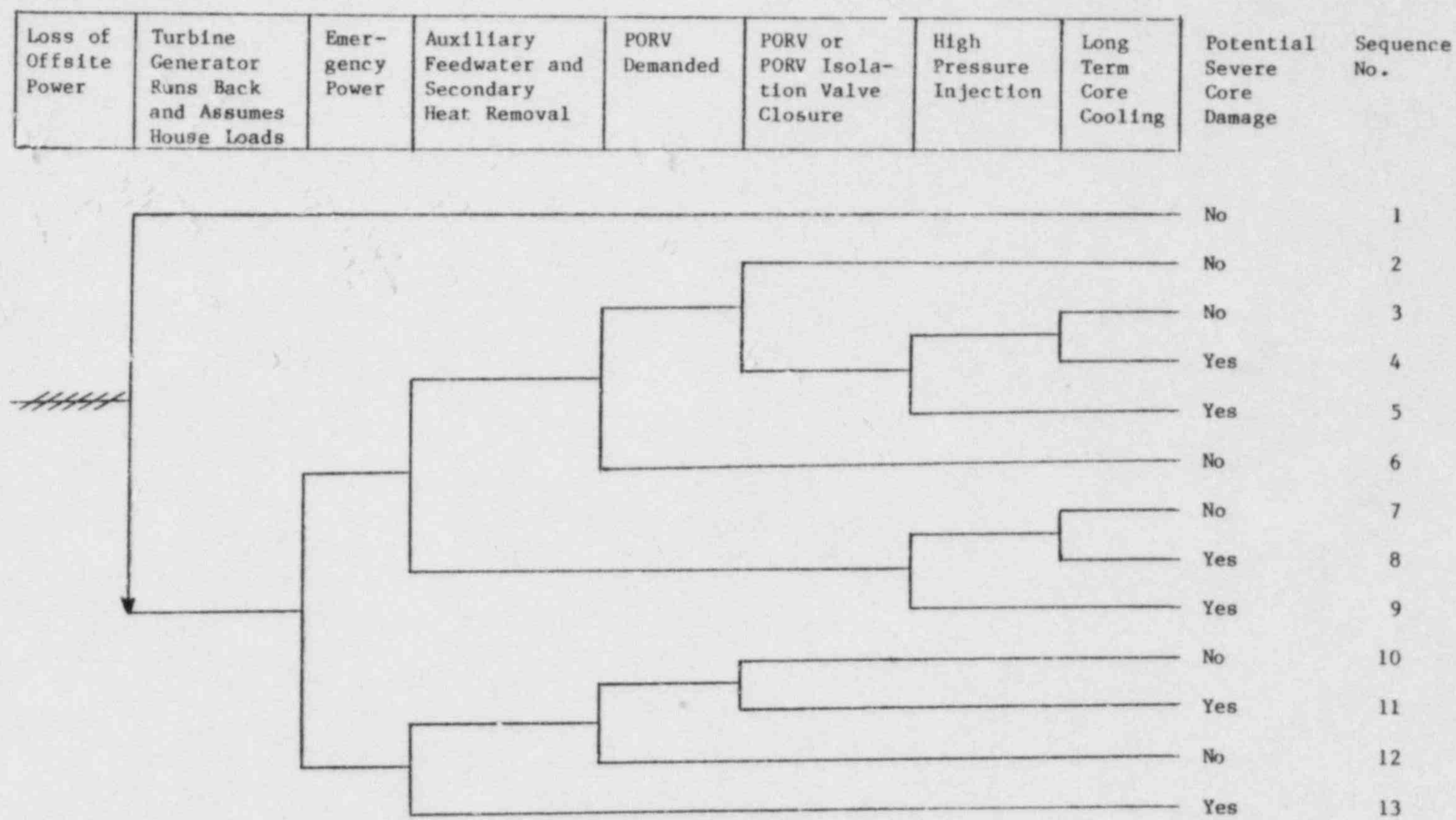
Offsite power sources were restored.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads when the unit generator is not available.



NSIC 158228 - Actual Occurrence for Loss of Offsite Power at Prairie Island 2



NSIC 158228 - Sequence of Interest for Loss of Offsite Power at Prairie Island 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158228

LER NO.: 80-020

DATE OF LER: July 29, 1980

DATE OF EVENT: July 15, 1980

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Switchyard and breakers, transmission lines

CAUSE: Electrical storm

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP

REACTOR NAME: Prairie Island 2

DOCKET NUMBER: 50-306

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 530 MWe

REACTOR AGE: 5.6 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Pioneer

OPERATORS: Northern States Power Co.

LOCATION: 28 miles SE of Minneapolis, Minnesota

DURATION: N/A

PLANT OPERATING CONDITION: Unit 1 at cold shutdown and Unit 2 at full power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158229

Date: July 28, 1980

Title: Scram Discharge Volume Fails to Drain and Alarms Clear at Dresden 3

The failure sequence was:

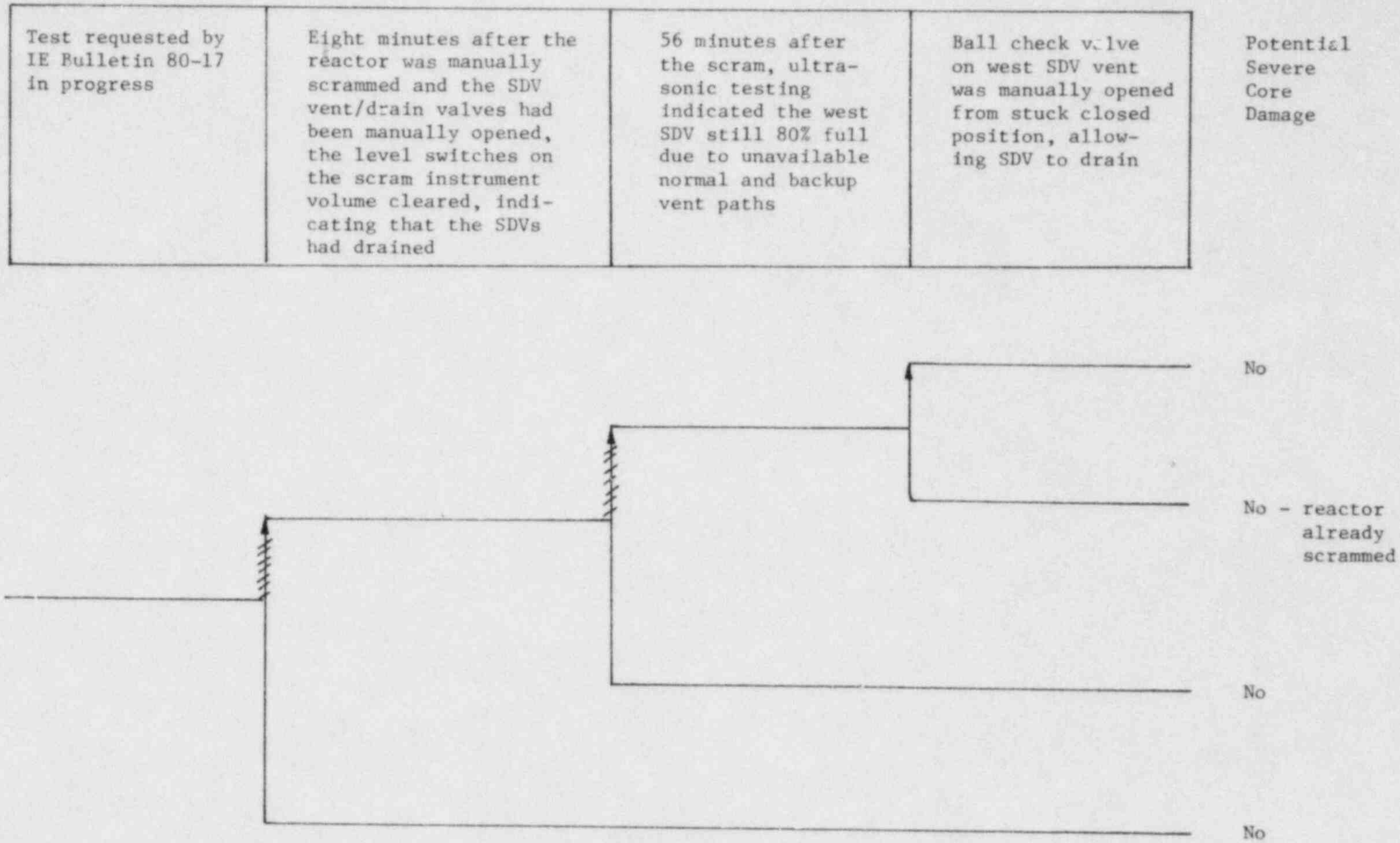
1. Data on BWR scram system was being gathered as requested by IE Bulletin 80-17.
2. The reactor was manually scramed, and the system was aligned to obtain data.
3. Approximately 8 minutes after scram the scram instrument volume (SIV) level instrumentation indicated that the scram discharge volumes (SDVs) had drained.
4. Ultrasonic test, however, showed that the 4-inch piping of the west SDV was still 80% full.
5. The ball check valve (vacuum breaker) on the west SDV alternate vent path was found to be stuck closed.
6. The cause for the failure of west side SDV drainage is believed to be due to unavailability of the normal vent path combined with unavailability of the alternate vent path. Both SDVs are normally vented via a common header that is piped to a vented tank, the RBEDT (reactor building equipment drain tank). [At the time of this event the level in the RBEDT was reported to be above normal, such that the SDV vent header was emersed and could not be vented. The east SDV would also have been prevented from draining if its alternate vent path was not functioning (i.e., if its ball check valve was stuck closed).]

Corrective action:

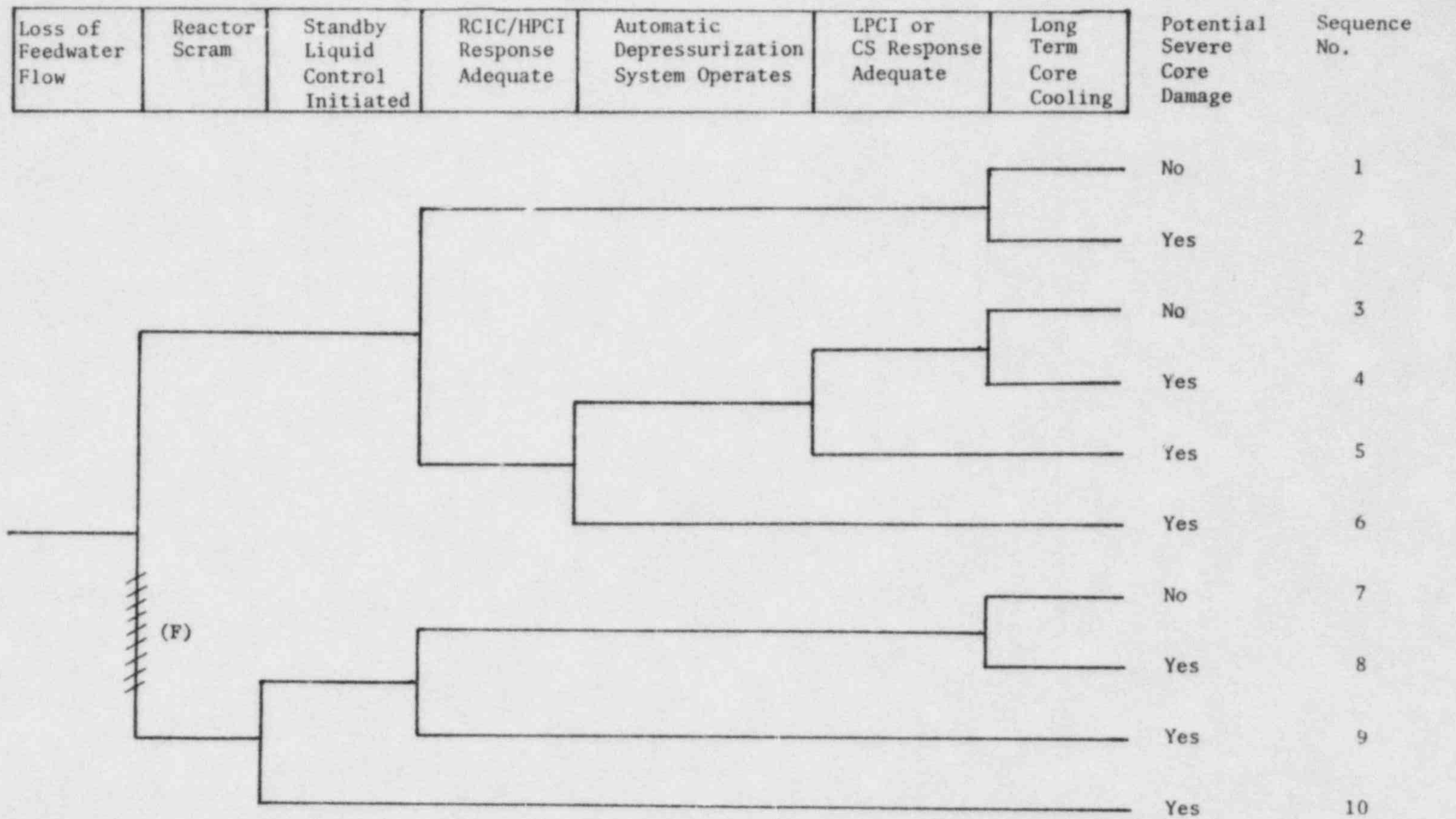
1. The west ball check valve was manually opened establishing a positive venting path, and the SDV drained.
2. A daily ultrasonic test to verify SDV drainage and scram volume availability was immediately implemented.
3. Both east and west ball check valves were removed and cleaned and were shown to operate satisfactorily.
4. An alternate continuous vent path was established that only depends on the SDV vent valves and not the ball valves.
5. Installation of a continuous water level monitoring system was to be investigated.

Design purpose of failed system or component:

The scram system is designed to rapidly insert the control rods in the event of required shutdown.



NSIC 158229 - Actual Occurrence of the Scram Discharge Volume Fails to Drain and Alarms Clear at Dresden 3



NSIC 158229 - Sequence of Interest for Scram Discharge Fails to Drain and Alarms Clear at Dresden 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158229

LER NO.: 80-031

DATE OF LER: July 28, 1980

DATE OF EVENT: July 19, 1980

SYSTEM INVOLVED: Scram system

COMPONENT INVOLVED: Scram discharge volume

CAUSE: Mechanical failure of vent check valve and level switch

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Failure of scram discharge volume (SDV) to drain and SDV level instrument to transmit correct signal

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 9.5 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 9 miles east of Morris, Illinois

DURATION: 24 hours (estimated) (service failure duration would have been controlled by emptying of the RBEDT. The duration is based on an assumed RBEDT drainage once per day (24 h).

PLANT OPERATING CONDITION: 0% power (test scram just initiated)

TYPE OF FAILURE: Inadequate performance;
made inoperable;

DISCOVERY METHOD: Test required by IE Bulletin 80-17

COMMENT: The 4-inch piping of the west SDV was reported to be 80% full. Based on SDV piping information for Dresden 3 available at NSIC, it appears that overall the west SDV was on the order of 40% full. Even with the CRD seal leakage and bypass flow accumulating nominally in excess of 5 gpm, the SDV may have been able to accommodate at least a partial scram. See also NUREG-0090, Vol. 3, No. 2.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158231

Date: May 8, 1980

Title: ADS Valves Fail to Open at Dresden 3

The failure sequence was:

1. During startup on April 25, 1980, following the refueling outage the 3A target relief valve and the 3E relief valves failed to open during surveillance testing. The air operator on the 3A valve was not properly attached. The leak-off line on the 3E electromatic relief valve was obstructed. An orderly shutdown commenced. Valves 3A and 3E were repaired.
2. Startup resumed April 26, 1980; however, electromatic relief valves 3B and 3E failed to open at rated pressure during surveillance. Shutdown commenced and the operators were adjusted.
3. Startup resumed but was terminated when valve 3B again failed. The unit was brought to cold shutdown and 3B was examined and repaired by replacing the pilot valve gasket and adjusting the stroke of the adjustment arm on the pilot valve.
4. Startup resumed again; however, valve 3B again failed to open at rated pressure. Shutdown was initiated and the valve pilot assembly was replaced.
5. Startup began again; however, valve 3B still did not open at rated pressure nor did valve 3C (April 29, 1980). The unit was brought to shutdown and the valve problems investigated. The cutout switch for the solenoid coils was binding on valve 3C. This switch was repaired. The 3B valve was readjusted and the 3/4-in. pilot valve bleed-off discharge line was replaced with a 1-in. line. Similar adjustments and line-size changes were made on all the electromatic relief valves.
6. The unit was started successfully on May 3, 1980, with all valves operating satisfactorily. HPCI was tested and proven operable on each test throughout the event.

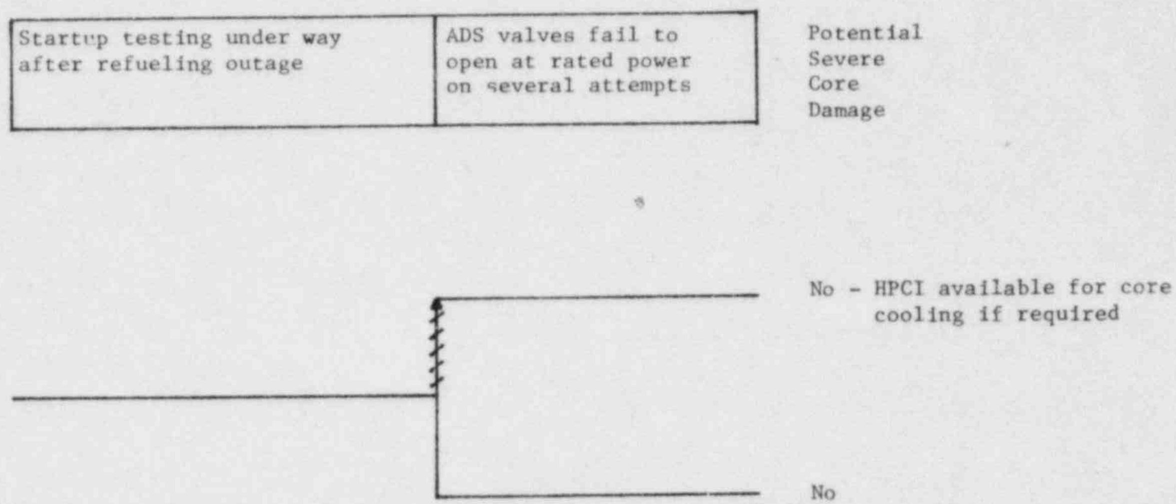
Corrective action:

All malfunctioning valves were repaired or replaced and tested to demonstrate operability.

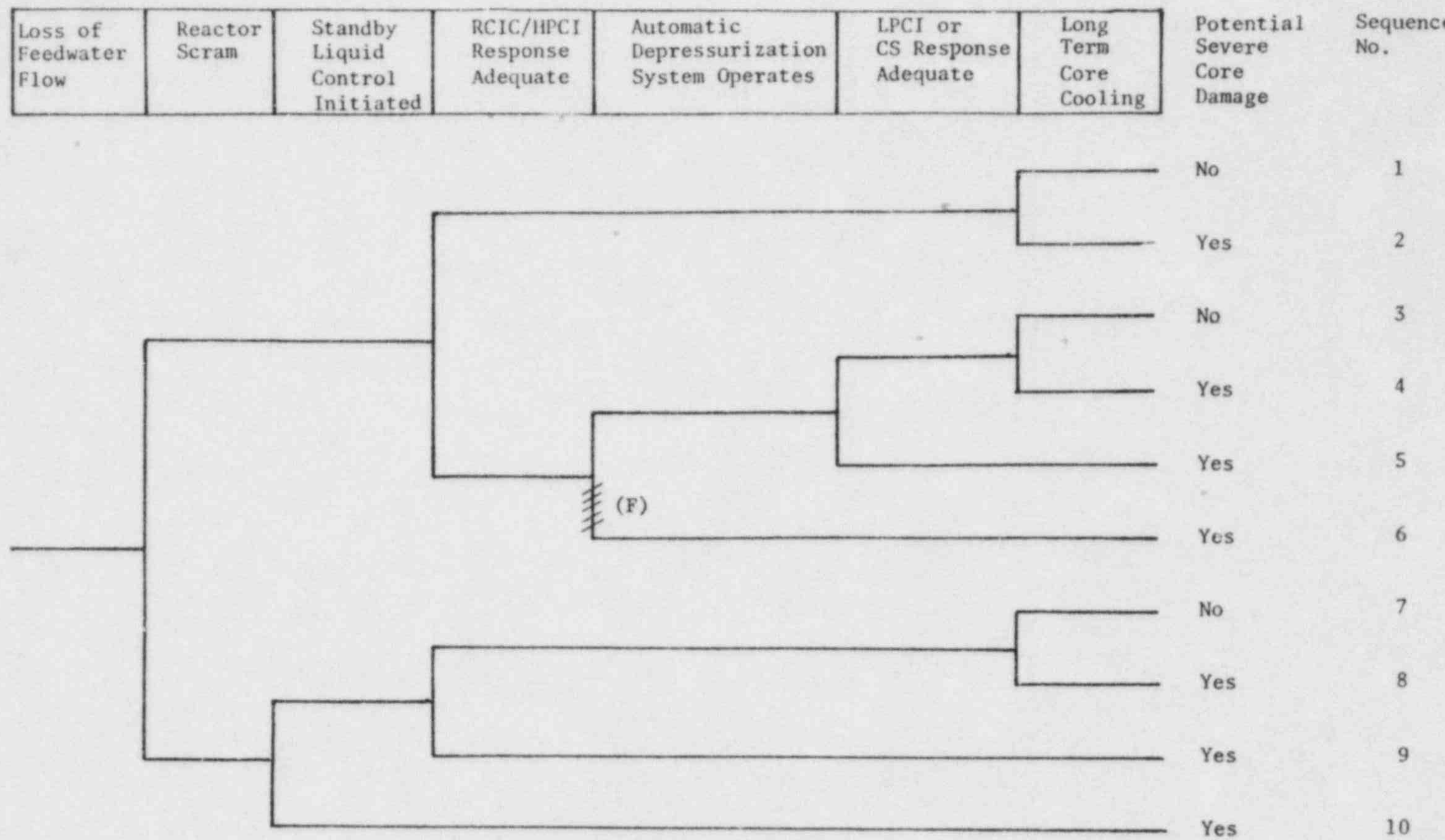
Improved valve design and system configuration was being evaluated for implementation at the next refueling outage.

Design purpose of failed system or component:

The ADS system provides for RCS depressurization to permit use of LPCI and core spray in the event RCIC and HPCI are unavailable for core cooling.



NSIC 158231 - Actual Occurrence of ADS Valves Fail to Open at Dresden 3



NSIC 158231 - Sequence of Interest for ADS Valves Fail to Open at Dresden 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158231

LER NO.: 80-021

DATE OF LER: May 8, 1980

DATE OF EVENT: April 25, 1980

SYSTEM INVOLVED: Automatic depressurization system

COMPONENT INVOLVED: Relief valves

CAUSE: Design and equipment problems

SEQUENCE OF INTEREST: LOFW

ACTUAL OCCURRENCE: Automatic depressurization valves fail to open at Dresden 3

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 9.2 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 9 miles east of Morris, Illinois

DURATION: 984 h (estimated), based on one-half of time between event and last startup on February 2, 1980.

PLANT OPERATING CONDITION: Starting up after refueling

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Testing

COMMENT: Thoroughout the event 4 out of 5 valves failed, but a maximum of 2 out of 5 failed during any one test.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158232

Date: June 17, 1980

Title: LOOP at Indian Point 2

The failure sequence was:

1. With the plant at full power, lightning struck a 345 kV/138 kV transmission tower between two substations.
2. A transmission tower shield line fell, which faulted four feeder lines, resulting in the loss of all external grid system supplies to the Buchanan Substation.
3. Two gas turbines previously brought on line as part of Con Edison's storm watch contingency operations were insufficient to supply the load connected to the "islanded" Buchanan Substation and tripped off following the loss of all 138 kV supplies to the Buchanan Substation.
4. This loss of power at Buchanan directly caused the two 6.9 kV buses normally fed from offsite power to deenergize, along with their associated 480 V buses at IP2. The remaining four 6.9 kV buses (and the two remaining safeguards 480 V buses) which are normally supplied directly from the IP2 unit output via the unit auxiliary transformer remained energized since the unit was still online.
5. The three emergency diesels automatically started, as designed, upon the loss of a 480 V bus caused by loss of the 6.9 kV buses.
6. Loss of one of the two 480 V buses associated with the loss of the two offsite 6.9 kV buses also caused a loss of the rod position indication system and a subsequent turbine runback to 70% power per design; core T_{ave} increased.
7. The condenser steam dump system operated resulting in a decreased core T_{ave}.
8. The operator began a manual plant shutdown (manual turbine trip). An auto trip from the LOOP occurred first by the opening of breaker 7 in the North 345 kV ring bus at Buchanan, thereby disconnecting the Indian Point 2 main generator from the electrical grid and deenergizing the remaining 6.9 kV buses. This caused an auto-generator/turbine trip, loss of power to all four reactor coolant pumps via loss of the remaining 6.9 kV buses and the deenergizing of the remaining 480 V buses.
9. When the main generator tripped, the idling emergency diesel generators closed onto and energized the 480 V buses.
10. With loss of all RCPs, RCS natural circulation cooling was begun. Gas turbine GT-1 was started and placed on standby.
11. A spurious signal from steam line break instrumentation initiated engineered safegards equipment. (Mismatched flow through atmospheric steam dump valves.)
12. All ESF systems functioned properly.

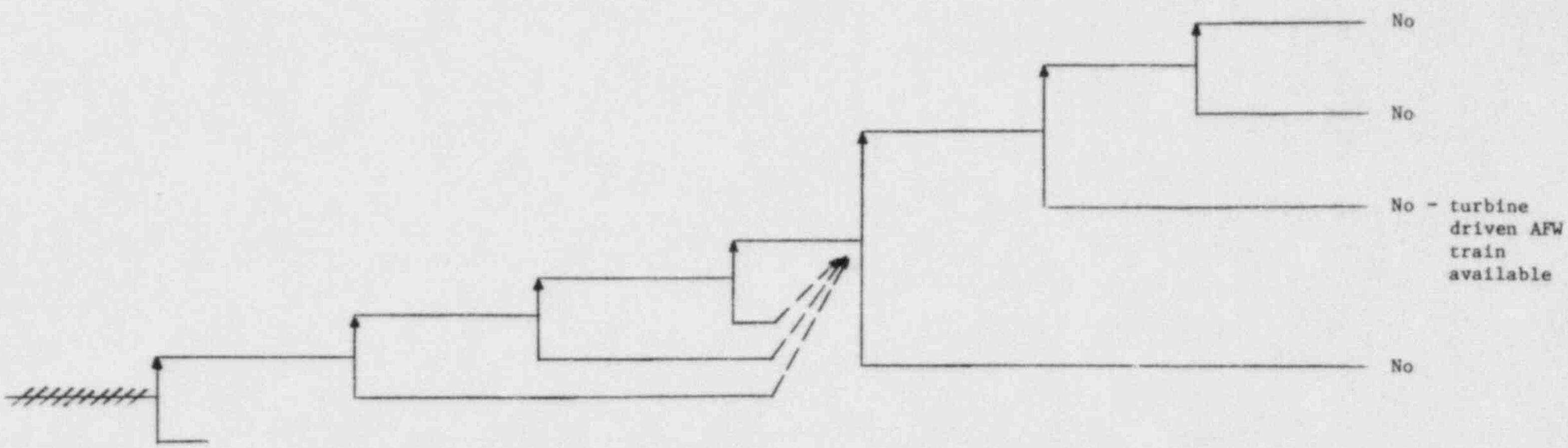
Corrective action:

Offsite power was restored.

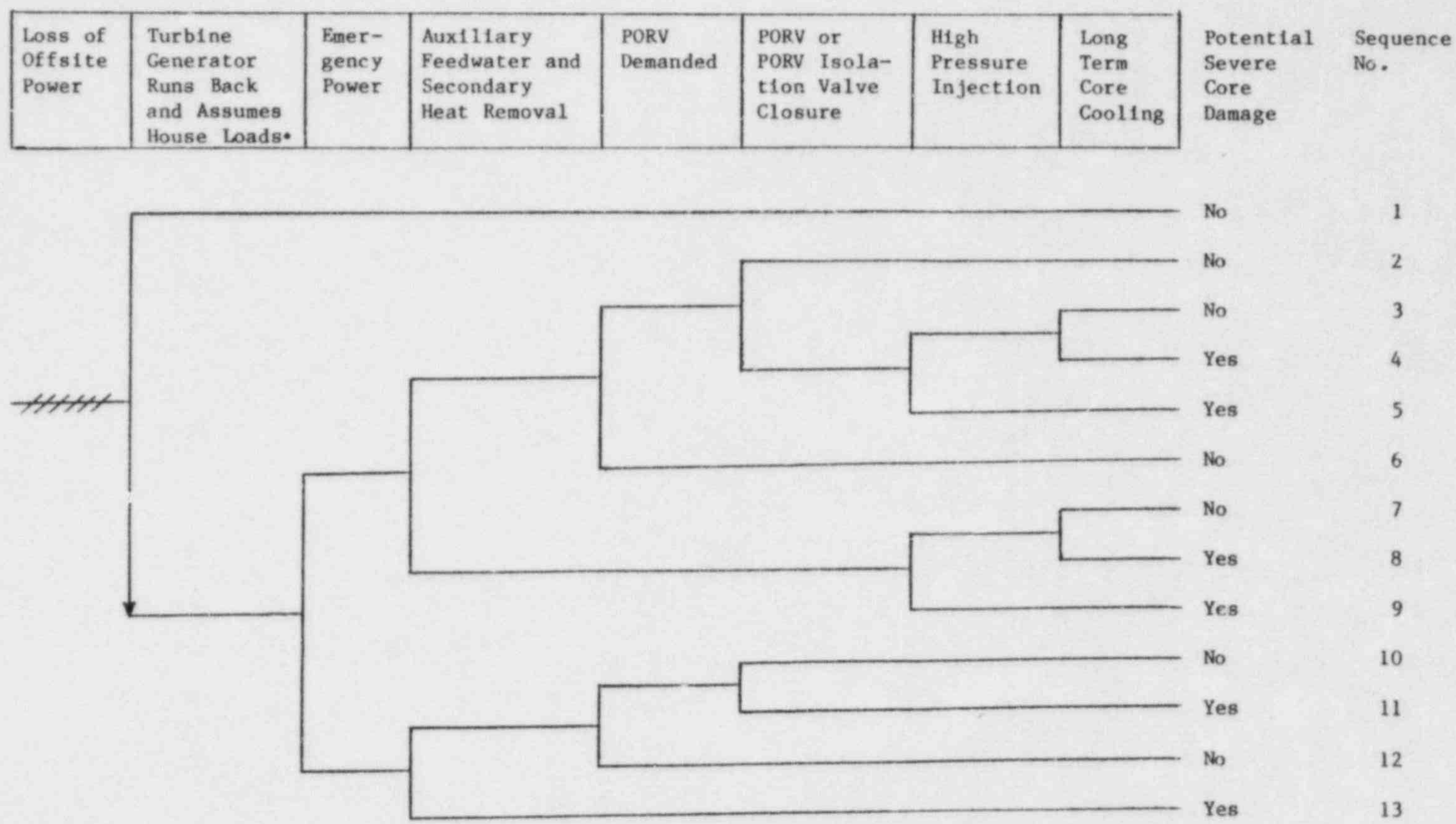
Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads when the unit generator is not available.

Reactor at power and lightning strike results in loss of four feeder lines	Two previously started gas turgine generators trip due to loss of 138-kV supply at Buchanan substation	Loss of power to two 6.9-kV and two 480-V buses results in diesel generator start	Loss of the 480-V buses results in 70% turbine run-back and loss of rod position indication	Operator initiates manual turbine trip	Generator trip and subsequent reactor/turbine trip from breaker opening on north ring bus; loss of all 6.9-kV buses	Emergency generators energize safety-related buses	Spurious safety injection due to steam line differential pressure	Potential Severe Core Damage



NSIC 158232 - Actual Occurrence for Loss of Offsite Power at Indian Point 2



NSIC 158232 - Sequence of Interest for Loss of Offsite Power at Indian Point 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158232

LER NO.: 80-006/99T Rev. 1

DATE OF LER: June 17, 1980

DATE OF EVENT: June 3, 1980

SYSTEM INVOLVED: Offsite power system

COMPONENT INVOLVED: Transmission lines

CAUSE: Lightning strike on transmission tower

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP

REACTOR NAME: Indian Point 2

DOCKET NUMBER: 50-247

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 873 MWe

REACTOR AGE: 7.0 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: United Engineers and Constructors

OPERATORS: Consolidated Edison

LOCATION: 25 miles north of New York, New York

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operation event

COMMENT: LER revised Sept. 5, 1980; see LER-80-006/99X-1.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158233

Date: June 25, 1980

Title: Loss of Reactor Coolant Pumps and Top Head Bubble Incident at St. Lucie 1

The failure sequence was:

1. With the reactor at 100% power, moisture in the vicinity of the solenoid for valve HCV-14-6 resulted in a short circuit across the solenoid valve terminal board. This caused one of the two in-series containment isolation valves in the component cooling water (CCW) common return line from all reactor coolant pumps (RCPs) to fail closed. This resulted in unavailability of component cooling water to all RCPs.
2. The operators manually tripped the plant and RCPs seven minutes later.
3. Natural circulation flow using the steam generators (SGs) was initiated and plant cooldown was began.
4. The CCW flow was restored by jumping the failed valve at 1-1/2 hours after its failure but plant cooldown was continued.
5. The cooldown rate did not exceed Tech. Spec. limits, but unknown to the operator, the top head was not cooling down as fast as the primary loops.
6. While reducing pressurizer pressure (using auxiliary spray from the charging system) to initiate the shutdown cooling system (SCS), the top head water flashed to steam and formed a large steam bubble. The bubble was not initially detected by the operators.
7. Pressure and level oscillations occurred in the reactor coolant system (RCS) as a result of the steam bubble. Pressurizer level fluctuations continued for approximately 6 h. Each time charging was shifted to the cold leg, the steam bubble slowly condensed and shrank.
8. The plant finally cooled down to the SCS entry condition and entered SCS at 235 psi. At this time the SCS relief valves opened, slowly discharging RCS water to the refueling water tank, a result of a partially opened valve in the recirculation line from LPSI pump 1B. (The LPSI pumps double as SCS pumps). The second LPSI was subsequently initiated and operated in the injection mode to restore RCS pressure until the shutdown cooling loops were isolated from the RWST (approximately 1-1/2 h).
9. Conduction cooling of the top head eventually collapsed the bubble.

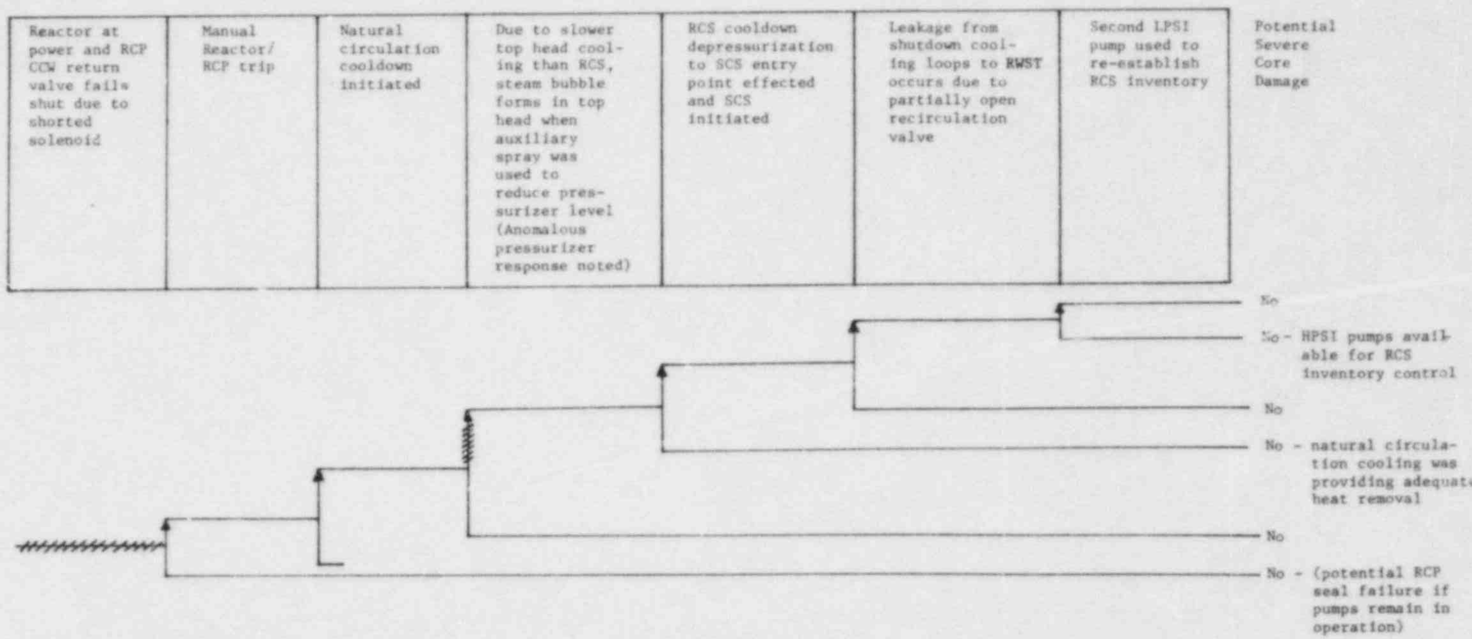
Corrective action:

1. The solenoid and terminal block were replaced.
2. The RCP seals required replacement.
3. The blowdown line which was the source of steam that caused the short was modified by replacing flanged (leaking) joints with welded joints.

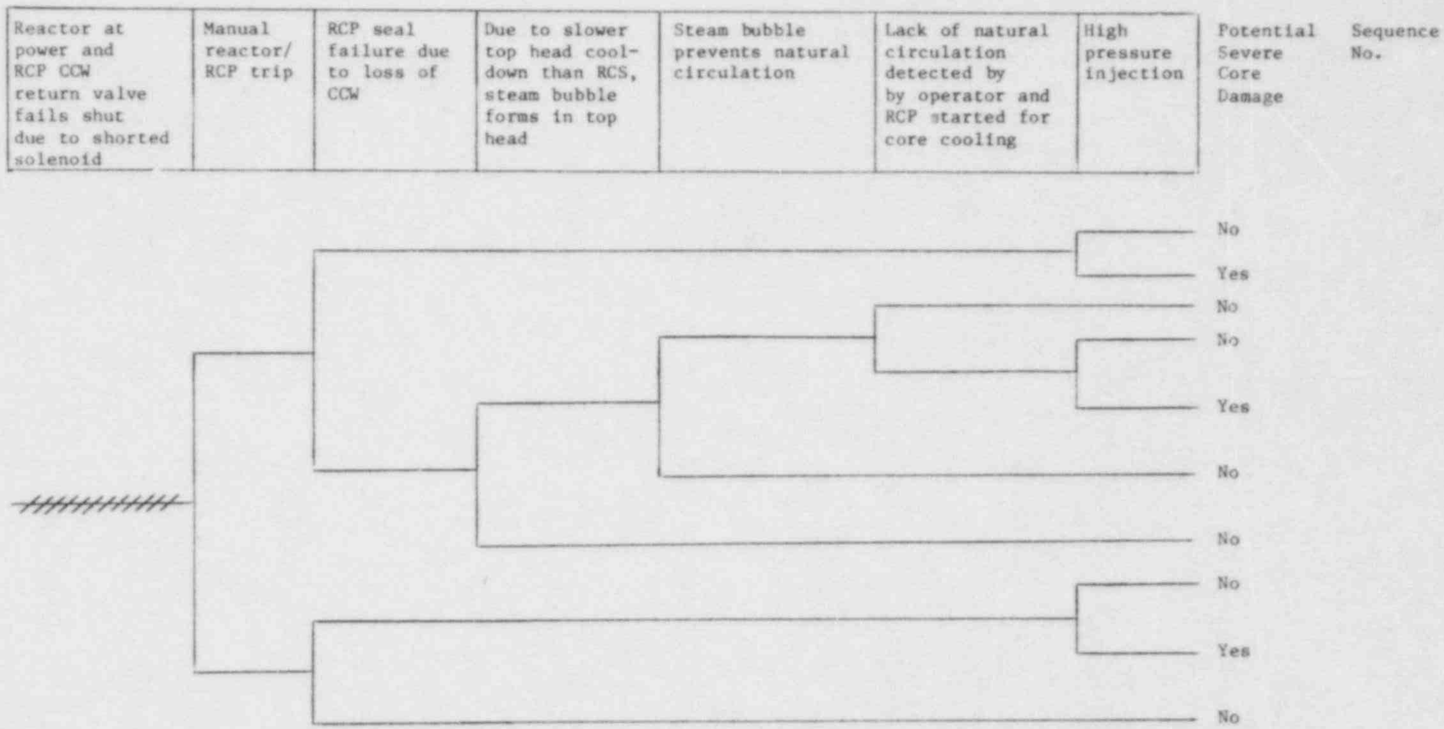
4. A backup nitrogen supply was provided to the RCP CCW isolation valves.
5. A manual jacking device was made more accessible.

Design purpose of failed system or component:

1. The CCW RCP discharge line is common to all RCPs and is part of the RCP seal cooling system.
2. The LPSI recirculation line is used to prevent deadheading of the pump.



NSIC 158233 - Actual Occurrence for Loss of Reactor Coolant Pumps and Top Head Bubble Incident at St. Lucie 1



NSIC 158233 - Sequence of Interest for Loss of Reactor Coolant Pumps and Top Head Bubble Incident at St. Lucie 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158233

LER NO.: 80-029

DATE OF LER: June 25, 1980

DATE OF EVENT: June 11, 1980

SYSTEM INVOLVED: Component cooling water, reactor coolant

COMPONENT INVOLVED: Solenoid valve

CAUSE: Short circuit resulted in a failed-closed solenoid valve

SEQUENCE OF INTEREST: Loss of RCP seal cooling

ACTUAL OCCURRENCE: Loss of RCP seal cooling

REACTOR NAME: St. Lucie 1

DOCKET NUMBER: 50-335

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 802 MWe

REACTOR AGE: 4.1 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Florida Power and Light

LOCATION: 12 miles SE of Ft. Pierce, Florida

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Inadequate performance;
failed to start

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158279

Date: July 14, 1980

Title: Degraded Emergency Feedwater System During a Loss of Offsite Power at Arkansas Unit 2

The failure sequence was:

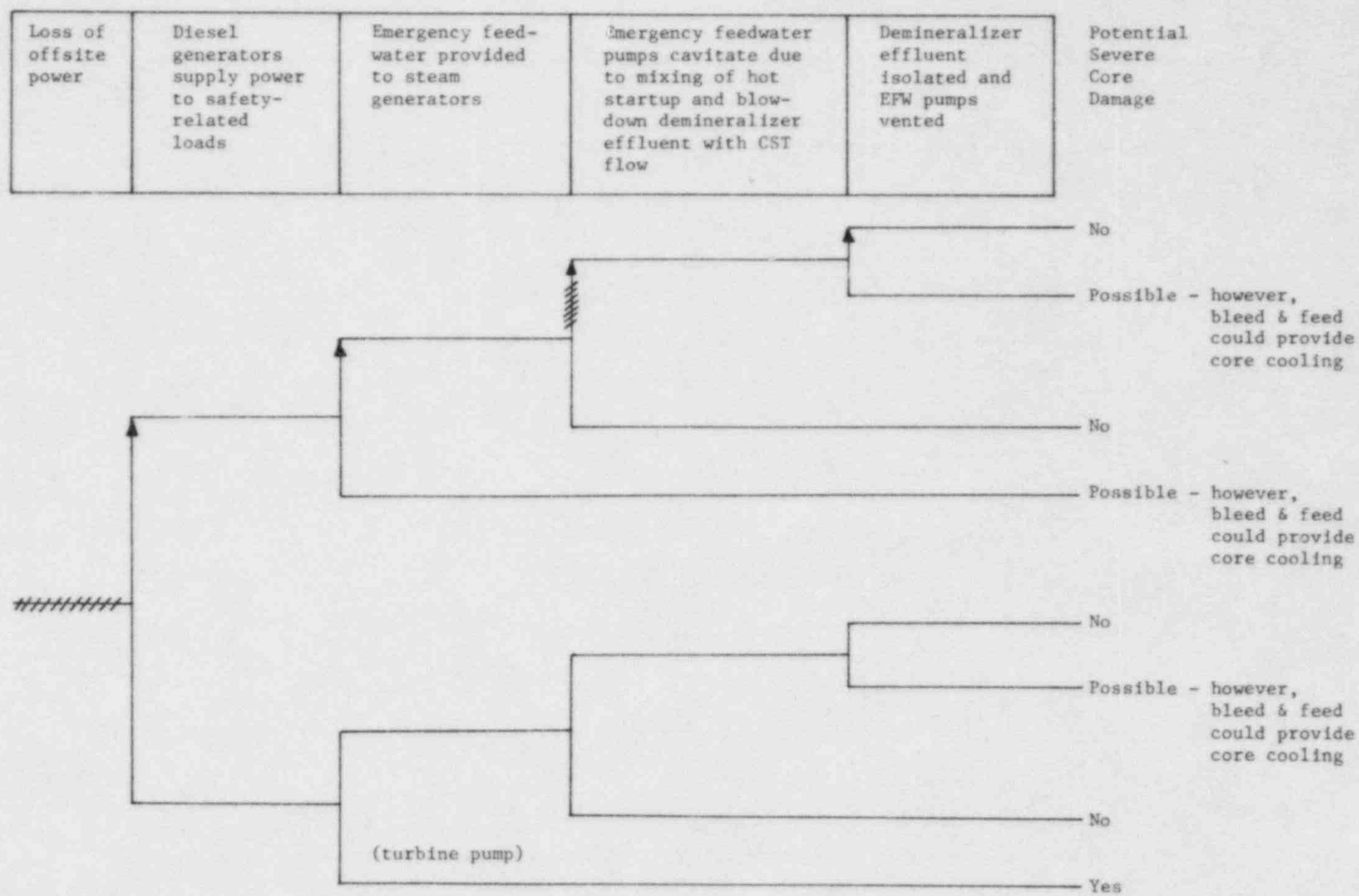
1. With the reactor at full power with atmospheric dump valves isolated due to vibration and failure to close problems, tornado activity resulted in the sequential loss of four of five offsite power lines. Protective relaying disconnected the remaining offsite power line from the bus tie autotransformer. (Offsite power was available through manual connection from the 161 kV transmission system.)
2. Both diesel generators started and powered safety-related loads.
3. Natural circulation was established with both emergency feedwater (EFW) pumps using a common suction from the condensate storage tanks and the startup and blowdown demineralizer effluent.
4. Approximately 15 min after the LOOP, emergency feedwater flow became erratic with flow rate oscillating between 80% and 100% of rated flow due to cavitation. This was caused by flashing of the startup and blowdown demineralizer effluent.
5. The startup and blowdown demineralizer effluent was isolated from the EFW pump suction and both EFW pumps were alternately stopped, vented, and restarted.
6. The process computer was unavailable during the event. This is believed to have been caused by protective trips from low voltage.

Corrective action:

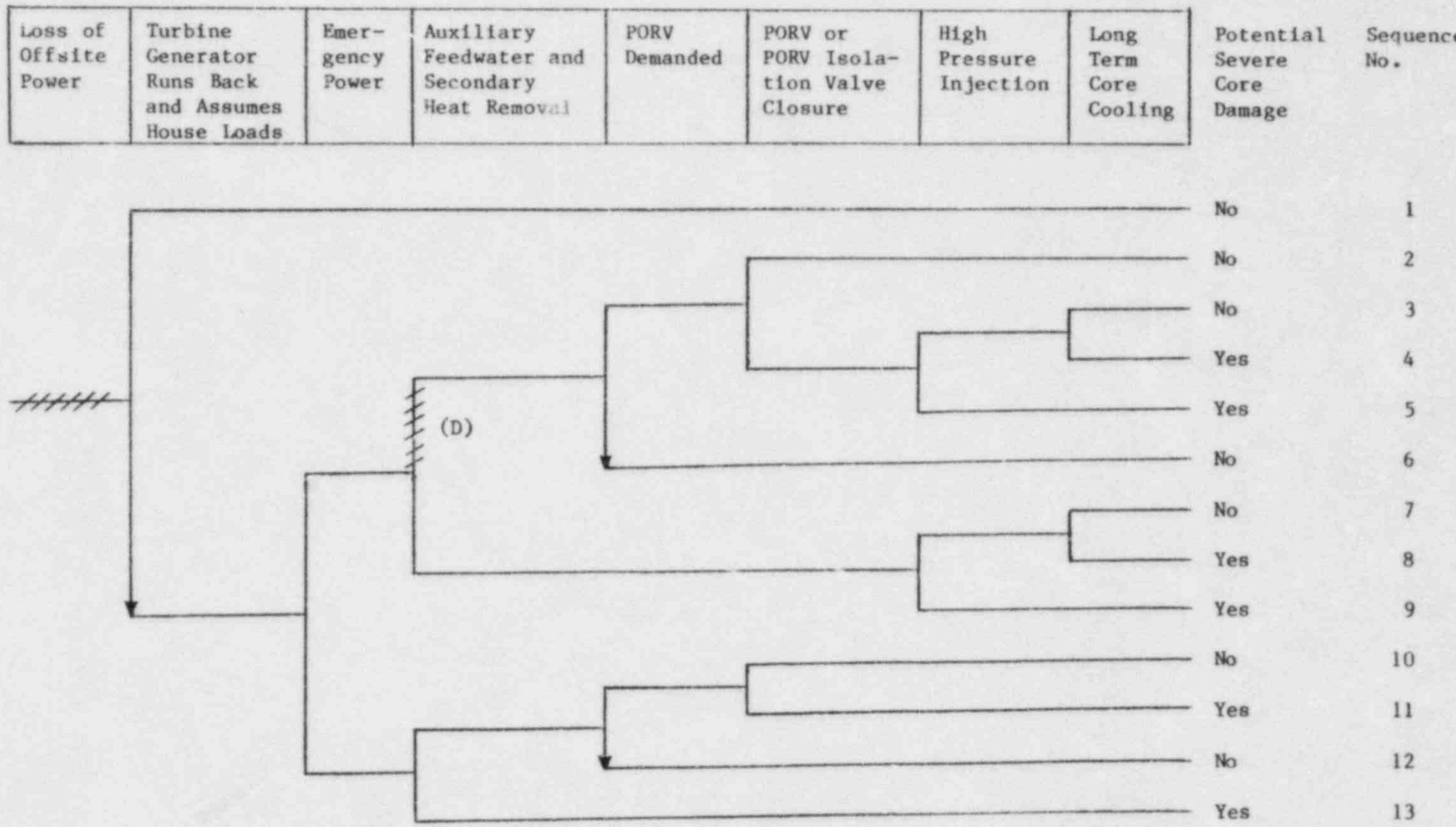
Offsite power was restored via startup transformers ST-2 and ST-3. The EFW system operating procedures were revised to require isolation of the startup and blowdown effluent prior to exceeding 5% power.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads when the unit generator is not available. The EFW system provides water to the steam generators for RCS when the main feedwater system is unavailable.



NSIC 158279 - Actual Occurrence for Degraded Emergency Feedwater System During a Loss of Offsite Power at Arkansas Unit 2



NSIC 158279 - Sequence of Interest for Degraded Emergency Feedwater System During a Loss of Offsite Power at Arkansas Unit 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158279

LER NO.: 80-018

DATE OF LER: July 14, 1980

DATE OF EVENT: April 7, 1980

SYSTEM INVOLVED: Offsite power, emergency feedwater system

COMPONENT INVOLVED: Transmission lines, both EFW pumps

CAUSE: Tornado damage and failure of operating procedures to require isolation of the startup and blowdown effluent thereby allowing flashing to occur in EFW pump suction lines.

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP and FW pump cavitation from hot demineralizer effluent causing flashing.

REACTOR NAME: Arkansas Nuclear Unit 2

DOCKET NUMBER: 50-368

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 912 MWe

REACTOR AGE: 1.3 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power and Light

LOCATION: 6 miles NW of Russelville, Arkansas

DURATION: N/A

PLANT OPERATING CONDITION: 100% rate power

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158650

Date: June 3, 1980

Title: Failure of Service Water System Plus Subsequent Auxiliary Feedwater System Unavailability at Calvert Cliffs 1

The failure sequence was:

1. With the plant at 100% power, the No. 12 service water subsystem was removed from service for heat exchanger saltwater side cleaning. The heat exchanger service water side outlet valve was closed.
2. A complete failure occurred in one of the No. 11 instrument air compressor aftercooler tubes. The air compressor after cooler is cooled by a common portion of the service water system (during safeguards actuation, the common portion of the service water system is isolated from the two redundant safety-related portions).
3. The tube leak allowed compressed air to enter the service water system. The constant vent valves on the operating header were designed to maintain the service water system free of air. However, air accumulated in the idled heat exchanger because the rate of leakage exceeded the capacity of that of the two constant vent valves.
4. When the idled heat exchanger was returned to service, the air bubble was released and, since the service water systems are not independent in the turbine building. Both No. 11 and No. 12 service water pumps lost suction.
5. The operator received low pressure alarms on both service water subsystems and verified that valve lineups were correct.
6. The reactor was manually tripped due to high main turbine bearing temperatures. All reactor coolant pumps, auxiliary feedwater pumps and the main condenser remained operable and were utilized to remove decay heat.
7. During the shutdown, No. 12 condensate storage tank (CST) was used to supply AFW to the steam generators.
8. When low level was indicated in No. 12 CST, flow was switched to No. 11 CST and No. 12 CST was isolated.
9. Later, the main feed system was restored to operation on auxiliary steam and the AFW system was shut down. The control room operator incorrectly designated the CST valves to be realigned.
10. Due to the valve realignment error, No. 11 and 12 CSTs were both isolated, rendering the auxiliary feedwater system unavailable.

Corrective action:

1. Service water was isolated from the affected compressor; the system was vented, and normal flow was restored.
2. The No. 12 CST was properly aligned for service 3 h after the operator error was made.

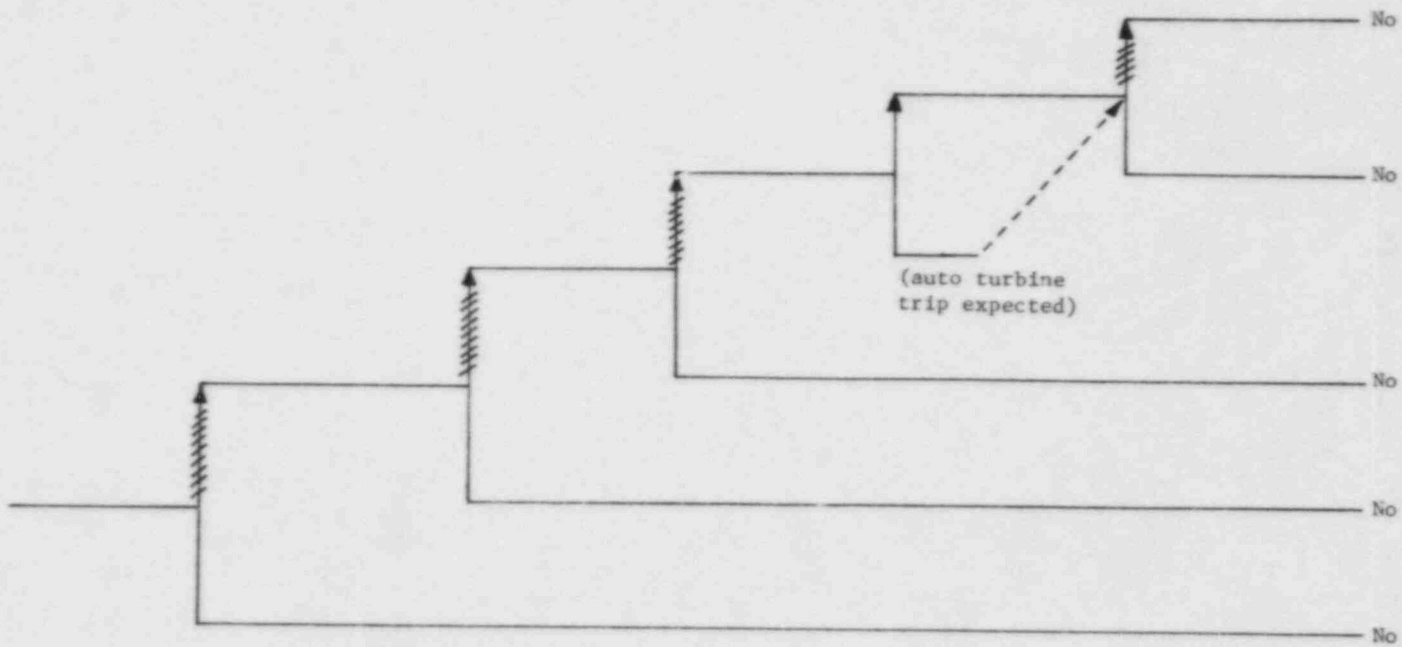
3. A design change was made to increase the capacity of the continuous vents at high places in the system.
4. Operating procedures were modified to eliminate unisolated dead legs during maintenance and to require venting prior to returning subsystems to service.

Design purpose of failed system or component:

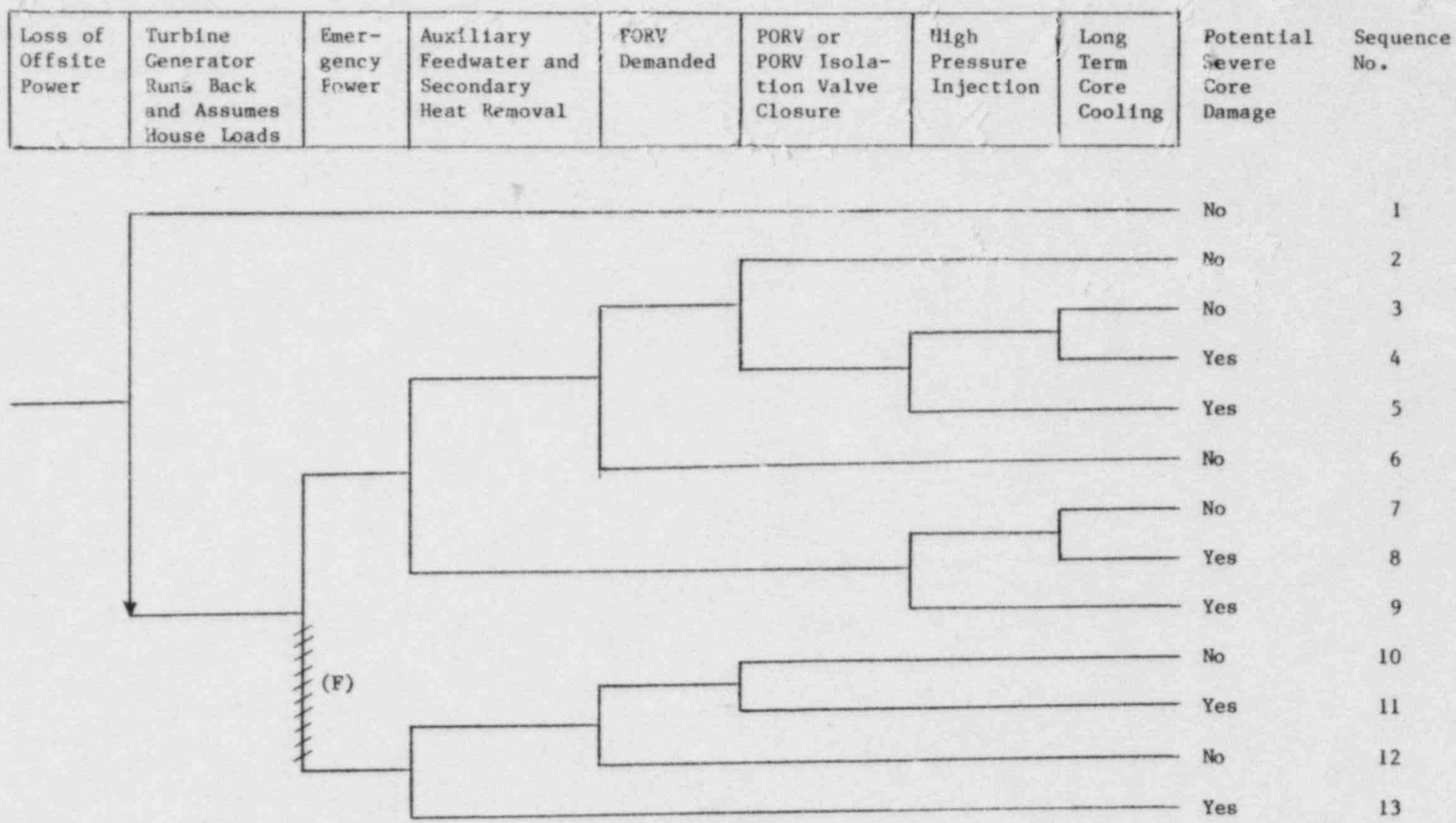
The service water system provides cooling water for turbine-side components as well as the diesel generators, containment coolers, and spent fuel pool coolers.

The auxiliary feedwater system provides water for steam generator cooling in the event the main feedwater system is unavailable.

Unit at full power and service water heat exchanger maintenance in progress	Air compressor aftercooler air leak results in accumulation of air in isolated heat exchanger	Air bubble travels to both service water subtrains when heat exchanger returned to service	Loss of suction pressure to service water pumps and loss of service water cooling	Manual reactor trip due to high main turbine bearing temperatures. Auxiliary feedwater system provides SG cooling	Condensate storage tank valves incorrectly realigned when AFW flow is terminated and MFW system started for SG cooling	Potential Severe Core Damage



NSIC 158650 - Actual Occurrence for Failure of Service Water System and Subsequent Auxiliary Feedwater Unavailability at Calvert Cliffs 1



NSIC 158650 - Sequence of Interest for Failure of Service Water System and Subsequent Auxiliary Feedwater Unavailability at Calvert Cliffs 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158650

LER NO.: 80-027

DATE OF LER: June 3, 1980

DATE OF EVENT: May 20, 1980

SYSTEM INVOLVED: Service water system, auxiliary feedwater system

COMPONENT INVOLVED: Air compressor aftercooler, condensate storage tank

CAUSE: Air leakage to service water system due to aftercooler tube failure, operator error in realigning valves when switching from one CST to another

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Loss of service water and subsequent unavailability of auxiliary feedwater

REACTOR NAME: Calvert Cliffs Unit 1

DOCKET NUMBER: 50-317

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 845 MWe

REACTOR AGE: 5.6 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Baltimore Gas Electric

LOCATION: 40 miles south of Annapolis, Maryland

DURATION: 3 h

PLANT OPERATING CONDITION: 100% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event and operator inspection

COMMENT: See also NSIC 157124 (for Arkansas Nuclear One, 50-368, April 24, 1979).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 158860

Date: April 30, 1980

Title: Loss of Two Essential Buses and Loss of Decay Heat Removal Capability at Davis-Besse 1

The failure sequence was:

1. The reactor was in cold shutdown in preparation for refueling with the following equipment/system status:
   a. the head was detensioned but not removed (water level below the vessel flange)
   b. decay heat was being removed using decay heat pump No. 2
   c. decay heat pump No. 1 was out of service for maintenance with its associated piping drained
   d. the manway covers on the top of the steam generators had been removed.
2. The unit electrical lineup had been revised in preparation for work on buses "A" and "C". Buses E2 and F2 were supplied from breaker HBBF2. Essential distribution panels Y1 and Y3 were on their alternate feed (YBR) which is supplied by F2.
3. The ground fault relay on breaker HBBF2 actuated (possibly due to vibration caused by construction personnel in the switchgear room) and tripped the breaker.
4. This deenergized essential distribution panels Y1 and Y3, which resulted in full SFAS actuation in levels 1 through 5.
5. The SFAS actuation isolated the RCS letdown line and caused the suction of decay heat removal pump No. 2 to transfer to the emergency sump. During the time the BWST outlet valves and emergency sump outlet valves were stroking, water gravity flowed into the emergency sump (approximately 1500 gallons). The decay heat pump was injecting BWST water into the RCS and increased RCS inventory approximately 3500 gallons. (The high pressure injection pumps and containment spray pump breakers had been racked out as required and hence did not actuate).
6. The closing of the BWST outlet valve caused the decay heat pump to draw suction from the emergency sump which resulted in air being drawn into the pump suction. The pump was shut down to stop the injection and to prevent pump damage due to loss of suction.
7. The emergency sump valves were closed and power was removed from their operators. Decay heat removal loop No. 2 was refilled from the BWST, vented, and returned to service. The electrical lineup was restored with buses E2 and F2 separated.
8. Decay heat removal was unavailable for approximately 2-1/2 hours. During that time interval, reactor coolant temperature increased from 90°F to 170°F.

Corrective action:

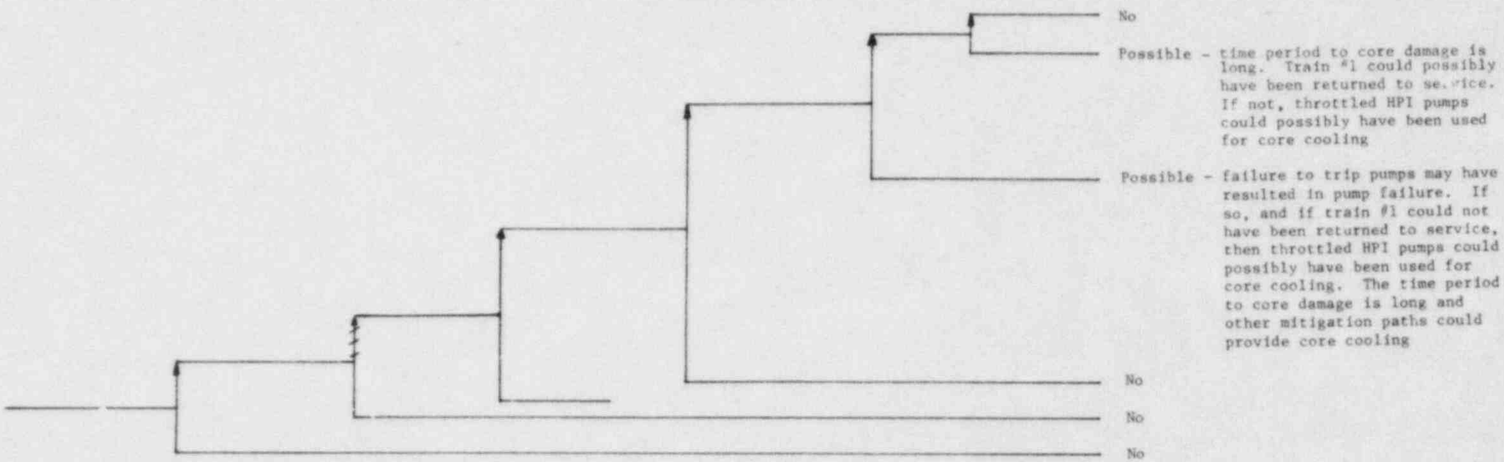
Plant procedures have been revised to ensure power is removed from the emergency sump valves during mode 5 and mode 6 operation.

The instrument ac system procedure was revised to allow inverters to be supplied from the dc bus when the normal feed for the regulated rectifiers from motor control centers E12A or F12A are to be deenergized.

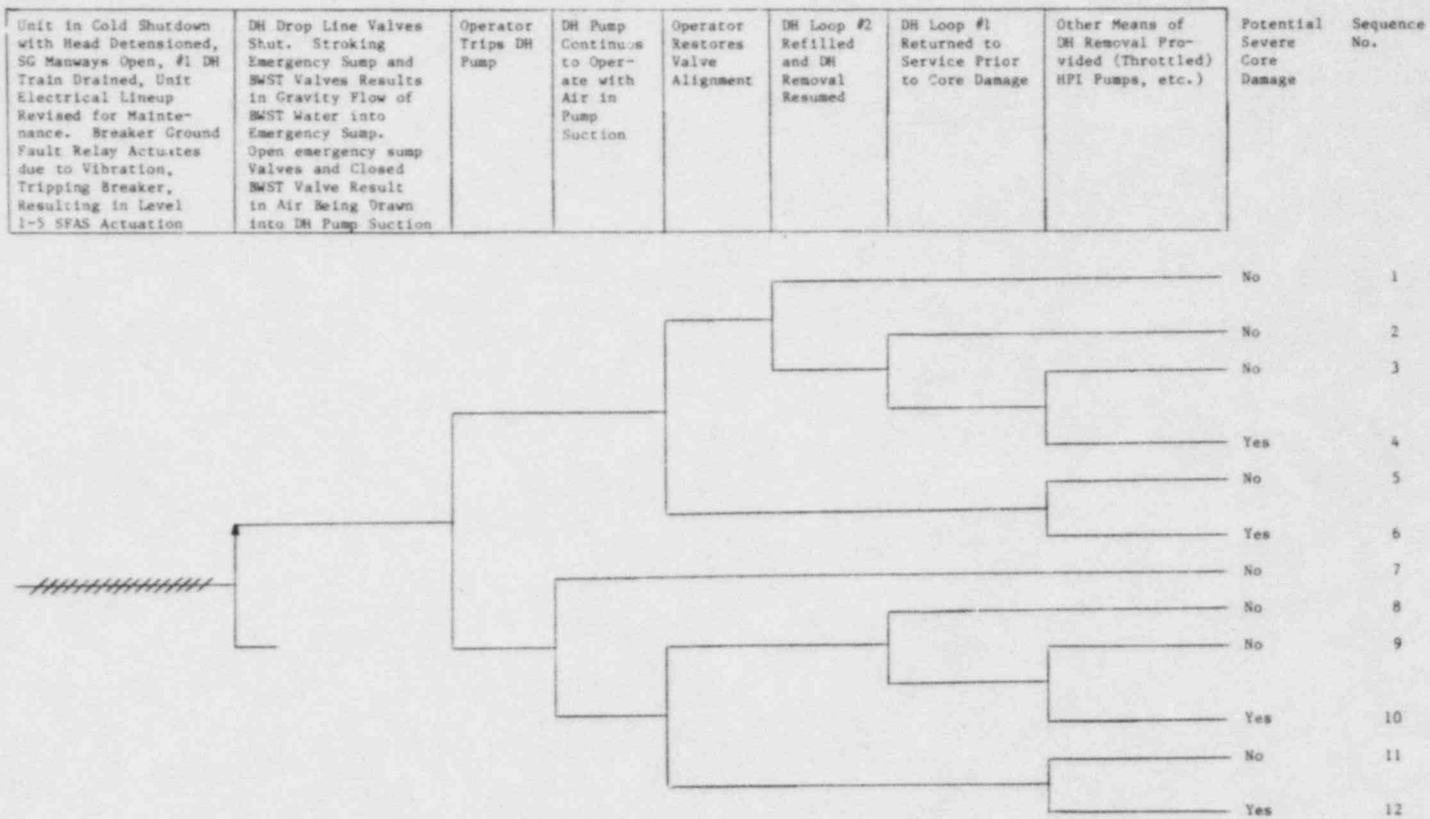
Design purpose of failed system or component:

The uninterruptable buses provide a continuous source of power to control and instrumentation circuitry which cannot tolerate short term power interruptions.

Reactor in cold shutdown with head detensioned, steam generator manway covers removed, and DH & pump #1 out of service for maintenance with its loop drained (DH pump #2 in operation)	Unit electrical lineup revised in preparation for maintenance. Buses E2 and F2 supplied by breaker HBBF2. Essential distribution panels Y1 and Y3 on alternate feed from F2	Ground fault relay on breaker HBBF2 actuates due to vibration, tripping breaker	Deenergized distribution panels Y1 and Y3 cause level 1-15 SFAS actuation, which transfers suction from RCS to emergency sump	Emergency sump valves not locked out. Stroking emergency sump and BWST valves results in gravity flow of BSWT water into emergency sump. Open emergency sump valves and closed BWST valve result in air being drawn into pump suction	Operator trips decay heat pump	Emergency sump valves closed and power removed from operation. Decay heat loop #2 refilled from BWST, vented, and returned to service	Potential Severe Core Damage



NSIC 158860 - Actual Occurrence for Loss of Two Essential Busses and Loss of Decay Heat Removal Capability at Davis-Besse 1



NSIC 158860 - Sequence of Interest for Loss of Two Essential Buses and Loss of Decay Heat Removal Capability at Davis-Besse 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 158860

LER NO.: 80-029

DATE OF LER: April 30, 1980

DATE OF EVENT: April 19, 1980

SYSTEM INVOLVED: Decay heat removal system, essential power system

COMPONENT INVOLVED: Essential bus, decay heat pump

CAUSE: Tripped breaker resulted in loss of two essential buses and loss of decay heat removal

SEQUENCE OF INTEREST: Loss of two essential buses while in cold shutdown

ACTUAL OCCURRENCE: Loss of two essential buses and loss of decay heat removal capability

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 2.7 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison

LOCATION: 21 miles east of Toledo, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: Cold shutdown

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 159134

Date: July 7, 1980

Title: Loss of Offsite Power at Arkansas Nuclear 1

The failure sequence was:

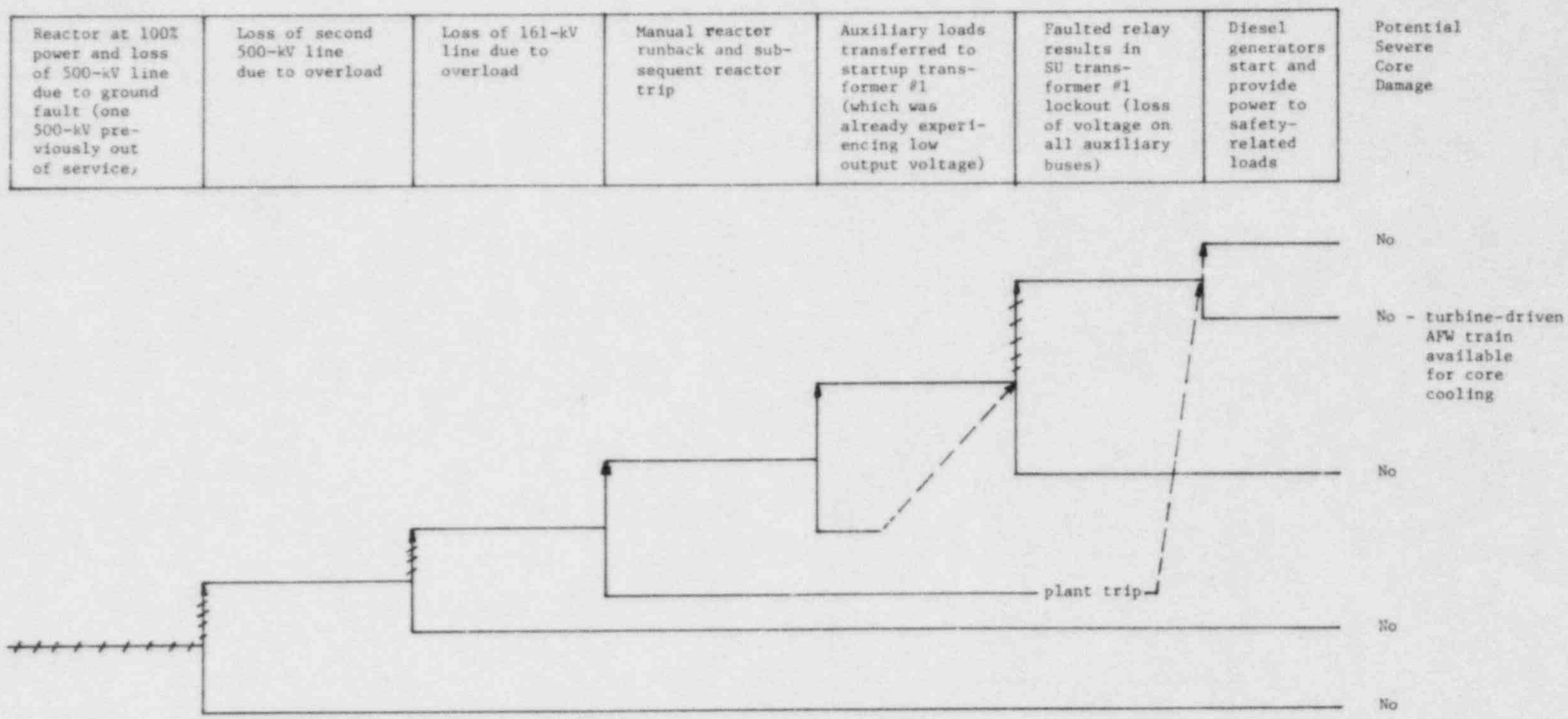
1. With the unit at 100% power, switchyard circuit breakers tripped apparently due to a ground fault, isolating the Mabelvale 500 kV line. Unit 2 and Unit 1 generation was transferred to the Fort Smith 500 kV line. (The Mayflower line was out of service.)
2. The Fort Smith 500 kV line tripped open at the Fort Smith and due to feeder overload. (The Mabelvale line became available but failed to close onto the ring bus because of a lack of synchonization.)
3. The Morrilton East 161 kV line tripped on overload, leaving only Unit 1 generator output feeding the auto transformer and the Russellville East line carrying Unit 1 power generation. Startup transformer No. 1 remained energized from the autotransformer and startup transformer No. 2 remained tied to the 161 kV bus.
4. A manual runback was initiated following an automatic generator runback. The reactor subsequently tripped.
5. Auxiliary loads were transferred to startup transformer No. 1, which was already experiencing low output voltage.
6. The autotransformer bank locked out due to a faulted relay, locking out startup transformer No. 1 and Unit 2 startup transformer No. 3.
7. This resulted in a loss of all auxiliary bus voltage.
8. The unit diesel generators started and provided power to safety-related loads. (Power remained available by manual transfer from startup transformer No. 2.)

Corrective action:

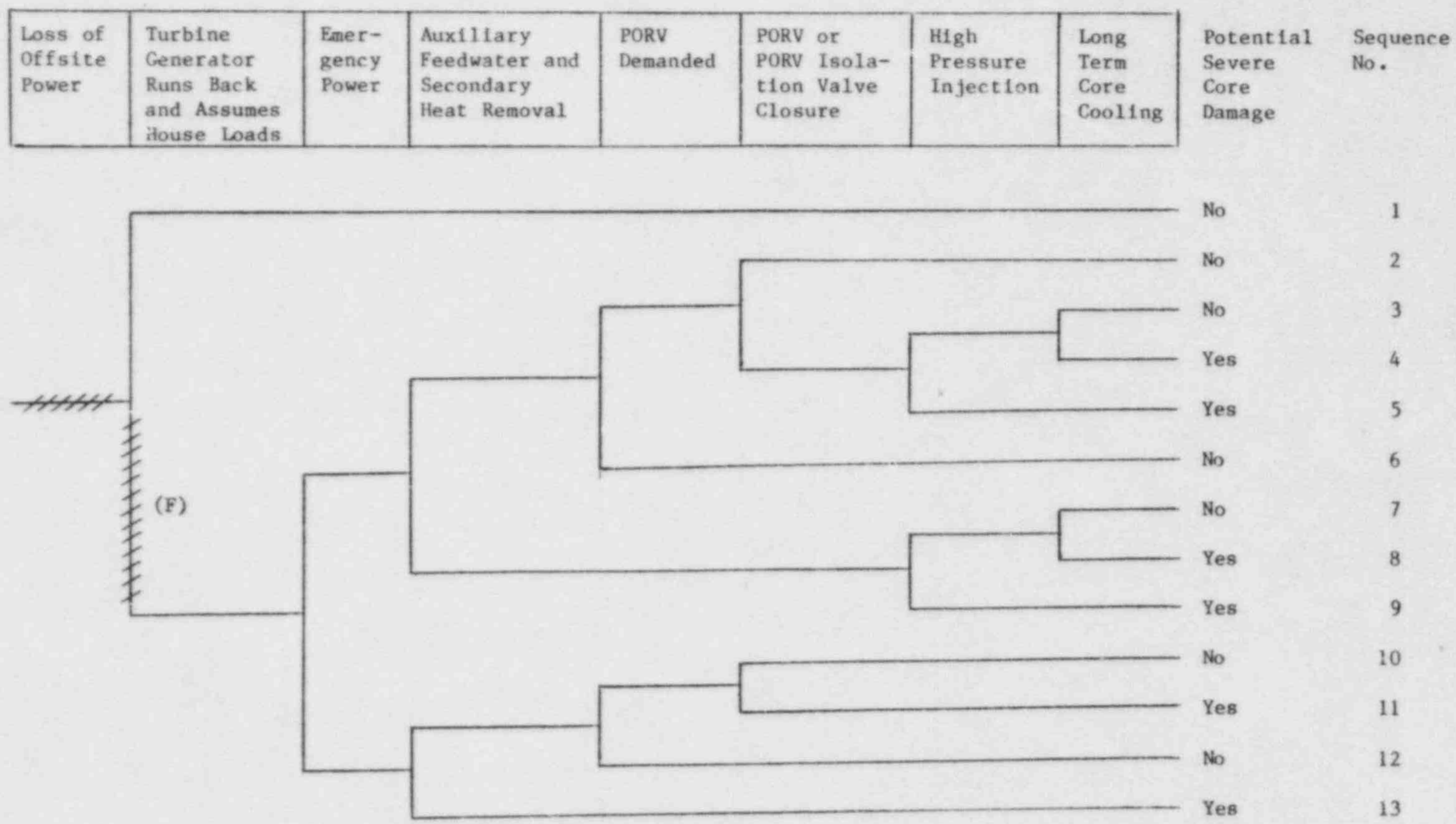
Approximately 65 minutes later house loads were transferred to startup transformer No. 1.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety related loads when the unit generator is unavailable.



NSIC 159134 - Actual Occurrence for Loss of Offsite Power at Arkansas Nuclear 1



NSIC 159134 - Sequence of Interest for Loss of Offsite Power at Arkansas Nuclear - Unit 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 159134

LER NO.: 80-022

DATE OF LER: July 7, 1980

DATE OF EVENT: June 24, 1980

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Transmission lines, relay

CAUSE: Sequential loss of transmission lines due to ground fault and overload, transformer lockout due to unspecified relay fault.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Arkansas Nuclear 1

DOCKET NUMBER: 50-313

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 850 MWe

REACTOR AGE: 5.9 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power and Light

LOCATION: 6 miles NW of Russellvile, Arkansas

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT: See NSIC 159136 (Arkansas Nuclear 2, 50-368, LER 80-042, July 7, 1980).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 159136

Date: July 7, 1980

Title: Loss of Offsite Power at Arkansas Nuclear 2

The failure sequence was:

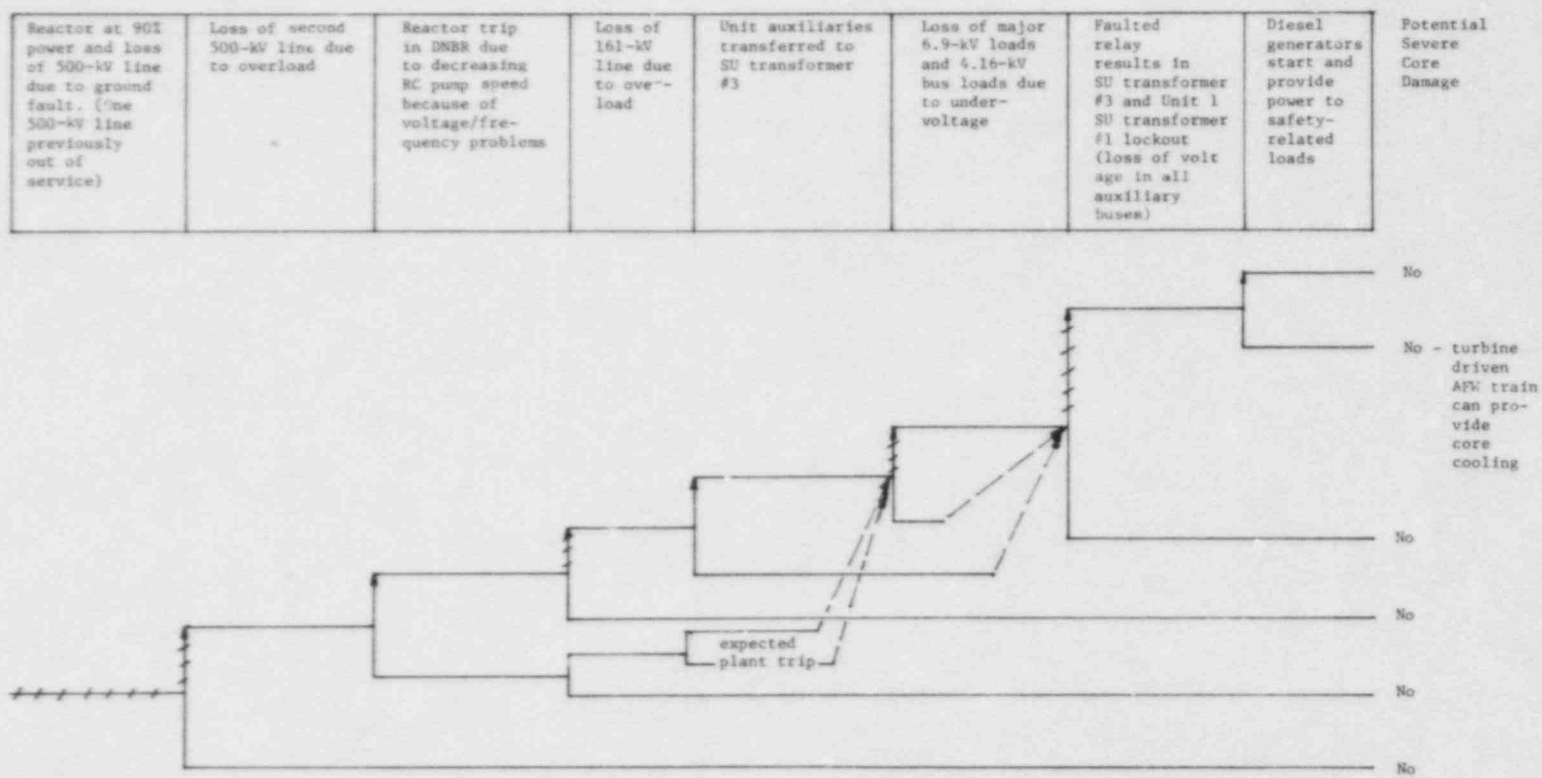
1. With the unit at 91% power, switchyard circuit breakers tripped apparently due to a ground fault, isolating the Mabelvale 500 kV line. Unit 2 and Unit 1 generation transferred to the Fort Smith 500 kV line. (The Mayflower line was out of service.)
2. The Fort Smith 500 kV line tripped open at the Fort Smith end due to feeder overload. (The Mabelvale line became available but failed to close onto the ring bus because of a lack of synchonization.)
3. Unit 2 tripped on DNBR due to decreasing RCP pump speed, a result of frequency and voltage upsets on the grid.
4. The Morrilton East 161 kV line tripped on overload, leaving only Unit 1 generator output feeding the autotransformer and the Russellville East line carrying Unit 1 power generation. Startup transformer No. 3 remained energized from the autotransformer, and startup transformer No. 2 remain tied to the 161 kV bus.
5. The unit auxiliaries were transferred to startup transformer No. 3. (The 6.9 kV buses had been transferred to startup transformer No. 3 prior to the start of the event.)
6. Undervoltage relays on the 6.9 kV buses operated and shed the RC pumps and circulating water pumps.
7. The 4.16 kV auxiliary circuit indicated low voltage and stripped the 4.16 kV buses from the startup transformer.
8. Subsequently the auto transformer bank locked out due to a faulted relay, locking out startup transformer No. 3 and the Unit 1 startup transformer No. 1.
9. This resulted in a loss of all auxiliary bus voltage.
10. The unit diesel generators started and provided power to safety-related loads. (Power remained available by manual transfer from startup transformer No. 2.)

Corrective action:

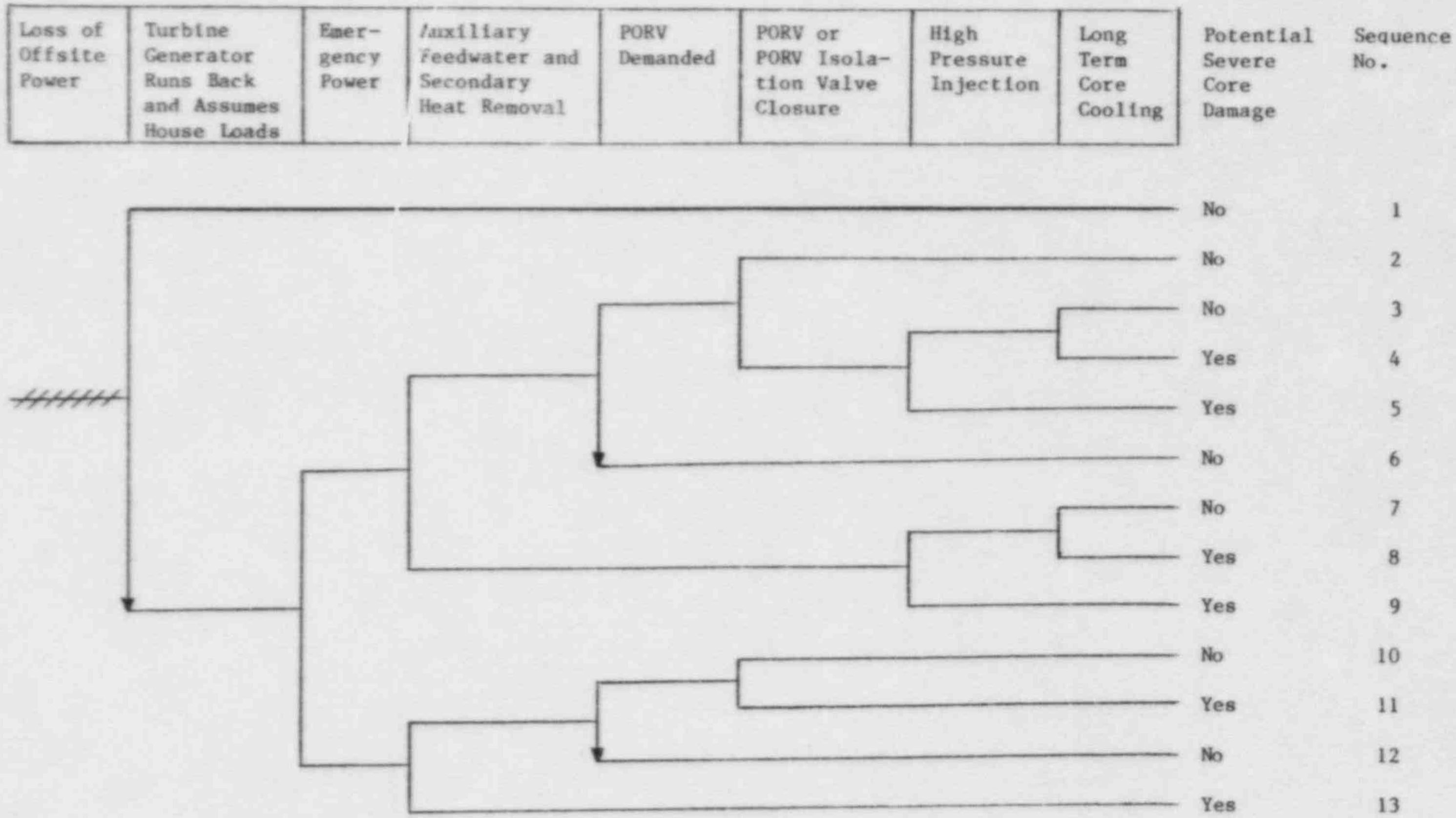
Buses 2 A1 and 2 H1 were manually energized from startup transformer No. 2 four minutes later. Approximately 55 minutes later all house loads were transferred to startup transformer No. 3 and both emergency diesels were secured.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety related loads when the unit generator is unavailable.



NSIC 159136 - Actual Occurrence for Loss of Offsite Power at Arkansas Nuclear 2



NSIC 159136 - Sequence of Interest for Loss of Offsite Power at Arkansas Nuclear 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 159136

LER NO.: 80-042

DATE OF LER: July 7, 1980

DATE OF EVENT: June 24, 1980

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Transmission lines, relay

CAUSE: Sequential loss of transmission lines due to ground fault and overload, transformer lockout due to unspecified relay fault

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Arkansas Nuclear 2

DOCKET NUMBER: 50-386

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 912 MWe

REACTOR AGE: 1.6 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power and Light

LOCATION: 6 miles NW of Russellville, Arkansas

DURATION: N/A

PLANT OPERATING CONDITION: 91% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT: See NSIC 159134 (Arkansas Nuclear 1, 50-313, LER-80-022, July 7, 1980).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 159347

Date: September 2, 1980

Title: Unavailability of All High Steam Flow Signals at Surry 2

The failure sequence was:

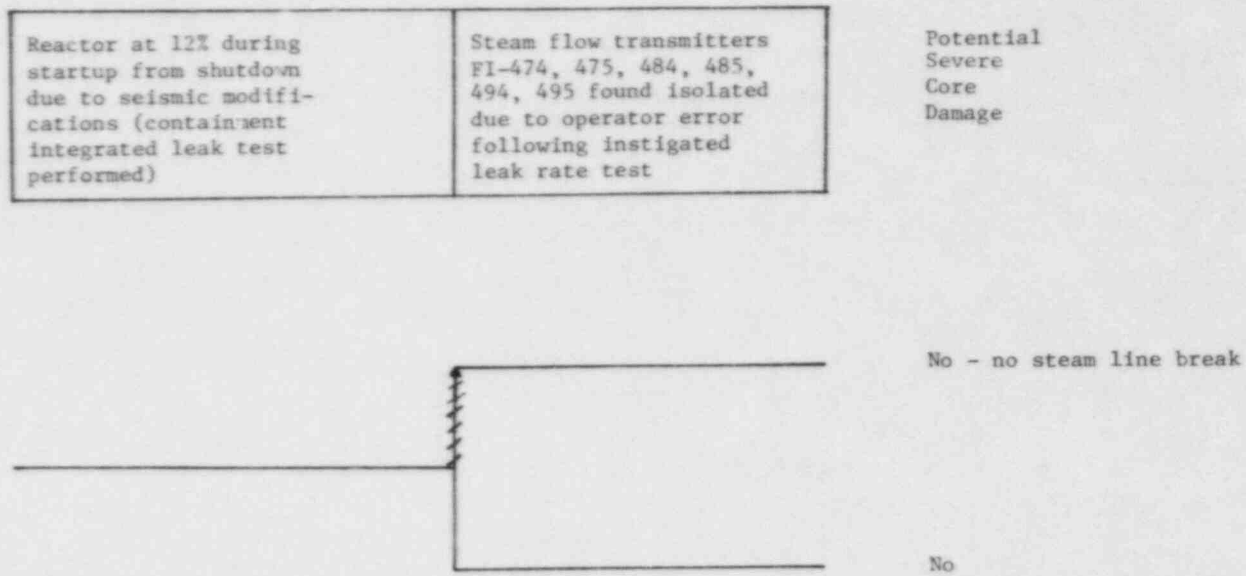
1. With the reactor at 12% power, steam flow transmitters FI-474, 475, 484, 485, 494, and 495 were discovered isolated. These transmitters had been isolated during the containment integrated leak rate test.
2. Associated bistables were immediately tripped.

Corrective action:

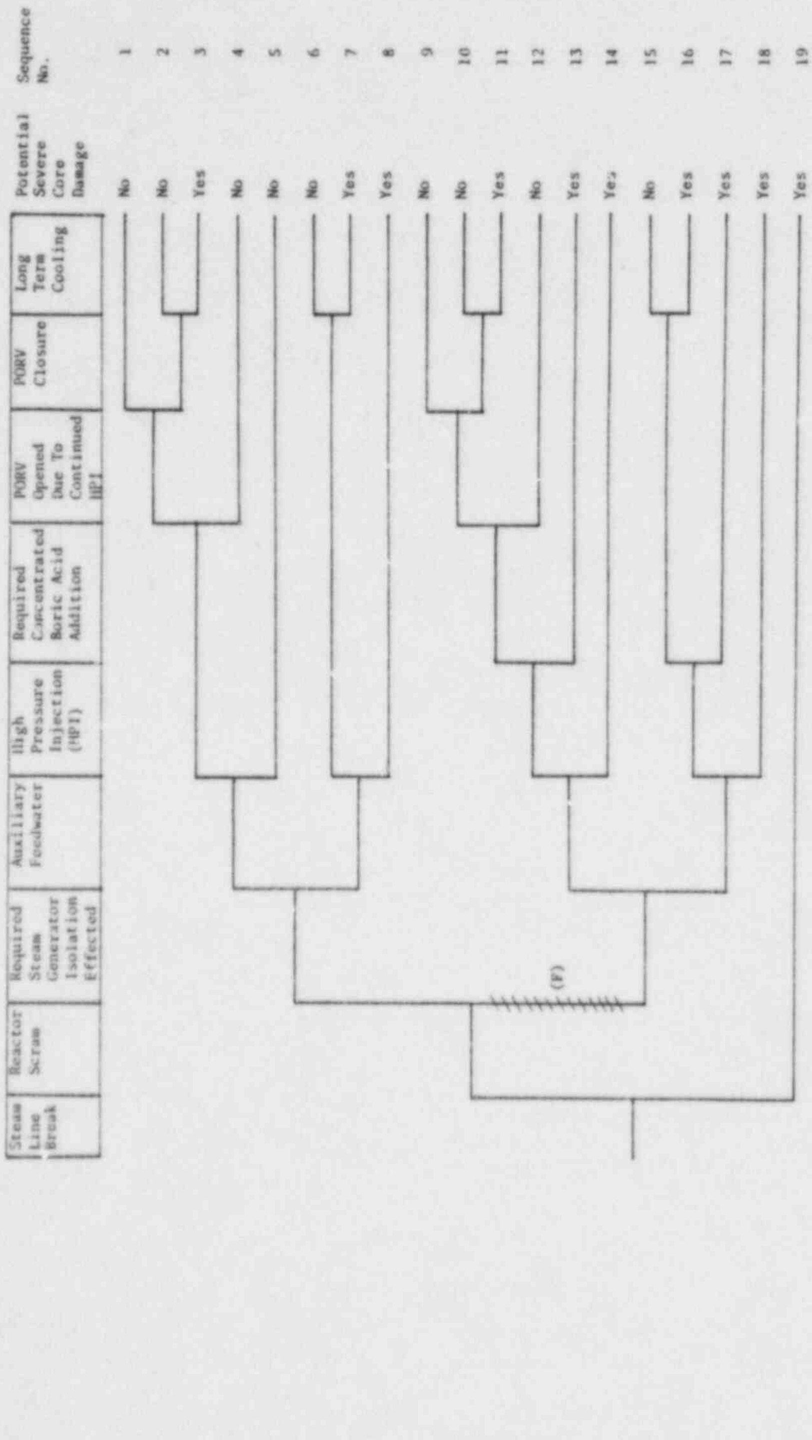
1. The transmitters were subsequently unisolated.

Design purpose of failed system or component:

The steam flow transmitters provide signals used in coincidence with either low RCS temperature or low main steam line pressure to initiate safety injection (low pressurizer pressure or level and main steam line/main steam header differential pressure also initiates safety injection). The high steam flow transmitters in coincidence with low RCS temperature or low main steam line pressure also initiates trip of the MSIVs (a containment high pressure also initiates trip of these valves).



NSIC 159347 - Actual Occurrence for Unavailability of all High Steam Flow Signals at Surry 2



NSIC 159347 - Sequence of Interest for Unavailability of All High Steam Flow Signals at Surry 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 159347

LER NO.: 80-011

DATE OF LER: September 2, 1980

DATE OF EVENT: August 19, 1980

SYSTEM INVOLVED: High steam flow instrumentation for safety injection initiation

COMPONENT INVOLVED: Steam flow transmitters

CAUSE: Human error - transmitters inadvertently left isolated

SEQUENCE OF INTEREST: Steam line break

ACTUAL OCCURRENCE: Transmitters discovered isolated during startup

REACTOR NAME: Surry 2

DOCKET NUMBER: 50-281

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 822 MWe (net)

REACTOR AGE: 7.5 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Virginia Electric & Power

LOCATION: 17 miles NW of Newport News, Virginia

DURATION: 8 h (estimated)

PLANT OPERATING CONDITION: 12% power, startup

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operator observation

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 160453

Date: September 23, 1980

Title: Loss of Emergency Power System at Davis-Besse 1

The failure sequence was:

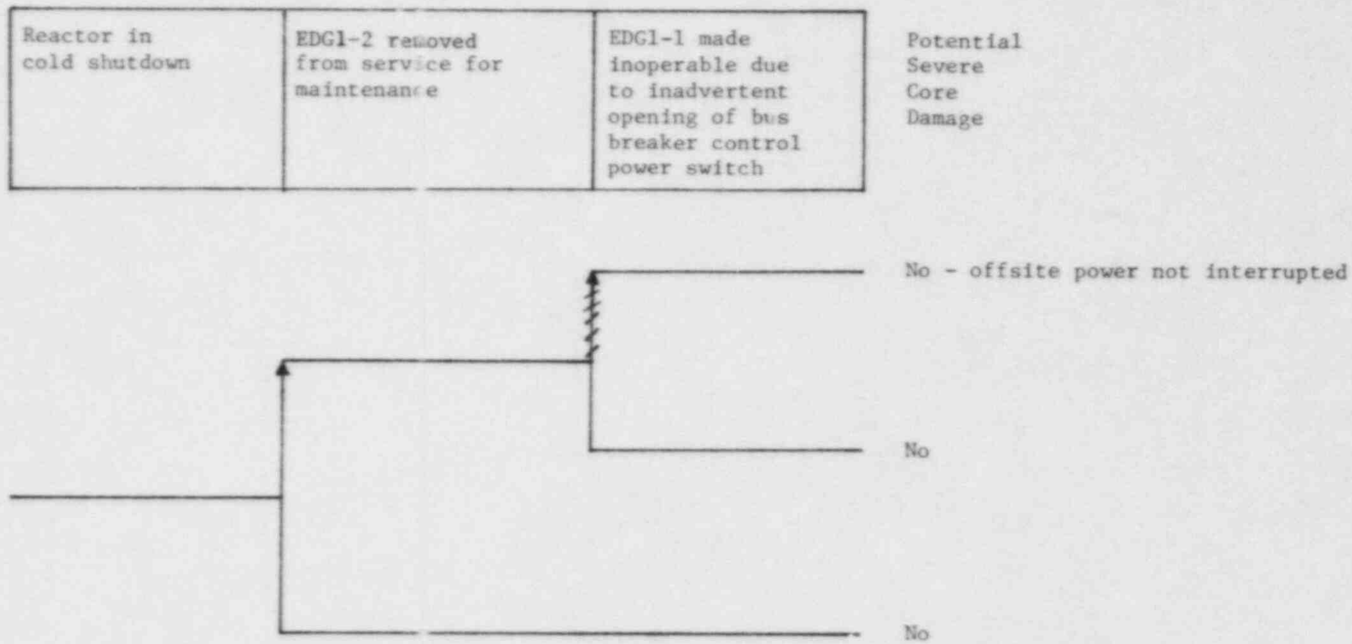
1. With the reactor in cold shutdown, EDG 1-2 was removed from service for maintenance.
2. A service representative opened a knife switch in the switchgear room thereby removing control power from all the 4160 volt bus C1 breakers. This would have prevented EDG1-1 from automatically starting if it had been needed. During the 8 min that control power was off the bus, the station did not have an operable EDG.

Corrective action:

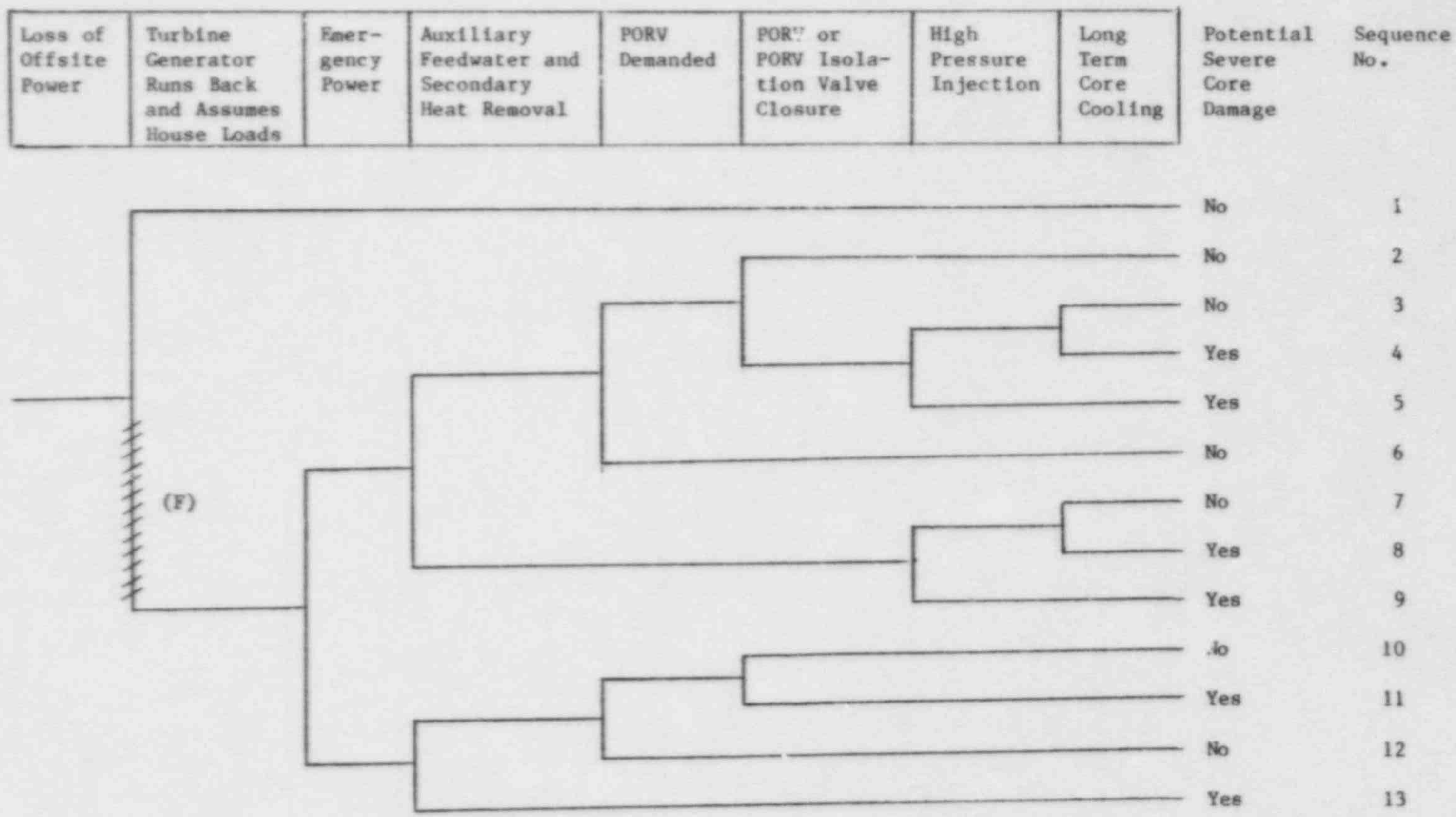
Control room operators noticed the breaker indicating lights had gone out. The shift supervisor went to the switchgear room, determined the cause of the problem, and reclosed the switch.

Design purpose of failed system or component:

The emergency power system provides ac power when offsite sources are unavailable.



NSIC 160453 - Actual Occurrence for Loss of Emergency Power at Davis-Besse 1



NSIC 160453 - Sequence of Interest for Loss of Emergency Power at Davis-Besse 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 160453

LER NO.: 80-065

DATE OF LER: September 23, 1980

DATE OF EVENT: August 26, 1980

SYSTEM INVOLVED: Emergency power system

COMPONENT INVOLVED: Essential bus breakers

CAUSE: Inadvertent opening of control power knife switch

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Diesel generators disabled

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 3.0 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison Co.

LOCATION: 21 miles east of Toledo, Ohio

DURATION: 8 min

PLANT OPERATING CONDITION: Mode 5, cold shutdown

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Control room operator observation

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 160497

Date: October 20, 1980

Title: Inadvertent Opening of Safety Relief Valve at Pilgrim 1

The failure sequence was:

1. Operating pressure in the nitrogen supply line to the solenoid valve that operates the safety relief valve was in excess of the design specifications for the solenoid valve, resulting in leakage through the solenoid valve seat into the diaphragm of the safety relief valve.
2. The reactor was operating at steady state at 95% power. At the associated high reactor pressure, the leakage through the valve seat was enough to lift the relief valve.
3. The reactor was manually scrammed.
4. Attempts were made to close the relief valve, per operating procedures, which entailed switching the valve to the open position (which opens the nitrogen supply valve) and then to the closed position.
5. The high nitrogen pressure (160 psi) delivered when the nitrogen supply valve opened, was above the set point (135 psi) at which the solenoid valve could respond to a "close" signal when the reactor is pressurized.
6. The relief valve remained open until the reactor depressurized.
7. The unit was successfully taken to cold shutdown.

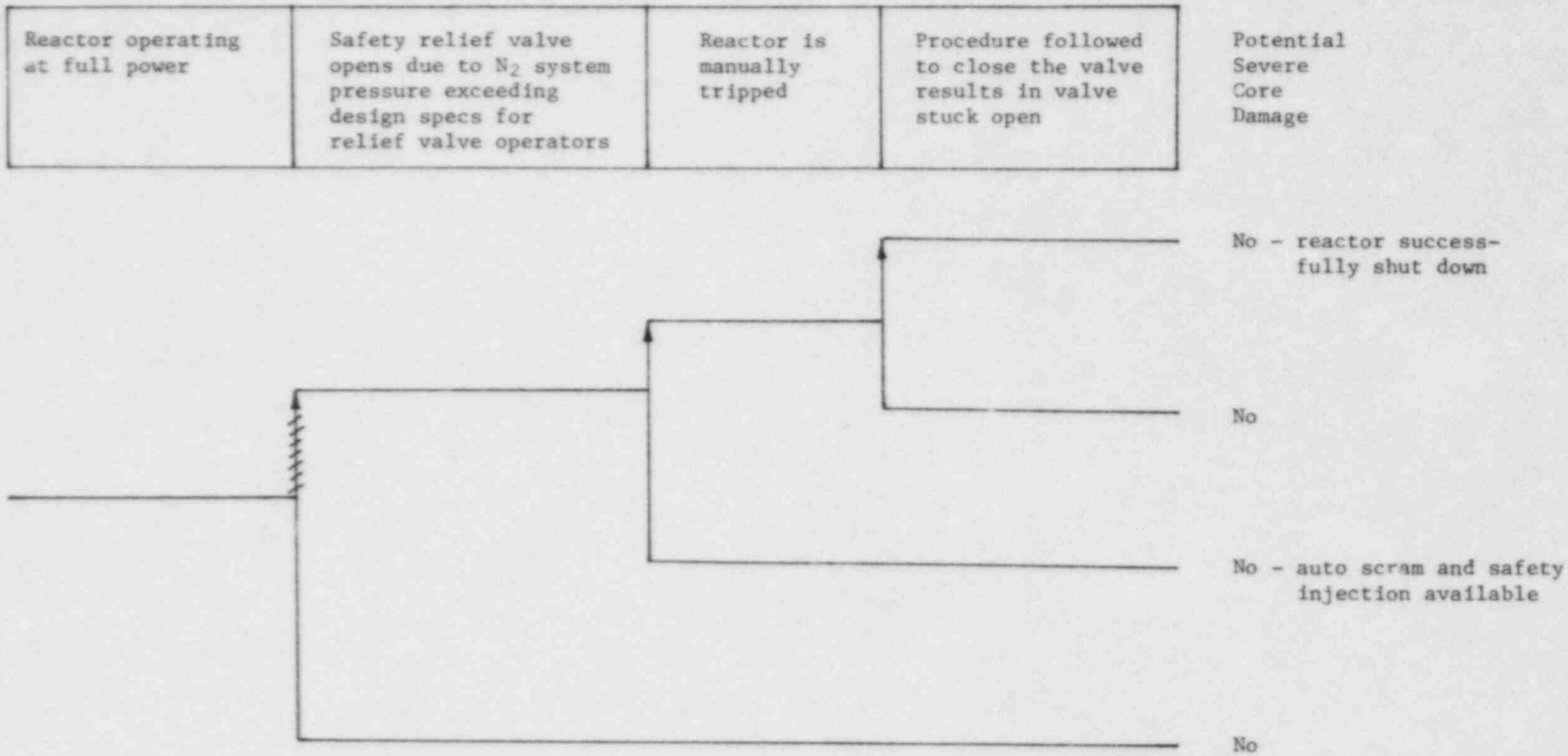
Corrective action:

1. Operation at reduced nitrogen supply pressure (110 psi) was to be implemented.
2. Station procedures were to be revised to require an operator verification check of nitrogen pressure once per shift.

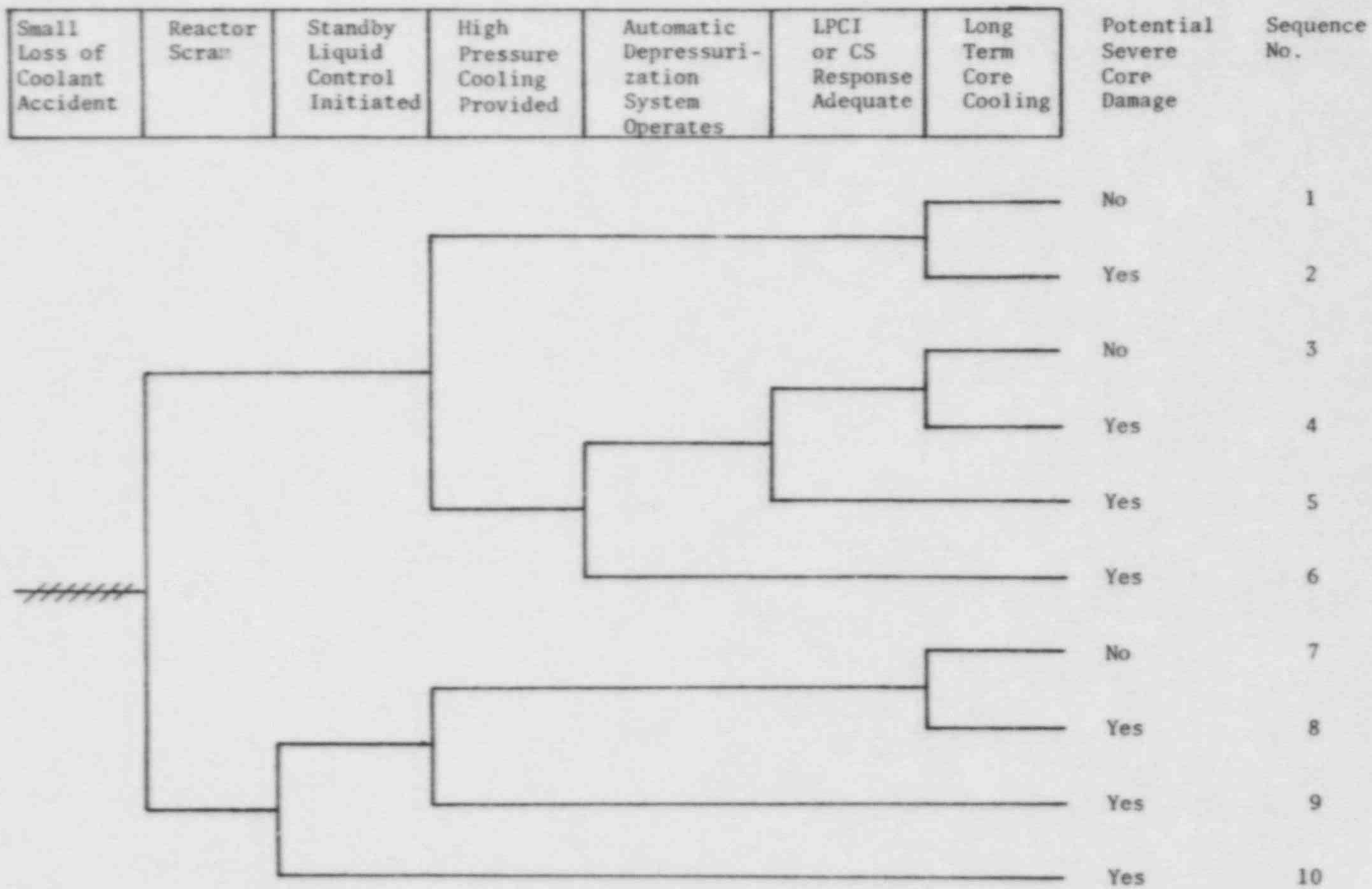
Design purpose of failed system or component:

The nitrogen system provides control gas to actuate the opening and closing of reactor relief valves upon automatic control demands as well as manual demands.

The reactor relief system provides overpressure protection for the RCS.



NSIC 160497 - Actual Occurrence for Inadvertent Opening of Safety Relief Valve at Pilgrim 1



NSIC 160497 - Sequence of Interest for Inadvertent Opening of Relief Valve at Pilgrim 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 160497

LER NO.: 80-069

DATE OF LER: October 20, 1980

DATE OF EVENT: October 7, 1980

SYSTEM INVOLVED: Pressure relief system

COMPONENT INVOLVED: Relief valve

CAUSE: Excessive nitrogen pressure to valve operator

SEQUENCE OF INTEREST: Main steam line break

ACTUAL OCCURRENCE: Stuck open relief valve

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 8.3 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison Co.

LOCATION: 4 miles SE of Plymouth, Massachusetts

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Design basis event

DISCOVERY METHOD: Operational event

COMMENT: All safety systems were available. See Accession No. 160559 for a similar event at Pilgrim 1 (50-293, LER 80-080, Nov. 14, 1980) but with a different cause.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 160532

Date: October 22, 1980

Title: Component Cooling Water Inoperable at Pilgrim 1

The failure sequence was:

1. At full power (100%) the "B" loop of the reactor building closed cooling water system (RBCCWS) was out of service for maintenance.
2. For unspecified reasons a 480 V breaker tripped open twice within 10 min, disabling the "A" loop of the RBCCWS.
3. This resulted in a complete loss of the RBCCWS while the "A" loop was disabled. The "A" loop was immediately restored.

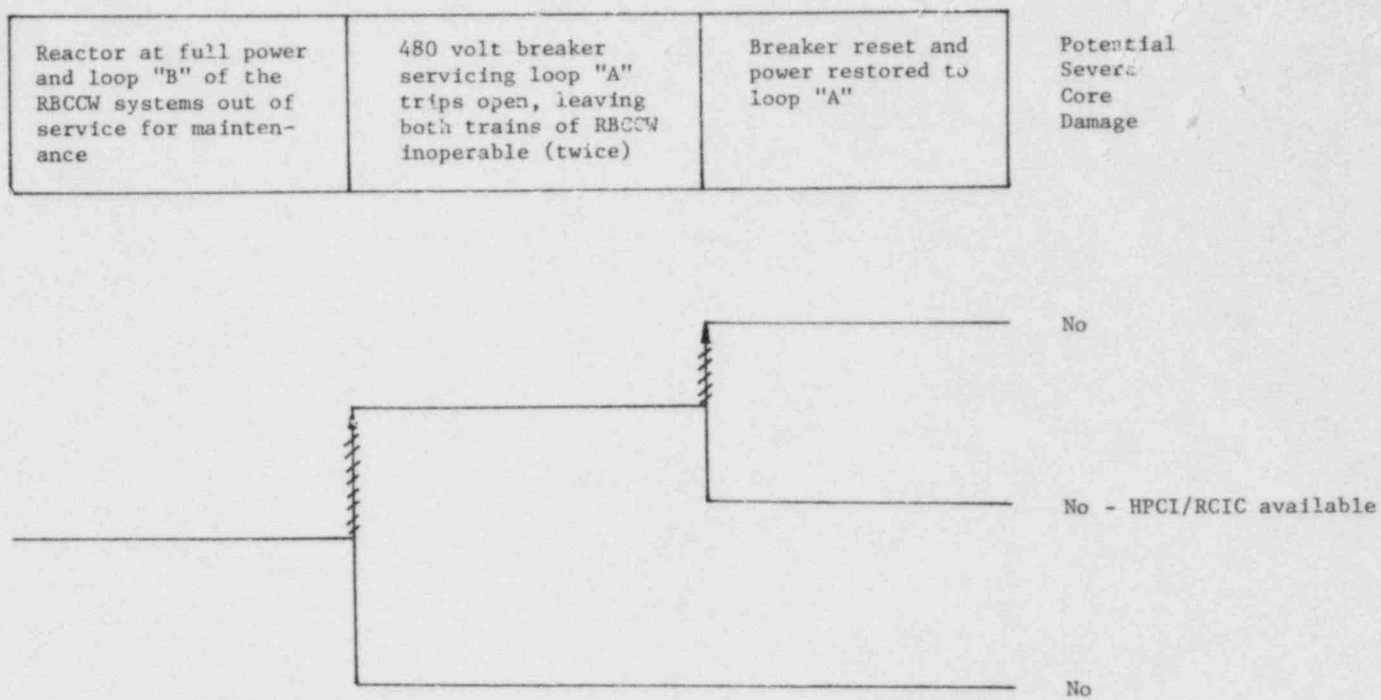
Corrective action:

1. The 480 V trip setting was temporarily raised.
2. A detailed review of 480 V bus was requested.

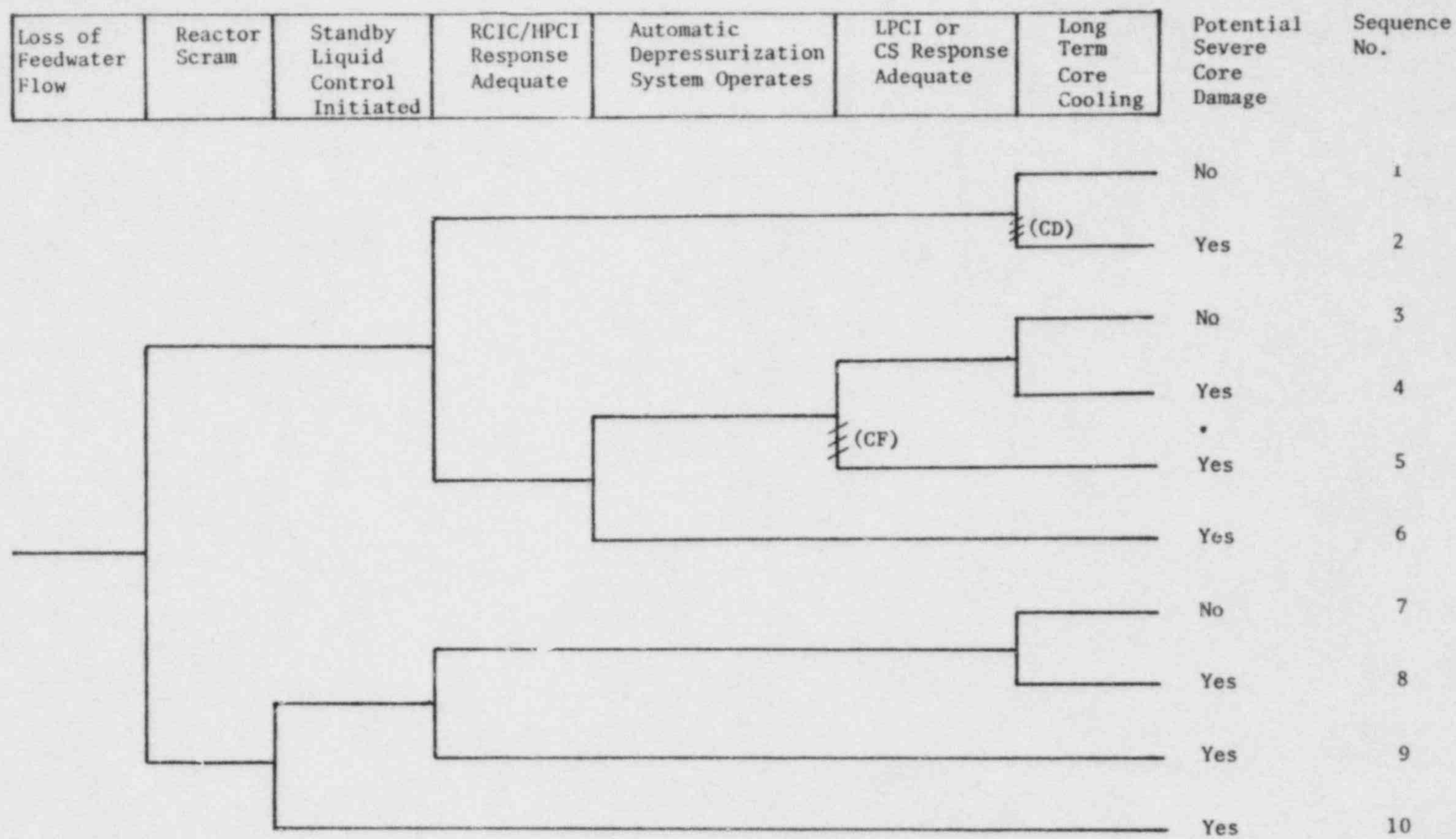
Design purpose of failed system or component:

The RBCCWS supplies cooling water to the following systems:

Loop A	Loop B
Two RHR pump lube oil coolers	Two RHR lube oil coolers
One fuel pool heat exchanger	One fuel pool heat exchanger
One RHR heat exchanger	One RHR heat exchanger
Two recirculating pump MG set fluid coupling oil and bearing coolers	Two control rod drive pump oil and bearing coolers
Two RCIC pump area cooling coils	Two recirculating pump seal water coolers
Two RHR pump area cooling coils	Two recirculating motor lube oil coolers
Four MG set area cooling coils	Two cleanup recirculating pump seal water coolers
One core spray pump motor thrust bearing	One cleanup nonregenerative heat exchanger
Sample coolers	Two HPCI pump area cooling coils
	Two control rod drive pump area cooling coils
	Six drywell air cooling coils
	Two drywell air cooling coils
	One core spray pump motor thrust bearing



NSIC 160532 - Actual Occurrence of Component Cooling Water Inoperable at Pilgrim 1



NSIC 160532 - Sequence of Interest for Component Cooling Water Inoperable at Pilgrim 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 160532

LER NO.: 80-070

DATE OF LER: October 22, 1980

DATE OF EVENT: October 10, 1980

SYSTEM INVOLVED: Reactor building closed cooling water system (RBCCWS)

COMPONENT INVOLVED: Circuit breaker

CAUSE: Spurious trips

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Component cooling water inoperable

REACTOR NAME: Pilgrim I

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 8.3 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison Co.

LOCATION: 4 miles SE of Plymouth, Massachusetts

DURATION: 10 min

PLANT OPERATING CONDITION: 100% full power

TYPE OF FAILURE: Inadequate performance;
made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 160559

Date: November 14, 1980

Title: Inadvertent Opening of Relief Valve at Pilgrim 1

The failure sequence was:

1. The reactor was at full power.
2. Liquid nitrogen passing by its ambient air vaporizer froze an inline nitrogen supply regulator in place while it was in the fail-open position, due to oversight in system design.
3. The full pressure nitrogen supply actuated a reactor vessel relief valve's solenoid, which caused the "A" relief valve to open.
4. The valve was manually cycled 4 to 5 times in an attempt to close it, but the valve remained open.
5. The high nitrogen pressure at the relief valve's solenoid prevented closure of the relief valve while the reactor was at operating pressure.
6. The operator reduced core power to 50% and initiated a manual scram.
7. The reactor continued to depressurize. At 340 psig reactor pressure the "A" relief valve was manually cycled and closed.

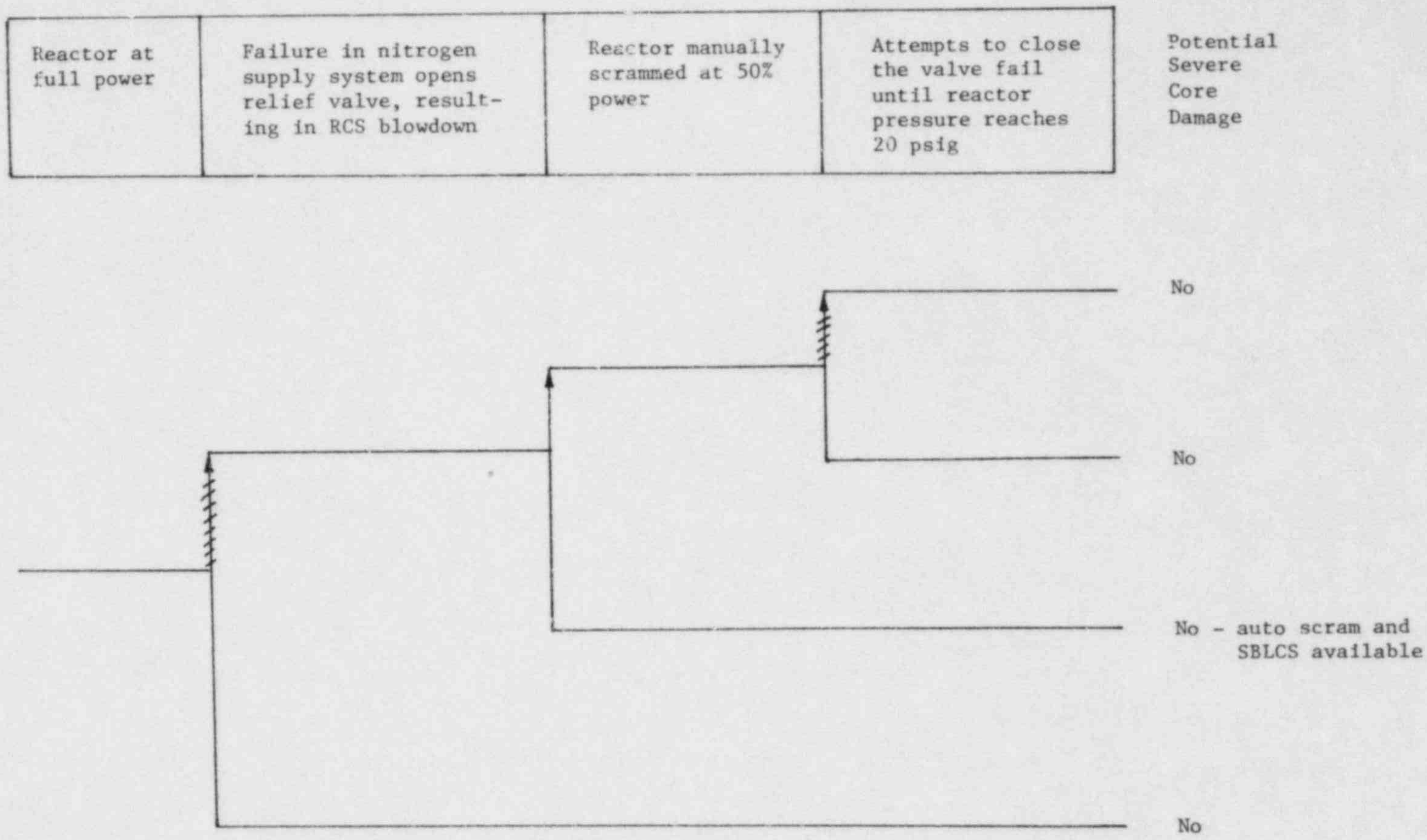
Corrective action:

1. The relief valve's solenoid actuator supply was switched from nitrogen to compressed air while changes in the nitrogen system were considered.
2. Replacement of the nitrogen regulator valve was considered along with reduction in the nitrogen supply tank pressure and installation of a nitrogen header relief valve.

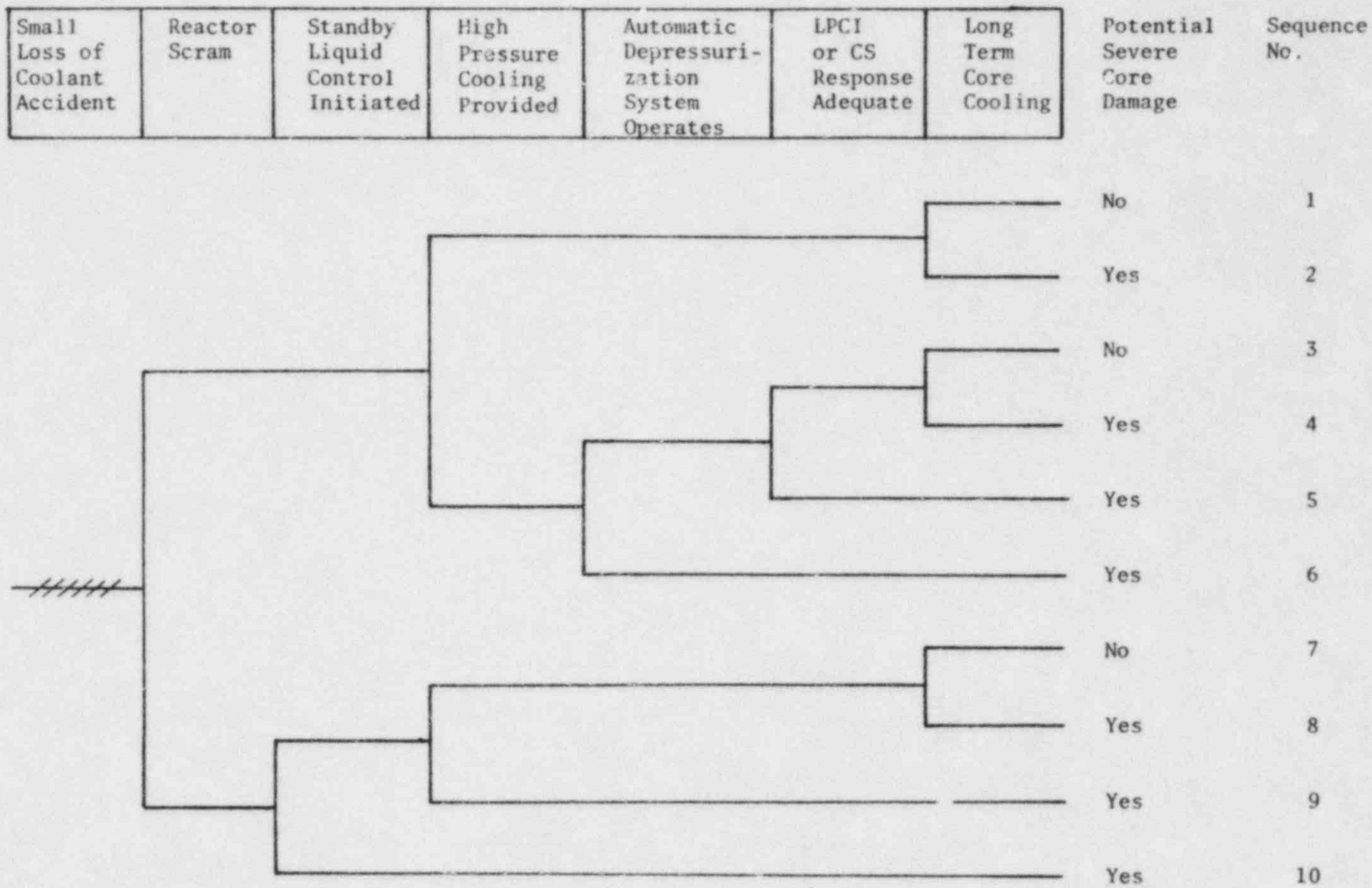
Design purpose of failed system or component:

The nitrogen system provides motive power for opening and closing reactor relief valves (among other control functions) upon automatic demands as well as manual demands.

The reactor relief valve provides RCS pressure protection.



NSIC 160559 - Actual Occurrence for Inadvertent Opening of Reactor Relief Valve at Pilgrim 1



NSIC 160559 - Sequence of Interest for Inadvertent Opening of Relief Valve at Pilgrim 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 160559

LER NO.: 80-080

DATE OF LER: November 14, 1980

DATE OF EVENT: October 31, 1980

SYSTEM INVOLVED: Pressure relief system

COMPONENT INVOLVED: Relief valve

CAUSE: Excessive nitrogen pressure to valve operator

SEQUENCE OF INTEREST: Main steam line break

ACTUAL OCCURRENCE: Inadvertent opening and sticking of a reactor relief valve

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 8.4 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison Co.

LOCATION: 4 miles SE of Plymouth, Massachusetts

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Operational event

COMMENT: See accession 160497 for a similar event at Pilgrim 1, LER 80-69, but with a different cause.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 160846

Date: March 20, 1980

Title: Loss of 24 Volt dc Nonnuclear Instrument Power Supply at Crystal River 3

The failure sequence was:

1. With the reactor at 100% power, the 24 V dc nonnuclear instrumentation "X" power supply was lost due to a failed voltage buffer card.
2. Because of this, the integrated control system received erroneous signals and demanded (1) a reactor power increase, (2) a feedwater runback, and (3) the opening of the power-operated relief valve.
3. The open PORV did not prevent reactor coolant pressure from reaching the over-pressure trip set point, which initiated a reactor trip.
4. The loss of power caused the PORV to stay open. Primary coolant was discharged into the reactor coolant drain tank and then into the reactor containment building (43,000 gal) after the drain tank rupture disk opened. RCS pressure dropped resulting in high pressure injection being automatically actuated at 3 min (HPI continued for 84 min).
5. At 4 min the operator tripped the reactor coolant pumps as required by the HPI initiation.
6. The "A" once through steam generator effectively boiled dry at approximately 8 min.
7. The operator shut the block valve which stopped the loss of reactor coolant through the PORV at 2 to 5 min.
8. The operator initiated emergency feedwater at approximately 9 min.
9. The RCS pressure increased due to continued HPI at 1100 gpm, and a code safety valve opened at 10 min.
10. Power was restored to the nonnuclear instrumentation about 21 min into the transient.
11. At 29 min the operator throttled the HPI flow to about 250 gpm.
12. Until about 2 h after the incident began, the safety valve periodically released coolant at 2300 psi to the containment. About 40,000 gal of coolant was dumped into the reactor building.
13. At about 5 h 7 min into the transient, decay-heat closed-cycle cooling pump DCP-1A failed, rendering the "A" decay-heat closed-cycle cooling loop inoperable. The failure was due to failure of the pump motor coupling.
14. The plant was stabilized and maintained in hot standby on natural circulation until forced RCS flow was initiated about 6 h 16 min

into the transient. The plant was subsequently taken to cold shutdown.

15. Throughout the event, core coverage was maintained; no fuel damage occurred, and the RCS remained within code allowable limits.

Corrective action:

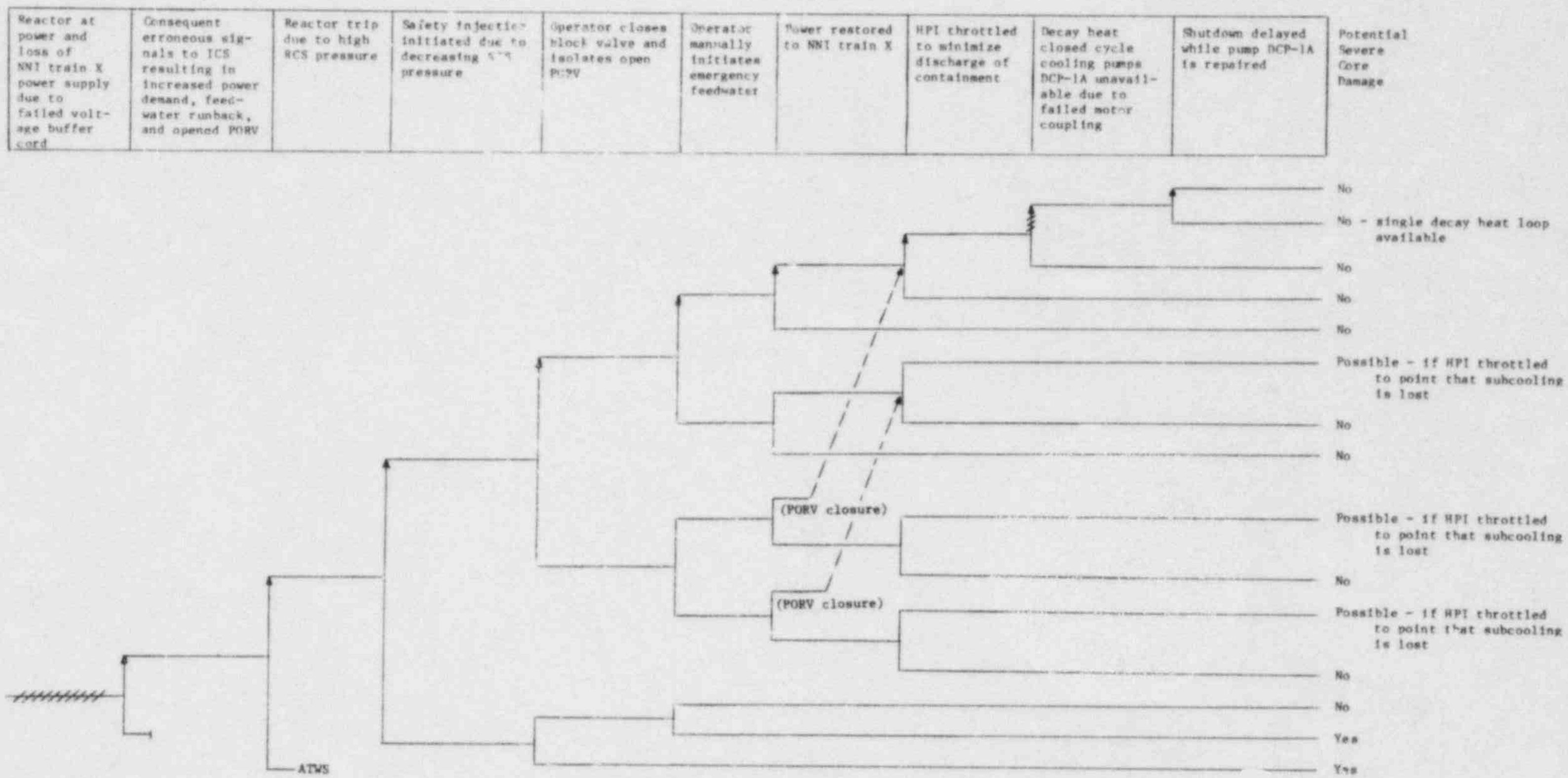
The plant was shut down for resolution of the problem and implementation of corrective actions. These actions included:

- Complete testing and inspection of the nonnuclear instrumentation system for similar failures.
- Installation of new redundant channels for indication of twenty-three key plant parameters to provide more reliable information to the operator.
- Comprehensive operator training in response actions for NNI and ICS failures.
- Installation of positive position indication on the power operated relief valve and the two code safety valves.
- Modification of the NNI power supply to provide more reliable power.
- Evaluation of NNI power supply reliability in response to IE Bulletin 79-27 (Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation).
- Modification of the control circuitry for the PORV and pressurizer spray valves so that the valves will not open in the event of loss of NNI power.

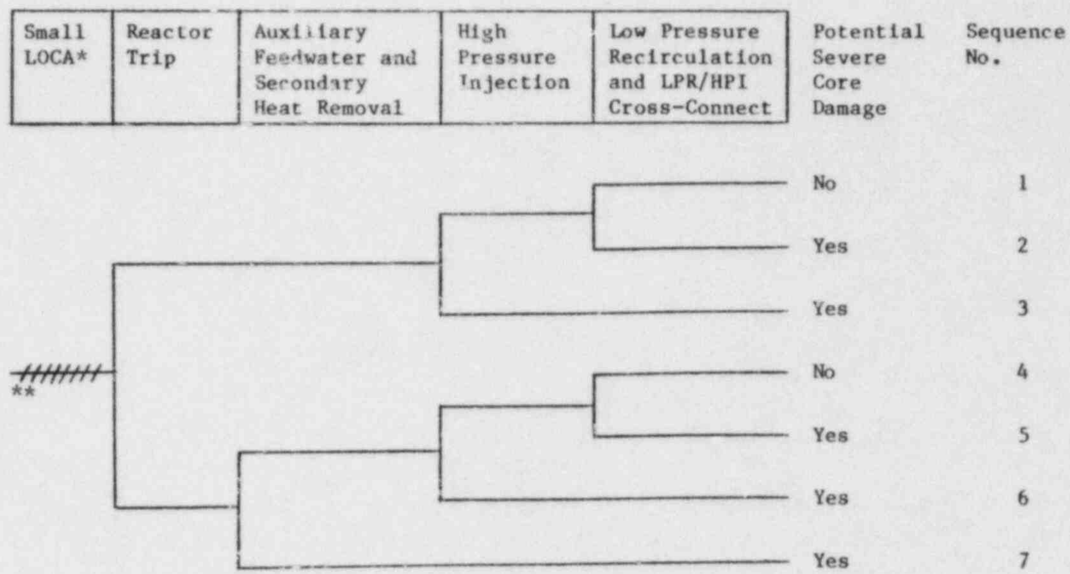
The NRC initiated an extensive review program regarding the effect of the Crystal River event.

Design purpose of failed system or component:

The nonnuclear process instrumentation provides the required input signals of process variables for the reactor protection, regulating, and auxiliary systems. It performs the required process control functions in response to those systems and provides instrumentation for startup, operation, and shutdown of the reactor system under normal and emergency conditions. Inputs to the reactor protective system by the nonnuclear instrumentation are reactor outlet temperature, coolant flow, and pressure. Inputs provided to the integrated control system are reactor outlet temperature, inlet differential temperature, coolant flow, feedwater temperature, feedwater flow, steam generator level, steam generator outlet pressure, and turbine header pressure.



NSIC 160846 – Actual Occurrence for Loss of 24-V Nonnuclear Instrument Power Supply at Crystal River 3



NSIC 160846 - Sequence of Interest for Loss of 24-V dc Nonnuclear Instrumentation Power Supply at Crystal River 3

*due to loss of NNI-X power supply

**continued small LOCA requires operator error in not closing PORV block valve

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 160846

LER NO.: 80-010 Rev. 1

DATE OF LER: March 20, 1980

DATE OF EVENT: February 26, 1980

SYSTEM INVOLVED: Nonnuclear instrumentation

COMPONENT INVOLVED: 24 V dc power supply

CAUSE: A combination of misaligned connector pins on printed circuit boards and a technician working on saturation meter circuit utilizing these boards resulted in loss of the power

SEQUENCE OF INTEREST: Loss of NNI-X power supply

ACTUAL OCCURRENCE: Failure of nonnuclear instrumentation power supply, reactor trip, turbine trip, and engineered safeguards actuation

REACTOR NAME: Crystal River Unit 3

DOCKET NUMBER: 50 302

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 3.1 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Gilbert Associates

OPERATORS: Florida Power Corporation

LOCATION: 7 miles NW of Crystal River, Florida

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 160926

Date: October 31, 1980

Title: Stuck Open Relief Valve at Pilgrim 1

The failure sequence was:

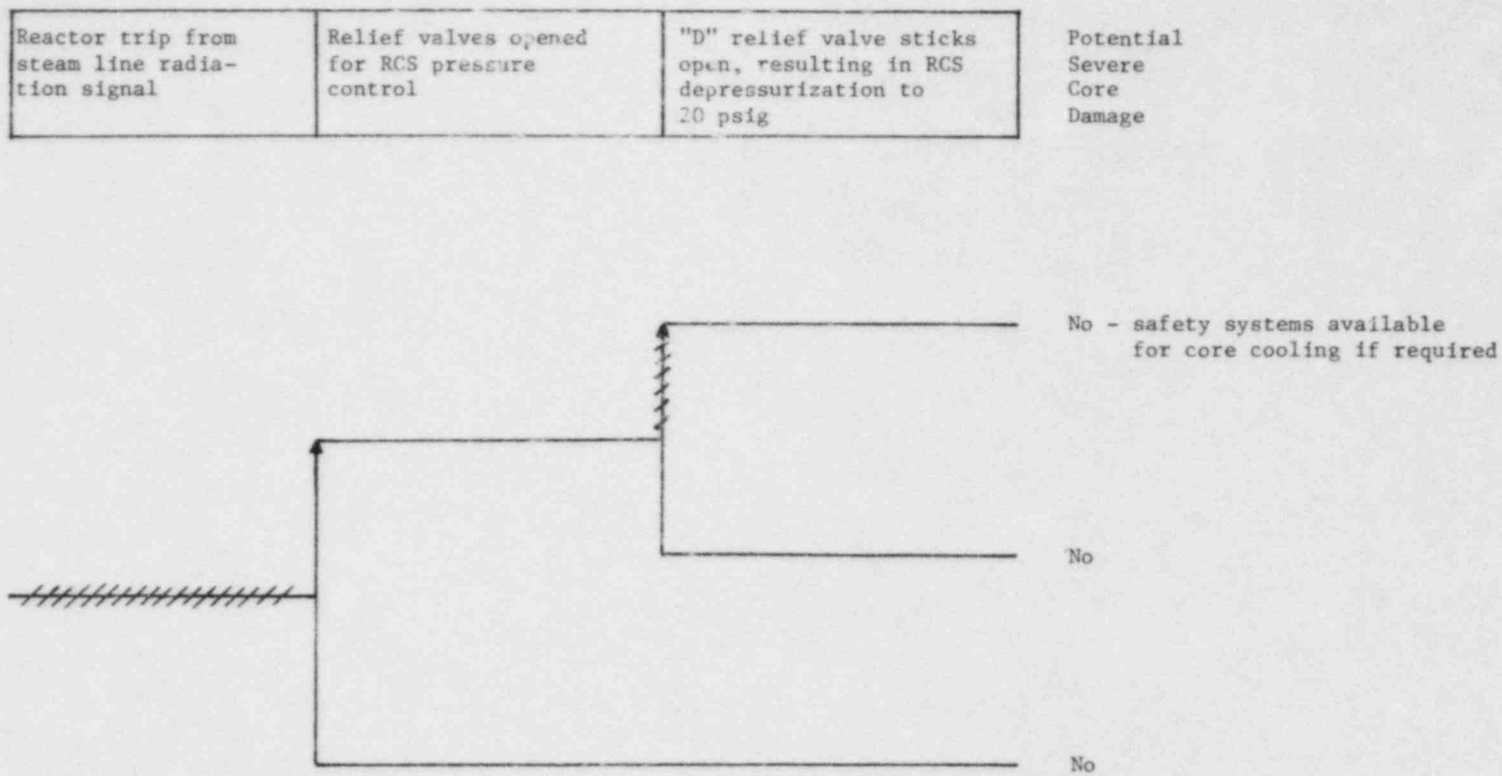
1. With the reactor at full power, a scram occurred due to main steam line high radiation.
2. To control pressure, "C" relief valve opened for 1 minute and closed successfully. HPCI and RCIC were placed in full flow test.
3. The MSIVs had closed during the event when the reactor depressurized beyond the MSIV closure set point of 880 psig. The "D" relief valve was opened to reduce the reactor pressure enough to be able to reset the scram and reopen the MSIVs.
4. The "D" relief valve stuck open.
5. Repeated attempts to energize/deenergize the relief valve solenoid valve from the control room failed to close the "D" relief valve and the plant depressurized to 20 psig.

Corrective action:

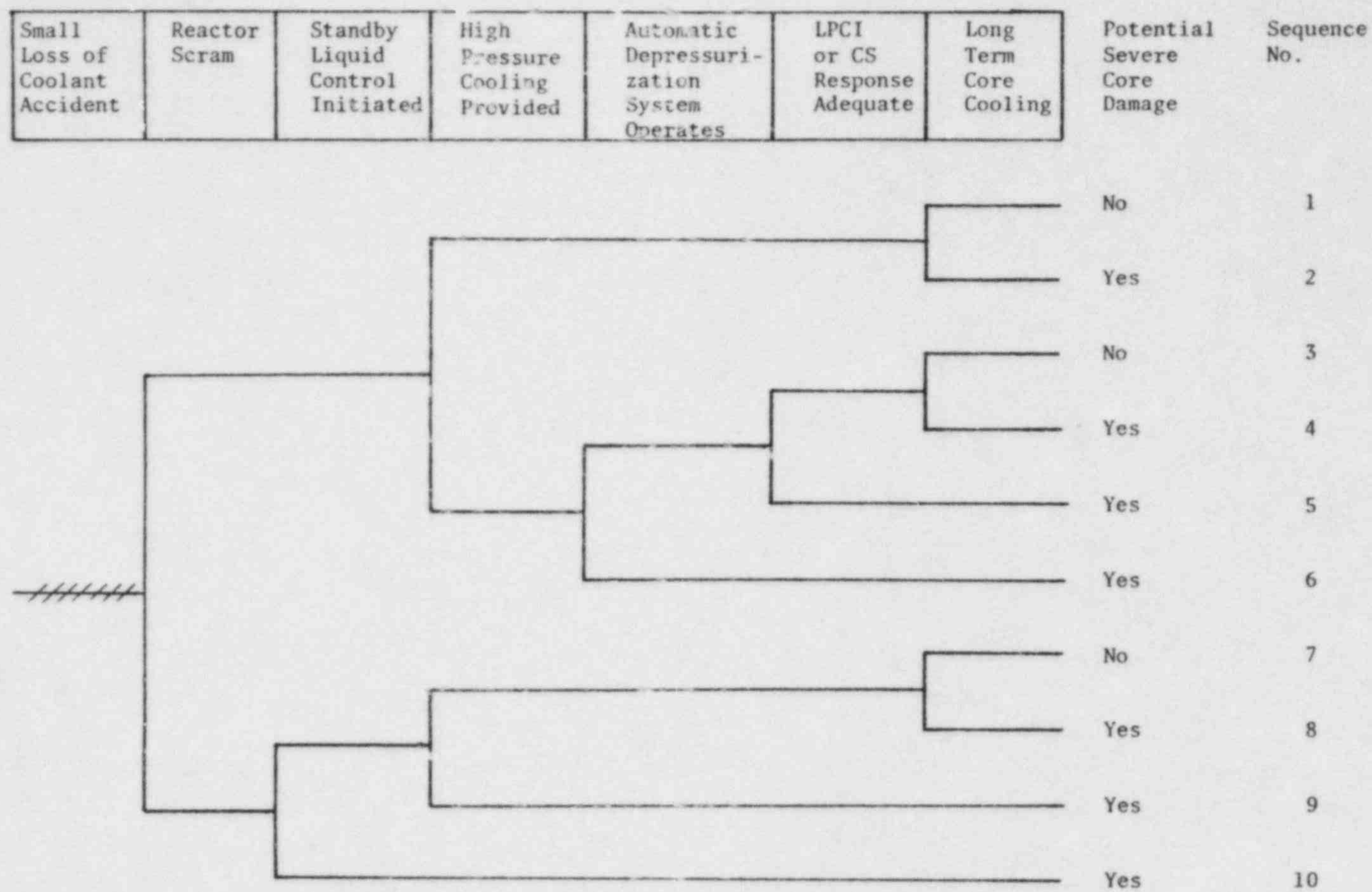
1. The plant was shutdown and the valve assembly removed and tested. The utility concluded that the probable cause of the failed valve was scoring on the main piston.
2. A spare valve assembly and solenoid were installed and the unit was returned to service.
3. The damaged valve was repaired.

Design purpose of failed system or component:

The pressure relief system provides overpressure protection for the RCS.



NSIC 160926 - Acutal Occurrence for Stuck-Open Relief Valve at Pilgrim 1



NSIC 160926 - Sequence of Interest for Stuck Open Relief Valve at Pilgrim 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 160926

LER NO.: 80-079

DATE OF LER: October 31, 1980

DATE OF EVENT: October 1, 1980

SYSTEM INVOLVED: Pressure relief system

COMPONENT INVOLVED: Relief valve

CAUSE: Mechanical failure in valve operator

SEQUENCE OF INTEREST: Main steam line break

ACTUAL OCCURRENCE: Stuck open relief valve

REACTOR NAME: Pilgrim 1

DOCKET NUMBER: 50-293

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 655 MWe

REACTOR AGE: 8.3 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Boston Edison Co.

LOCATION: 4 miles SE of Plymouth Massachusetts

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 161601

Date: November 28, 1980

Title: Loss of Service Water Cooling to the Diesel Generators at Salem 1

The failure sequence was:

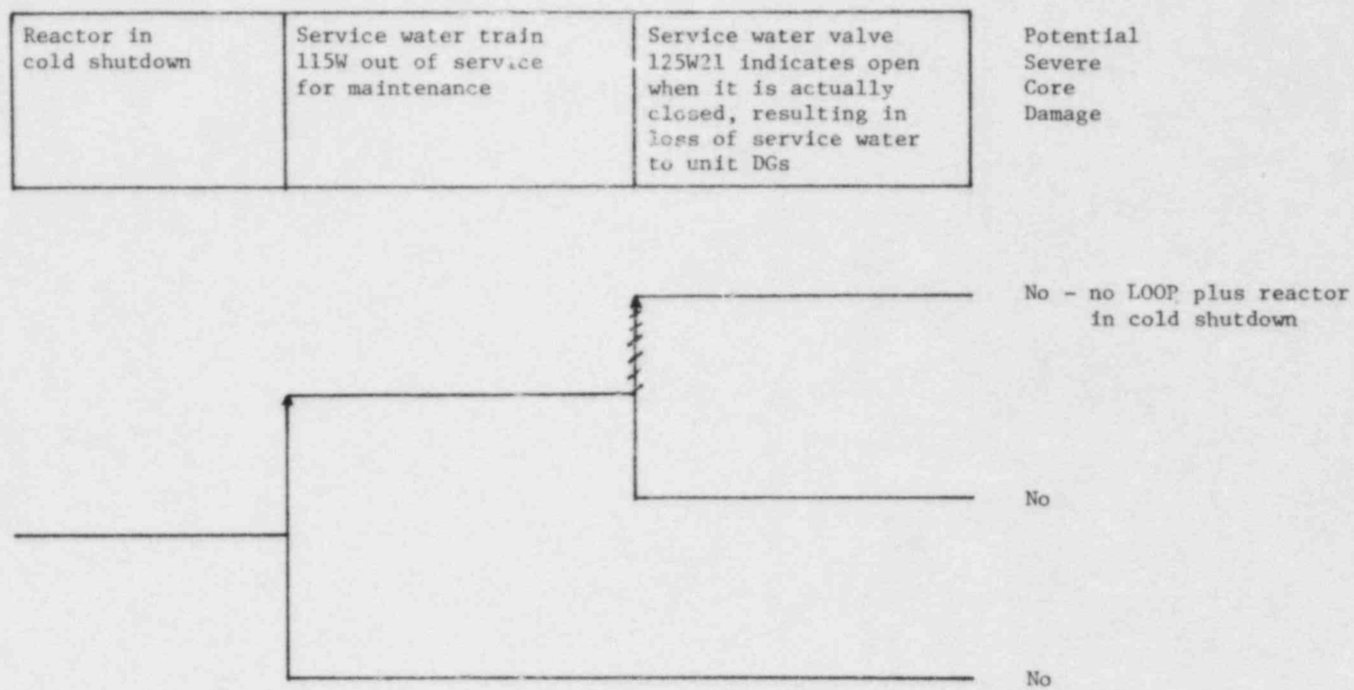
1. With the reactor in cold shutdown, service water header train 11SW was out of service for repairs.
2. The diesels were found to be overheating.
3. Train 12SW flow valve 12SW21 was found indicating open when it was actually closed, resulting in loss of service water to the DGs.

Corrective action:

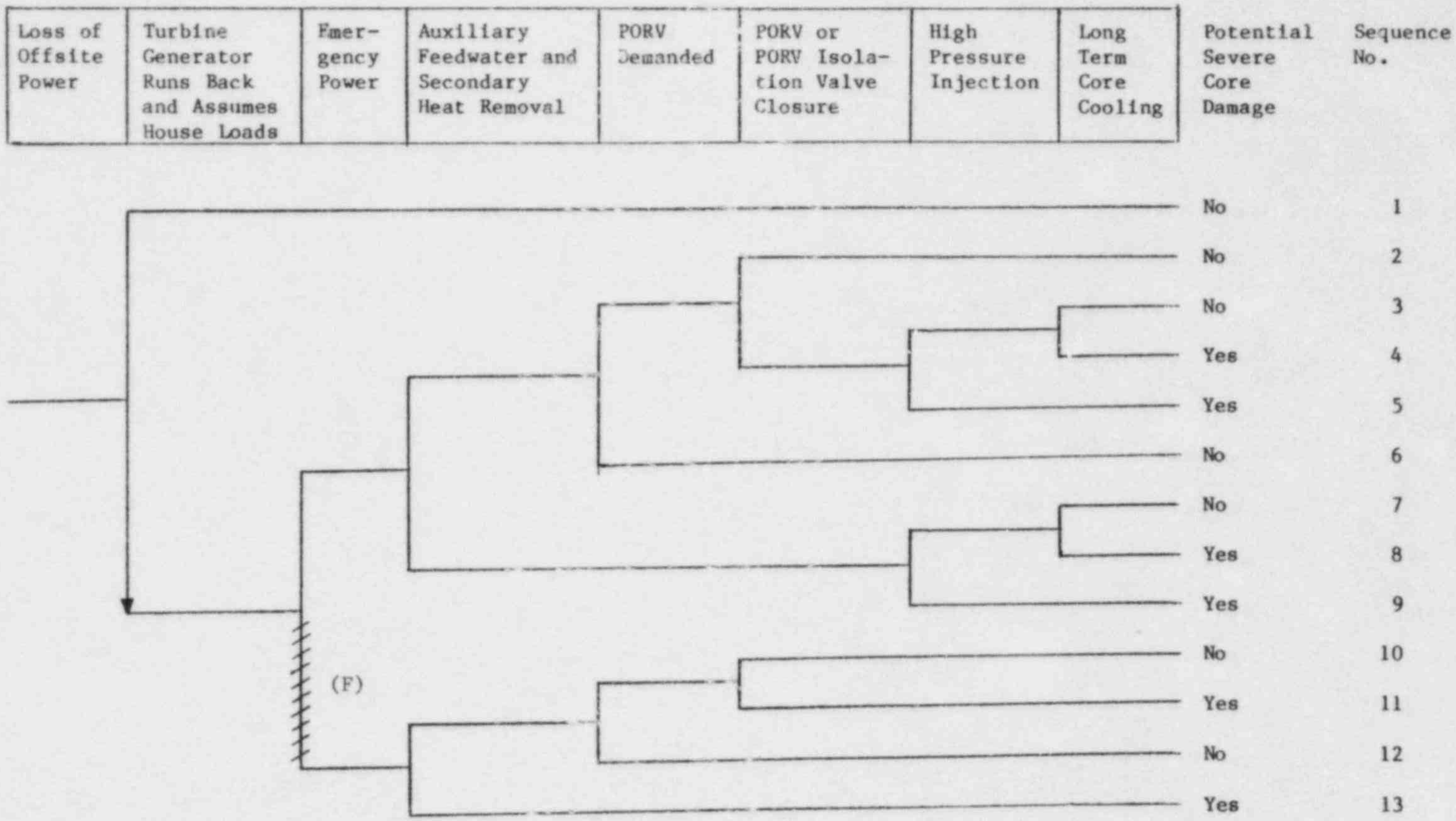
Valve 12SW21 was opened and service water flow restored. The valve positioner and indicator were realigned.

Design purpose of failed system or component:

Valve 12SW21 controls the cooling flow from No. 12SW service water header to the diesel generators.



NSIC 161601 - Actual Occurrence for Loss of Service Water Cooling to Diesel Generators at Salem 1



NSIC 161601 - Sequence of Interest for Loss of Service Water System to Diesel Generators at Salem 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 161601

LER NO.: 80-060

DATE OF LER: November 28, 1980

DATE OF EVENT: October 8, 1980

SYSTEM INVOLVED: Emergency power system

COMPONENT INVOLVED: Valve positioner and indicator

CAUSE: Alignment error

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Loss of service water cooling to diesel generators

REACTOR NAME: Salem 1

DOCKET NUMBER: 50-272

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1090 MWe

REACTOR AGE: 3.8 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Public Service Electric and Gas

OPERATORS: Public Service Electric and Gas

LOCATION: 20 miles south of Wilmington, Delware

DURATION: 4-1/4 h

PLANT OPERATING CONDITION: Refueling shutdown

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 161649

Date: November 17, 1980

Title: Two Diesel Generators Inoperable at Sequoyah 1

The failure sequence was:

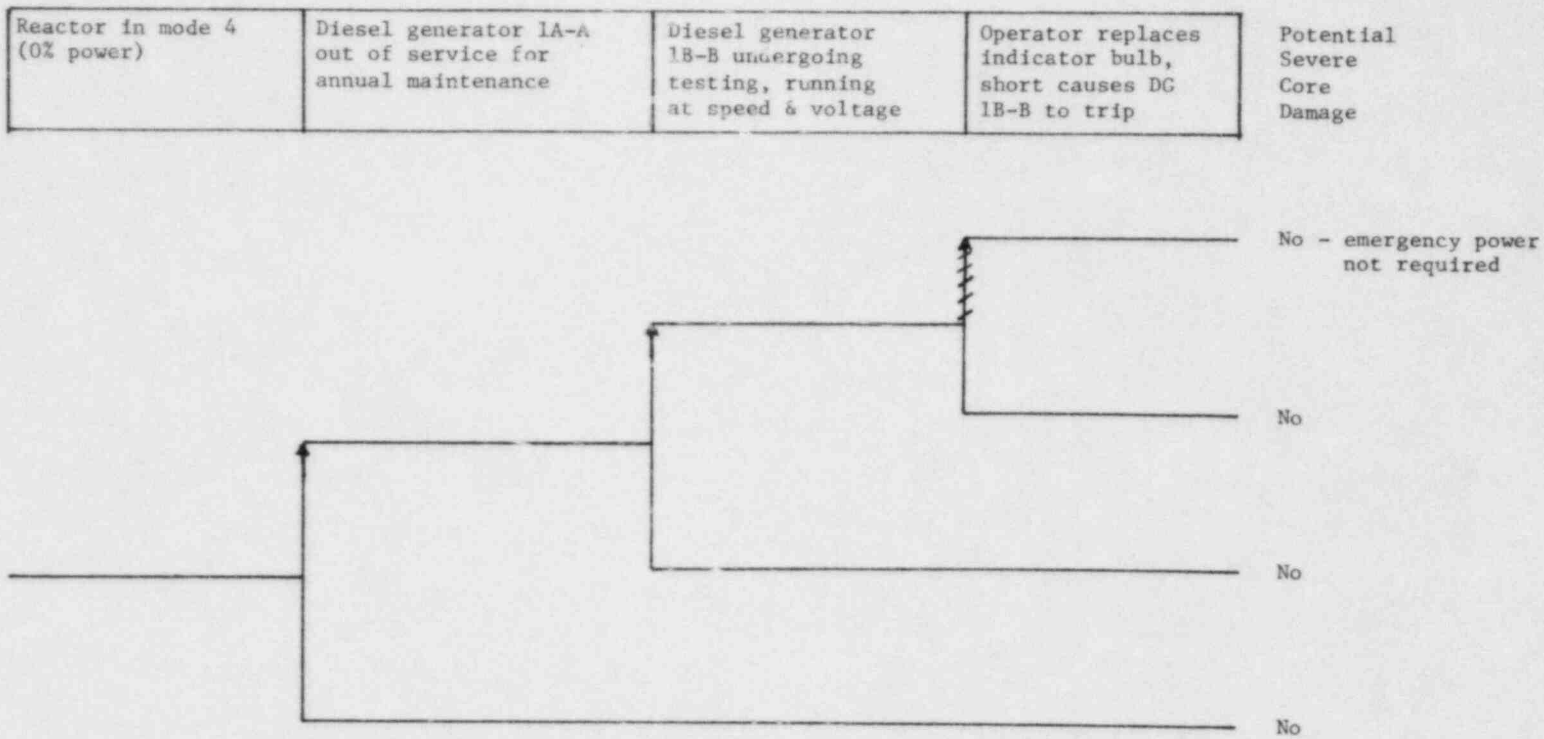
1. With the reactor in mode 4 (0% power), diesel generator 1A-A was out of service for annual maintenance.
2. Diesel generator 1B-B was operating at speed and voltage during testing.
3. The diesel generator (1B-B) tripped when the operator replaced an indicator bulb, a result of an internal short in the lamp socket which tripped a relay and subsequently tripped the diesel.

Corrective action:

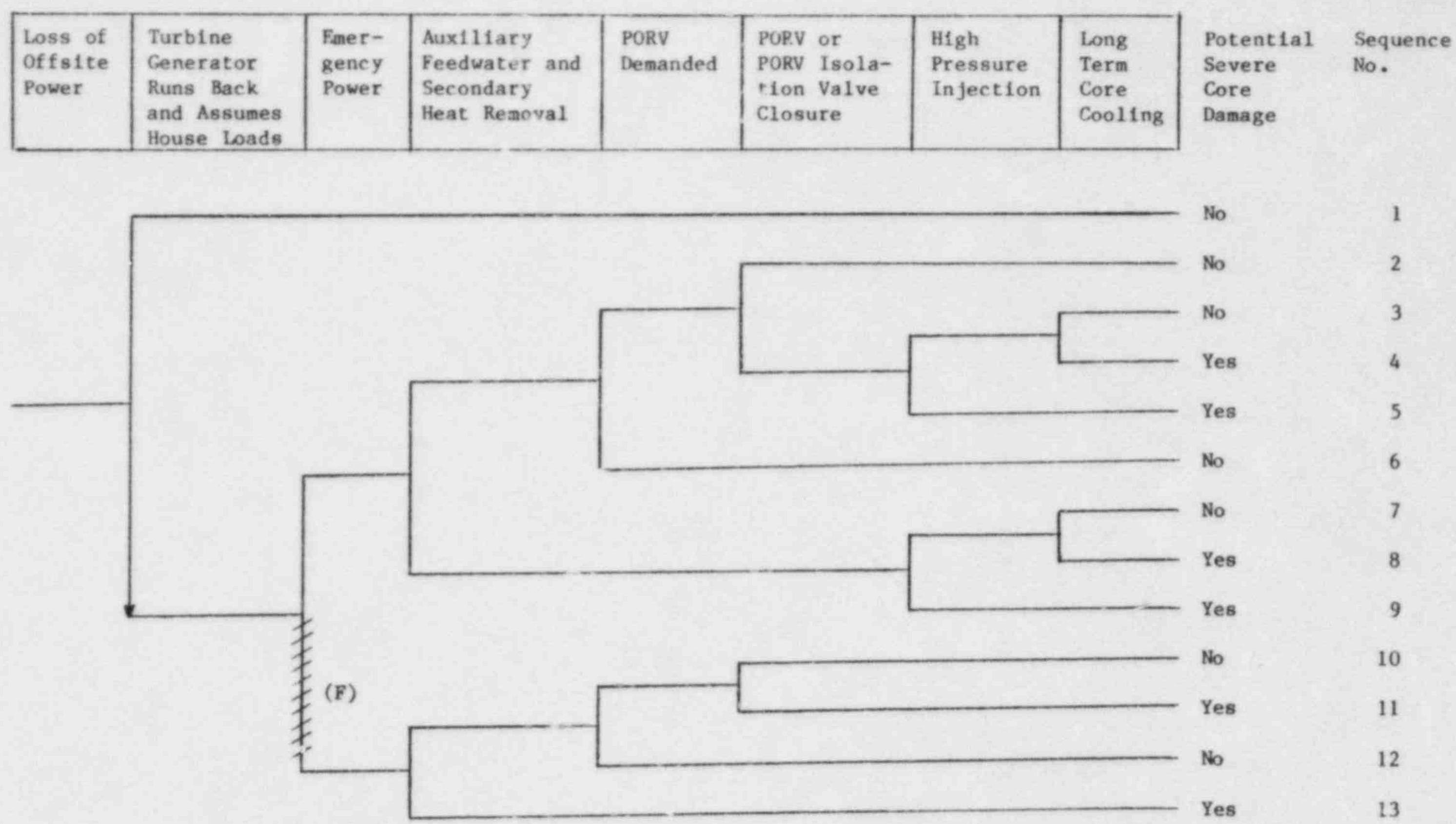
The lamp socked was repaired and DG 1B-B returned to service.

Design purpose of failed system or component:

The diesel generators provide emergency power for safe shutdown when the normal sources of AC power are not available.



NSIC 161649 - Actual Occurrence for Two Diesel Generators Inoperable at Sequoyah 1



NSIC 161649 - Sequence of Interest for Loss of Emergency Power System at Sequoyah 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 161649

LER NO.: 80-171

DATE OF LER: November 17, 1980

DATE OF EVENT: October 16, 1980

SYSTEM INVOLVED: Emergency power system

COMPONENT INVOLVED: Diesel generators

CAUSE: Internal short in indicator light socket for one diesel while other was unavailable due to maintenance.

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Unavailability of two diesel generators

REACTOR NAME: Sequoyah Unit 1

DOCKET NUMBER: 50-327

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1148 MWe

REACTOR AGE: 0.3 year

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Tennessee Valley Authority

OPERATORS: Tennessee Valley Authority

LOCATION: 9.5 miles NE of Chattanooga, TN

DURATION: 2 h 8 min

PLANT OPERATING CONDITION: Mode 4, 0% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Diesel tripped as a result of a short circuit during testing.

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 161906

Date: November 24, 1980

Title: RCIC and HPCI Inoperable at Quad-Cities 2

The failure sequence was:

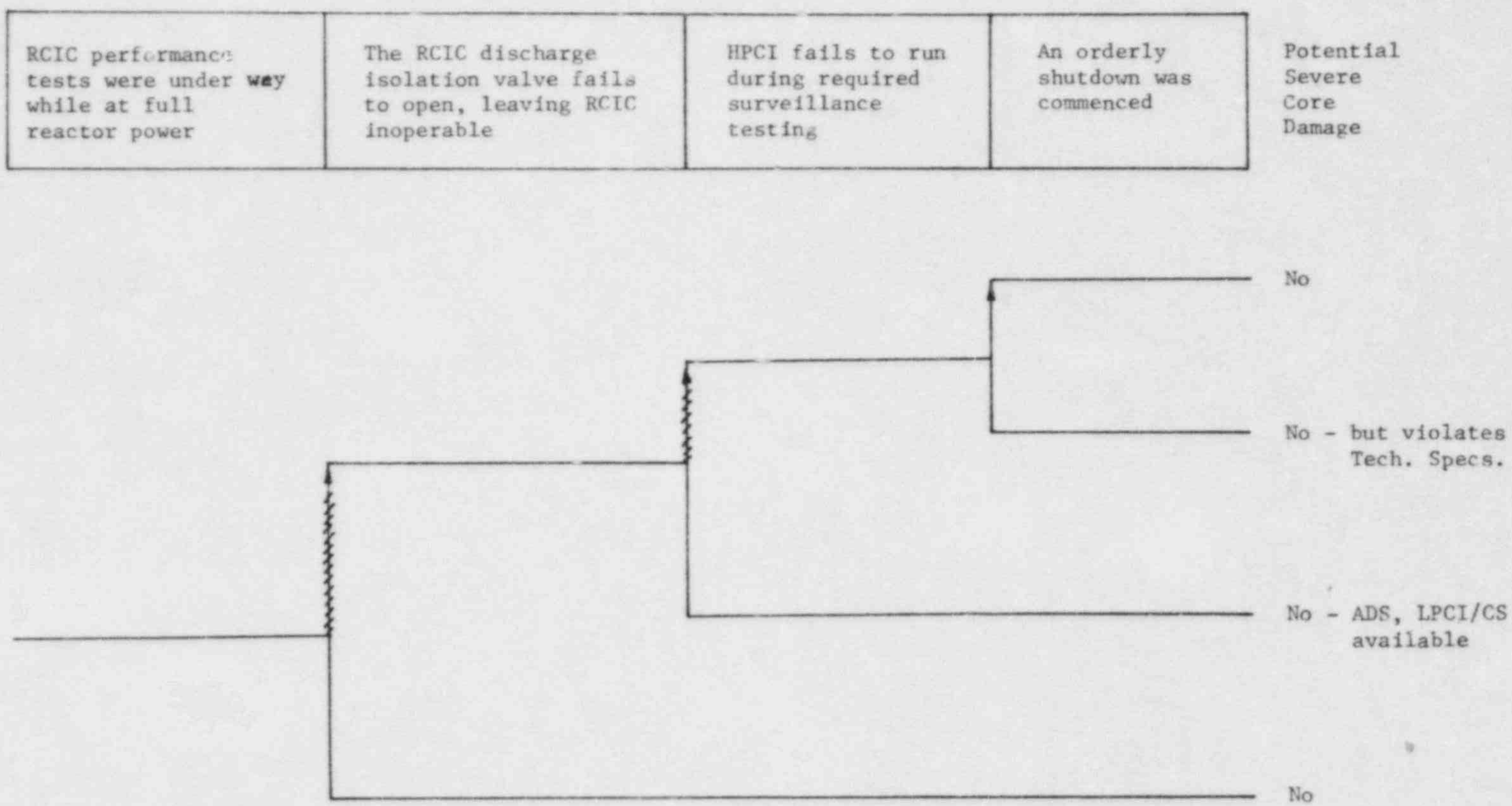
1. During performance test of the monthly operability at 99% power, the RCIC discharge isolation valve would not open from the control room. This caused RCIC to be inoperable. A torque switch on the valve was faulty.
2. HPCI was subsequently declared inoperable due to an oil leak in the HPCI turbine stop valve actuator cover.
3. An orderly shut down was commenced.

Corrective action:

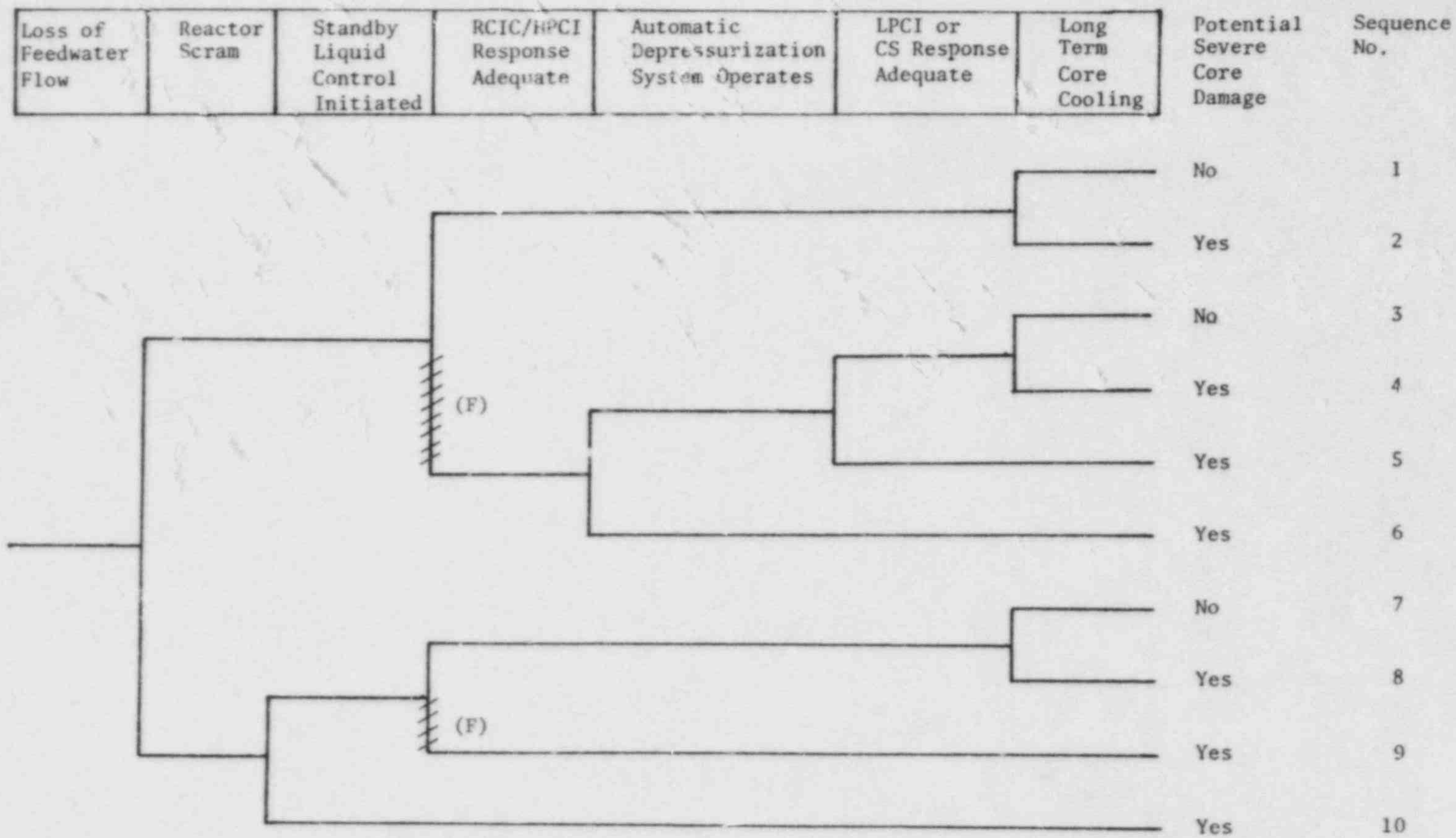
1. The RCIC torque switch was replaced before shutdown was reached and the shutdown was terminated.
2. The HPCI valve actuator cover was replaced with a cover from Unit 1.
3. About 10 h was required to correct the simultaneous loss of RCIC and HPCI.

Design purpose of failed system or component:

1. RCIC provides for reactor water level control given a loss of feedwater.
2. HPCI provides for water level control given a small break LOCA.



NSIC 161906 - Actual Occurrence of RCIC and HPCI Inoperable at Quad Cities 2



NSIC 161906 - Sequence of Interest for RCIC and HPCI Inoperable at Quad Cities 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 161906

LER NO.: 80-077

DATE OF LER: November 24, 1980

DATE OF EVENT: November 16, 1980

SYSTEM INVOLVED: RCIC and HPCI system

COMPONENT INVOLVED: Torque switch; turbine stop valve

CAUSE: Failure of the torque switch; oil leak

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: RCIC and HPCI inoperable at Quad-Cities 2

REACTOR NAME: Quad-Cities 2

DOCKET NUMBER: 265

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 789 MWe

REACTOR AGE: 8.6 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison Co.

LOCATION: 20 miles NE of Moline, Illinois

DURATION: 360 h (estimated)

PLANT OPERATING CONDITION: 99% power

TYPE OF FAILURE: Failed to start;
made inoperable

DISCOVERY METHOD: Testing

COMMENT: ADS, LPCI, and core spray were available as backup systems.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 162083

Date: December 26, 1980

Title: Loss of Refueling Water Tank Level Instrumentation at Arkansas Nuclear Unit 2

The failure sequence was:

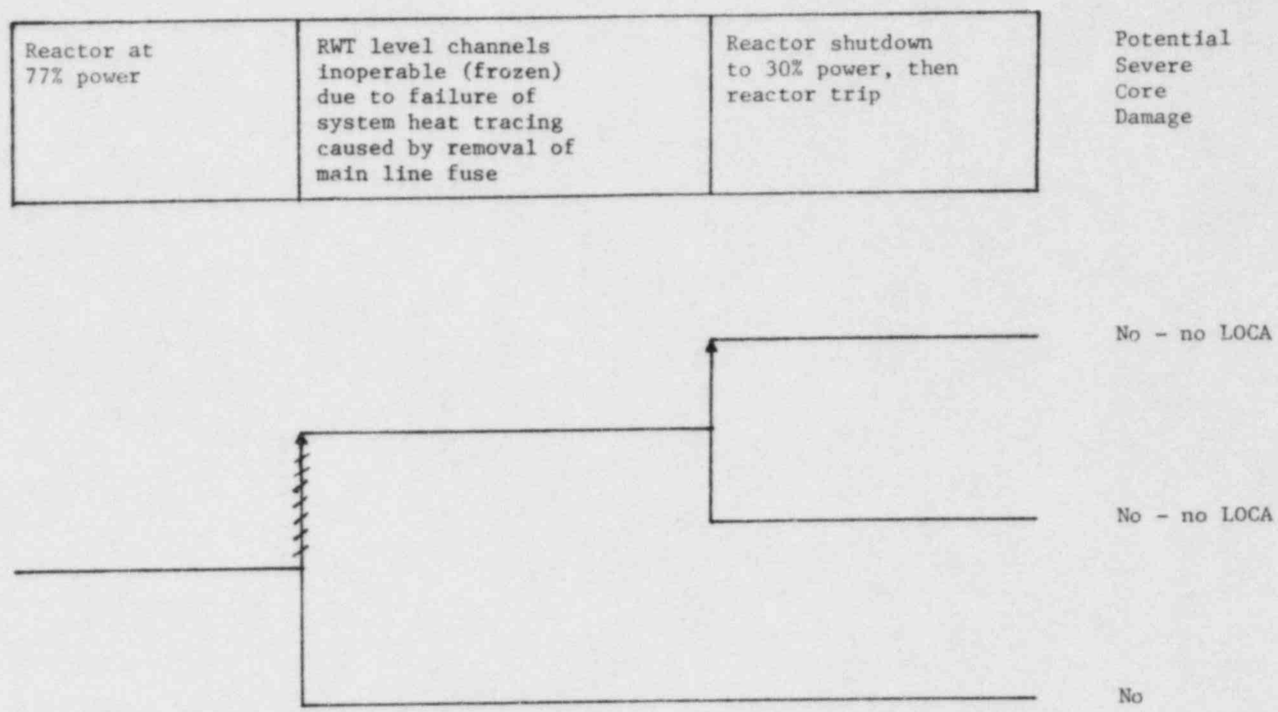
1. With the reactor at 77% power, refueling water tank level channels C and D were observed indicating greater than 100%, while channel B indicated 82%. Previous checks indicated approximately 94%.
2. All four channels were found to be frozen. The system heat tracing circuit was de-energized because the main line fuse had been removed. The ambient temperature was 15°F.
3. Reactor shutdown was initiated to meet Tech. Spec. requirements. A trip was required at 30% power.

Corrective action:

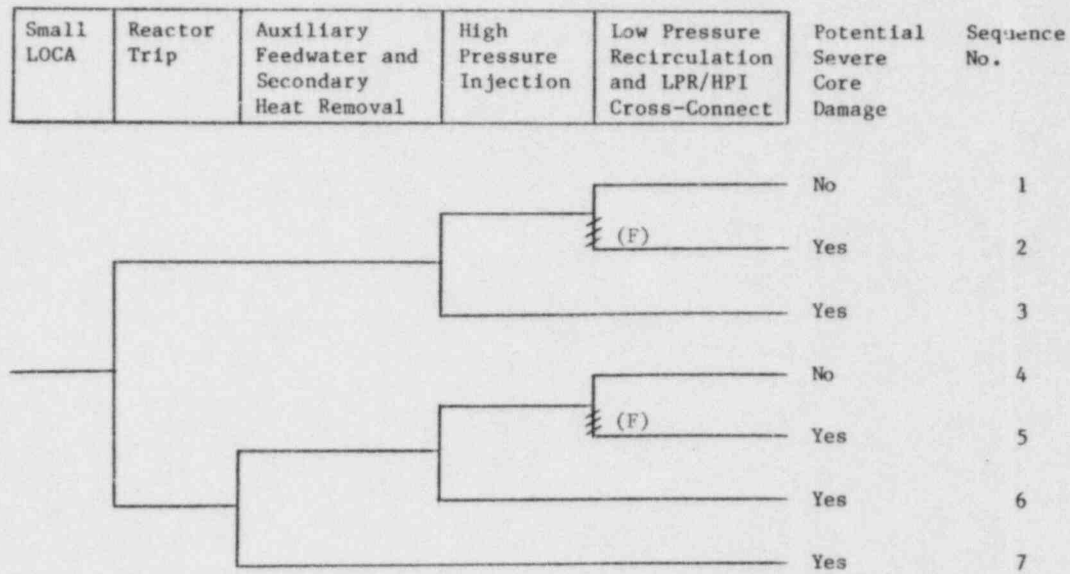
The fuse was replaced, the transmitters thawed out, and the RWST indication returned to operable status. Long-term corrective actions were being evaluated at the time of the report.

Design purpose of failed system or component:

The RWST level instruments monitor RWST level and initiate the ECCS switchover from the RWST to the containment sump during safety injection.



NSIC 162083 - Actual Occurrence for Loss of RWT Level Instrumentation at Arkansas Unit 2



NSIC 162083 - Sequence of Interest for Loss of RWT Level Instrumentation at Arkansas Unit 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 162083

LER NO.: 80-091

DATE OF LER: December 26, 1980

DATE OF EVENT: December 20, 1980

SYSTEM INVOLVED: Low pressure recirculation

COMPONENT INVOLVED: RWST level instruments

CAUSE: Personnel error — fuse removed

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Level transmitters unavailable due to freezing

REACTOR NAME: Arkansas Nuclear Unit 2

DOCKET NUMBER: 50-368

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 912 MWe

REACTOR AGE: 2.1 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power & Light Co.

LOCATION: 6 miles NW of Russellville, Arkansas

DURATION: 8 h (estimated)

PLANT OPERATING CONDITION: 77% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD Operator observation

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 163356

Date: January 16, 1981

Title: Output Breakers for Both Station Batteries Opened at Palisades

The failure sequence was:

1. With the reactor at full power, a battery charge started the previous day was being terminated.
2. As part of this procedure, the breakers associated with the in-service chargers were to be opened and the breakers associated with the idle chargers were to be closed. Battery breaker position was not annunciated in the control room.
3. However, plant electricians closed the idle charger breakers and opened the station battery breakers.
4. Because of the misalignment, a required dc bus voltage adjustment could not be made. Investigation of this by a supervisor led to the discovery of the incorrect breaker alignment (approximately 1 h later), and the breaker alignment was restored.

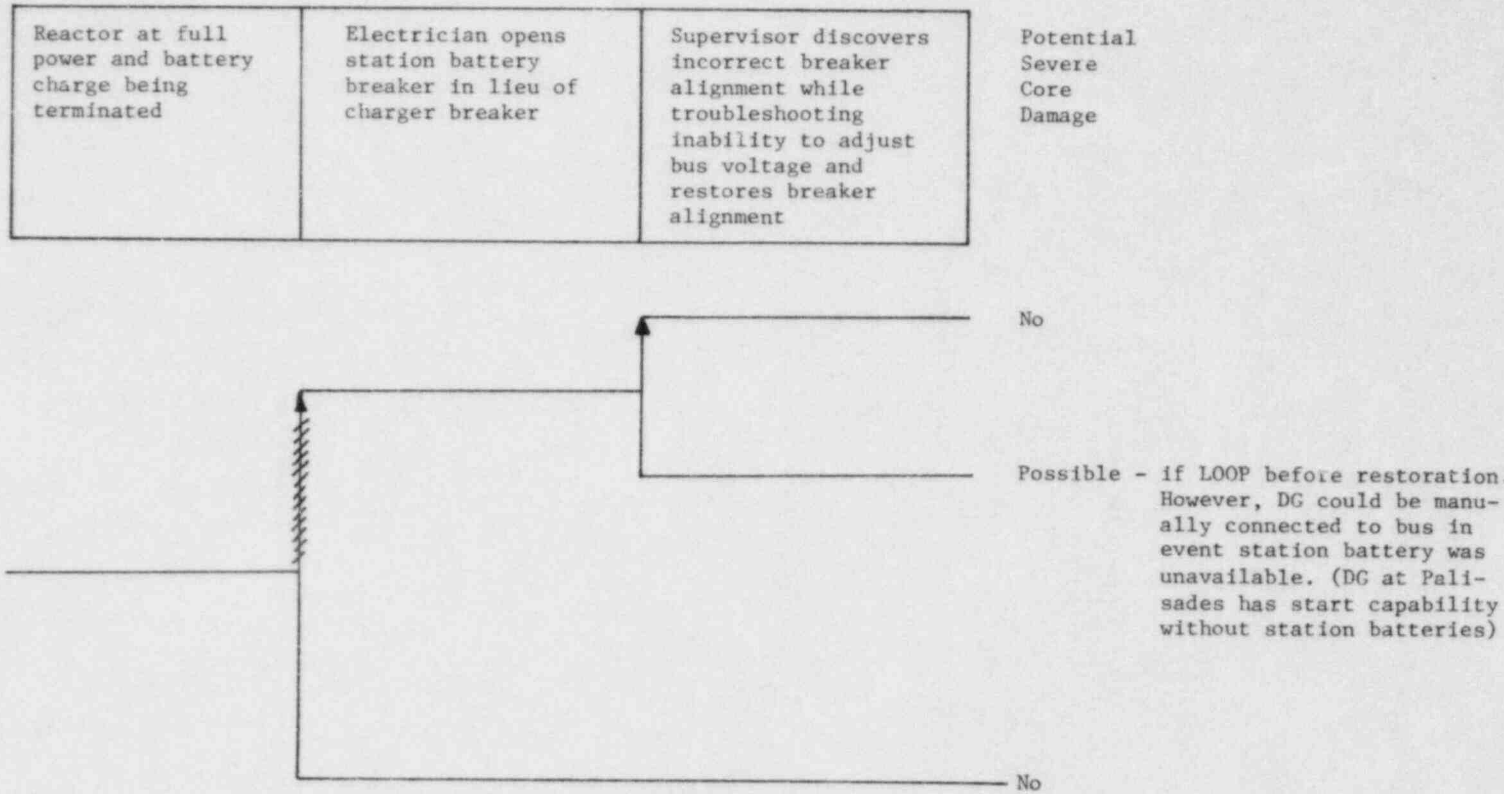
Corrective action:

1. Proper breaker alignment was restored.
2. The following short-term corrective actions were to be implemented:
   a. Daily audits of plant operations by a corporate management representative.
   b. A committee consisting of a member of corporate management (in addition to the corporate representative referenced in item 1 above), a senior reactor operator, and another qualified engineer was to review all safety-related surveillance and maintenance procedures and other maintenance procedures which cover work to be conducted in vital areas before they are used again, to ensure that:
      (1) each procedure is specifically identified as being safety related, or as having the potential to affect safety-related equipment;
      (2) authorization to perform work is required from plant manangement;
      (3) special notification of work performed is made to the Shift Supervisor;
      (4) system conditions to perform work are defined;
      (5) minimum personnel skill level is defined; and
      (6) return-to-normal verification requirements are specified.
   c. All personnel who perform safety-related work or other work in vital areas were to be reinstructed on the importance of strict adherence to procedures and the necessity for performance of all assigned duties in a disciplined and professional manner.
   d. Immediately upon their completion, all activities involving the maniuplation of safety-related circuits or systems were to be verified by a second qualified individual.

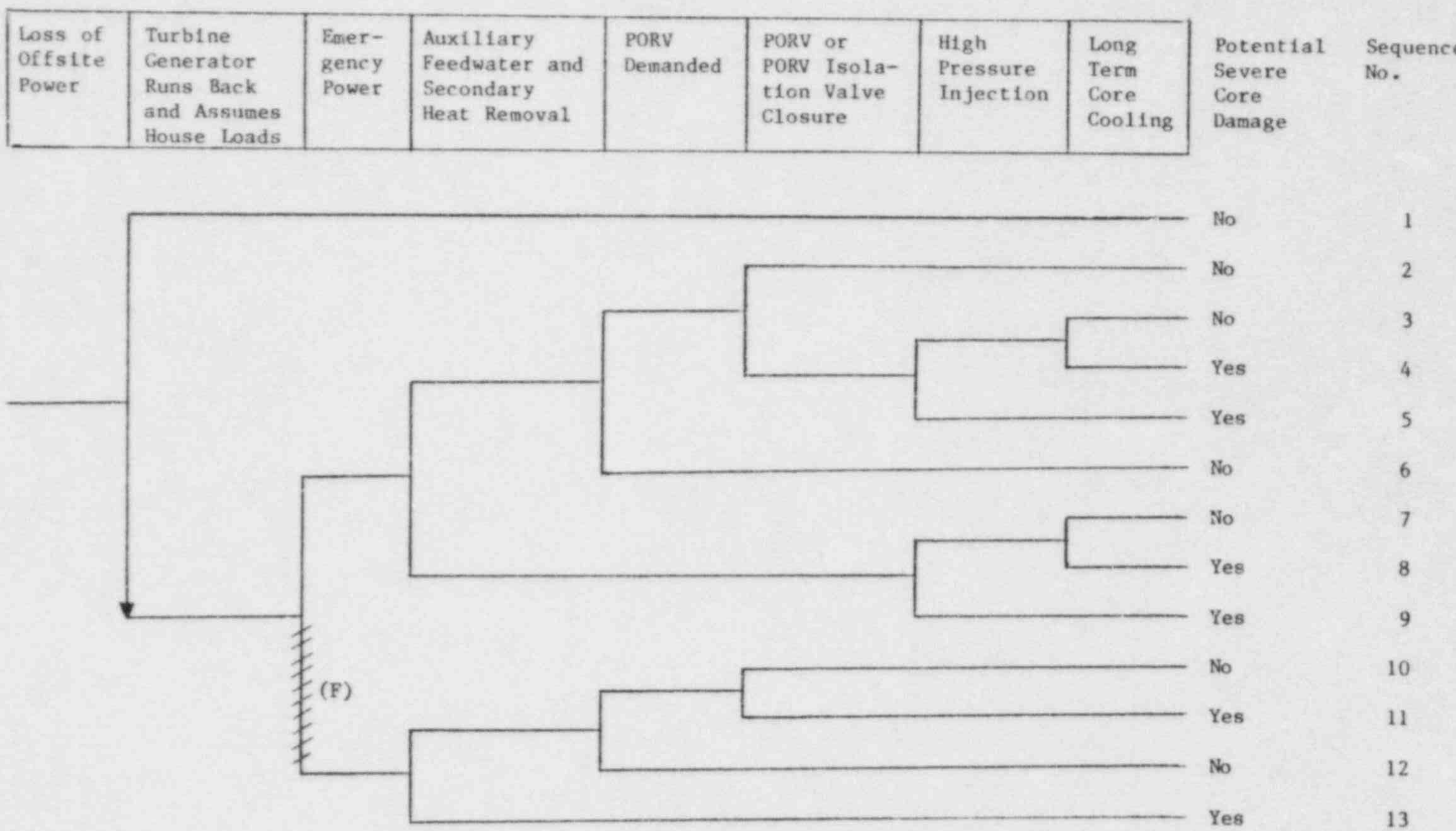
5. The specific circuitry involved in the January 6, 1981, event was to be reviewed to determine if control room indications are required to show when an abnormal lineup exists. Palisades subsequently stated its intent to annunciate in the control room whenever a station battery is disconnected from its bus.

Design purpose of failed system or component:

The batteries provide power to the emergency buses when power is unavailable through the chargers from ac sources. This includes power for connecting the diesel generators to the emergency buses, operation of other circuit breakers, and equipment powered through vital inverters.



NSIC 163356 - Actual Occurrence for Output Breakers for Both Station Batteries Opened at Palisades



NSIC 163356 - Sequence of Interest for Output Breakers for Both Station Batteries Opened at Palisades

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 163356

LER NO.: 81-001

DATE OF LER: January 16, 1981

DATE OF EVENT: January 6, 1981

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Station batteries

CAUSE: Battery breakers inadvertently opened

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Battery breakers inadvertently opened

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 9.6 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power

LOCATION: 5 miles south of South Haven, Michigan

DURATION: 1 hour

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT: On Palisades, DG start is possible without the station batteries; however, DG loading requires batteries, or loading must be performed manually.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 163405

Date: July 28, 1980

Title: Failure of 76 Control Rods to Insert at Browns Ferry 3

The failure sequence was:

1. The reactor was being shut down for maintenance on the feedwater system following steady state operation at 400 MWe (37% power).
2. Power was reduced to 390 MWe by decreasing recirculation flow and inserting 10 control rods.
3. Normal operating procedures at Browns Ferry involve scramming the reactor from about 30% power, so the operator initiated a manual scram.
4. All the west bank rods scrammed full-in except rod 30-23 which settled at position "02."
5. However, due to excess water in the east bank scram discharge headers, 75 out of 88 east bank withdrawn rods failed to fully insert, coming to rest at various positions from 46 to 02.
6. This first scram was sufficient to reduce the reactor power from 36% to less than 2%, but not to subcriticality. Water level was maintained by the feedwater control system.
7. The scram discharge volume was drained for periods of 93 and 53 s respectively and the reactor was manually scrammed two additional times. On each occasion additional rod insertion occurred.
8. The scram discharge volume was allowed to drain again for 160 s.
9. Finally the reactor auto scrammed on high drain volume when the auto scram bypass was removed. The remaining withdrawn control rods then inserted at normal speed to the fully inserted position.
10. The total time that elapsed between the initial scram and final insertion of all the control rods was 14 min.

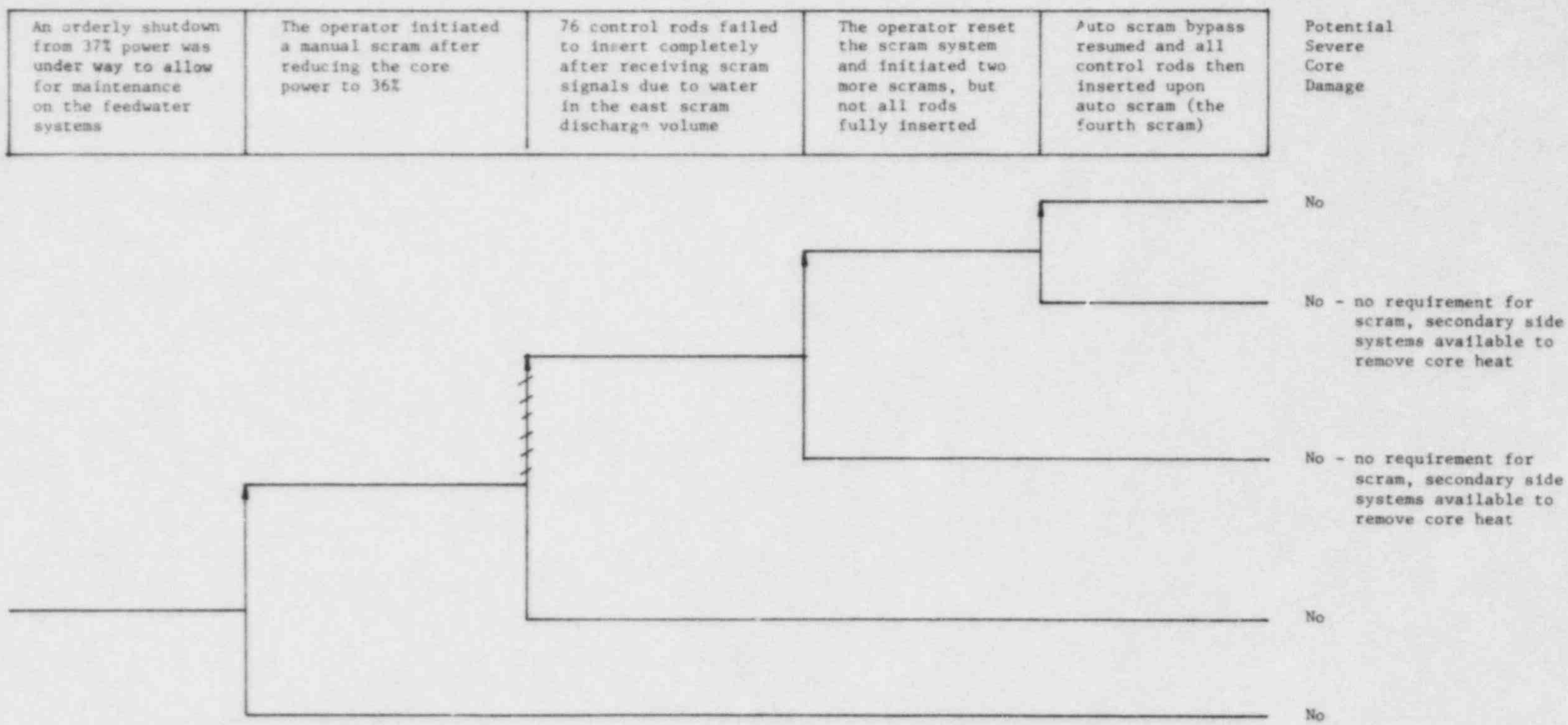
Corrective action

1. The operator manually scrammed all rods. Scram discharge volumes were drained and the bypass was removed from the high-level-drain scram interlock.
2. An investigation was initiated to identify the problem. No problems were identified with the electrical/control system, hydraulic control unit valve lineup, vent and drain piping (i.e., no line blockage found), or seal inleakage to the SDV. The cause for the excess water in the east SDV was not identified.
3. The following measures were to be incorporated promptly to ensure adequate drainage of the scram volume headers:
   a. Monitor the scram discharge headers daily.
   b. Increase the test frequency of the scram discharge volume level switches to once per month.
   c. Develop procedures to verify that the scram discharge volume is free of residual water following each scram and before startup.

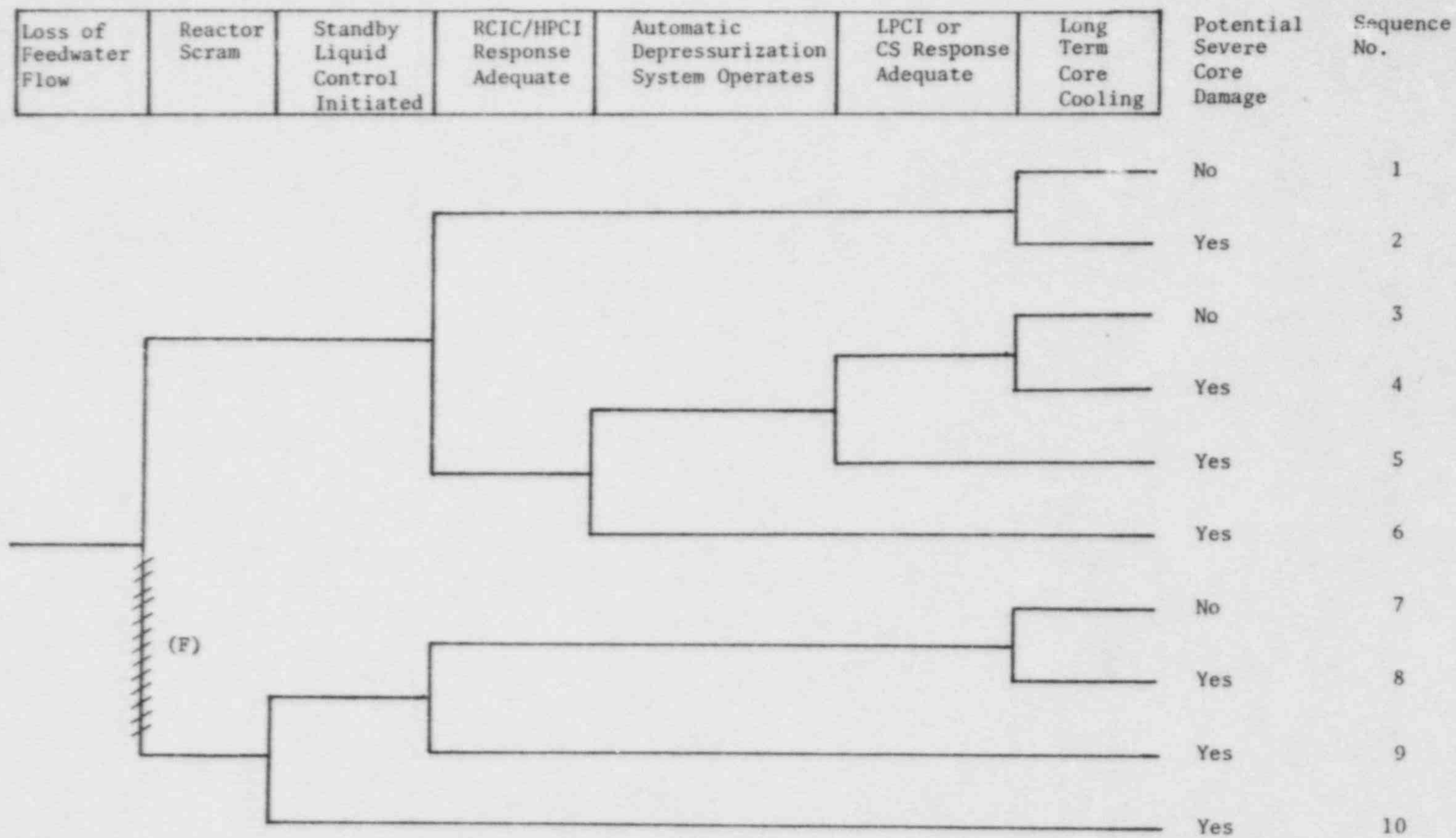
d. Visually inspect each hydraulic control unit daily to ensure that the manual valves are in their normal position.
e. Verify daily the normal position indication of SDV vent valves and drain valves.
f. Additional requirements of IE 80-17 will be implemented in accordance with that document.

Design purpose of failed system or component:

The scram system is designed to rapidly insert control rods for prompt reduction of the reactor power.



NSIC 163405 - Actual Occurrence of Failure of 76 Control Rods to Insert at Browns Ferry 3



NSIC 163405 - Sequence of Interest for Failure of 76 Control Rods to Insert at Browns Ferry 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 163405

LER NO.: 80-024 Rev. 1

DATE OF LER: July 28, 1980

DATE OF EVENT: June 28, 1980

SYSTEM INVOLVED: Reactivity control

COMPONENT INVOLVED: Scram discharge volume

CAUSE: Insufficient drainage from east SDV

SEQUENCE OF INTEREST: Control rods fail to insert

ACTUAL OCCURRENCE: Failure of 76 control rods to insert

REACTOR NAME: Browns Ferry 3

DOCKET NUMBER: 50-296

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: 3.9 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Tennessee Valley Authority

OPERATORS: Tennessee Valley Authority

LOCATION: 10 miles NW of Decatur, Alabama

DURATION: 1/2 of interval from last scram, 240 h (estimated)

PLANT OPERATING CONDITION: 37% power

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Operational event

COMMENT: The analysis done on the control rod pattern that resulted from the scram showed that if the scram had occurred at 100% power, the maximum power after the scram would have been 345 MW(t) [see Nuclear Safety, 23(5), September-October 1982]. The after-scram power level would likely have been even higher if scram had been demanded at 100% power because some or all of the east side control rods (5) that were manually inserted as part of the power reduction procedure could have also failed to fully insert.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 163478

Date: July 10, 1980

Title: HPCI and RCIC Fail to Inject Following Scram at Hatch 1

The failure sequence was:

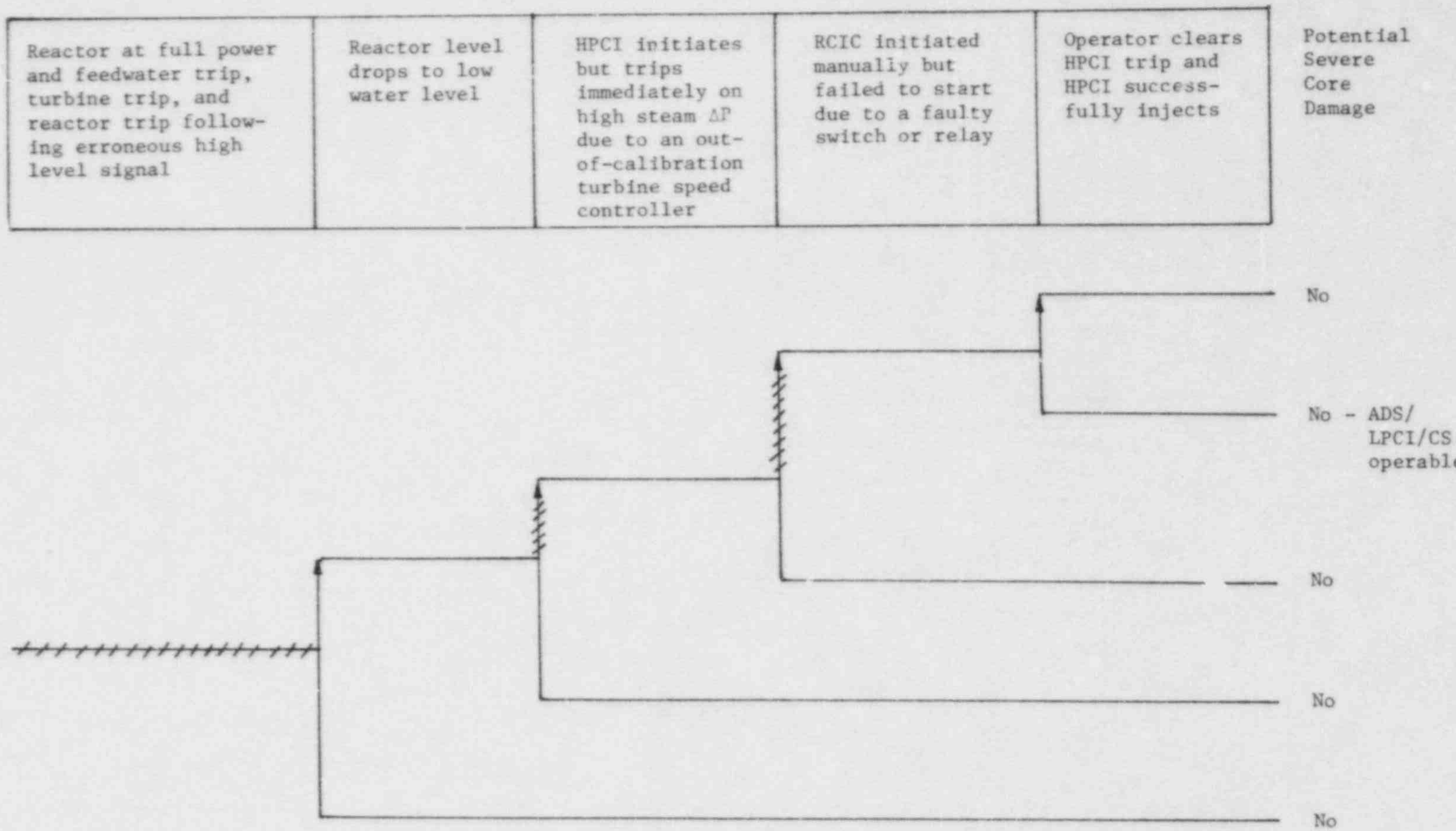
1. At 99.4% power, an erroneous reactor high water level signal initiated tripping of the feedwater pumps and the turbine/generator, which subsequently resulted in a reactor scram.
2. The reactor water level dropped to the low-low level set point.
3. The feedwater pump started but tripped since the MSIVs were closed.
4. On low low level the HPCI system received an auto-initiation signal but failed to inject water into the reactor due to automatic isolation of the HPCI turbine caused by the initial high steam pressure to the HPCI turbine.
5. RCIC was manually initiated during the event, but failed to start and remained inoperable throughout the event.
6. About 5 min after the event began, the operators cleared the HPCI steam supply isolation trip and HPCI auto-initiated successfully and injected water into the core.

Corrective action:

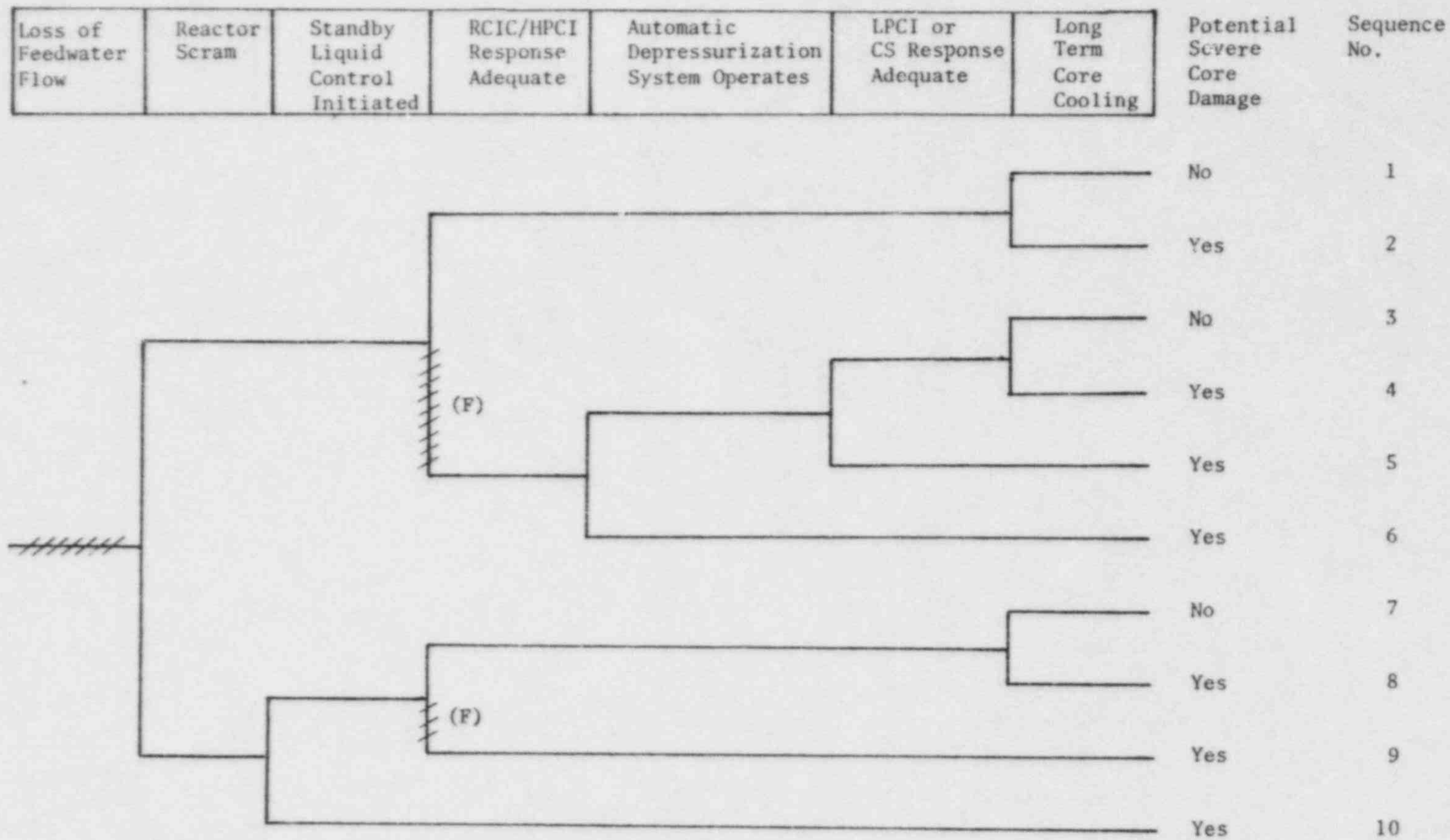
1. The HPCI isolation, due to steam line high pressure differential, was caused by the turbine speed controller being out of calibration. The system was recalibrated and HPCI tested satisfactorily.
2. The RCIC failure, determined to be due to a faulty limit switch and/or relay, was corrected by replacement of both components. The system was then tested satisfactorily.
3. Calibration surveillance and test procedures were being revised to include cold quickstarts of both HPCI and RCIC.

Design purpose of failed system or component:

RCIC provides RCS makeup following reactor trip or loss of feedwater. HPCI provides cooling to the core given a small break LOCA.



NSIC 163478 - Actual Occurrence of HPCI and RCIC Failure to Inject Following Scram at Hatch 1



NSIC 163478 - Sequence of Interest for RCIC and HPCI Failure to Inject Following Scram at Hatch 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 163478

LER NO.: 80-069

DATE OF LER: July 10, 1980

DATE OF EVENT: June 26, 1980

SYSTEM INVOLVED: HPCI and RCIC systems

COMPONENT INVOLVED: HPCI speed controller; RCIC trip switch and relay

CAUSE: Failed components, improper calibrating procedures

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: HPCI and RCIC fail to inject following scram

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 777 MWe

REACTOR AGE: 5.8 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Georgia Power Co.

LOCATION: 11 miles north of Baxley, Georgia

DURATION: N/A

PLANT OPERATING CONDITION: Power not specified prior to scram

TYPE OF FAILURE: Failed to start

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 163499

Date: July 17, 1980

Title: Reactor Coolant Pump Seal Failure at Arkansas Nuclear 1

The failure sequence was:

1. With the reactor at 86% power, reactor coolant pump (RCP) "C" seal failed, resulting in excessive RCS leakage to the containment.
2. A controlled power reduction was begun, and approximately one-half hour later letdown was secured to reduce RCS inventory loss. RCS leak was estimated to be 10-20 gpm.
3. RCS leak rate increased during the power reduction and the plant was subsequently rapidly taken off line. RCP "C" was tripped after the turbine was taken off line but with the reactor critical.
4. RCS leak rate increased substantially when RCP "C" was tripped, and the RCP "C" lift pumps were started and stopped four times in succession in an attempt to reduce the leak rate. On the fourth attempt a reduction in leak rate was noticed. RCS leak rate had increased to a maximum of approximately 350 gpm.
5. The reactor was manually tripped and HPI pumps B and C started and all HPI valves opened to provide RCS makeup. The RCP "C" seal return line was isolated to prevent inventory loss through that line and RCP seal flow increased to quench the steam/water leaking by the failed seal.
6. A one-half psi increase in containment pressure occurred and the reactor building emergency coolers were put in service to minimize the pressure increase.
7. One HPI pump was secured and the HPI valves closed 1.3 hours after the seal failure. Two HPI pumps were used to provide continued RCS makeup from the BWST.
8. Individual SLBIC trains were inadvertently initiated twice during the cooldown, resulting in start of the turbine-driven EFW pump. This pump was subsequently stopped and the auxiliary feedwater pump lined up to feed the steam generators.
9. During the RCS cooldown, containment entry was required to isolate the two core flood tanks to prevent their discharging into the RCS below 600 psig. A decrease in core flood tank level of 18 in. and 12 in. occurred prior to effecting isolation.
10. Throughout the incident a greater than 100°F margin to saturation existed. Approximately 60,000 gallons of water collected in containment.

Corrective action:

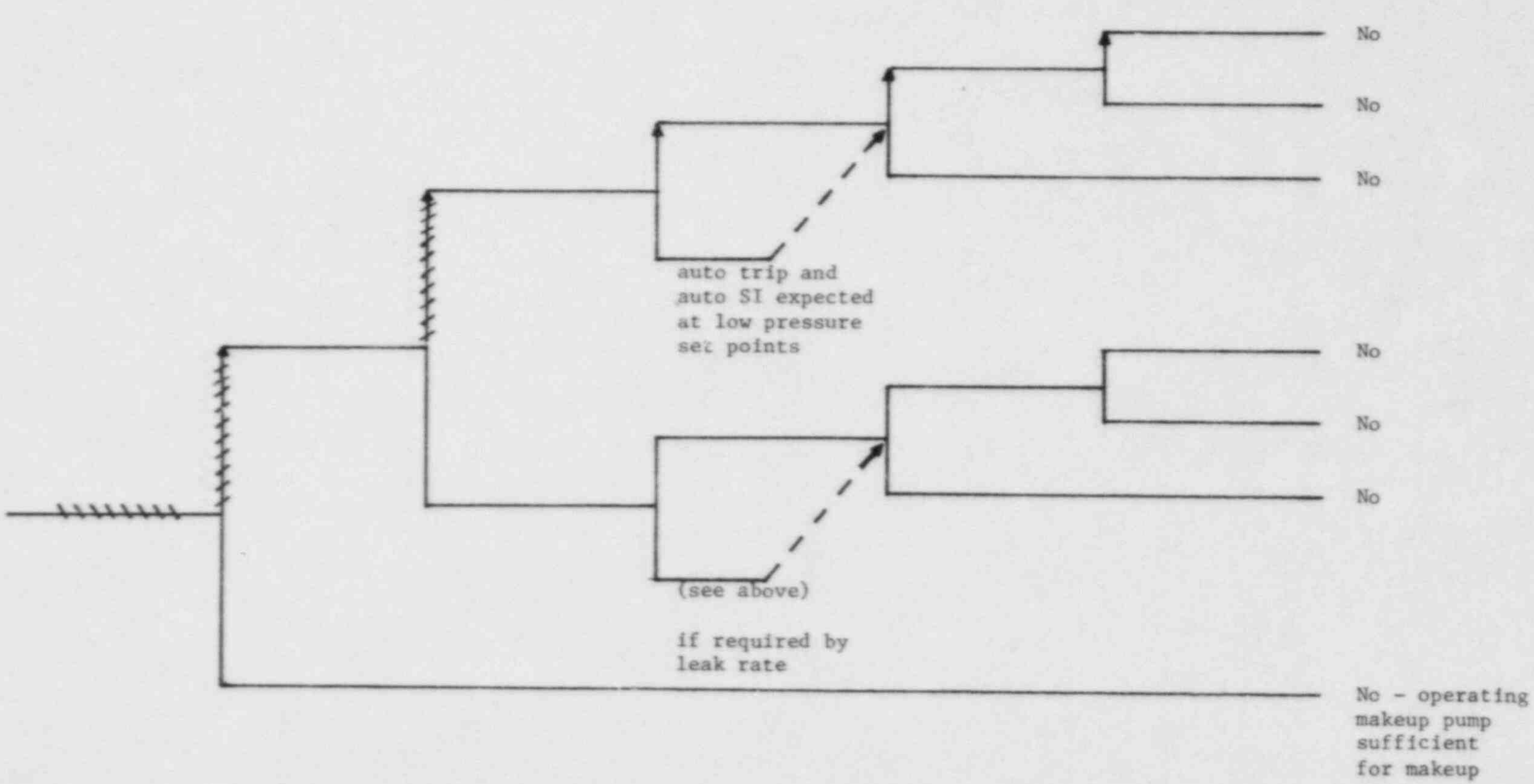
1. The failed seal was examined and extensive damage observed. The cause of the damage could not be identified with certainty.

2. All remaining RCP seals were inspected and evidence of high temperature operation observed. All seals were replaced.
3. The CFT isolation valve controls were relocated outside containment to facilitate access.

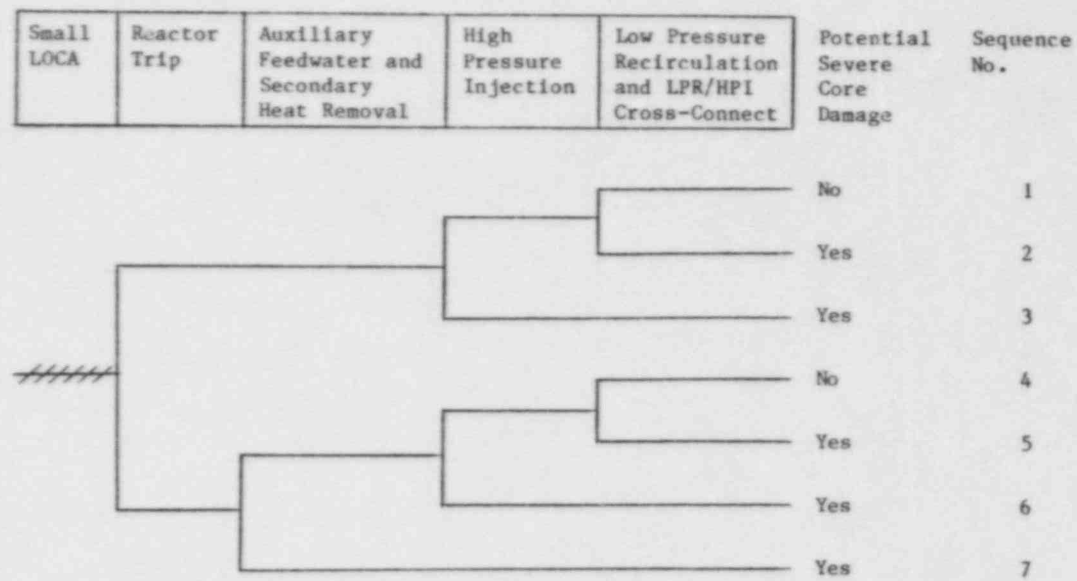
Design purpose of failed system or component:

The RCP seals provide a pressure boundary against RC system pressure at the rotating shaft of the pump.

Reactor at 86% power and RCP "C" seal failure (10-20 gpm initial leak rate)	Leakage rate increases during power reduction - plant taken off line and RCP "C" tripped	RCP leak rate increases to 350 gpm. RCP "C" lift pumps operated, re-sulting in some reduction in leak rate	Reactor trip and manual HPI initiation to provide RCS makeup	One HPI pump secured and HPI valves closed. Two HPI pumps provide makeup	Core flood isolation valves closed to prevent CFT discharge at RCS pres-sure of 600 psig	Potential Severe Core Damage



NSIC 163499 - Actual Occurrence for Reactor Coolant Pump Seal Failure at Arkansas Nuclear 1



NSIC 163499 - Sequence of Interest for Reactor Coolant Pump Seal Failure at Arkansas Nuclear 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 163499

LER NO.: 80-015 Rev. 1

DATE OF LER: July 17, 1980

DATE OF EVENT: May 10, 1980

SYSTEM INVOLVED: Reactor coolant system

COMPONENT INVOLVED: Reactor coolant pump seals

CAUSE: Seal failure

SEQUENCE OF INTEREST: Small break LOCA

ACTUAL OCCURRENCE: RCP "C" seal failure

REACTOR NAME: Arkansas Nuclear 1

DOCKET NUMBER: 50-313

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 850 MWe

REACTOR AGE: 5.8 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power & Light

LOCATION: 6 miles NW of Russellville, Arkansas

DURATION: N/A

PLANT OPERATING CONDITION: 86% power

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Operational event

COMMENT: See also: NSIC 165417 Arkansas Nuclear 1, 50-313, LER 80-015R, April 13, 1981, and Nuclear Safety, Vol. 22, No. 2, March-April 1981, pp. 237-238.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 164149

Date: February 12, 1981

Title: Isolable Small Break LOCA at Robinson 2

<u>The failure sequence was:</u>

1. With the plant at 100%, the "A" electrohydraulic control oil pump developed a seal leak and a plant shutdown was begun. The "B" EHC pump was already out of service because of vibration problems.
2. The "B" feedwater and condensate pumps were stopped due to erratic feedwater pump behavior.
3. Immediately following the opening of the generator output breakers (6% power), the turbine governor valve spiked open (apparently due to the electrohydraulic problems) and generated a momentary high steam flow signal. This, in combination with an existing low T signal resulted in train B safequards initiation. (Train A did not initiate nor did the MSIVs close, apparently because of the short duration of the high steam flow signal). The SI signal tripped the reactor.
4. The "A" SI train was manually actuated and the MSIVs were closed.
5. The "A" containment fire alarm was received shortly after the SI actuation.
6. During the automatic isolation of the letdown line on safety injection, relief valve CVC-RV-203 bellows ruptured, either because of the relatively slower closure of valves upstream of the relief valve compared with those downstream of it, or because of leakage past the upstream valves. In addition, a pressure surge due to the isolation valves closing caused a drain cap on a partially open drain valve to be blown off.
7. Having determined that a spurious SI had occurred and unaware of the above failure, the operators reset SI and feedwater isolation and restored letdown.
8. Containment pressure and dewpoint increased and RCS pressure decreased. Letdown was secured approximately 15 min later. A containment entry was made in an attempt to determine the leakage path. Approximately 3000 gallons of water were in the containment sump at that time.
9. After letdown was isolated, pressurizer pressure continued to decrease. A second safety injection occurred on low pressure. Both trains of safeguards equipment actuated.
10. Four hours after the first containment entry a second entry was made and the leaking drain line identified. The two upstream level control valves were leaking at approximately 5-7 gpm. The drain valve was closed.

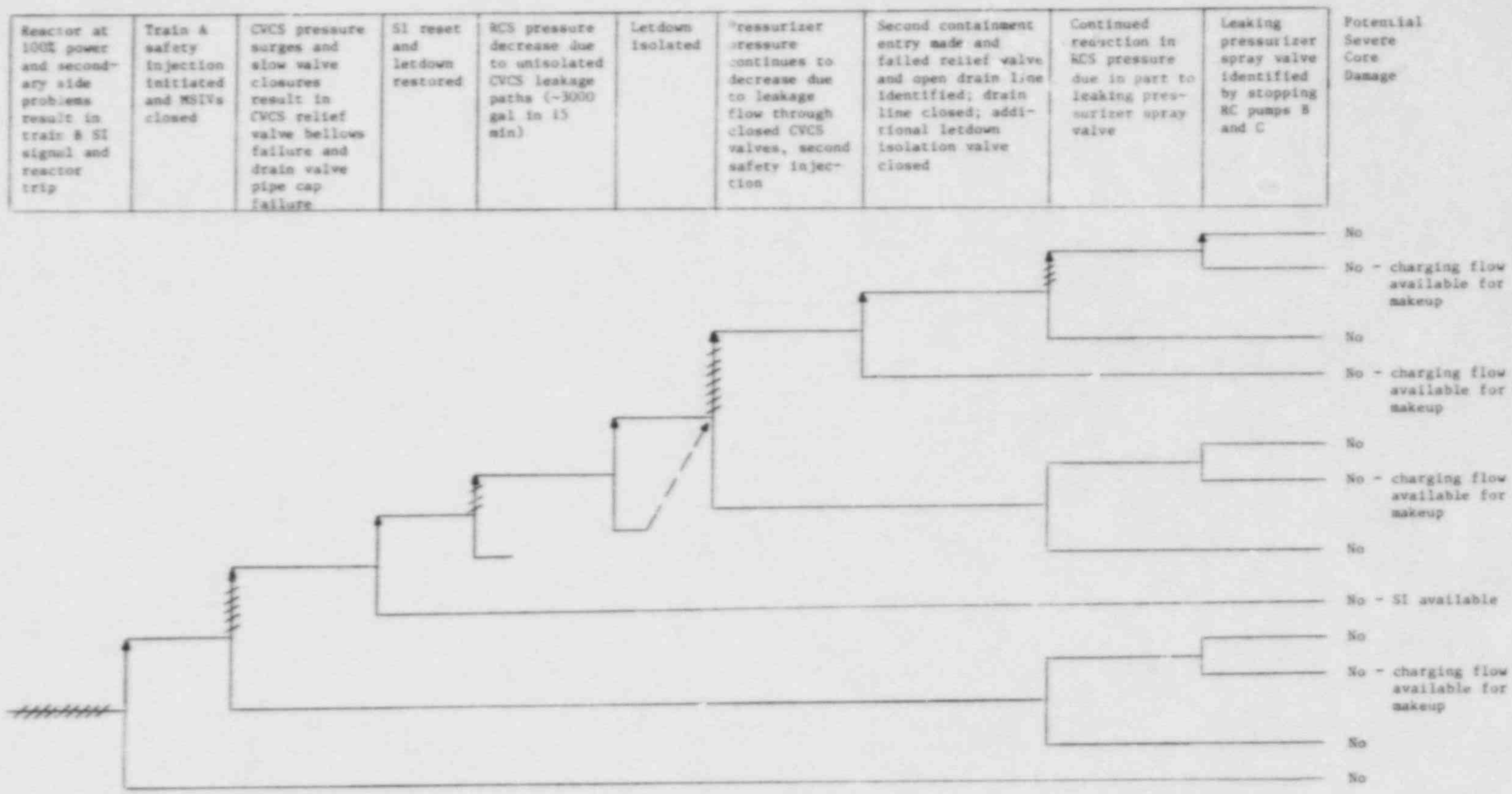
11. After the drain valve was closed RCS pressure still continued to decrease in part as a result of a partially opened pressurizer spray valve. (The pressurizer spray valve position is indicated by demand in lieu of stem position, which delayed identification of the cause of the depressurization.)
12. The leak rate could not be accurately determined but it was estimated to have been approximately 100 gpm while letdown was unisolated. A total of about 4500 to 6000 gallons of water was leaked to the containment sump during the event.

Corrective action:

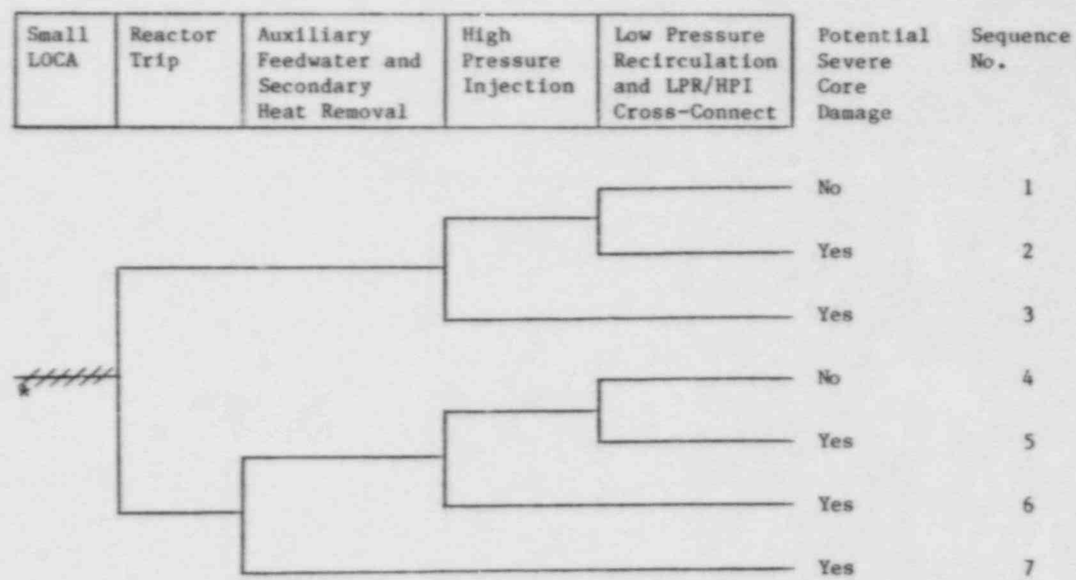
1. The leaking drain valve was closed and a new pipe cap installed. All other similar valve/pipe cap combinations were verified closed.
2. Corrective action concerning the failed relief valve was not identified.

Design purpose of failed system or component:

The reactor coolant system transfers heat generated in the core to the steam generators. The letdown line provides a means of removing reactor coolant from the RCS for boron concentration changes and for purification.



NSIC 164149 - Actual Occurrence for Isolable Small Break LOCA at Robinson 2



NSIC 164149 - Sequence of Interest for Isolable Small Break LOCA at Robinson 2

*auto isolation on safety injection closure of upstream valves

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 164149

LER NO.: 81-005

DATE OF LER: February 12, 1981

DATE OF EVENT: January 29, 1981

SYSTEM INVOLVED: Containment chemical volume control system

COMPONENT INVOLVED: Relief valve, drain valve, pipe cap, pressurizer spray valve

CAUSE: Bellows rupture, drain valve vibrated open, pipe cap blew off, leaking spray valve

SEQUENCE OF INTEREST: Small break LOCA

ACTUAL OCCURRENCE: Small break LOCA

REACTOR NAME: H. B. Robinson 2

DOCKET NUMBER: 50-261

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 700 MWe

REACTOR AGE: 10.4 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Carolina Power & Light

LOCATION: 5 miles NW of Hartsville, South Carolina

DURATION: N/A

PLANT OPERATING CONDITION: 100% power (6% at time of trip)

TYPE OF FAILURE: Small LOCA

DISCOVERY METHOD: Operational event

COMMENT: Additional information: "Engineering Evaluation of the H. B. Robinson Reactor Coolant System leak on January 29, 1981," Office for Analysis and Evaluation of Operational Data, March 23, 1981.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 164453

Date: February 2, 1981

Title: Brief Loss of Offsite Power and Degraded Load Shed Capability at San Onofre 1

The failure sequence was:

1. With the reactor in cold shutdown, an operator intended to transfer kV buses 1-C and 2-C from auxiliary transformer C to auxiliary transformers A and B.
2. The operator opened the bus tie breaker by mistake instead of the auxiliary transformer C bus supply breakers.
3. Buses 1-C and 2-C de-energized as auxiliary transformer C de-energized, de-energizing CVCS, RHR, CCW and SWC pumps.
4. Both diesel generators started and were available for loading.
5. Bus 1C source breaker 11C02 did not open to shed loads on bus 1C because dc bus control power to the undervoltage logic scheme was deenergized due to a mispositioned switch.

Corrective action:

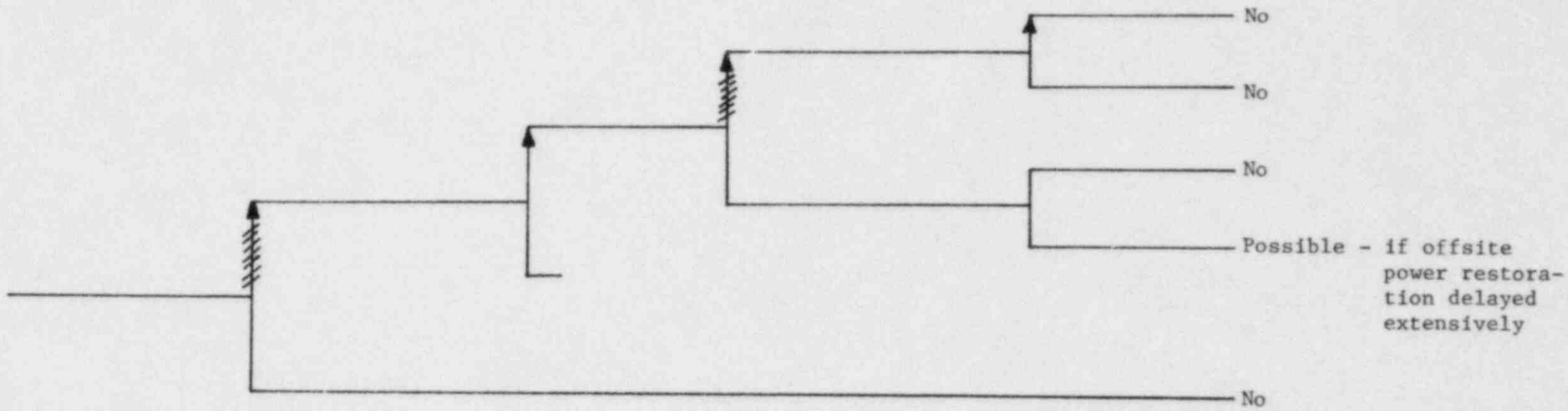
Power was restored by reenergizing auxiliary transformer C within seconds. The station was to implement administrative controls on safety-related dc control circuits prior to the end of the refueling outage.

Design purpose of failed system or component:

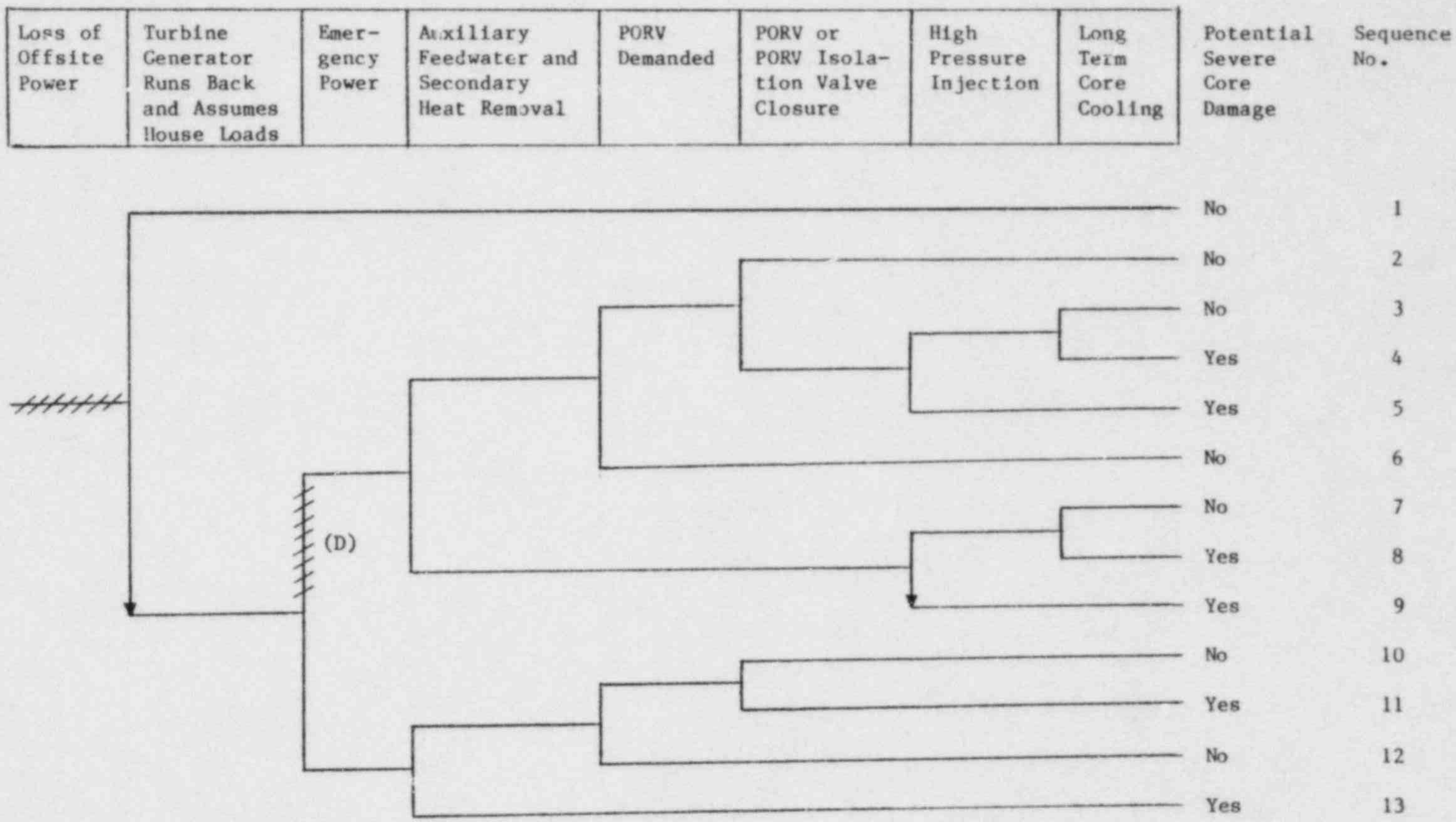
Offsite power provides the preferred source of power to safety-related loads when the unit generator is not available.

Load shedding is provided to strip safety-related buses of all large loads prior to initiation of the diesel loading sequence. This prevents overloading of the diesel due to starting surges.

Reactor in shutdown and operator intends to transfer 1-C and 2-C **buses** from auxiliary transformer C to auxiliary transformers A and B	Operator error results in opening of bus tie breaker in lieu of transformer C bus supply breakers	Loss of power to **buses** 1-C and 2-C	Diesel generators start and are available for loading. Bus 1C source breaker fails to open due to unavailable undervoltage scheme	Offsite power restored	Potential Severe Core Damage



NSIC 164453 - Actual Occurrence for Brief Loss of Offsite and Degraded Load Shed Capability in San Onofre 1



NSIC 164453 - Sequence of Interest for Brief Loss of Offsite Power and Degraded Load Shed at San Onofre 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 164453

LER NO.: 80-038 Rev. 1

DATE OF LER: February 2, 1981

DATE OF EVENT: November 22, 1980

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: 4 kV bus circuit breakers

CAUSE: Operator error

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP and degraded load shed capability

REACTOR NAME: San Onofre 1

DOCKET NUMBER: 50-206

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 436 MWe

REACTOR AGE: 13.4 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Southern California Edison

LOCATION: 5 miles south of San Clemente, California

DURATION: N/A

PLANT OPERATING CONDITION: Cold shutdown

TYPE OF FAILURE: Failed to start

DISCOVERY METHOD: Operational event

COMMENT: The utility believes the dc control switch was mispositioned during the current refueling outage, although this cannot be confirmed. See also NSIC No. 161910 (San Onofre, 50-206, LER 80-038, Dec. 9, 1980).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 164617

Date: January 30, 1981

Title: Partial Loss of dc Power and Diesel Generator Trip at Millstone 2

The failure sequence was:

1. With the reactor at 100% power, 125 volt dc emergency bus A was deenergized when the main feeder breaker connecting the battery and its charger outputs to this bus was inadvertently opened by a plant equipment operator intending to take ground readings on the bus.
2. The deenergization of the bus resulted in the removal of control power to the reactor trip breakers, causing a reactor scram. All control room annunciators were lost.
3. The turbine trip which normally follows a reactor trip did not occur because of the unavailability of dc bus A.
4. The A diesel generator started on loss of control power to the air-start solenoids. (Because of the loss of control power, the diesel would not have auto-closed onto its bus, if required.)

Time Approximately 30 Seconds

5. Turbine was manually tripped.
6. The fast transferring of the in-house loads from the normal station service transformer (NSST) to the reserve station service transformer (RSST) which normally follows a turbine trip did not occur because the transfer logic is powered from dc system A.
7. The failure of the fast transfer left open the two breakers through which offsite power is fed to the "B" 6.9 kV bus and 4.16 kV emergency bus. This resulted in the loss of offsite power to the "B" buses.
8. The loss of offsite power to the "B" 4.16 kV emergency bus resulted in the starting and loading of the B diesel generator.
9. Since the two breakers through which power is fed to the "A" 6.9 kV bus and 4.16 kV emergency bus did not operate because dc control power was not available, offsite power remained available to the "A" train via the normal transformer.
10. The automatic opening of the main generator switchyard breakers which normally follows a turbine trip did not occur because the initiating signal to open the breakers could not be generated as a result of the loss of dc system A. The main generator started to motor.

Time Approximately 50 Seconds

11. The "A" 125 volt dc bus was reenergized by closing the main feeder breaker.
12. With dc control available, the "A" 4.16 kV emergency bus and 6.9 kV bus were transferred from the normal to the reserve transformer.

13. The "A" diesel generator shut down automatically as a result of a design feature which trips the diesel generator when dc control power is restored. (Local reset required prior to restart.)
14. Upon restoration of dc power to system A, the main steam isolation valves closed, tripping the main feedwater pumps. The auxiliary feedwater pumps were started and water was supplied to both steam generators.
15. Upon restoration of dc power, the "B" 6.9 kV bus was connected to the reserve transformer. This connection was immediately lost due to an overcurrent condition caused by attempting to start all the loads in the bus at the same time. The supply breaker from the reserve transformer to the "B" 4.16 kV emergency bus could not be closed because the breaker was locked-out when the offsite was previously lost.
16. The generator output breakers in the switchyard were opened by the reverse power time delay relay, isolating the main generator from the 345 kV switchyard.

Time 10 Minutes

17. The "B" diesel generator tripped automatically because of a service water flange leak (a result of improper gasket compression during a recent modification) which sprayed the electronic governor and caused the diesel generator to trip. Because of this, the "B" 4.16 kV emergency bus was deenergized.
18. The load shed signal was overridden and the "B" 4.16 kV emergency bus was reenergized from the reserve transformer.
19. Numerous instruments powered from the "B" main were not available as a result of blown fuses caused by increased currents to inductive loads during the diesel trip.
20. Difficulty was experienced with RCS pressure control following restoration of power. The existing combination of two reactor coolant pumps provided no significant pressurizer spray flow. The operator did not associate the ineffectiveness of spray with greatly reduced spray control but rather concluded that a "hard bubble" had resulted from a collection of noncondensable gases in the pressurizer. Approximately 2 h and 15 min into the event, pressurizer pressure increased to 2380 psia, causing both power operated relief valves to open for a short duration. The auxiliary spray valve was subsequently used to control pressure.

Corrective action:

1. The emergency procedure for loss of a dc bus was revised to reflect information gained during the event.
2. Circuit modifications were made to prevent the shutdown of a running diesel upon reapplication of dc power, following a complete loss of dc to all control circuits, and to the fail-to-start feature, the latter to preclude the possibility of forced shutdown via operation of the shutdown relay in the event a diesel which starts on an autostart signal does not achieve a speed of 250 rpm in 12 seconds.
3. Instrument loops were to be protected with manufacturer-recommended slow-blow fuses instead of the currently installed quick-blow fuses.

4. The utility proposed redundant power supplies for the control room annunciator system.
5. The diesel generator flange gasket was replaced.

Design purpose of failed system o component:

1. The diesel generator provides back-up power to safety-related loads when the main generator and offsite power sources are unavailable.
2. The dc buses provide control power for all breakers, input power for vital inverters, as well as providing power to a variety of uninterruptible loads.
3. The control room annunciator provides status indication for numerous plant parameters and system operating conditions.

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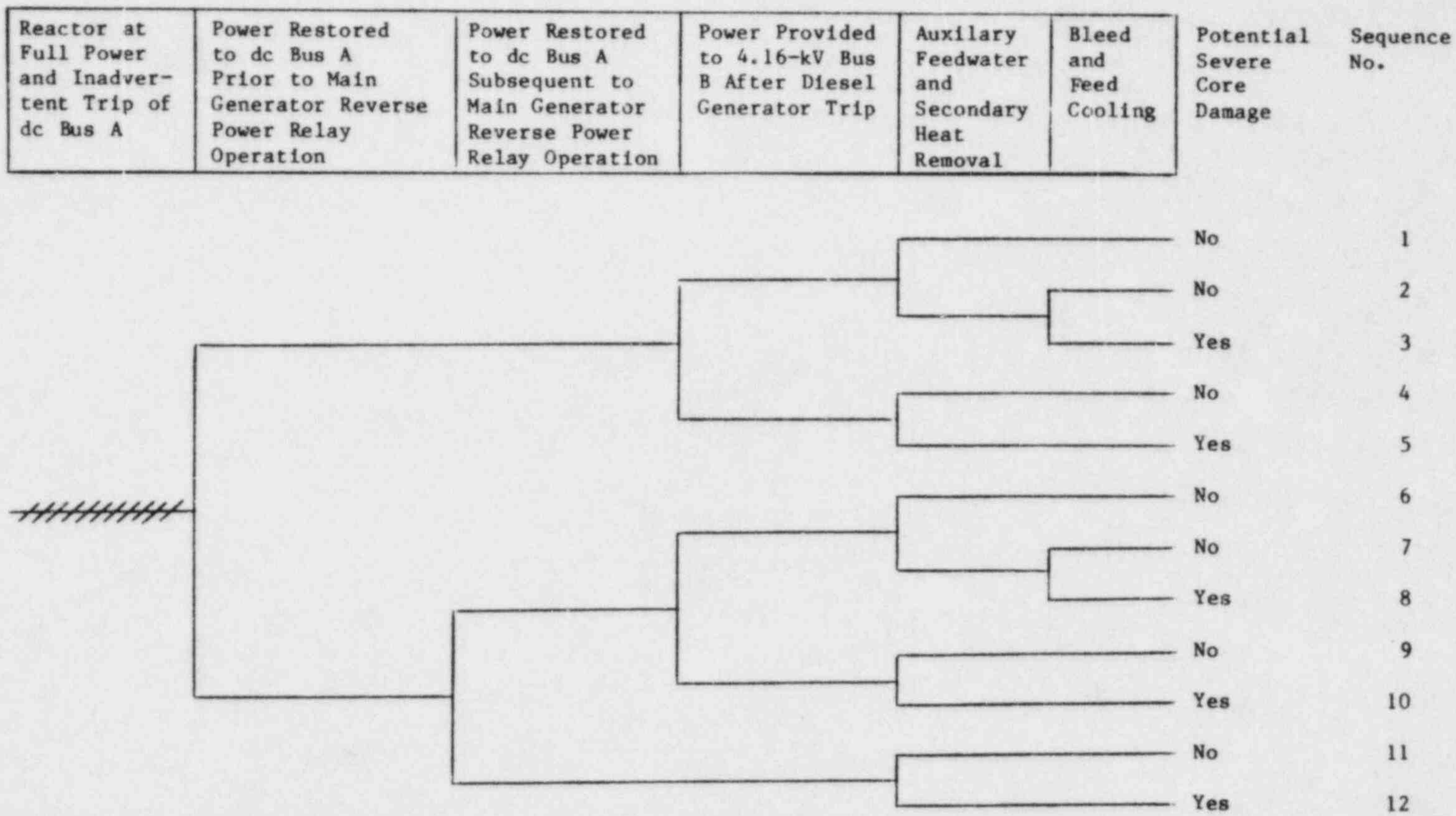


NSIC 164617 - Sequence of Interest for Loss of dc Bus and Diesel Generator Trip at Millstone 2

CAT GORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 164617

LER NO.: 81-005

DATE OF LER: January 30, 1981

DATE OF EVENT: January 2, 1981

SYSTEM INVOLVED: Offsite power, emergency 4.16 kV and dc power

COMPONENT INVOLVED: 125 V dc bus, diesel generator

CAUSE: Inadvertent trip of dc bus and diesel generator trip due to leaking service water flange

SEQUENCE OF INTEREST: Loss of dc bus

ACTUAL OCCURRENCE: Loss of dc bus and diesel generator trip

REACTOR NAME: Millstone 2

DOCKET NUMBER: 50-336

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 870 MWe

REACTOR AGE: 5.2 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Northeast Nuclear Energy Company

LOCATION: 5 miles SW of New London, Connecticut

DURATION: N/A

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Operational event

COMMENT: Had the "A" dc bus not been restored prior to completion of the main generator reverse power time delay relay operation (dc power was restored at ~50 seconds, the relay operated at ~60 seconds), then a station blackout would have occurred when the "B" diesel generator tripped. Power could have been made available to affected buses by manual operation of breakers. See also NRC Operating Reactor Event Memorandum No. 81-31: Loss of Direct Current (DC) Bus at Millstone Unit 2, March 31, 1981.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 164703

Date: May 18, 1981

Title: Loss of offsite power at La Crosse

The failure sequence was:

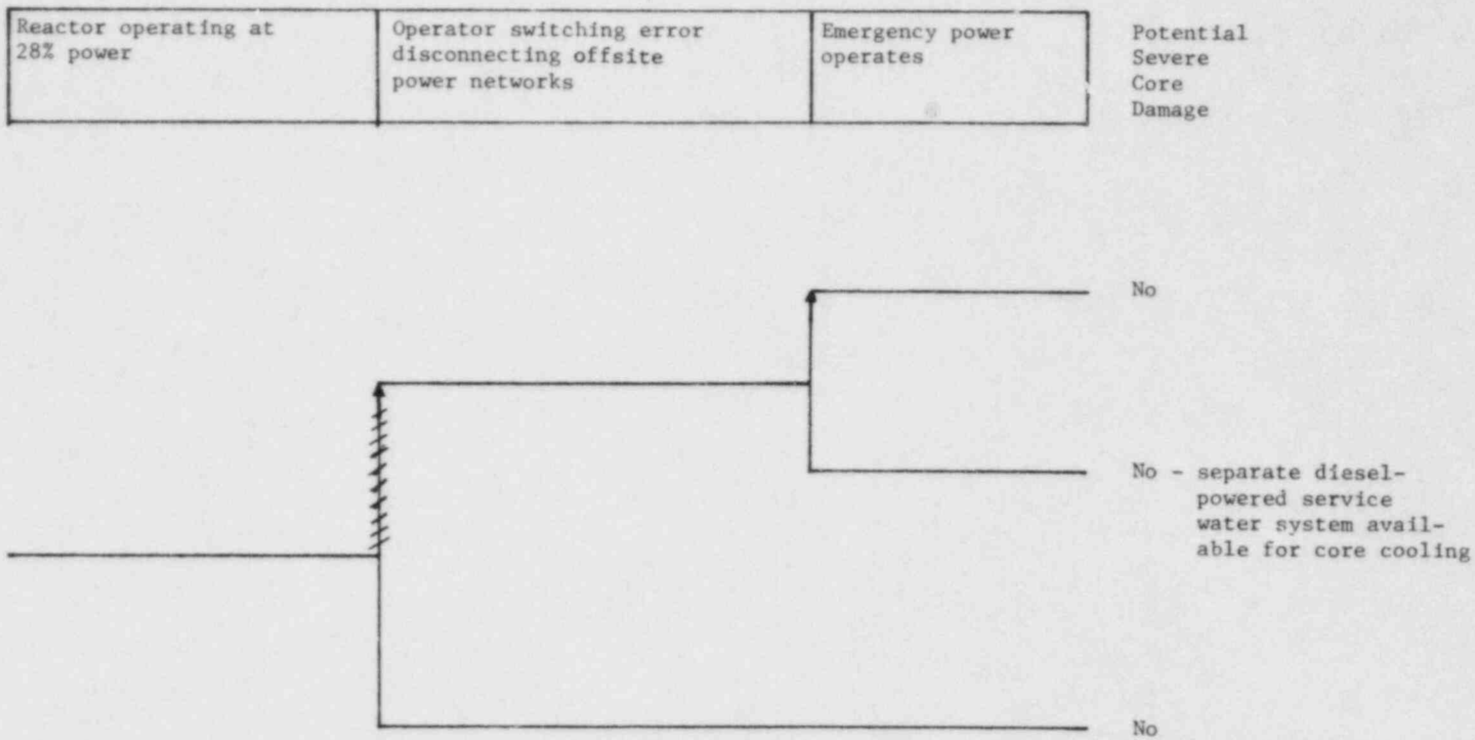
1. The reactor was operating at 28% power.
2. The switchyard operator was directed to open a switch in response to another situation, but he opened the wrong switch, disconnecting the auxiliary transformer from the 69 kV transmission line. This resulted in a loss of all offsite power.
3. The reactor scrammed and the diesel generators started automatically and supplied the required vital loads.

Corrective action:

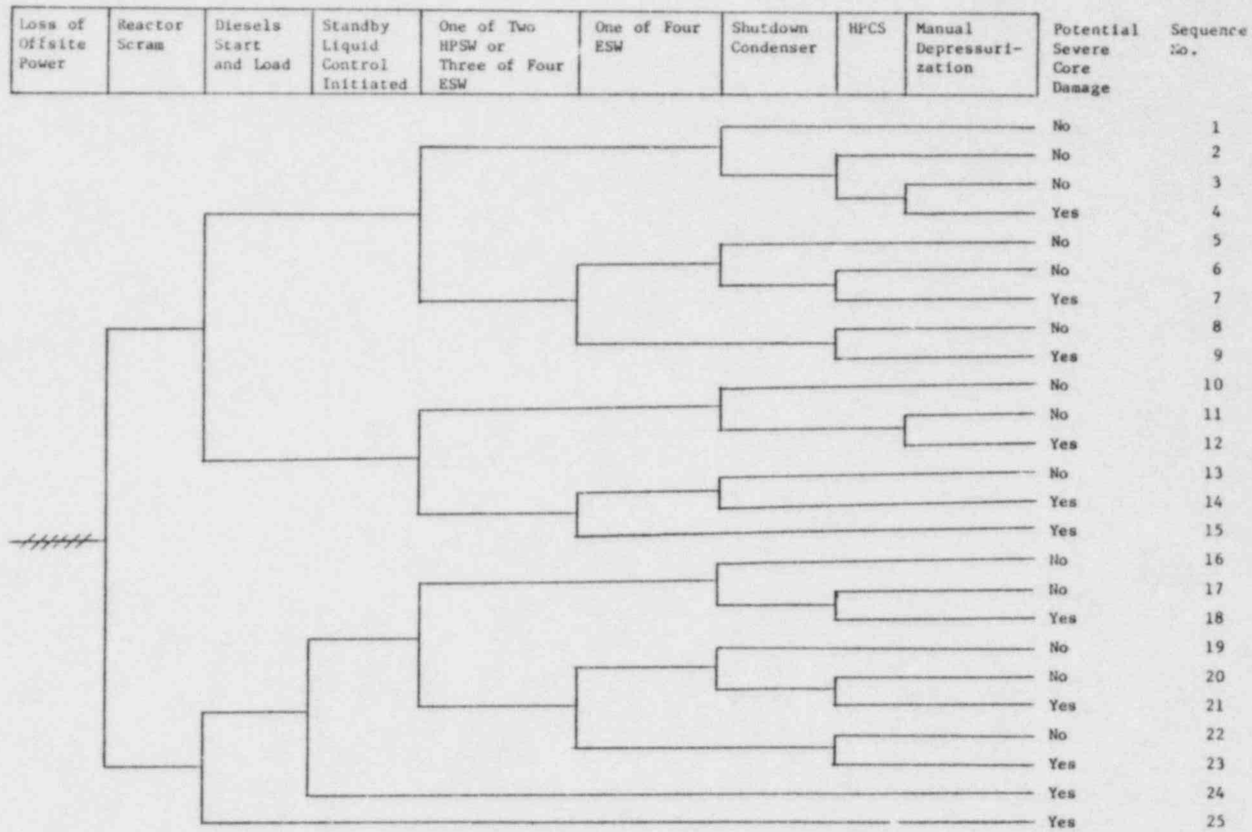
1. Offsite power was restored within 14 min.
2. Practices were revised to require outside switching order to be executed in the presence of a second person whenever possible.
3. Additional practical training was to be provided on the switchyard for all operators.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to plant loads when the unit generator is unavailable.



NSIC 164703 - Actual Occurrence for Loop at LaCrosse



NSIC 164703 - Sequence of Interest for LOOP at La Crosse

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 164703

LER NO.: 81-002 Rev. 1

DATE OF LER: May 18, 1981

DATE OF EVENT: February 1, 1981

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Switch

CAUSE: Operator error

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP

REACTOR NAME: La Crosse

DOCKET NUMBER: 50-409

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 50 MWe

REACTOR AGE: 13.6 years

VENDOR: Allis-Chalmers

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Dairyland Power Cooperative

LOCATION: 19 miles south of LaCrosse, Wisconsin

DURATION: 14 minutes

PLANT OPERATING CONDITION: 28% power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 164955

Date: March 26, 1981

Title: Loss of HPCI and RCIC Systems at Hatch 1

The failure sequence was:

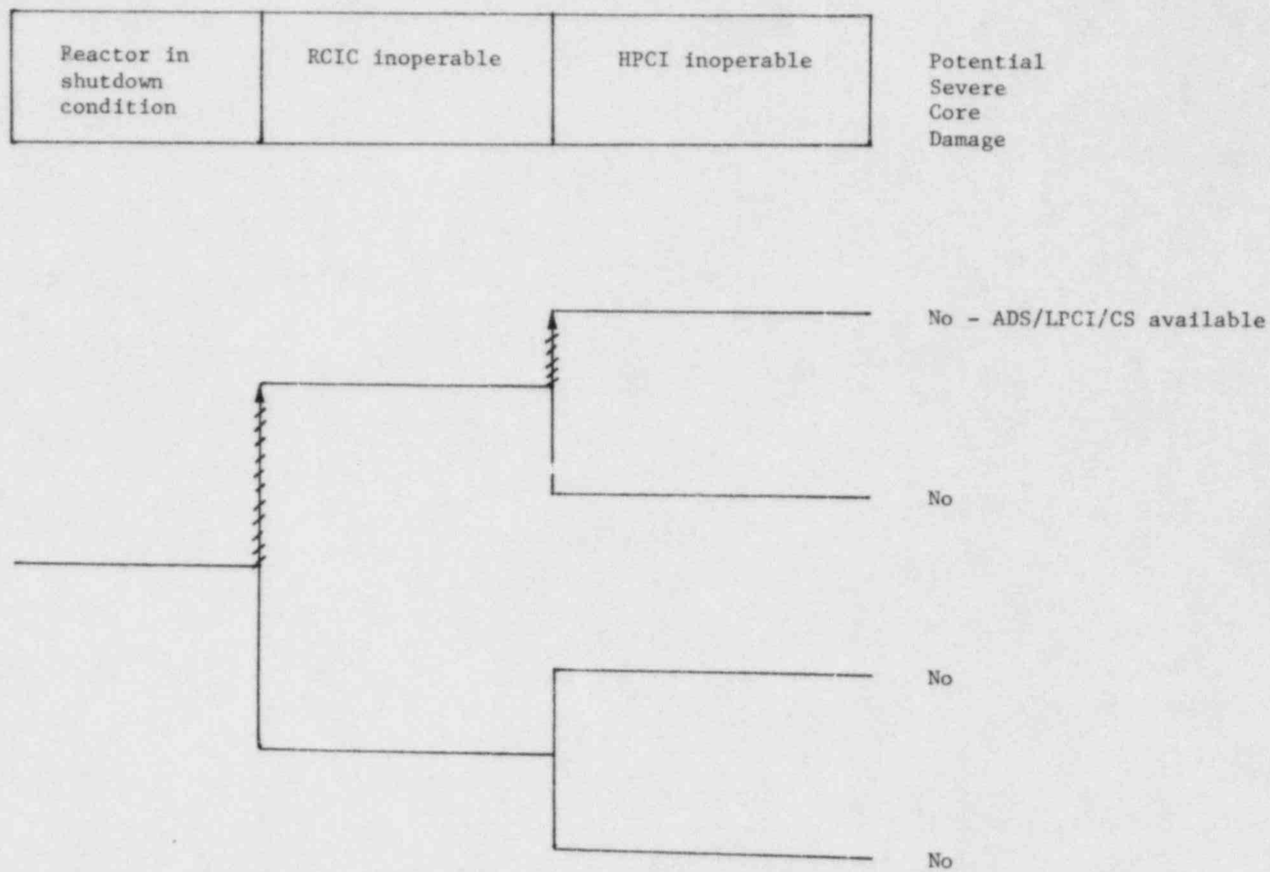
1. With the reactor in shutdown, RCIC had been removed from service to perform an overspread trip test which failed.
2. A HPCI system oil line had been stepped on causing damage to the line.
3. During a HPCI test, the HPCI turbine vibration caused the damaged oil line to break.
4. HPCI was declared inoperable.

Corrective action:

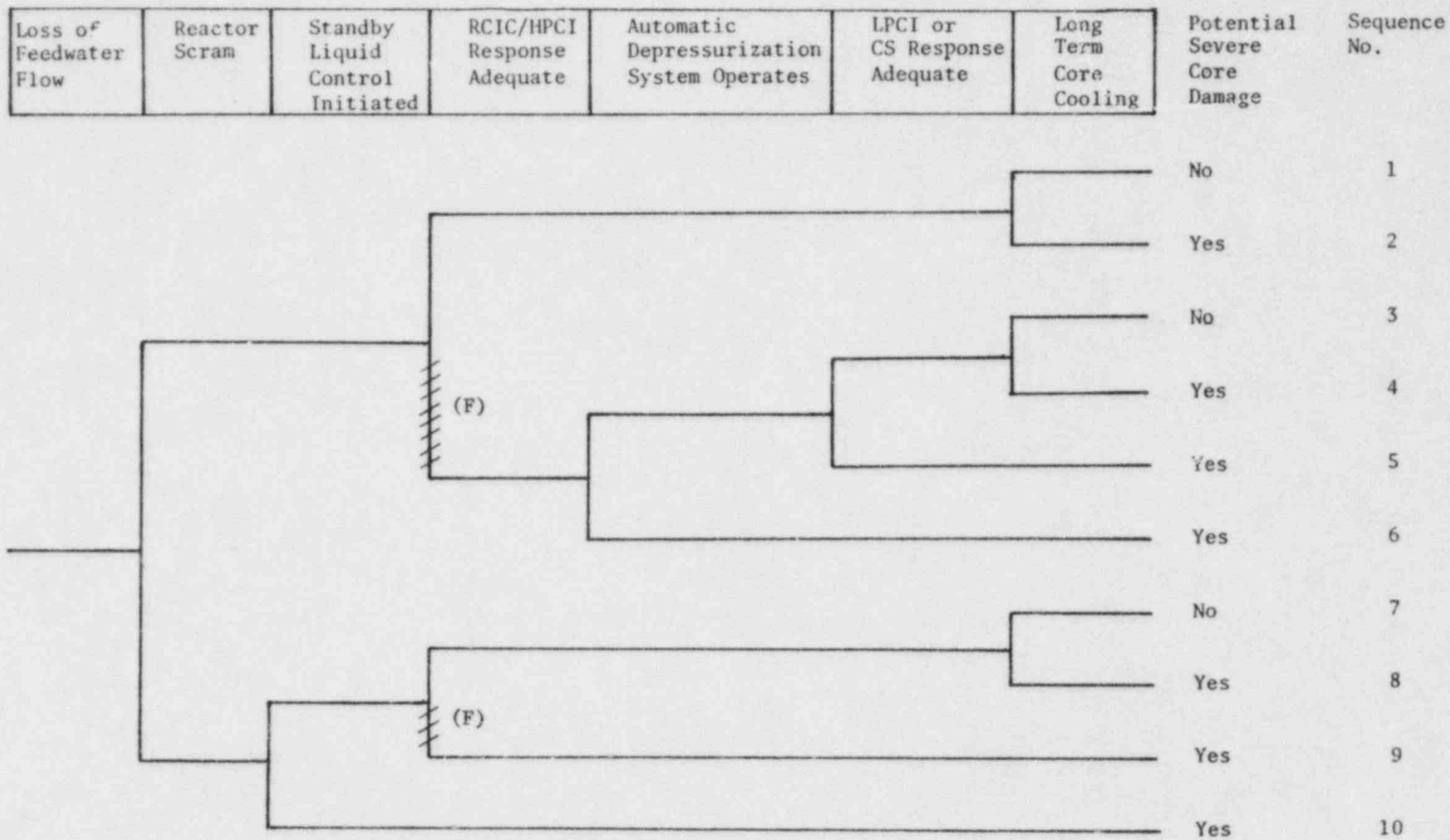
1. The line was repaired and the HPCI test performed satisfactorily.
2. Remaining HPCI and RCIC oil lines were inspected for damage.
3. Personnel were cautioned not to step on oil lines.

Design purpose of failed system or component:

HPCI and RCIC provide high pressure core cooling in the event normal cooling is lost.



NSIC 164955 - Actual Occurrence for Loss of HPCI and RCIC Systems at Hatch 1



NSIC 164955 - Sequence of Interest for Loss of RCIC at HPCI at Hatch 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 164955

LER NO.: 81-013

DATE OF LER: March 26, 1981

DATE OF EVENT: February 28, 1981

SYSTEM INVOLVED: HPCI and RCIC

COMPONENT INVOLVED: HPCI (broken oil line); RCIC (unknown)

CAUSE: Personnel error

SEQUENCE OF INTEREST: LOFW

ACTUAL OCCURRENCE: HPCI declared inoperable with RCIC inoperable

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 777 MWe

REACTOR AGE: 6.5 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Southern Services

OPERATORS: Georgia Power Co.

LOCATION: 11 miles north of Baxley, Georgia

DURATION: 8 h (estimated)

PLANT OPERATING CONDITION: Shutdown

TYPE OF FAILURE: Failed to start;
made inoperable

DISCOVERY METHOD: Surveillance testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 165438

Date: April 16, 1981

Title: Unavailability of Emergency Power at Hatch 1

The failure sequence was:

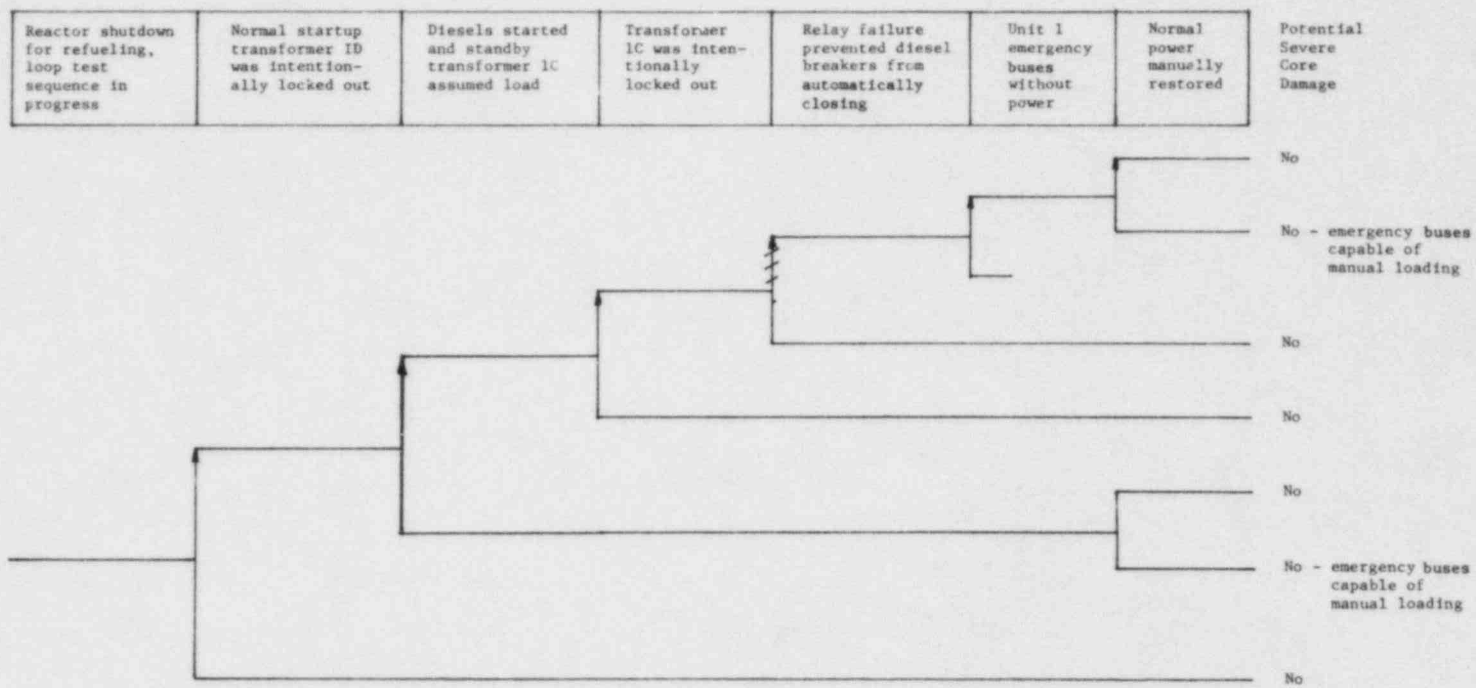
1. Unit 1 was shutdown for refueling and modification, while Unit 2 was operating at rated power.
2. As a part of a protective relay-breaker test, a loss of offsite power test sequence was performed.
3. The test was initiated by locking out the normal startup transformer 1D by tripping its protective relaying. The diesels started and the standby transformer 1C assumed emergency bus loads, both of which are correct responses.
4. The test then continued with locking out of the standby startup transformer 1C by tripping its protective relaying. The diesel breakers should have autoclosed in response but did not, due to a failure in disk rotation in two undervoltage relays in the lockout trip relay logic for the 1C transformer.
5. The emergency buses for Unit 1 remained deenergized until normal power was manually restored.
6. Unit 2 entered an LCO, since Unit 1 standby gas treatment operability (which is dependent upon diesel emergency power) is a requirement for Unit 2 operation (because of common operating areas).

Corrective action:

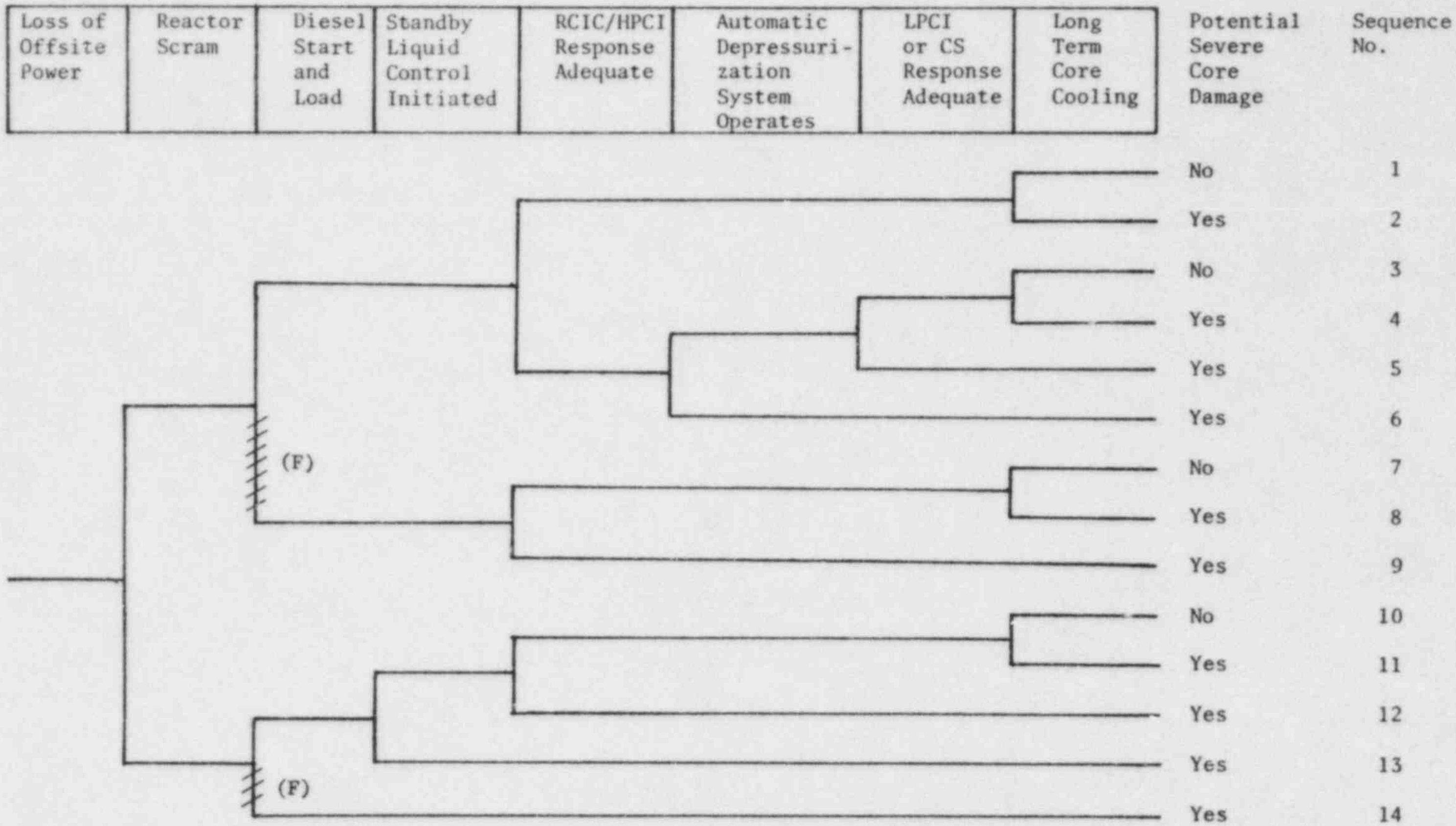
1. The failure was traced to the undervoltage relays, which were removed, checked and returned to operable status.
2. The LOOP test was rerun successfully.
3. The relays were checked for disk movement upon loss of PT voltage the next day for both units and were found working properly.
4. The present application of the relay remained under investigation.

Design purpose of failed system or component:

The emergency power system is designed to provide essential ac power in the event the normal plant power is unavailable.



NSIC 165438 - Actual Occurrence for Unavailability of Emergency Power at Hatch 1



NSIC 165438 - Sequence of Interest for Unavailability of Emergency Power at Hatch 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 165438

LER NO.: 81-026

DATE OF LER: April 16, 1981

DATE OF EVENT: April 5, 1981

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Relays

CAUSE: Mechanical failure

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Emergency buses failed to energize during LOOP test

REACTOR NAME: Hatch 1

DOCKET NUMBER: 50-321

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 777 MWe

REACTOR AGE: 6.7 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Georgia Power Co.

LOCATION: 11 miles north of Baxley, Georgia

DURATION: 360 h (estimated)

PLANT OPERATING CONDITION: Refueling

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Surveillance testing

COMMENT: Power could be restored to the buses, given failure of the auto closure, by manual loading of the diesels from the control room.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 165900

Date: April 21, 1981

Title: Inadvertent Opening of Pressurizer Relief Valve at Haddam Neck

The failure sequence was:

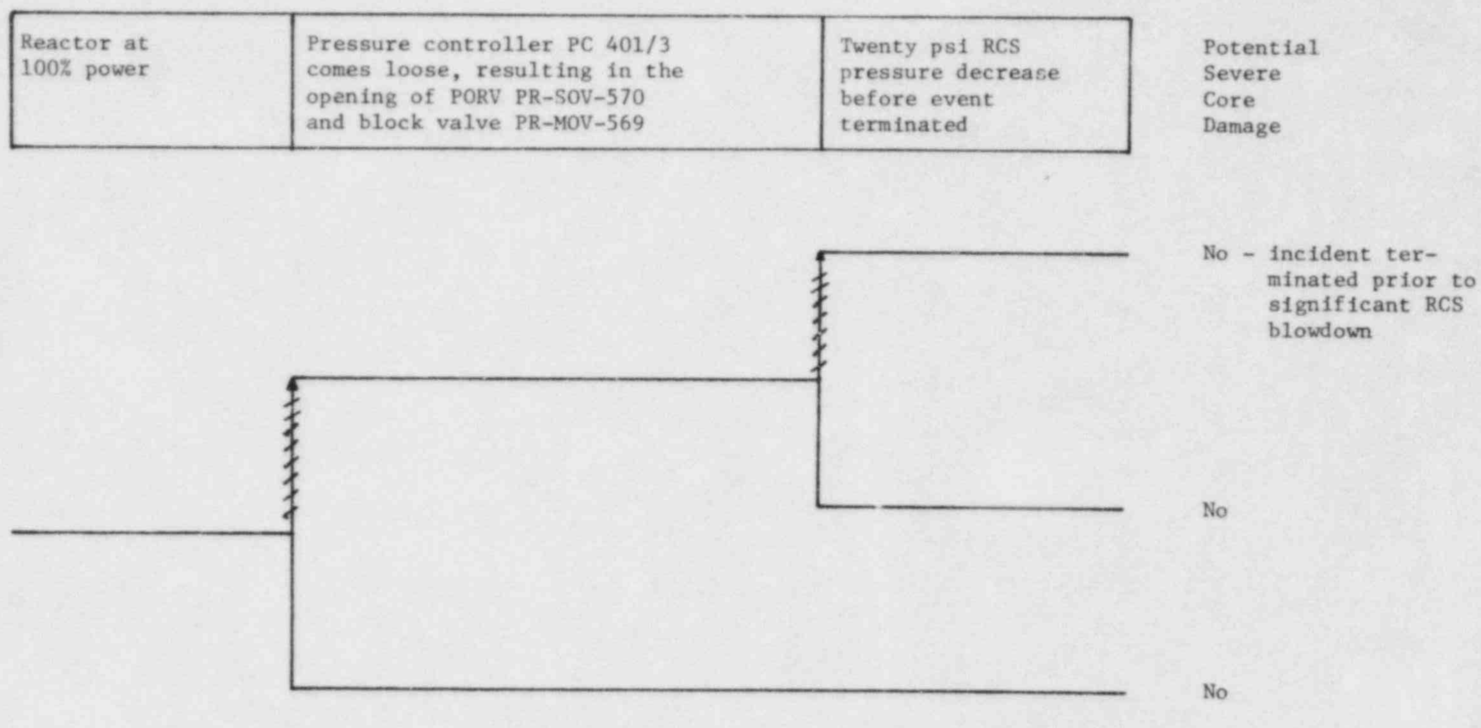
1. With the reactor at 100% power, pressurizer PORV PR-SOV-570 and its block valve PR-MOV-569 opened spuriously due to a connector coming loose from pressure controller PC 401/3.
2. This resulted in an RCS pressure decrease from 2000 to 1980 psi.

Corrective action:

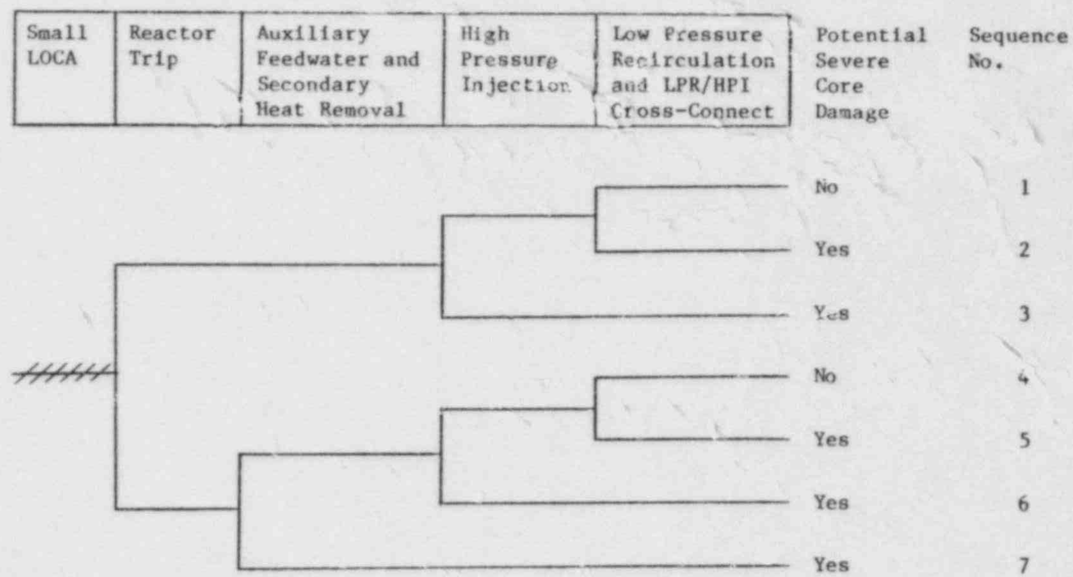
1. The connector was reinstalled. Actuation logic for the PORV and block valve were modified from a one channel pressure signal to 2 out of 3 channels to prevent spurious actuations.

Design purpose of failed system or component:

The PORV provides overpressure protection for the RCS and prevents challenges to the code safety valves.



NSIC 165900 - Actual Occurrence for Inadvertent Opening of Pressurizer Relief Valve and Block Valve at Haddam Neck



NSIC 165900 - Sequence of Interest for Inadvertent Opening of Pressurizer Relief Valve and Block Valve at Haddam Neck

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 165900

LER NO.: 81-003

DATE OF LER: April 21, 1981

DATE OF EVENT: April 3, 1981

SYSTEM INVOLVED: Pressurizer relief

COMPONENT INVOLVED: PORV and block valve

CAUSE: Loose electrical connector caused spurious signal

SEQUENCE OF INTEREST: Small break LOCA

ACTUAL OCCURRENCE: Small break LOCA

REACTOR NAME: Haddam Neck

DOCKET NUMBER: 50-213

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 580 MWe

REACTOR AGE: 13.7 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Conn Yankee Atomic Power Co.

LOCATION: 13 miles east of Meriden, Connecticut

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

TYPE OF FAILURE: Inadvertent opening

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 166072

Date: May 19, 1981

Title: Damaged RHR Heat Exchangers at Brunswick 1

The failure sequence was:

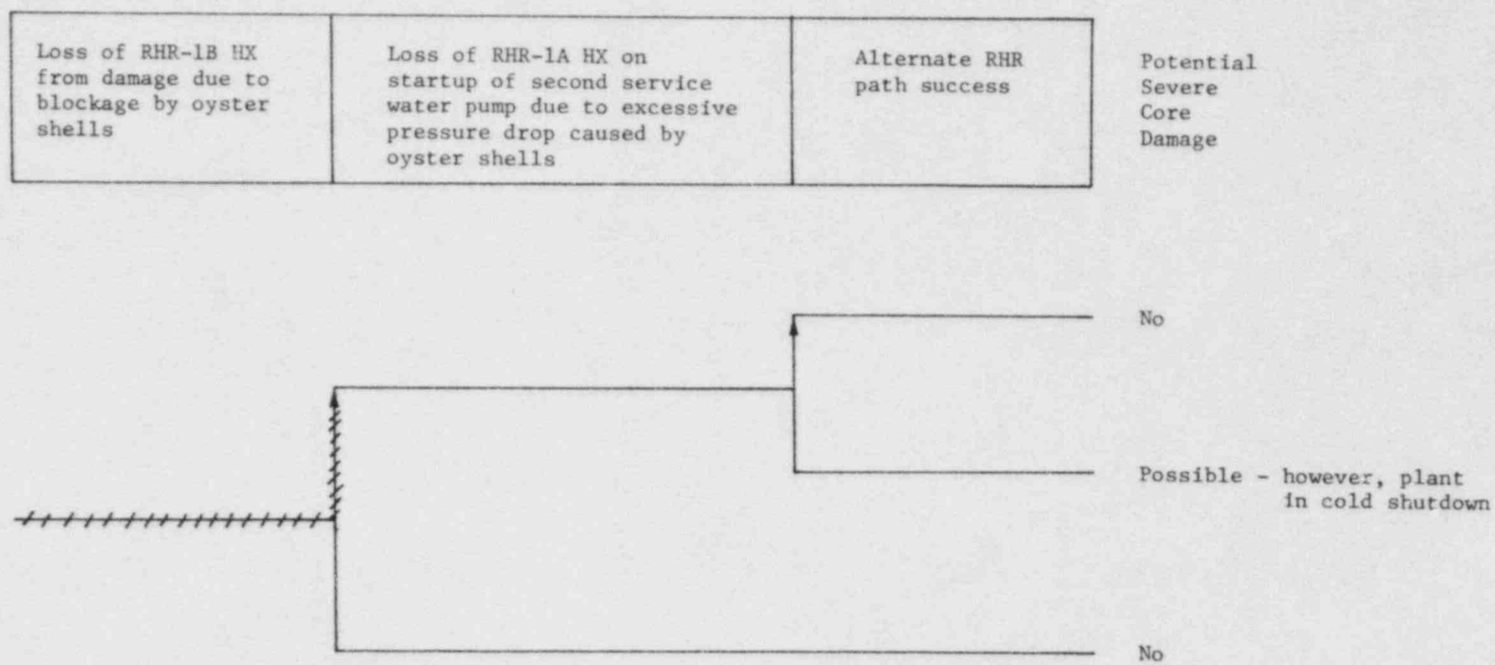
1. Oyster shells growing in the service water (SW) piping dislodged and accumulated in the tubing of the B RHR heat exchanger (HX). The shell buildup resulted from the service water chlorination system being out of service for an extended period of time.
2. The blockage in the HX tubes created a high pressure drop which ultimately displaced the baffle plate normally separating the SW inlet and outlet. The displacement allowed the service water to bypass the HX tubes, rendering the heat removal capability unavailable. This was discovered during a special inspection conducted during a cold shutdown April 19, 1981.
3. The 1B HX was removed from service for repair while the 1A RHR HX maintained the cold shutdown condition until April 25, 1981, when the second service water pump for the 1A RHR HX was started.
4. The increased flow caused the 1A HX baffle to displace in the same manner as the 1B HX, resulting in a loss of cooling. Shell buildup in the 1A HX was also responsible for the high pressure drop conditions leading to the baffle displacement.
5. Both RHR HXs for Brunswick 1 were unavailable for heat removal.

Corrective action:

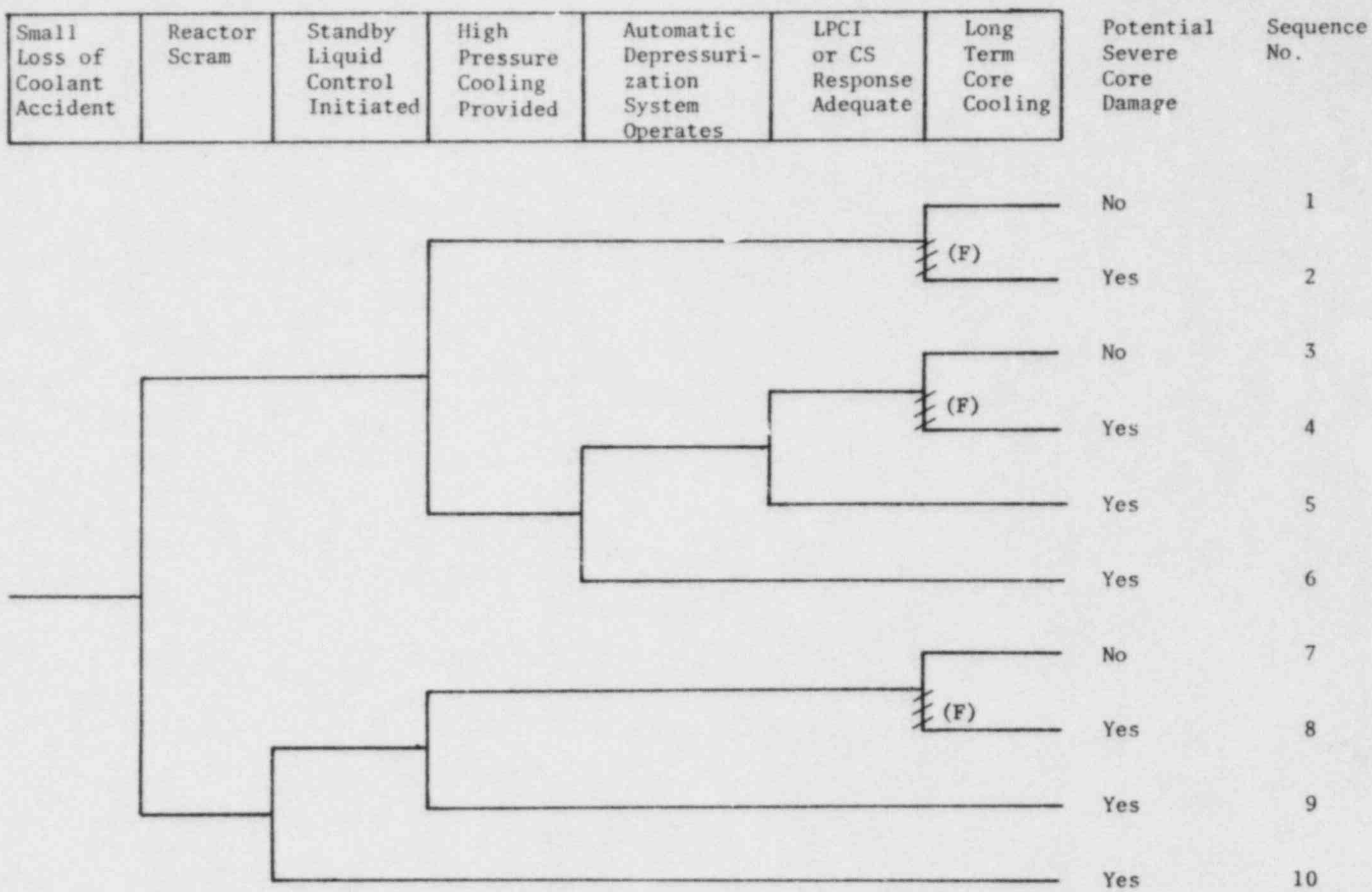
1. An alternate shutdown cooling lineup was established using fuel pool coolers, the condensate storage tank, and the core spray system.
2. Temporary repairs were performed on the 1A heat exchanger and it was returned to service while permanent repairs were still in progress on the 1B heat exchanger. Permanent repairs for the 1A HX were to follow return to service of the 1B HX.
3. Programs were being pursued to monitor safety-related HX performance, clean the HX tubes and SW piping, and resume the SW chlorination operation.
4. Plant procedures were to be revised to vent the RHR service water system regularly.

Design purpose of failed system or component:

The RHR system provides long-term core cooling for normal shutdown, maintenance of cold shutdown, and heat removal capability for emergency shutdown modes.



NSIC 166072 - Actual Occurrence for Loss of RHR Heat Exchangers at Brunswick 1



NSIC 166072 - Sequence of Interest for Loss of RHR at Brunswick 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 166072

LER NO.: 81-032

DATE OF LER: May 19, 1981

DATE OF EVENT: April 19, 1981

SYSTEMS INVOLVED: Residual heat removal

COMPONENT INVOLVED: RHR heat exchanger

CAUSE: Excessive pressure drop from aquatic growth

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: none

REACTOR NAME: Brunswick Unit 1

DOCKET NUMBER: 50-325

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 4.5 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: United Engineers & Constructors

OPERATORS: Carolina Power & Light

LOCATION: 3 miles north of Southport, North Carolina

DURATION: 512 h (estimated). This includes half the test interval (360 h) plus the period between the two failure discoveries (144 h) plus 8 additional hours to make the temporary repairs and return the 1A heat exchanger to service.

PLANT OPERATING CONDITION: Cold shutdown

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Special inspection

COMMENT: Further information: NSIC 166019 (Brunswick 1, 50-325, LER 81-0005, April 30, 1981). Feed and bleed using the main feedwater and main steam systems could potentially be an alternative for RHR cooling but may be difficult to implement due to the potential radioactive environment of the containment following a LOCA.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 166082

Date: May 7, 1981

Title: Loss of RCIC and HPCI Systems at Brunswick 2

The failure sequence was:

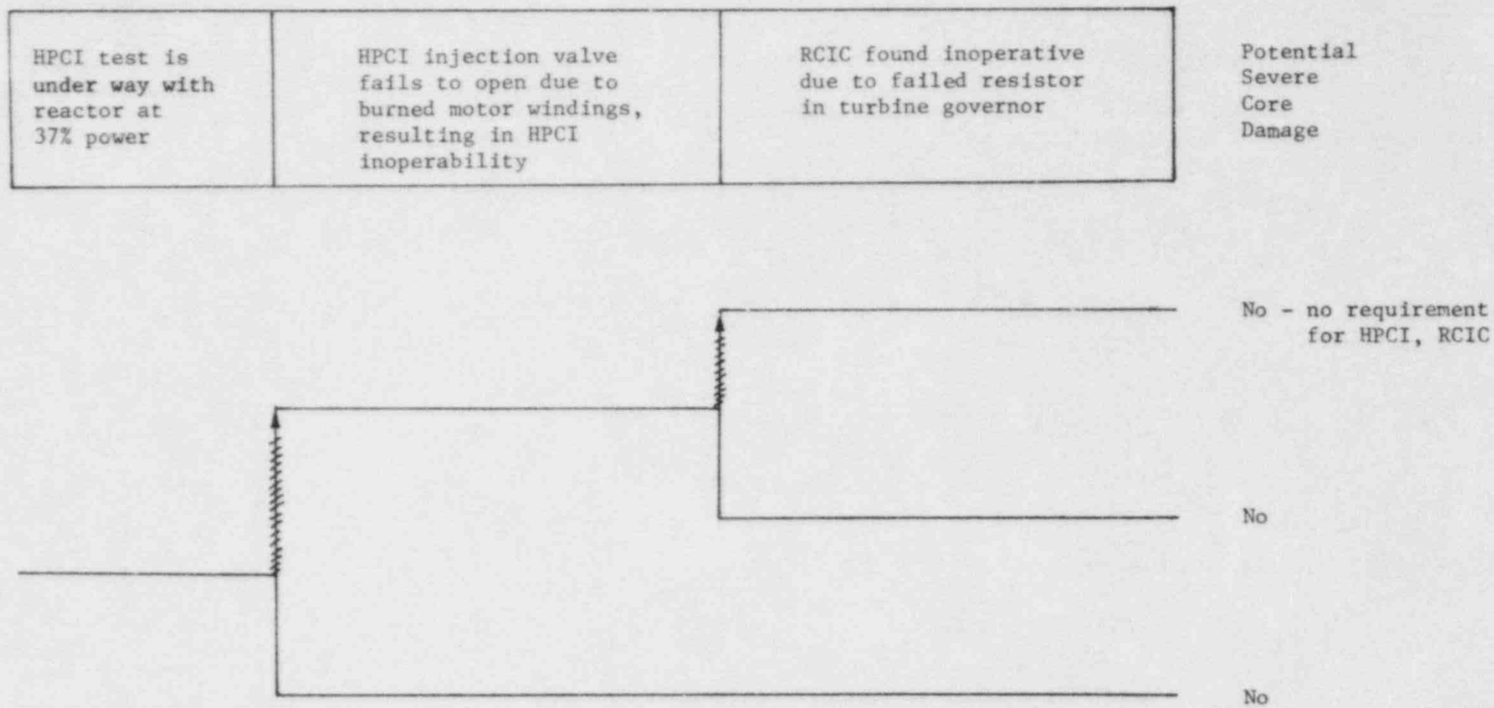
1. At approximately 37% power, during performance testing, the HPCI system injection valve failed to open due to burned windings in the valve motor operator.
2. The HPCI system was declared inoperable.
3. During subsequent testing of RCIC, a faulty resistor in the RCIC turbine governor controls caused a loss of speed control, followed by an overspeed trip, making the RCIC system inoperable.
4. Both HPCI and RCIC systems were inoperable.

Corrective action:

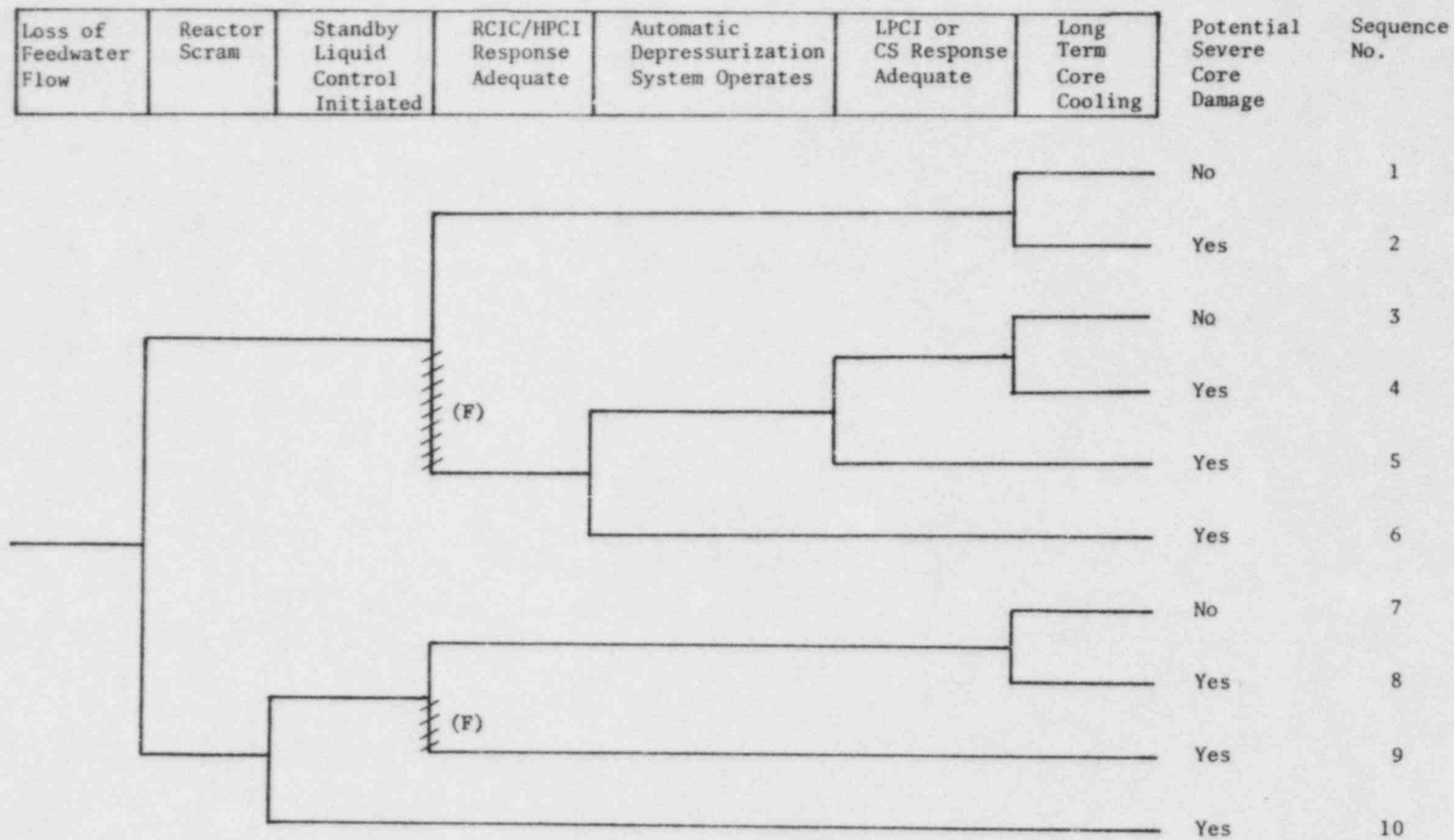
1. The faulty HPCI valve motor operator, which was found to have burned windings, was replaced, tested satisfactorily, and returned to service.
2. A thorough investigation of the HPCI failed motor windings did not reveal cause for the failure.
3. The faulty RCIC resistor was replaced, and the system was tested satisfactorily and returned to service.

Design purpose of failed system or component:

1. RCIC pro des reactor water level makeup following trip when the feedwate system is unavailable.
2. HPCI pro des reactor core cooling in the event of a small break LOCA.



NSIC 166082 - Actual Occurrence of Loss of RCIC and HPCI Systems at Brunswick 2



NSIC 166082 - Sequence of Interest for Loss of RCIC and HPCI at Brunswick 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 166082

LER NO.: 81-039

DATE OF LER: May 7, 1981

DATE OF EVENT: April 10, 1981

SYSTEM INVOLVED: HPCI and RCIC

COMPONENT INVOLVED: Valve operator and turbine governor

CAUSE: Mechanical failures

SEQUENCE OF INTEREST: LOFW

ACTUAL OCCURRENCE: HPCI and RCIC found inoperable upon testing

REACTOR NAME: Brunswick 2

DOCKET NUMBER: 50-324

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 821 MWe

REACTOR AGE: 6.1 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: United Engineers & Constructors

OPERATORS: Carolina Power & Light

LOCATION: 3 miles north of Southport, North Carolina

DURATION: 360 h (estimated)

PLANT OPERATING CONDITION: 37% power

TYPE OF FAILURE: Failed to start;
made inoperable

DISCOVERY METHOD: Surveillance testing

COMMENT: Additional information: NSIC 166083 (Brunswick 2, 50-324, LER 81-029, May 5, 1981).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 166384

Date: May 22, 1981

Title: LOOP at Monticello

The failure sequence was:

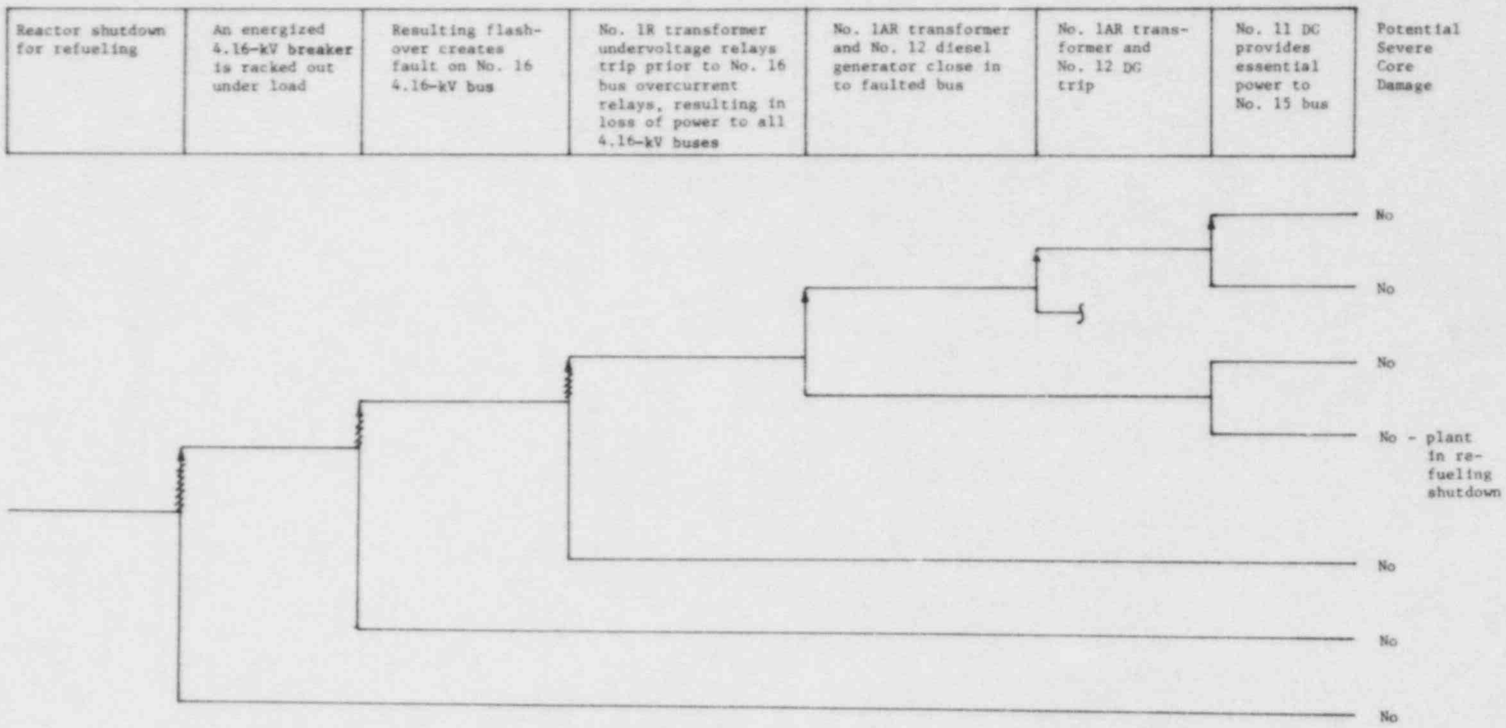
1. During a refueling outage an energized 4.16 kV breaker was racked out under load due to an operator error.
2. The resulting flashover created an electrical fault on No. 16 4.16 kV essential bus.
3. The No. 16 bus fault was picked up by the undervoltage relays on the 1R transformer, which then tripped off before the overcurrent relays on bus No. 16 could trip off.
4. Sensing of the undervoltage condition on the 1R transformer actuated automatic isolation of the entire 4.16 kV distribution system from the power source.
5. A momentary loss of all 4.16 kV busses occurred.
6. The No. 1AR transformer and the No. 12 diesel generator closed into the faulted bus and subsequently tripped due to undervoltage and overcurrent, respectively.
7. The No. 11 diesel generator came on line and successfully supplied power to the No. 15 4.16 kV essential bus.

Corrective action:

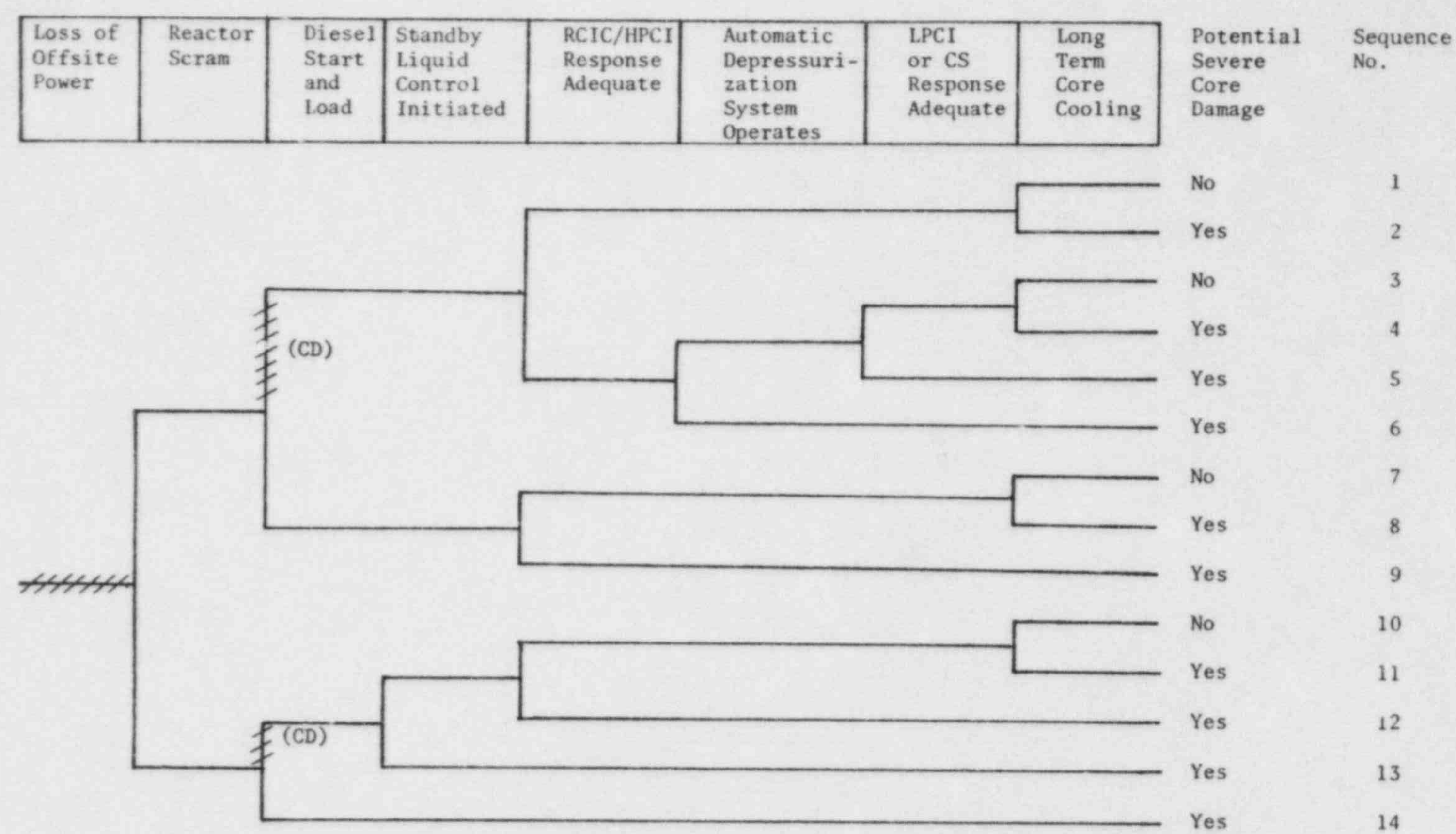
1. Operator action restored the unfaulted 4.16 kV buses to service by returning transformer 1R to service.
2. The faulted essential bus and damaged breaker were isolated.
3. Operator retraining program was revised to include proper breaker operations.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads when the unit generator is not operating.



NSIC 166384 - Actual Occurrence for LOOP at Monticello



NSIC 166384 - Sequence of Interest for LOOP at Monticello

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 166384

LER NO.: 81-009

DATE OF LER: May 22, 1981

DATE OF EVENT: April 27, 1981

SYSTEM INVOLVED: Electrical system

COMPONENT INVOLVED: 4.16 kV buses

CAUSE: Operator error initiating electrical fault

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP

REACTOR NAME: Monticello

DOCKET NUMBER: 50-263

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 545 MWe

REACTOR AGE: 10.4 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Northern States Power Co.

LOCATION: 30 miles NW of Minneapolis, Minnesota

DURATION: N/A

PLANT OPERATING CONDITION: Refueling

TYPE OF FAILURE: Inadequate performance;
made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 166650

Date: June 19, 1981

Title: Safety Injection Supply Valve Found Unlocked and Closed at Beaver Valley

The failure sequence was:

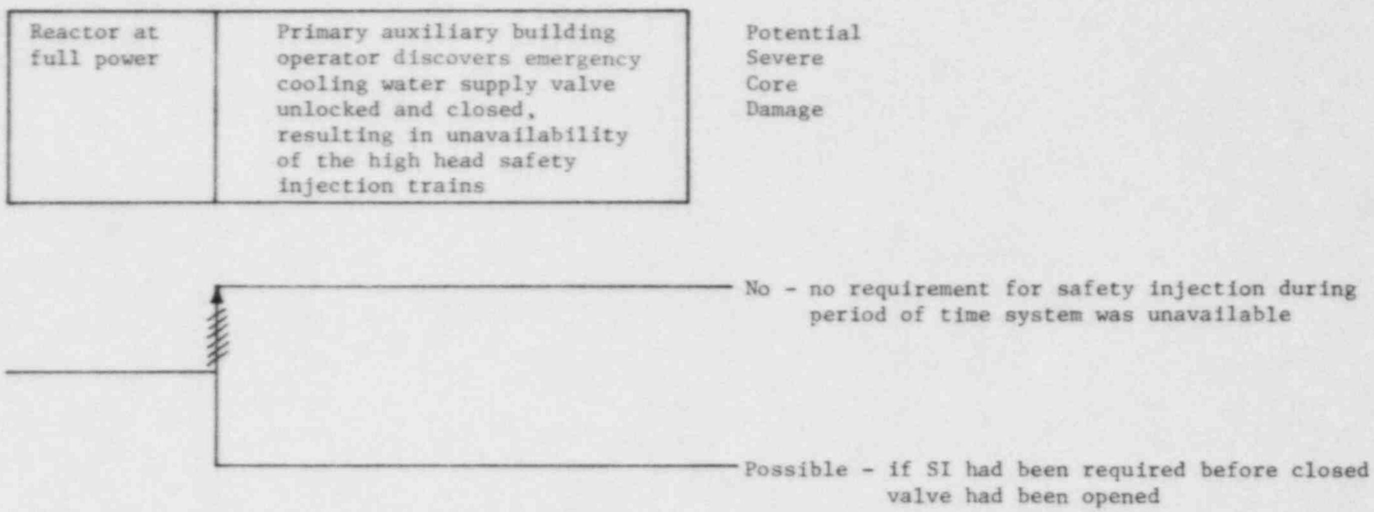
1. With the reactor at full power, the primary auxiliary building operator discovered the normally locked-open emergency cooling water supply valve to the safety injection pumps unlocked and closed.

Corrective action:

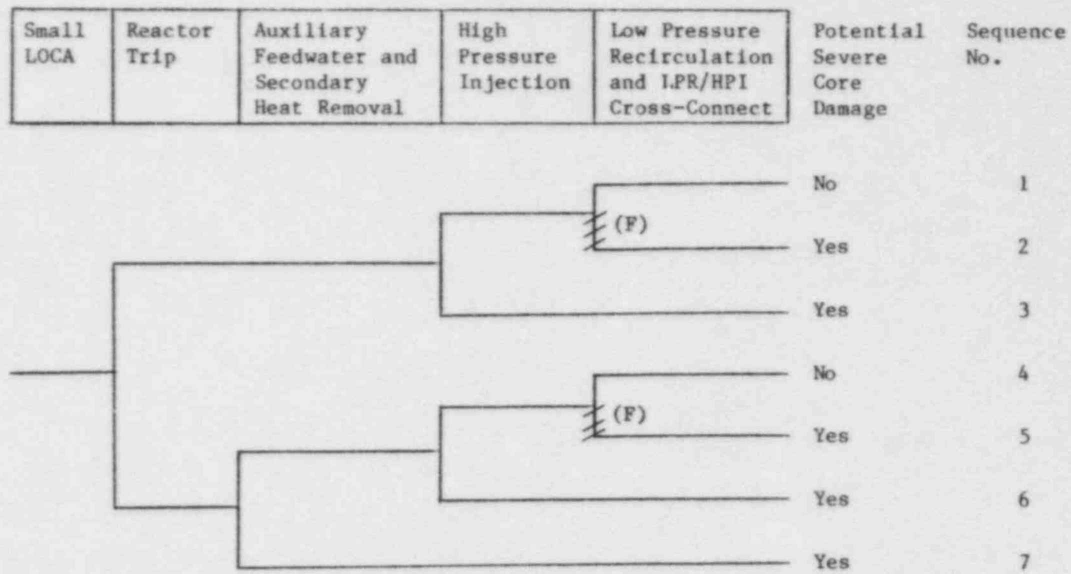
1. The valve was reopened and locked in that position.

Design purpose of failed system or component:

The high head safety injection pumps provide water for RCS makeup following a small break LOCA.



NSIC 166650 - Actual Occurrence for Safety Injection Supply Valve Found Unlocked and Closed at Beaver Valley



NSIC 166650 - Sequence of Interest for Safety Injection Supply Valve Found Unlocked and Closed at Beaver Valley

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 166650

LER NO.: 81-047

DATE OF LER: June 19, 1981

DATE OF EVENT: June 6, 1981

SYSTEM INVOLVED: High pressure safety injection

COMPONENT INVOLVED: Suction valve

CAUSE: Normally locked open valve found closed

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Unavailability of high head SI systems

REACTOR NAME: Beaver Valley

DOCKET NUMBER: 50-334

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 852 MWe

REACTOR AGE: 5.35 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Duquesne Light Co.

LOCATION: 5 miles east of East Liverpool, Ohio

DURATION: 4.62 h

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operator observation

COMMENT: (1) On June 5, 1981, the utility also discovered the chains used to lock the normally open emergency feedwater valves were missing and could not be found. Those valves were in their correct positions. (2) Additional information: Report to Congress on Abnormal Occurrences, NUREG-0090, Vol. 4, No. 3.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 166745

Date: July 10, 1981

Title: Two Shutdown Sequencers and Diesel Generator Fail at Palisades

The failure sequence was:

1. With the reactor at 99% power, shutdown sequencer 34-5 failed to operate during its monthly test. The prototype events recorder was attached.
2. Diesel generator 1-2 was tested satisfactorily.
3. Sequencer 34-6 was tested and failed to operate apparently due to binding in the clutch. The prototype events recorder was attached.
4. Diesel generator 1-1 was tested and failed to reach rated frequency.
5. Following cleaning of the 34-5 sequencer contacts, it was retested successfully. No events recorder was attached. The events recorder was then reattached and attempts to test-start loads associated with the sequences were unsuccessful. This was attributed to an inadequately isolated events recorder used during the testing.

Corrective action:

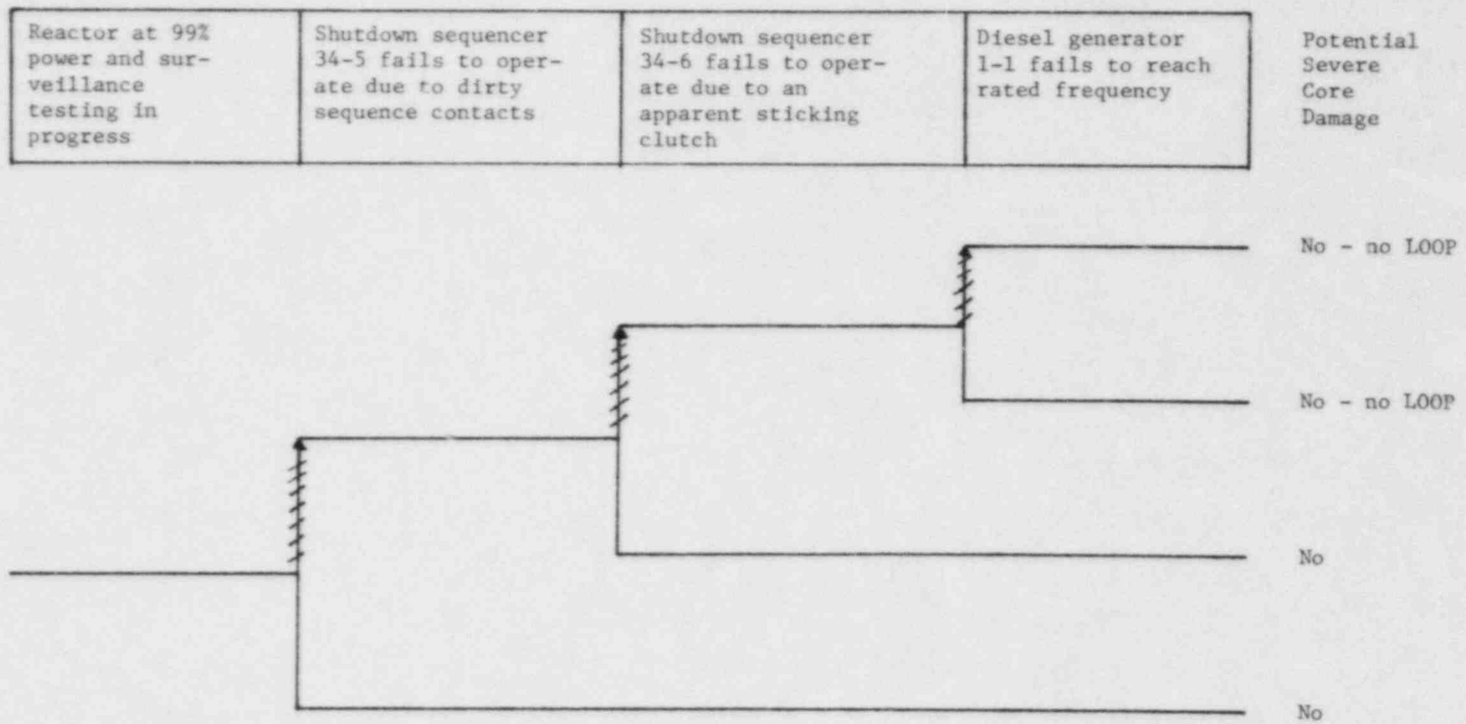
1. Sequencer 34-5 contacts were cleaned and the sequencer was returned to service.
2. Sequencer 34-6 failure was not repeatable; however the clutch was cleaned and lubricated and the sequencers were returned to service.

Design purpose of failed system or component:

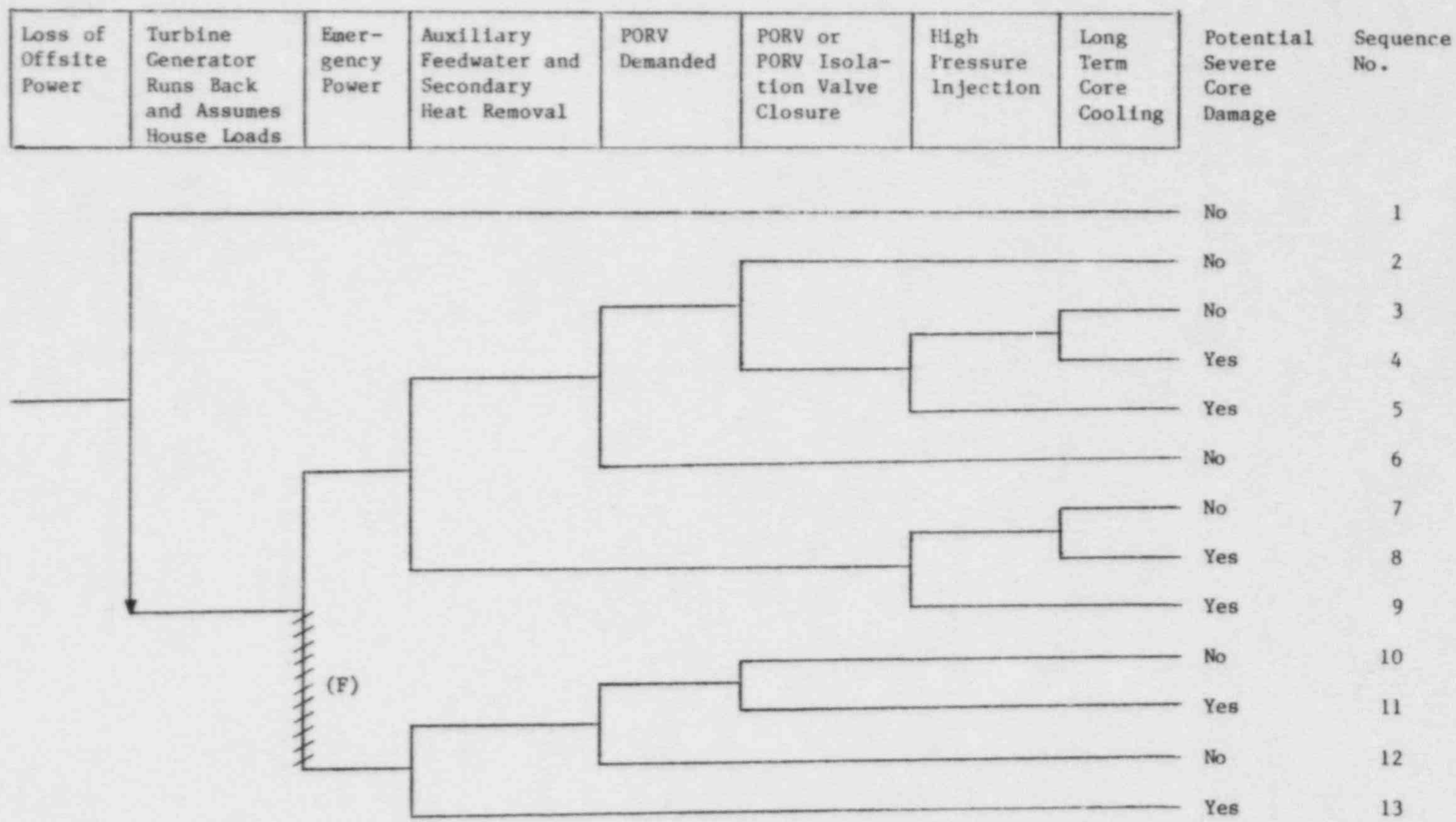
1. The diesel generators provide power to safety-related loads when the unit generator and offsite power sources are unavailable.
2. The sequencers function to automatically start selected loads following a loss of offsite power event; the loads are sequenced to prevent overloading of the diesel generators. Loads operated by the sequencers are tabulated below.

34-5 (left channel)	34-6 (right channel)
o Service water pump P-7B	o Service water pumps P7A and P7C
o Charging pump P-55C	o Charging pumps P-55A and P-55B
o Containment cooling RCIC fan V4A	o Containment cooling recirc. fans V1A, V2A, and V3A

Sequencer failure would require the loads tabulated above to be manually started following a loss of offsite power event.



NSIC 166745 - Actual Occurrence for Two Shutdown Sequencers and Diesel Generator Unavailable at Palisades



NSIC 166745 - Sequence of Interest for Two Shutdown Sequencers and Diesel Generator Unavailable at Palisades

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 166745

LER NO.: 81-025

DATE OF LER: July 10, 1981

DATE OF EVENT: June 26, 1981

SYSTEM INVOLVED: Emergency power systems

COMPONENT INVOLVED: Sequencers and diesel generator

CAUSE: Dirty contacts, sticking clutch

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Loss of DG auto load capability and one DG failure to reach correct frequency

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 10.1 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power

LOCATION: 5 miles south of South Haven, Michigan

DURATION: 15 days (estimated)

PLANT OPERATING CONDITION: 99% power

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Surveillance test

COMMENT: Manual start and load were available but potential DG overload was possible.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 167117

Date: July 7, 1981

Title: Switchyard Voltage Drops Below Low Limit at Rancho Seco

The failure sequence was:

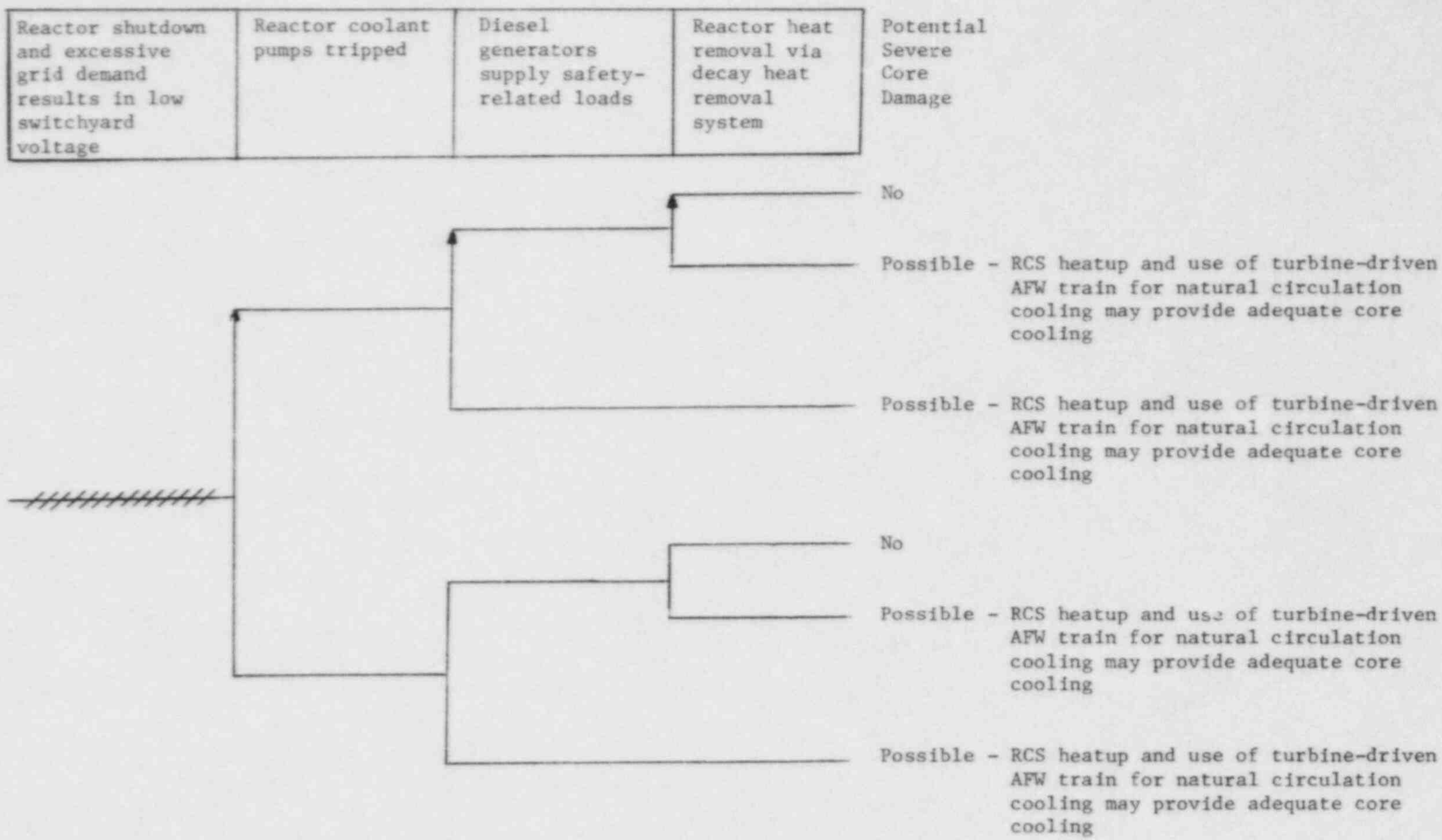
1. With the reactor in a heatup mode coming from cold shut down, excessive electrical demand resulted in a reduction in switchyard voltage to 207 kV. This is below the minimum voltage (214 kV) assumed for analysis.
2. The reactor coolant pumps were tripped to avoid excessive temperatures.
3. Two failures in the electrohydraulic control system for the turbine generator occurred.
4. The diesel generators were started and provided power to the safety-related buses.
5. The reactor was operated in the decay heat mode.

Corrective action:

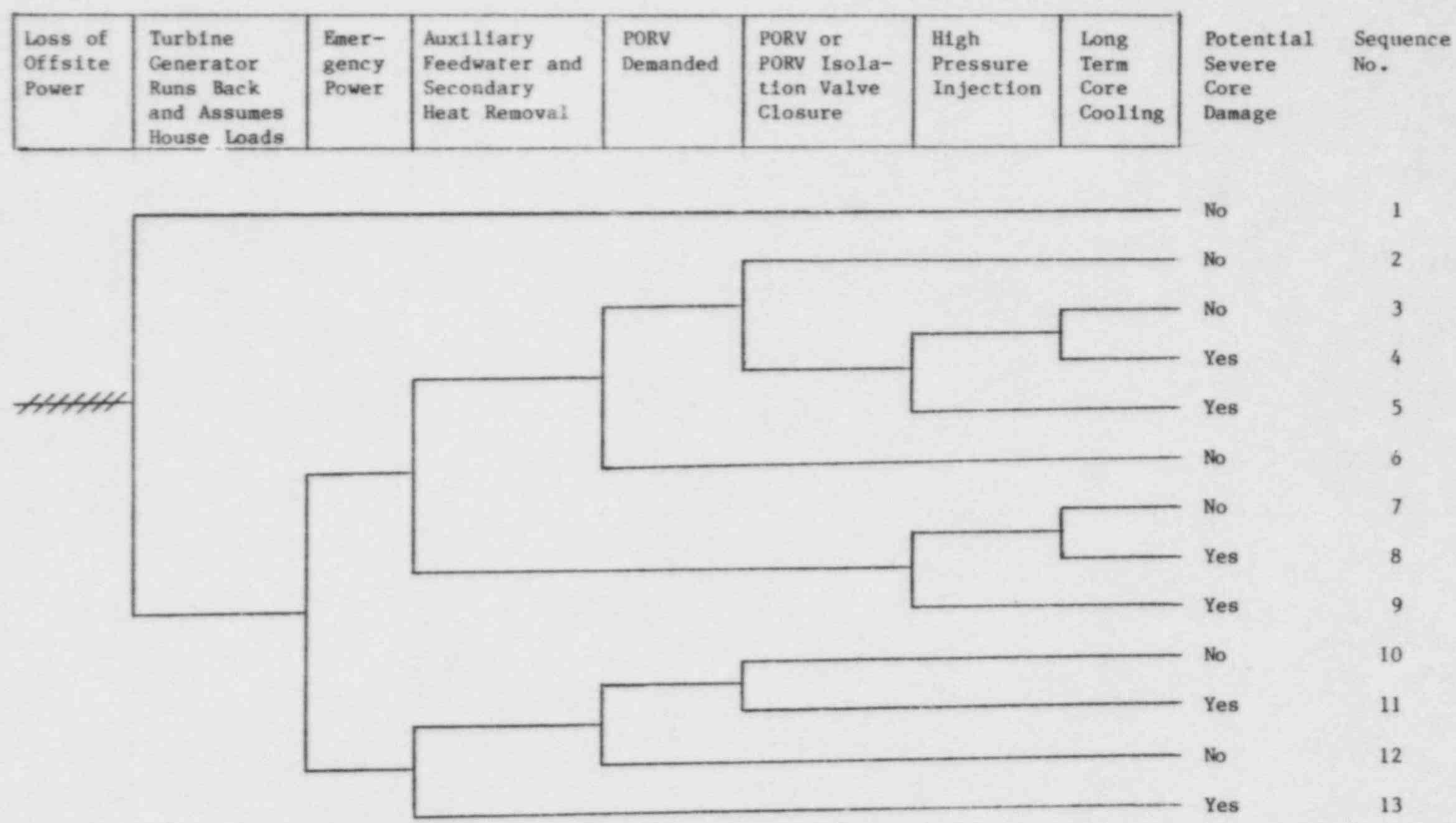
Direct switchyard voltage indications and alarms were to be installed to facilitate control room monitoring of switchyard voltages.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads. The vital buses have 2 power supplies, offsite power and emergency diesel generators. The unit generator cannot directly power these buses.



NSIC 167117 - Actual Occurrence for Effective Loop at Rancho Seco



NSIC 167117 - Sequence of Interest for Effective LOOP at Rancho Seco

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 167117

LER NO.: 81-034

DATE OF LER: July 7, 1981

DATE OF EVENT: June 19, 1981

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Switchyard

CAUSE: Low switchyard voltages due to excessive load demand

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Effective LOOP

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 50-312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 6.8 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 miles SE of Sacramento, California

DURATION: N/A

PLANT OPERATING CONDITION: Cold shutdown

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Operational event

COMMENT: The safety features equipment called upon were the DGs. They performed as designed and powered the vital buses. They did not experience inadequate performance or inoperability. Also see Accession 168548, Rancho Seco, 50-312, LER 81-039, Aug. 25, 1981.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 167611

Date: June 30, 1981

Title: Inadvertent Spray Initiation and Draining of Reactor Coolant System at Sequoyah 1

The failure sequence was:

1. The unit was in cold shutdown on RHR train and preparing to bring RHR train B on line.
2. To effect this, the unit operator sent an assistant unit operator (AUO) to open locally operated valves 1-HCV-74-37 and 531. Because the RHR containment spray valve 1-FCV-72-40 had been tested for operability earlier in the day, the auxiliary unit operator was also instructed to check it for closure.
3. A later telephone conversation between the unit operator and auxiliary unit operator apparently confused the AUO regarding the valves to be opened, and he opened all three valves which resulted in initiation of containment spray.
4. The RCS began to blow down through the B RHR train and spray line to the containment.
5. RCS pressure and pressurizer level decreased rapidly.
6. The operators, believing a LOCA had possibly occurred, tripped the operating RC pumps.
7. The containment was evacuated.
8. Standard emergency (LOCA) procedures were implemented:
   a. containment purge was terminated,
   b. the charging pumps were aligned to the RWST,
   c. RHR suction valve to the RWST was opened.
9. Pressurizer level began to increase.
10. Forty three minutes after the event began, the operators learned that the AUO had opened the spray valves and verified its being open from the control board indicators. The valve was closed and the RCS stabilized.
11. During the event, approximately 105,000 gallons of water had been sprayed into the containment; 40,000 gallons from the RCS and 65,000 gallons from the RWST.

Corrective action:

An extensive initial investigation subsequently resulted in the following actions:

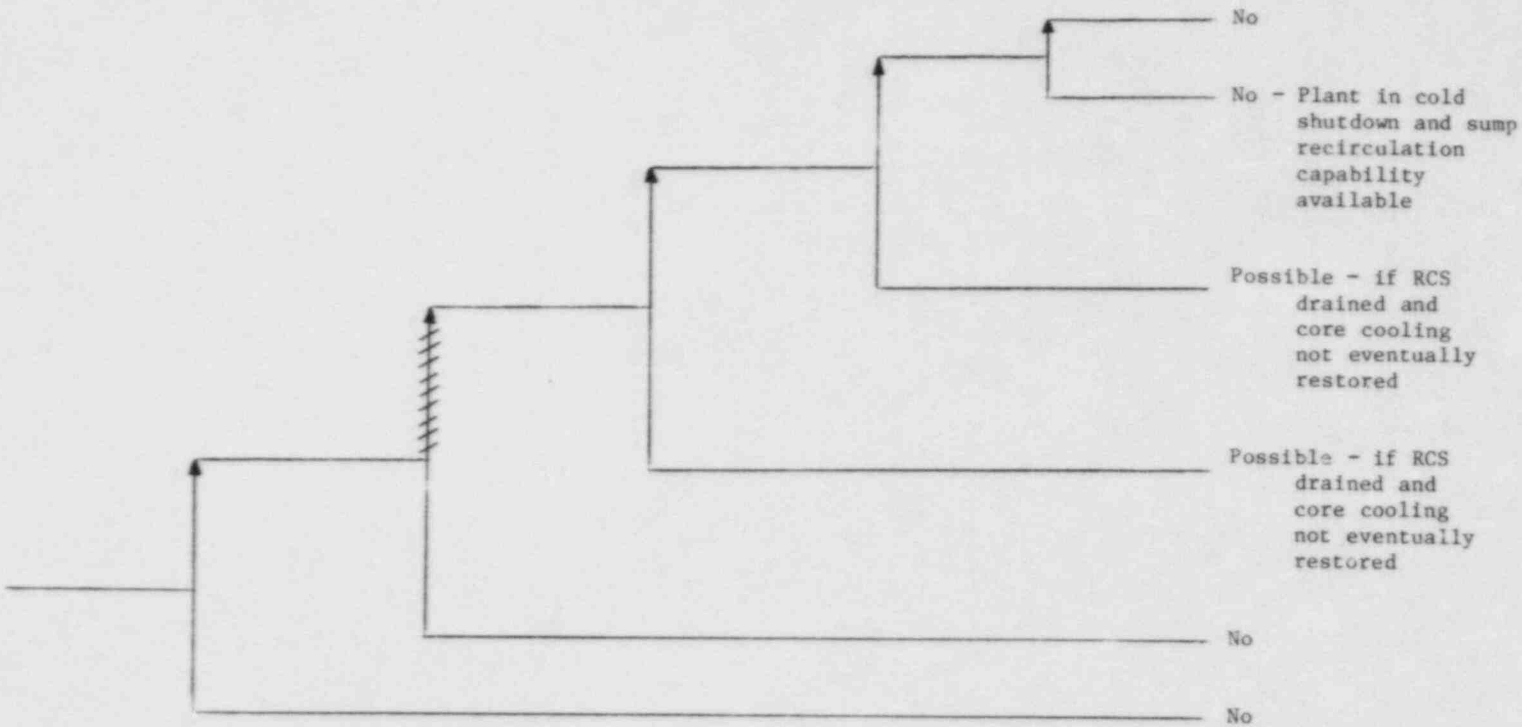
1. In order to clarify the duties and responsibilities of the shift employees including the shift engineer, the structure of the operating shift was revised and issued. The general responsibilities and authorities of each position are described in the job description provided to the individuals when they are appointed to the position. Administrative Instruction AI-2 has been revised to describe the responsibilities and authorities of each operating station.

2. In order to improve communications between shift personnel and between shift personnel and management, clearer lines of communication were provided between operating positions within a shift as well as between operating shifts and other sections by revising the shift structure, clarifying the communication paths, establishing work location routines, improving the maintenance of telephones, and investigating additional or different radio communications.
3. The environment in the main control room was improved by closer supervision and compliance with established policies regarding conduct, access, and housekeeping.
4. The Assistant Superintendent and Operations Supervisor met with each shift crew before restart to emphasize the conduct required by AI-2. These discussions stressed clear communications, control room atmosphere, authorities and responsibilities of operating personnel, and status control of safety-related systems. Discussions were also held with all key supervisors emphasizing the requirements to keep the shift engineer informed of work in progress and his responsibility to keep control of activities affecting safety.
5. An in-plant on-the-job training and certification system of nonlicensed operating personnel was established.
6. All future nonlicensed operating employees will, upon assignment to Sequoyah, receive on-the-job break-in training and examinations before assuming responsibility for any job position. A Sequoyah Standard Practice describing this break-in was issued and implemented.
7. In order to ensure that only qualified employees are assigned to perform functions that can affect the safety of operations, TVA evaluated nonlicensed operating employees, specifically the assistant unit operators and fourth-period student operators, to determine each individual's qualifications and competence in regard to performing operating functions that can affect the safety of operations. The result of this evaluation is a qualification status list which reflects the spectrum of nonlicensed operating personnel's operating experience at Sequoyah. This list will be used to fill vacant shift positions.

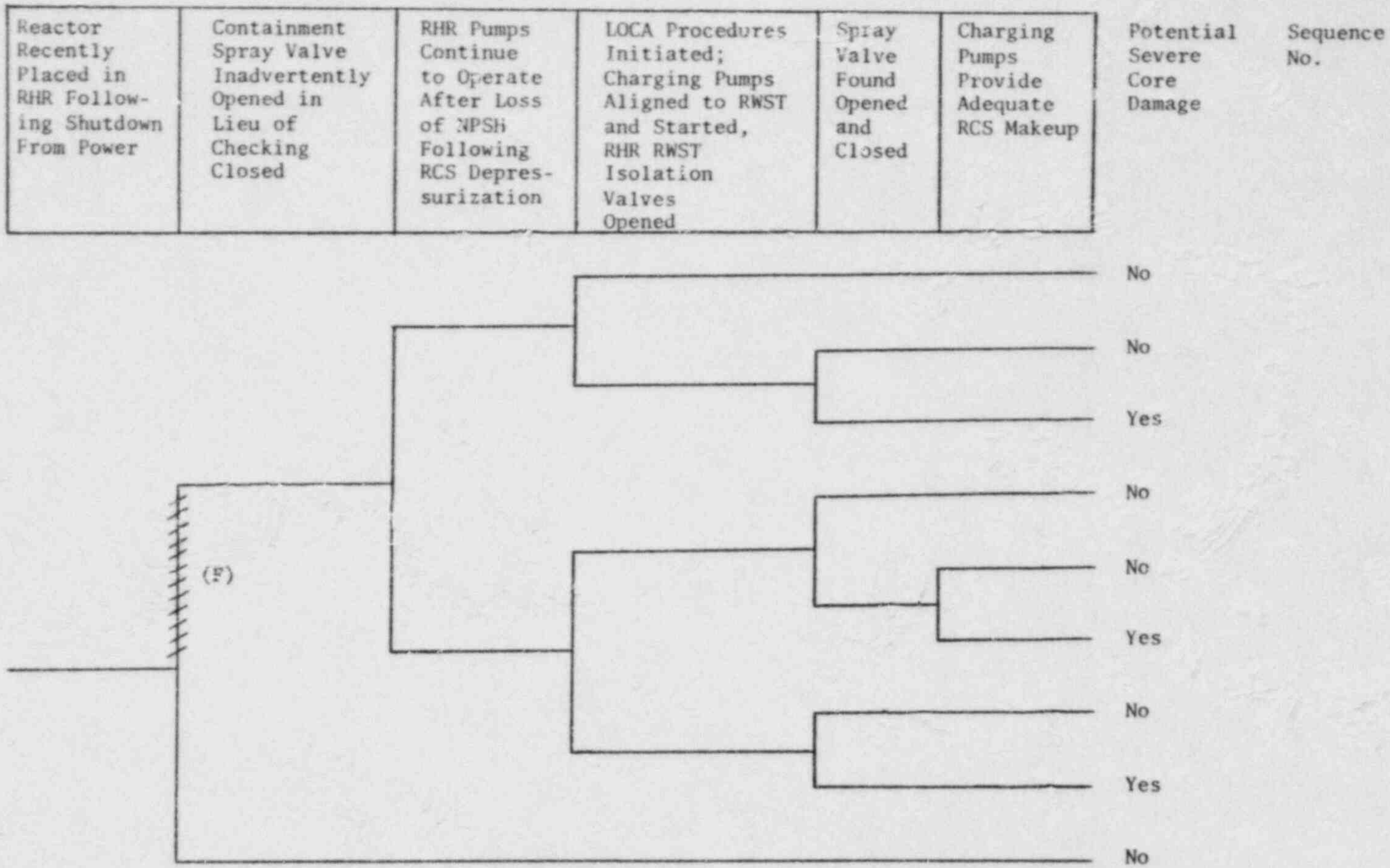
Design purpose of failed system or component:

The spray system provides water to the containment atmosphere for containment depressurization and radionuclide scavenging. The RHR system provides core cooling during plant shutdown.

Reactor at cold shutdown with RHR train A in operation	Assistant unit operator sent to open RHR train B valves 1-HCV-74-37 and -531 to prepare for starting train B and to check closed containment spray valve 1-FCV-72-40	After opening RHR valves, AUO opened spray valve in lieu of checking it closed, initiating containment spray through RHR system	Rapid RCS pressure and pressurizer level decrease results in initiation of LOCA emergency procedure	Charging pumps aligned to RWST; RHR suction valves to RWST opened	Spray valve discovered opened and closed	Potential Severe Core Damage



NSIC 167611 - Actual Occurrence for Inadvertent Spray Initiation and Draining of Reactor Coolant System at Sequoyah 1



NSIC 167611 - Sequence of Interest for Inadvertent Spray Initiation and Draining of Reactor Coolant System at Sequoyah 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 167611

LER NO.: 81-021 Rev. 1

DATE OF LER: June 30, 1981

DATE OF EVENT: February 11, 1981

SYSTEM INVOLVED: Residual heat removal and containment spray

COMPONENT INVOLVED: Spray isolation valve

CAUSE: Operator error

SEQUENCE OF INTEREST: Small break LOCA while on RHR

ACTUAL OCCURRENCE: Inadvertent spray initiation and RCS draining

REACTOR NAME: Sequoyah 1

DOCKET NUMBER: 50-327

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1128 MWe

REACTOR AGE: 0.6 year

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Tennessee Valley Authority

OPERATORS: Tennessee Valley Authority

LOCATION: 9.5 miles NE of Chattanooga, Tennessee

DURATION: N/A

PLANT OPERATING CONDITION: Cold shutdown

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 167624

Date: July 14, 1981

Title: Loss of Offsite Power and Failure of a Diesel Generator to Start at Crystal River 3

The failure sequence was:

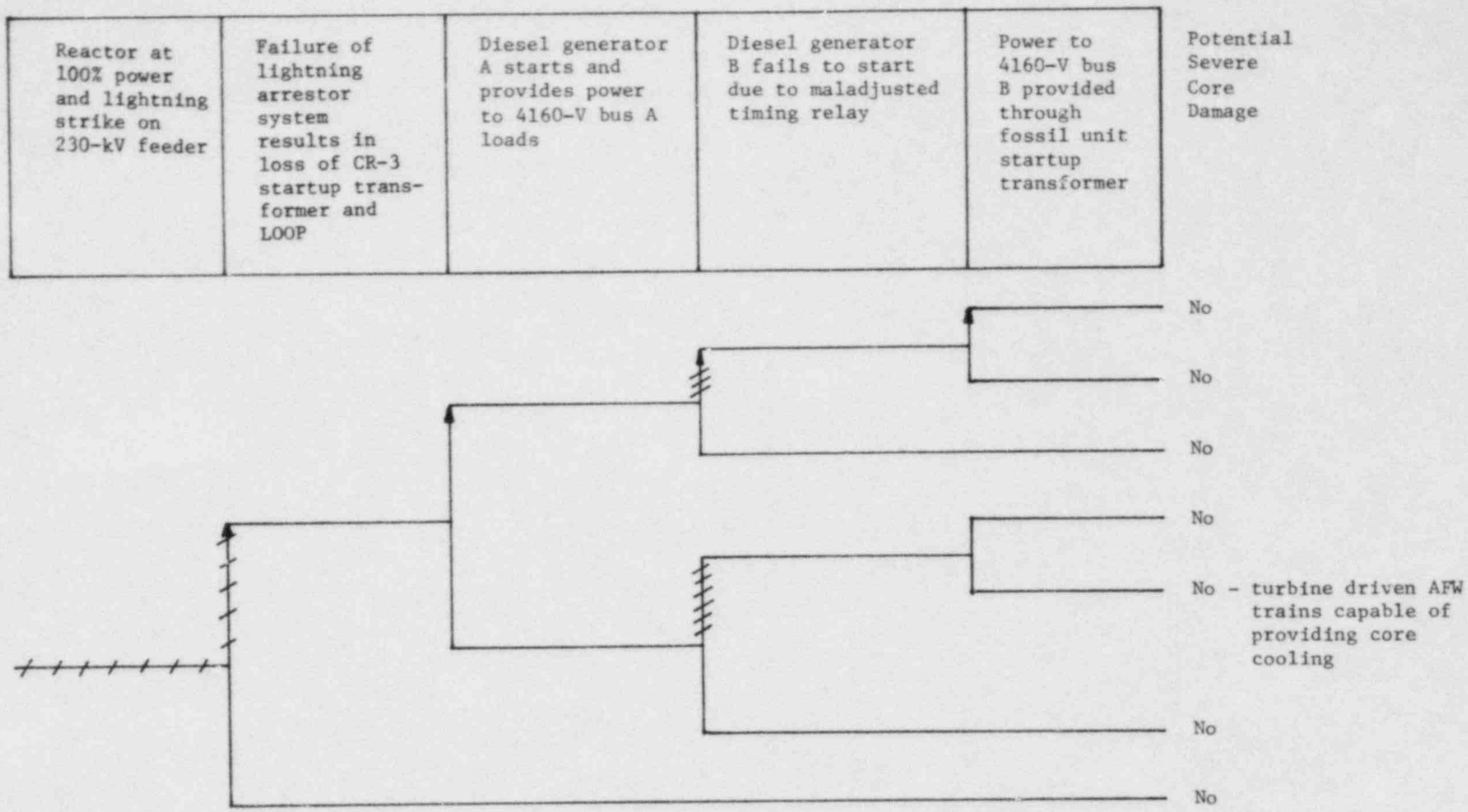
1. With the reactor at 100% power, lightning struck the 230 kV feeder line.
2. The lightning arrestor system failed to prevent loss of the CR-3 startup transformer, resulting in loss of all ac power.
3. Diesel generator A started and loaded the A 4169 volt bus.
4. Diesel generator B failed to start due to a maladjusted timing relay. Power was provided to the B 4180 volt bus from a fossil plant startup transformer via manual connection by the operator.
5. The "B" DG was successfully started manually about 10 minutes into the transient and was allowed to run on standby until Unit 3 transformer was energized.

Corrective action:

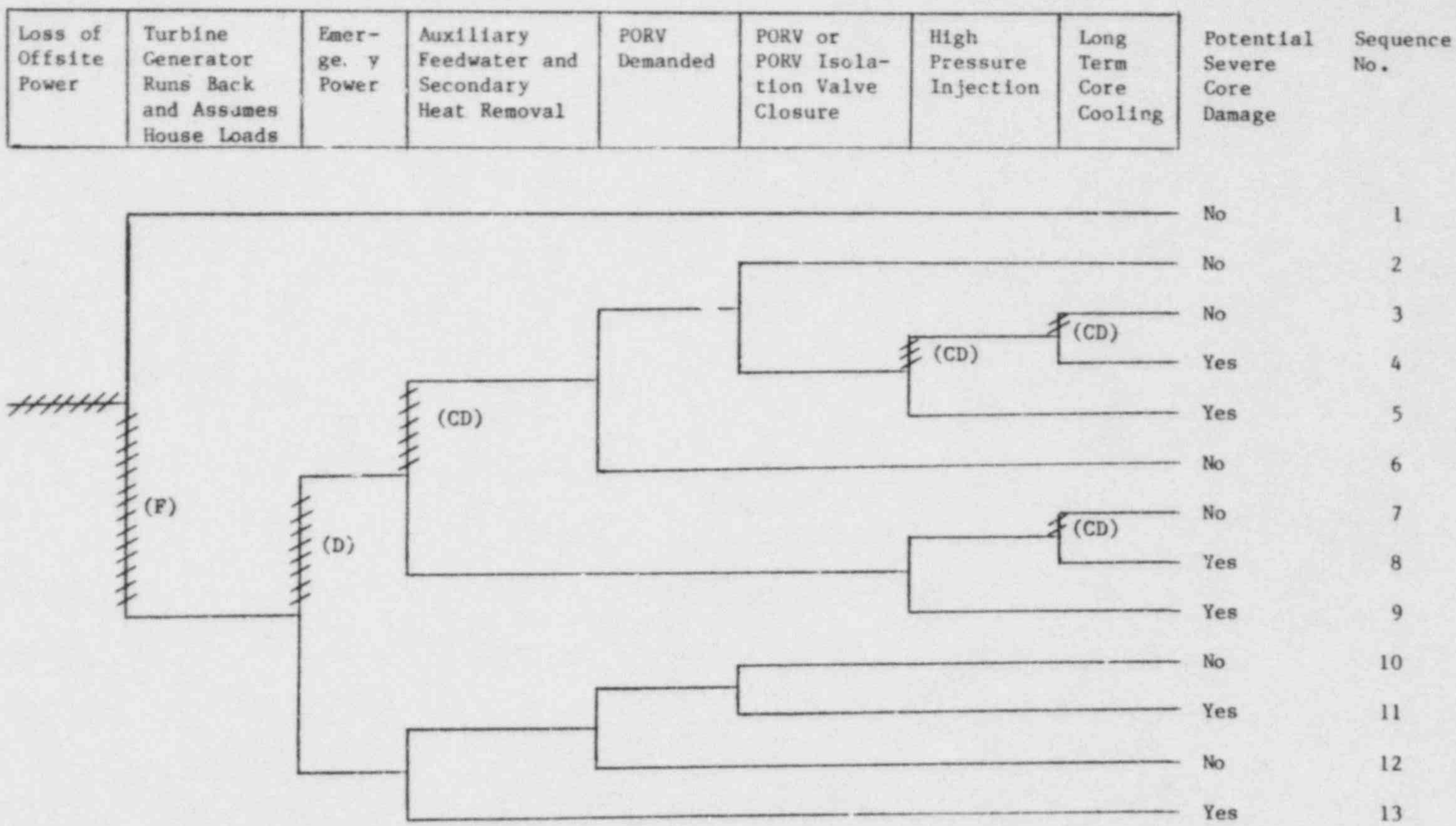
Offsite power was restored approximately 4-6 h after the loss of offsite power. The diesel was successfully started about 10 minutes into the transient but was not connected to the buses. The diesel timing relay was readjusted and incorporated into the surveillance program.

Design purpose of failed system or component:

Offsite power provided the preferred source of power to safety-related loads when the unit generator is unavailable. The diesel generators provide power to the safety-related buses when both offsite power and the unit generator are unavailable.



NSIC 167624 - Actual Occurrence for Loss of Offsite Power and Failure of a Diesel Generator to Start at Crystal River 3



NSIC 167624 - Sequence of Interest for Loss of Offsite Power and Diesel Generator Failure to Start at Crystal River 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 167624

LER NO.: 81-033

DATE OF LER: July 14, 1981

DATE OF EVENT: June 16, 1981

SYSTEM INVOLVED: Offsite power, emergency power

COMPONENT INVOLVED: Startup transformer, diesel generator

CAUSE: Lightning strike resulted in loss of the startup transformer, timing relay maladjustment resulted in diesel failure to start

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP plus diesel failure to start

REACTOR NAME: Crystal River 3

DOCKET NUMBER: 50-302

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: 4.4 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Gilbert Associates

OPERATORS: Florida Power Corporation

LOCATION: Red Level, Florida

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

TYPE OF FAILURE: Failed to start;
made inoperable:

DISCOVERY METHOD: Operational event

COMMENT: See also NSIC 167119 (Crystal River 3, 50-302, LER 81-030, July 1, 1981).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 168548

Date: August 25, 1981

Title: Switchyard Voltage Drops Below Low Limit at Rancho Seco

The failure sequence was:

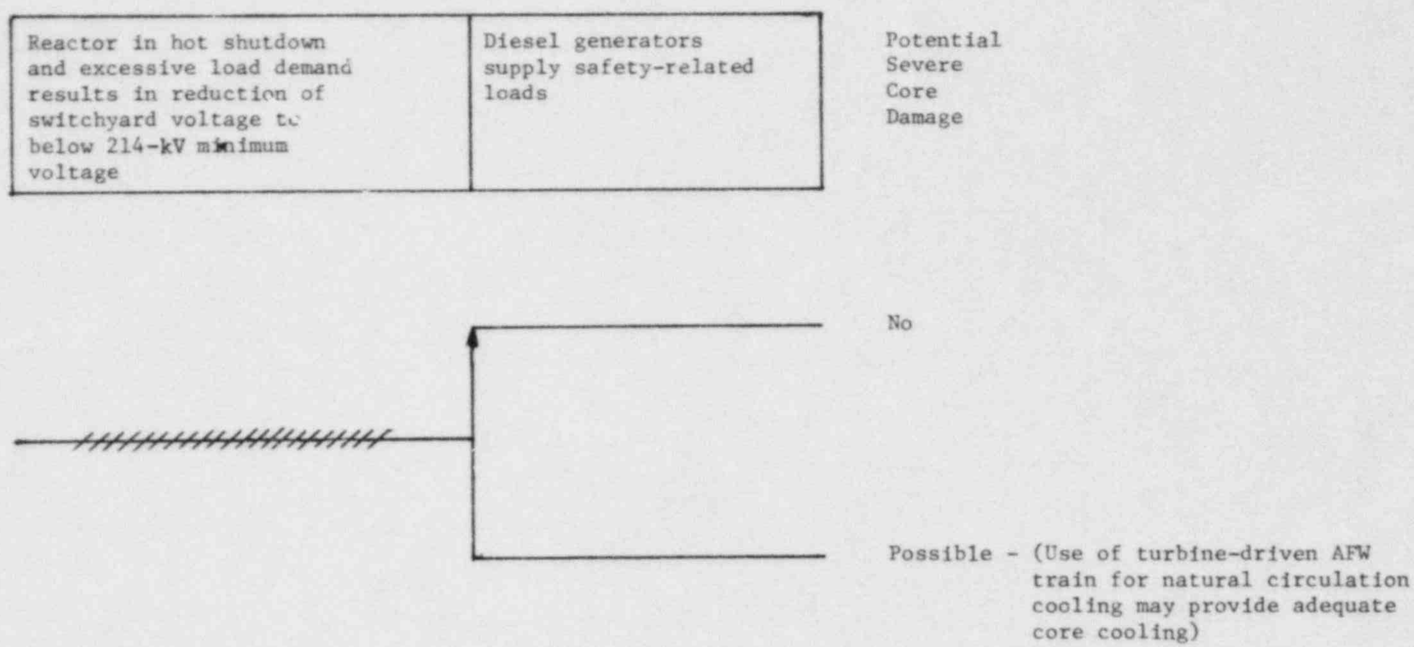
1. With the reactor in hot shutdown, excessive electrical demand resulted in a reduction in switchyard voltage to 206 kV. This is below the minimum voltage (214 kV) assumed for analysis.
2. The diesel generators were started and provided power to the safety-related buses.

Corrective action:

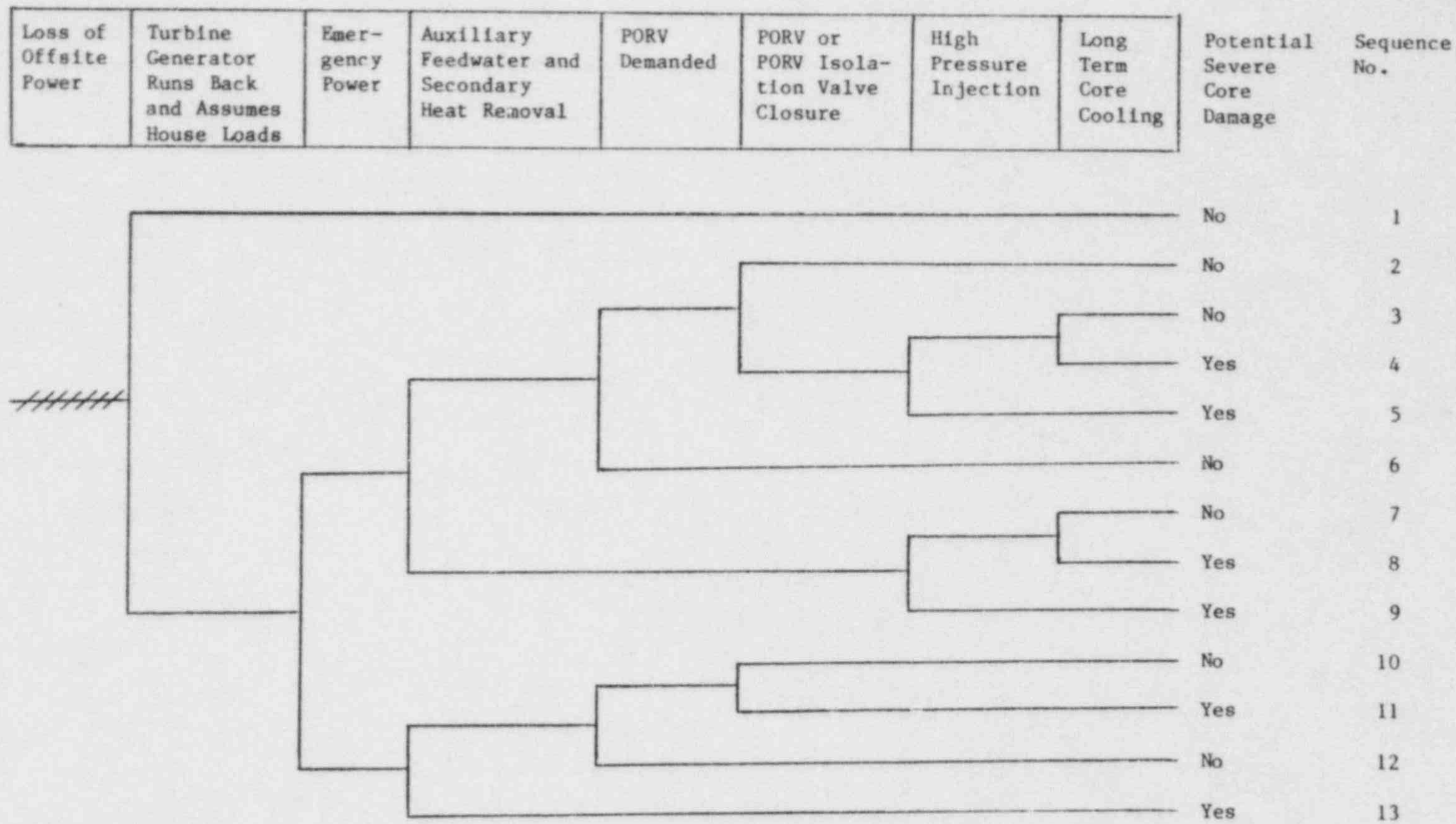
Direct switchyard voltage indications and alarms were to be installed to facilitate control room monitoring of switchyard voltages.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety-related loads. The vital buses have 2 power supplies, offsite and emergency diesel generators.



NSIC 168548 - Actual Occurrence for Effective LOOP at Rancho Seco



NSIC 168548 - Sequence of Interest for Effective LOOP at Rancho Seco

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 168548

LER NO.: 81-039

DATE OF LER: August 25, 1981

DATE OF EVENT: August 7, 1981

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Switchyard

CAUSE: Low switchyard voltages due to excessive load demand

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Effective LOOP

REACTOR NAME: Rancho Seco

DOCKET NUMBER: 50-312

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 918 MWe

REACTOR AGE: 6.9 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Sacramento Municipal Utility District

LOCATION: 25 miles SE of Sacramento, California

DURATION: N/A

PLANT OPERATING CONDITION: Shutdown

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Operational event

COMMENT: The safety feature equipment called upon was the DGs. They performed as designed and powered the vital buses. They did not experience inadequate performance or inoperability. Also see Accession 167117 (Rancho Seco, 50-312, LER 81-034, July 7, 1981).

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 168829

Date: September 14, 1981

Title: Both Safety Injection Trains Inoperable Due to Valve Problems at San Onofre 1

The failure sequence was:

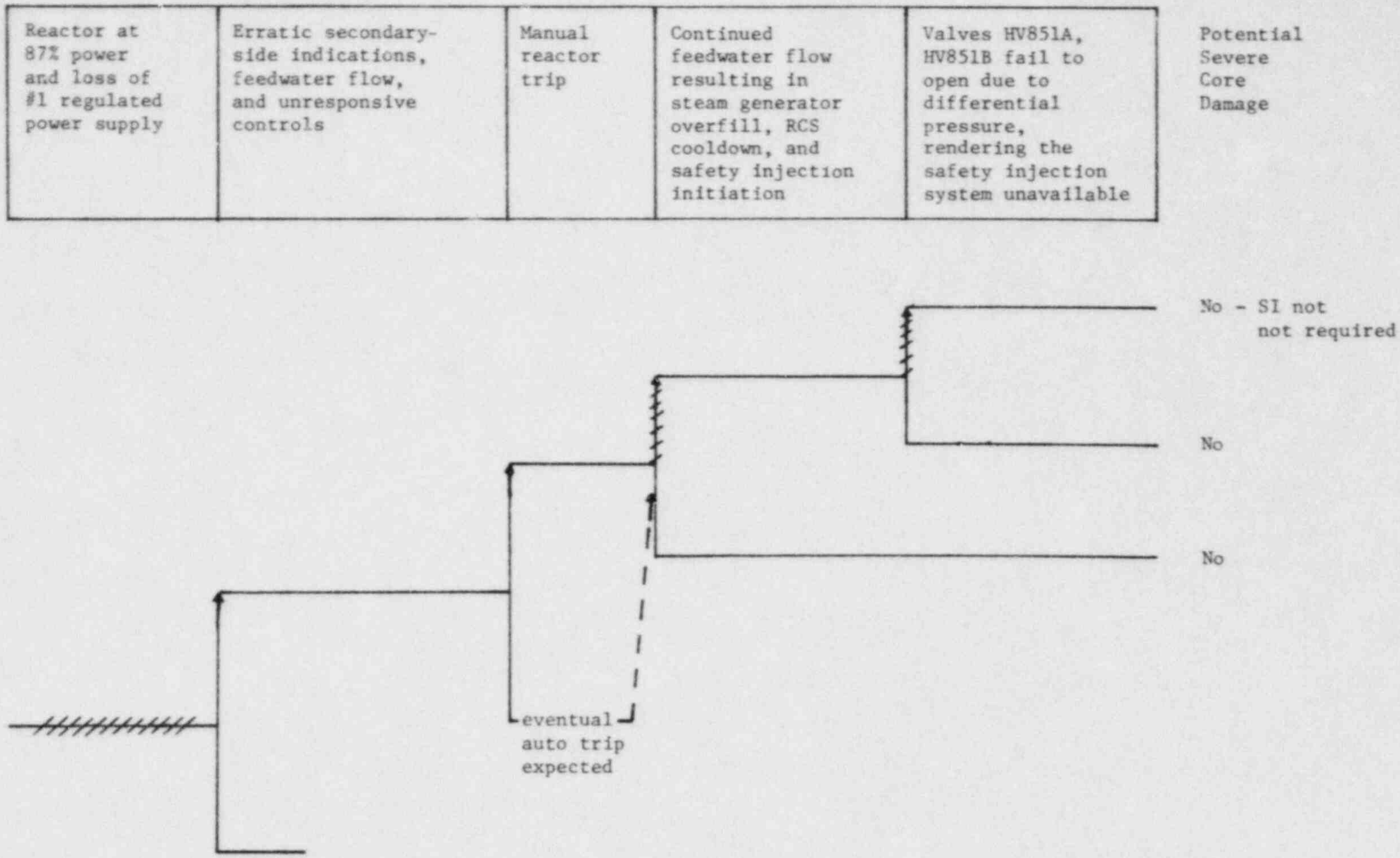
1. With the reactor at 87% power, No. 1 regulated power supply failed due to a filter choke failure.
2. This resulted in (1) failure of the dc power supplies for "A" steam generator steam flow, feed flow, and narrow range level transmitters; (2) failure of the feedwater flow and level controllers and consequent full open travel of the "A" feedwater control valve; (3) failure of all three loop steam flow and feed flow computers resulting in erratic steam and feed flow indications and erratic operation of the "B" and "C" feedwater control valves; and (4) automatic transfer of dc power supplies to the backup source providing for manual operation of all feedwater control valves.
3. The operator manually tripped the reactor because of erratic indications, erratic feedwater flow, and unresponsive controls.
4. Feedwater flow continued at varying rates (resulting in high steam generator levels) until reduced RCS pressure initiated a safety injection signal at 1735 psi.
5. Feedwater flow was terminated by automatic transfer of the feed pumps to safety injection service.
6. SI valves HV851A and HV851B failed to open as required because the differential pressure across the valves prevented their operation. Failure of these valves to open rendered the SI system inoperable. (Subsequent inspection, maintenance and testing indicated that consistent reliable valve operation could not be demonstrated with the feed pumps running).
7. RCS pressure decreased to 1700 psi and then recovered. Since feed pump discharge pressure is limited to approximately 1200 psi, no safety injection flow was required.

Corrective action:

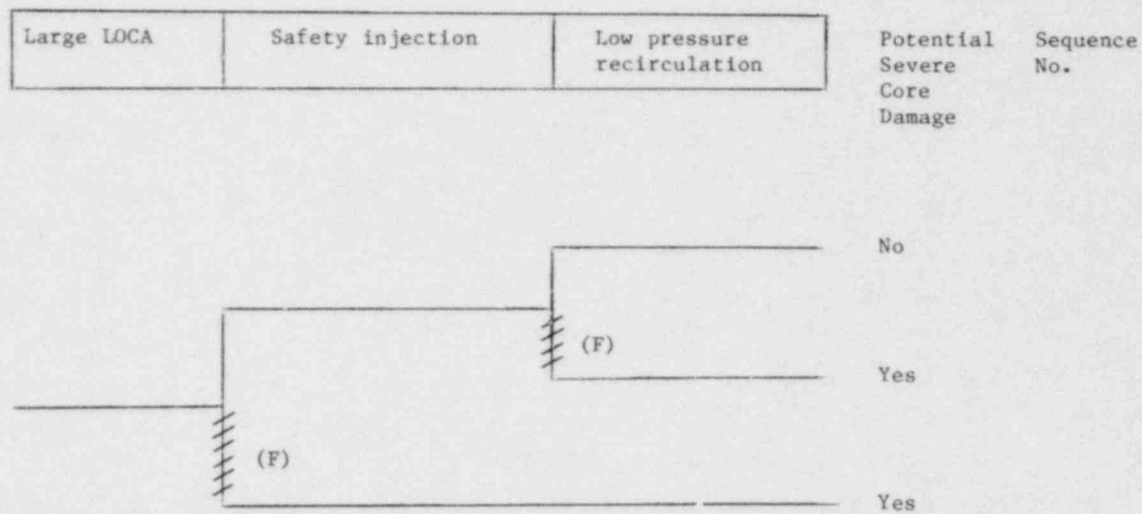
1. The No. 1 voltage regulator was first jumpered and then replaced.
2. The utility intended to implement changes requiring tripping of the feed pumps and relieving valve internal pressure prior to opening the safety injection valves.

Design purpose of failed system or component:

1. The regulated power supply provides an uninterruptible source of power for various instrumentation and controls.
2. The safety injection system provides a source of water for core cooling following a LOCA.



NSIC 168829 - Actual Occurrence for Both Safety Injection Trains Inoperable Due to Valve Problems at San Onofre 1



NSIC 168829 - Sequence of Interest for Both Safety Injection Trains Inoperable Due to Valve Problems at San Onofre 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 168829

LER NO.: 81-020

DATE OF LER: September 14, 1981

DATE OF EVENT: September 3, 1981

SYSTEM INVOLVED: Vital power, safety injection

COMPONENT INVOLVED: Regulated power supply, isolation valves

CAUSE: Filter choke failure, valves would not operate with normal differential pressure

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Reactor trip, excessive cooldown, and failure of safety injection system to operate

REACTOR NAME: San Onofre 1

DOCKET NUMBER: 50-206

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 436 MWe

REACTOR AGE: 14.2 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Southern California Edison

LOCATION: 5 miles south of San Clemente, California

DURATION: 871 days

PLANT OPERATING CONDITION: 87% power

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Operational event

COMMENT: NSIC 168830 provides additional information. Valve operability was satisfactory during installation testing prior to plant startup on April 11, 1977. Subsequently the valve surfaces degraded until they failed in 1981. There was no demand for valve operation in-service or full differential testing between installation and September 1981.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 169042

Date: August 13, 1981

Title: Degraded High Pressure Injection System During a Loss of Offsite Power at Arkansas Nuclear Unit 1

The failure sequence was:

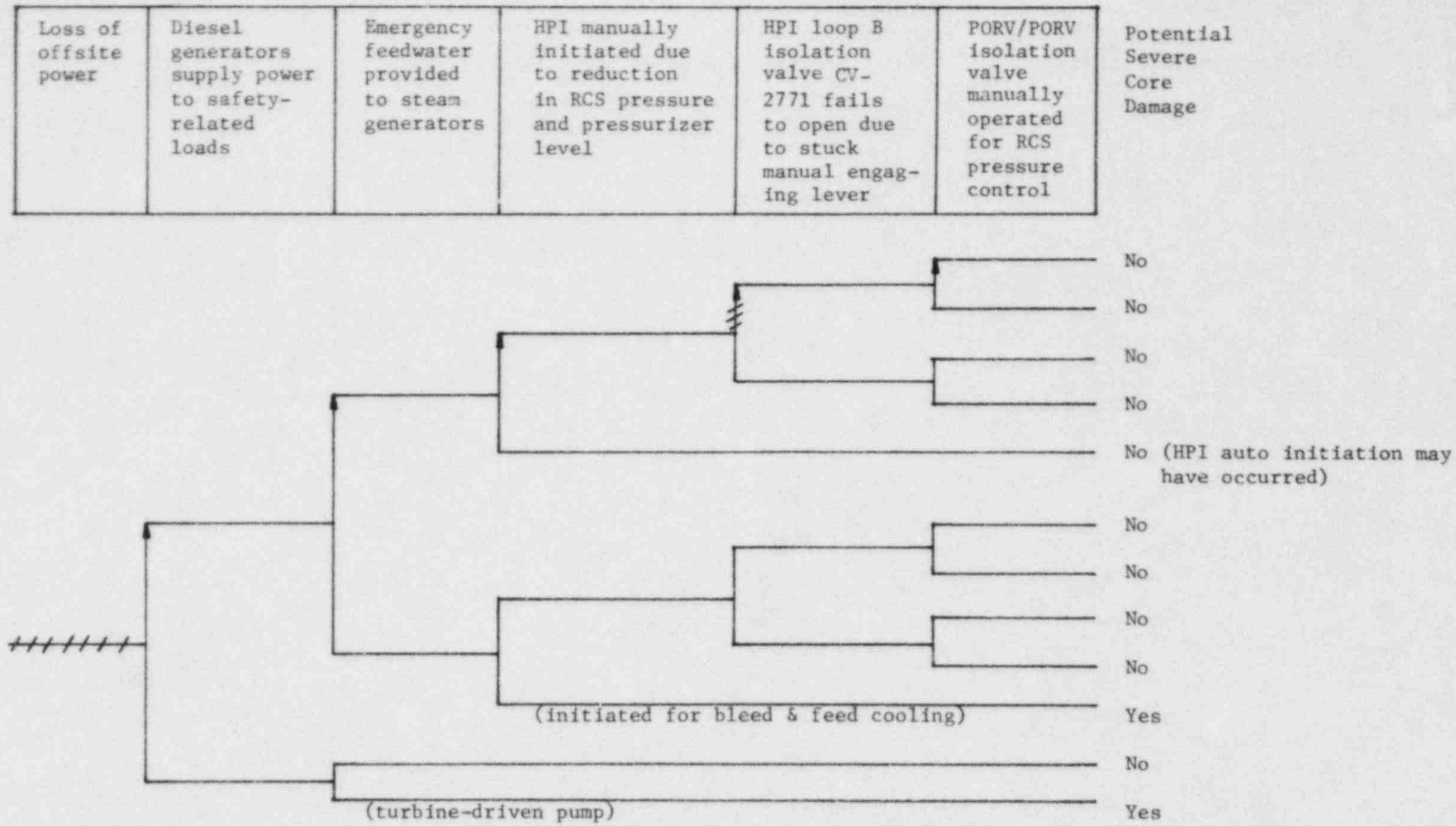
1. With the reactor at full power with atmospheric dump valves isolated due to vibration and failure to close problems, tornado activity resulted in the sequential loss of four of five offsite power lines. Protective relaying disconnected the remaining offsite power line from the bus tie autotransformer. (Offsite power was available through manual connection from the 161 kV transmission system).
2. Because of a prior tornado warning, diesel generator No. 1 had been started before the LOOP. Diesel generator No. 2 started and both provided power to safety-related loads.
3. Both emergency feedwater pumps provided water for steam generator cooling.
4. High-pressure injection was manually initiated to recover system pressure following the trip. HPI loop B isolation valve CV-1227 failed to open on HPI initiation. The valve failed to open because of a stuck manual engaging lever.
5. Because of continued HPI flow, pressurizer level increased to above maximum level indication. The pressurizer PORV and its associated block valve were opened and cycled to ensure the pressurizer was not solid and to reduce RCS pressure.
6. The process and trend computers were unavailable during the event because of incorrectly positioned power switches.
7. On the same day, but prior to the LOOP, maintenance and testing had been completed on EFW pump steam admission valve CV-2667. The valve did not fully close during testing. Steam admission valves also did not fully close during recovery from the LOOP. Also on the same day, quarterly tests of the HPI loop isolation valves had been performed and the valves declared operable. Forty-two minutes prior to the LOOP, turbine-driven EFW pump P7A had been returned to service following pump gland seal repacking.

Corrective action:

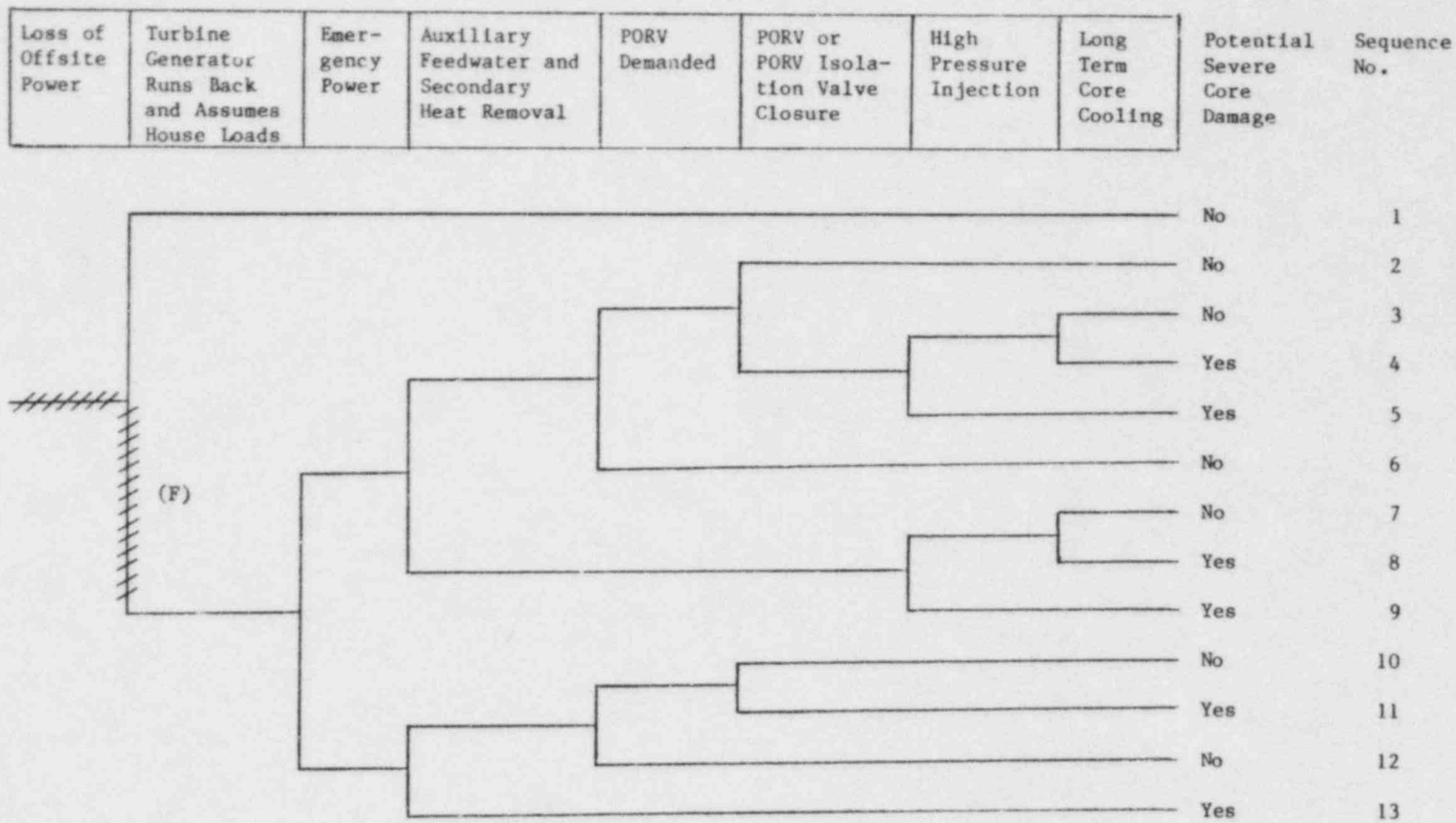
Offsite power was restored via startup transformer ST-2. The stuck manual engaging lever for HPI valve CV-1227 was freed. Procedures were revised to require stroking of such valves from the control room after manual operation.

Design purpose of failed system or component:

Offsite power provides the preferred source of power to safety related loads when the unit generator is not available. The HPI system provides water to the RCS for core cooling during off-normal situations.



NSIC 169042 - Actual Occurrence for Degraded HPI System During Loop at Arkansas Unit 1



NSIC 169042 - Sequence of Interest for Degraded HPI System During LOOP at Arkansas Unit 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 169042

LER NO.: 80-013 Rev. 1

DATE OF LER: August 13, 1981

DATE OF EVENT: April 7, 1980

SYSTEM INVOLVED: Offsite power, high-pressure injection

COMPONENT INVOLVED: Transmission lines, valve

CAUSE: Tornado damage to transmission lines, failure of valve to open due to stuck manual engaging lever

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: LOOP and failure of HPI valve to open

REACTOR NAME: Arkansas Nuclear Unit 1

DOCKET NUMBER: 50-313

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 850 MWe

REACTOR AGE: 5.7 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Arkansas Power and Light

LOCATION: 6 miles NW of Russellville, Arkansas

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

TYPE OF FAILURE: Made inoperable;
failed to open

DISCOVERY METHOD: operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 169587

Date: November 4, 1981

Title: Safety Injection Path to RCS Obstructed at Turkey Point 4

The failure sequence was:

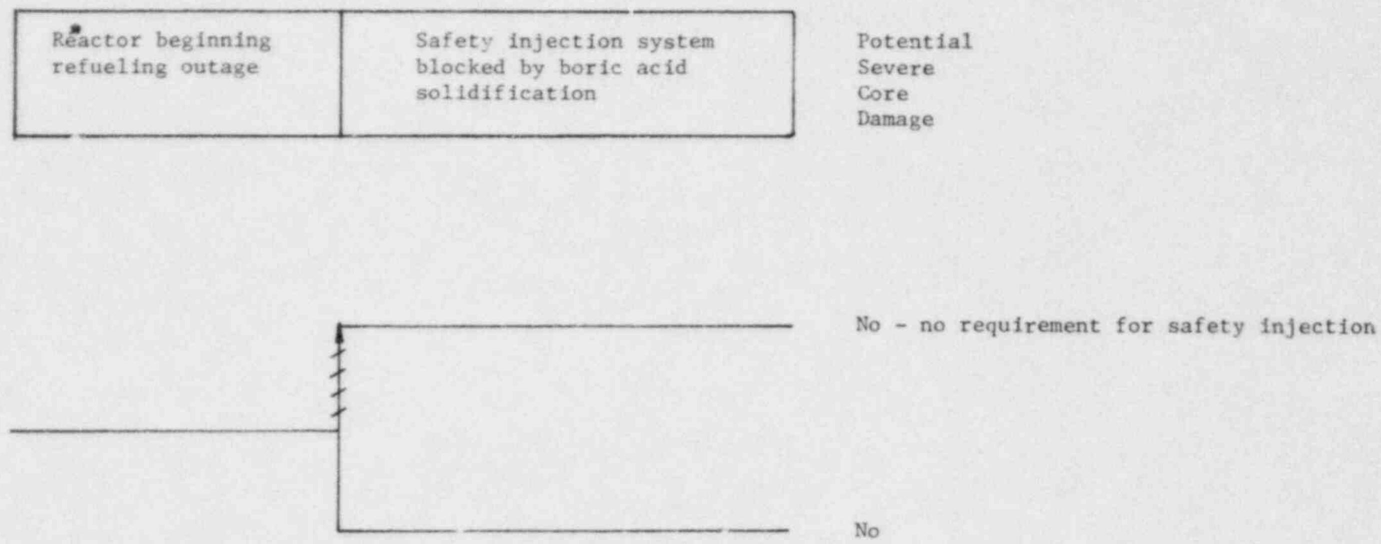
1. At the beginning of a refueling outage and while conducting a routine periodic test during partial draining of the RCS, the flow path to the RCS associated with the boron injection tank (BIT) was found obstructed.
2. The apparent cause of the obstruction was reduced temperature resulting from missing insulation in the vicinity of a 4-in. tee at the inlet to the BIT.
3. As a result of the obstruction, the safety injection flow path from all high head safety injection pumps to the RCS cold legs were blocked. A flow path to the hot legs was verified to be operable by establishing flow.

Corrective action:

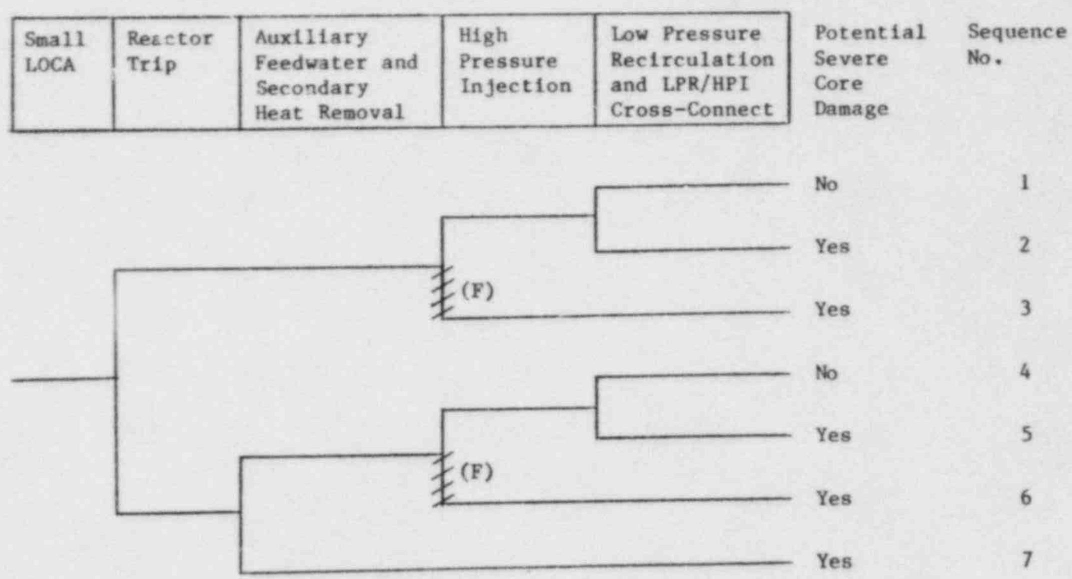
1. The flow path was reestablished by alternating pump operation and raising the thermostat setting for the related heat tracing.

Design purpose of failed system or component:

The safety injection system provides high-pressure replacement water to the RCS during a LOCA and steam line break. This system also provides boron injection for steam break return to power considerations.



NSIC 169587 - Actual Occurrence for Safety Injection Path to RCS Obstructed at Turkey Point 4



NSIC 169587 - Sequence of Interest for Safety Injection Path to RCS Obstructed at Turkey Point 4

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 169587

LER NO.: 81-011

DATE OF LER: November 4, 1981

DATE OF EVENT: October 21, 1981

SYSTEM INVOLVED: Safety injection

COMPONENT INVOLVED: 4-in. piping tee

CAUSE: Boric acid solidification blocked tee

SEQUENCE OF INTEREST: LOCA

ACTUAL OCCURRENCE: Unavailability of high head safety injection paths

REACTOR NAME: Turkey Point 4

DOCKET NUMBER: 50-251

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: 8.4 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power & Light

LOCATION: 25 miles south of Miami, Florida

DURATION: 360 h (estimated)

PLANT OPERATING CONDITION: Entering refueling shutdown

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 170098

Date: November 19, 1981

Title: Two Boron Injection Tank Inlet Valves Fail to Open at Salem 1

The failure sequence was:

1. With the reactor in hot shutdown following a reactor trip, replacement of inverter 1A cabinet fan unit fuses was in progress.
2. A voltage transient induced in the inverter control wiring resulted in loss of the inverter and loss of vital bus 1A.
3. An inadvertent safety injection occurred due to an apparent high steam flow indication caused by loss of the vital bus.
4. During the safety injection, boron injection tank inlet valves 1SJ4 and 1SJ5 failed to open due to boric acid crystallization on the valve stems.

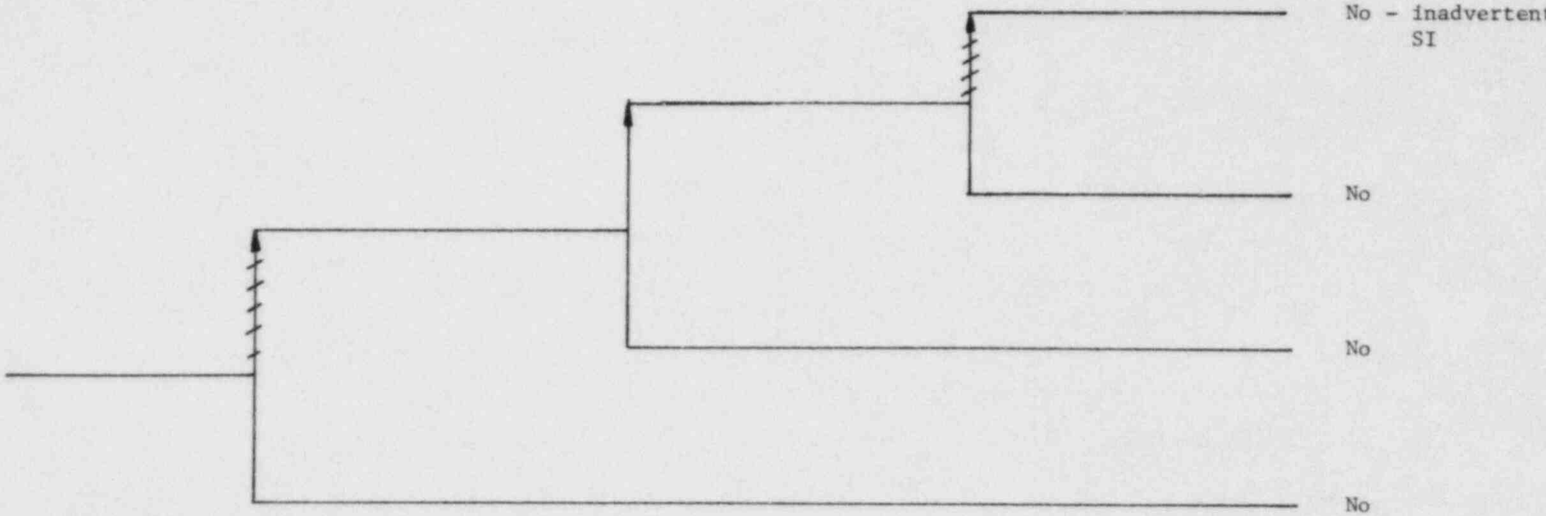
Corrective action:

1. The vital bus was removed from the inverter and connected to its alternate Solatron power source.
2. The BIT inlet valves were cleaned and stroked. The valve opeator torque settings were increased.

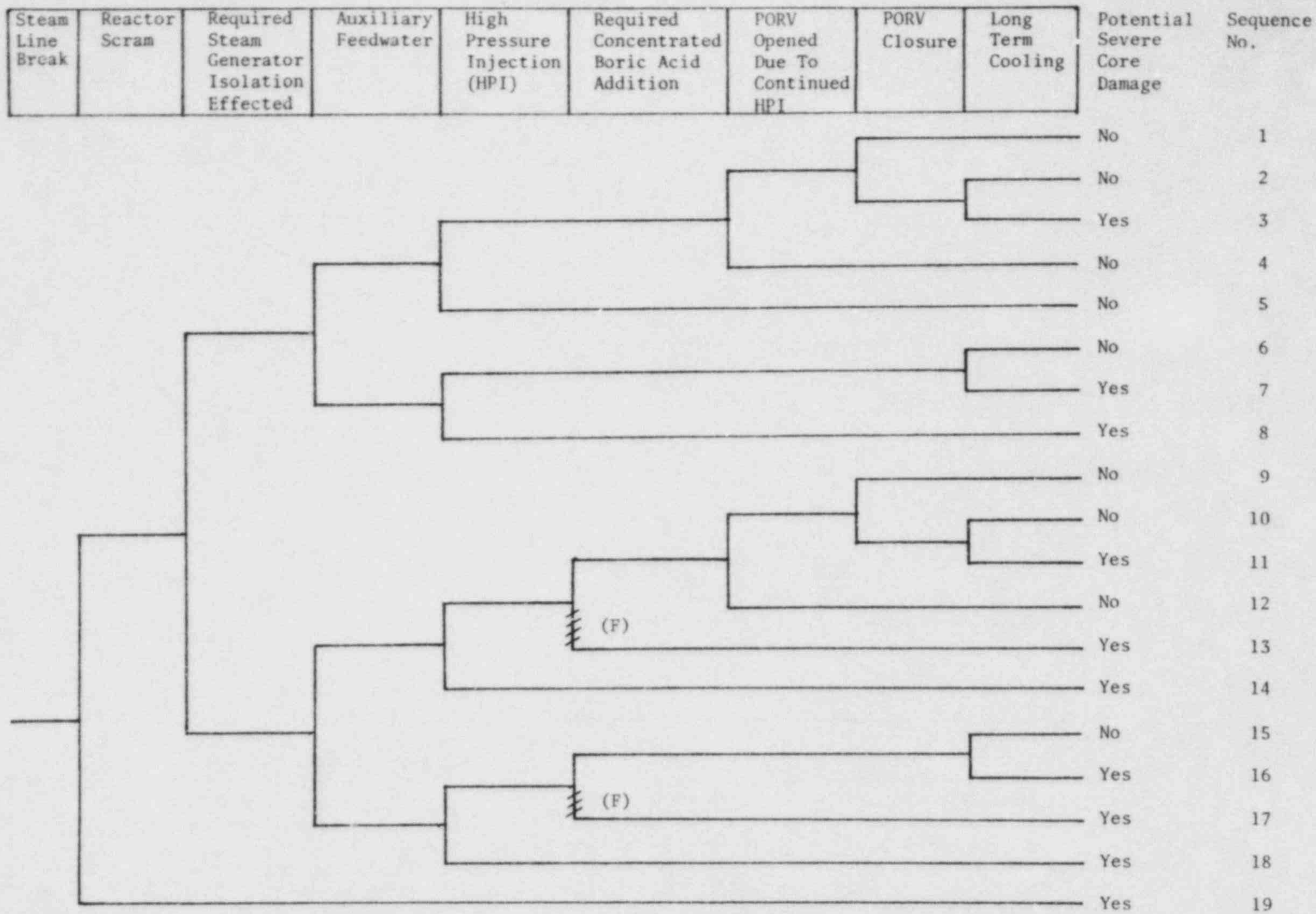
Design purpose of failed system or component:

The BIT provides a source of high concentration boric acid to mitigate a potential return-to-power condition following a steam line break.

Reactor in hot shutdown following a reactor trip	During replacement of inverter 1A cabinet fan unit fuses, a voltage transient was induced in the inverter control wiring, resulting in loss of 1A inverter and loss of the 1A vital bus	Safety injection occurs due to high steam flow indication caused by loss of vital bus	Boron injection tank inlet valves 1SJ4 and 1SJ5 fail to open fully due to boron crystal accumulations on the valves	Potential severe core damage



NSIC 170098 - Actual Occurrence for Two Boron Injection Tank Inlet Valves Fail to Open at Salem 1



NSIC 170098 - Sequence of Interest for Two Boron Injection Tank Inlet Valves Fail to Open at Salem 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 170098

LER NO.: 81-097

DATE OF LER: November 19, 1981

DATE OF EVENT: November 6, 1981

SYSTEM INVOLVED: Safety injection

COMPONENT INVOLVED: BIT inlet valves

CAUSE: Boric acid crystal buildup around valve stems

SEQUENCE OF INTEREST: Main steam line break (MSLB)

ACTUAL OCCURRENCE: BIT inlet valves failed to open on demand from inadvertent SI signal

REACTOR NAME: Salem 1

DOCKET NUMBER: 50-272

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1090 MWe

REACTOR AGE: 4.9 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Public Service Gas and Electric

OPERATORS: Public Service Gas and Electric

LOCATION: 20 miles south of Wilmington, Delaware

DURATION: 360 hours (estimated)

PLANT OPERATING CONDITION: Hot shutdown following trip

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Operational event

COMMENT: 1. Valves may have opened sufficiently to permit boron injection for MSLB mitigation. However, the utility reported BIT dilution only after stroking the valves during subsequent maintenance.

2. Additional information: LERS 81-111/03L, 81-112/03L, 81-110/03L.

3. During the inadvertent SI, fan coil unit No. 11 failed to start in the slow speed mode due to a failure to close of the automatic trip shutter on the low speed circuit breaker when the breaker was racked in. This resulted in the breaker tripping every time it was reset.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 170199

Date: October 27, 1981

Title: Unavailability of Diesel Generator and Component Cooling Water Train at Kewaunee

The failure sequence was:

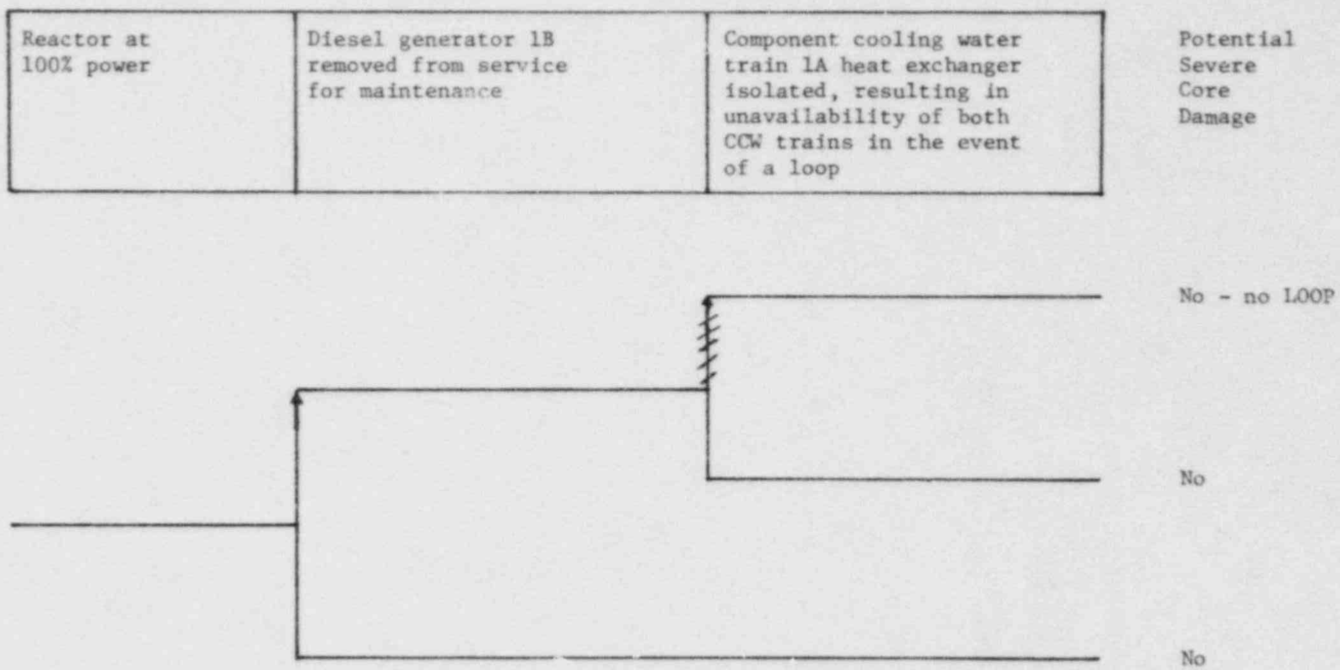
1. With the reactor at 100% power, diesel generator 1B was removed from service for maintenance.
2. Later in the shift, the 1A component cooling water heat exchanger was isolated and the supply motor operated valve breaker opened. This resulted in unavailability of both component cooling water trains in the event a LOOP.

Corrective action:

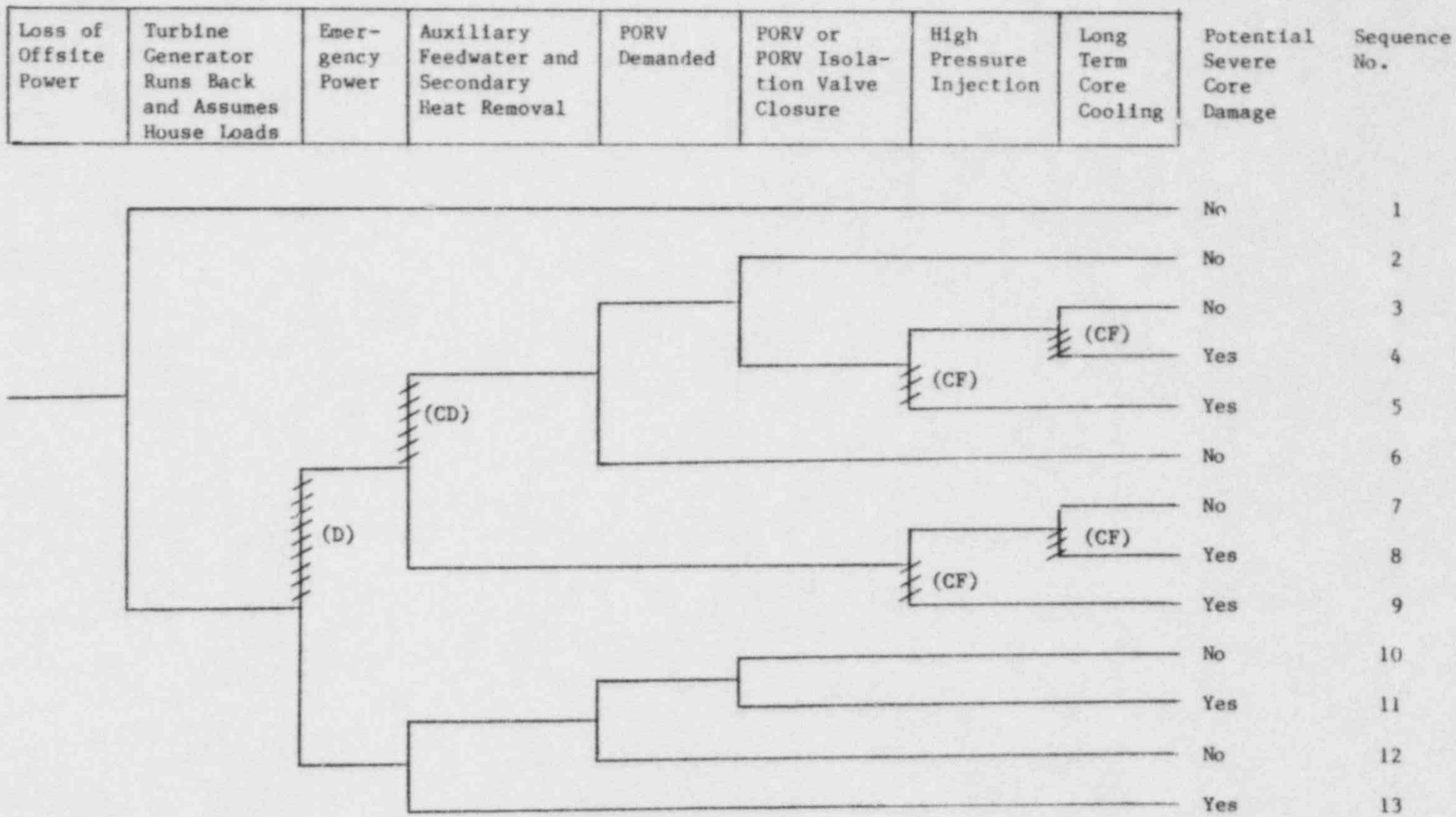
The unavailability was discovered when service water was restored.

Design purpose of failed system or component:

The diesel generator provides power to safety-related loads when the unit generator and offsite power sources are unavailable. The component cooling water system provides cooling water for the RC pumps, RHR heat exchangers, and safety injection pumps as well as other equipment.



NSIC 170199 - Actual Occurrence for Unavailability of Diesel Generator and CCW Train at Kewaunee



NSIC 170199 - Sequence of Interest for Unavailability of Diesel Generator and Component Cooling Water Train at Kewaunee

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 170199

LER NO.: 81-033

DATE OF LER: October 27, 1981

DATE OF EVENT: October 16, 1981

SYSTEM INVOLVED: Component cooiing water and emergency power

COMPONENT INVOLVED: CCW heat exchanger and diesel generator

CAUSE: Operator error

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Concurrent unavailability of diesel generator and alternate CCW train

REACTOR NAME: Kewaunee

DOCKET NUMBER: 305

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 535 MWe

REACTOR AGE: 7.6 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Flour Power

OPERATORS: Wisconsin Public Service Co.

LOCATION: 27 miles east of Green Bay, Wisconsin

DURATION: 50 min

PLANT OPERATING CONDITION: Full power

TYPE OF FAILURE: Made inoperable

DISCOVERY METHOD: Operator observation

COMMENT: Upon the concurrent occurrence of LOOP and SI, manual action would have been required to supply service water to the 1A heat exchanger to support post-LOCA recirculation operation.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 171202

Date: December 3, 1981

Title: Unavailability of Emergency Power System at Turkey Point 3

The failure sequence was:

1. With Units 3 and 4 in cold shutdown, maintenance was being performed on the "B" emergency diesel generator.
2. Diesel generator "A" was started and subsequently stopped due to the inability of its start motor to disengage (a result of a broken diaphragm in the starting air solenoid).
3. This rendered both diesel generators inoperable for approximately 30 min.

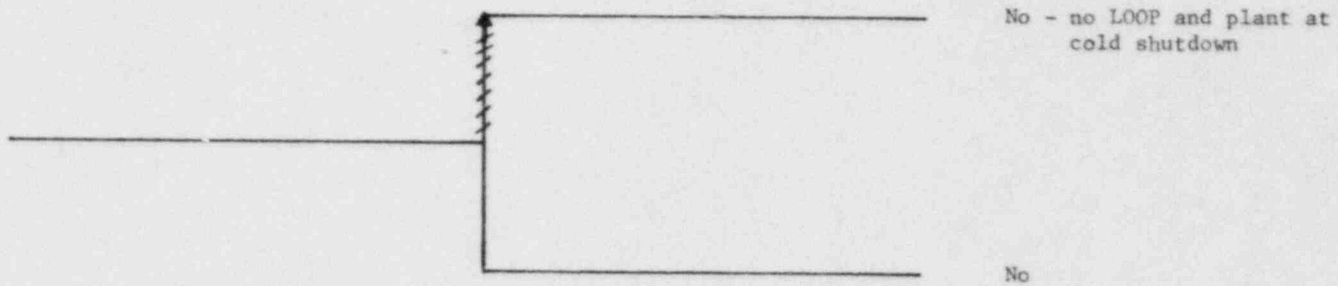
Corrective action:

1. The solenoid was replaced and the "A" diesel generator tested and returned to service.

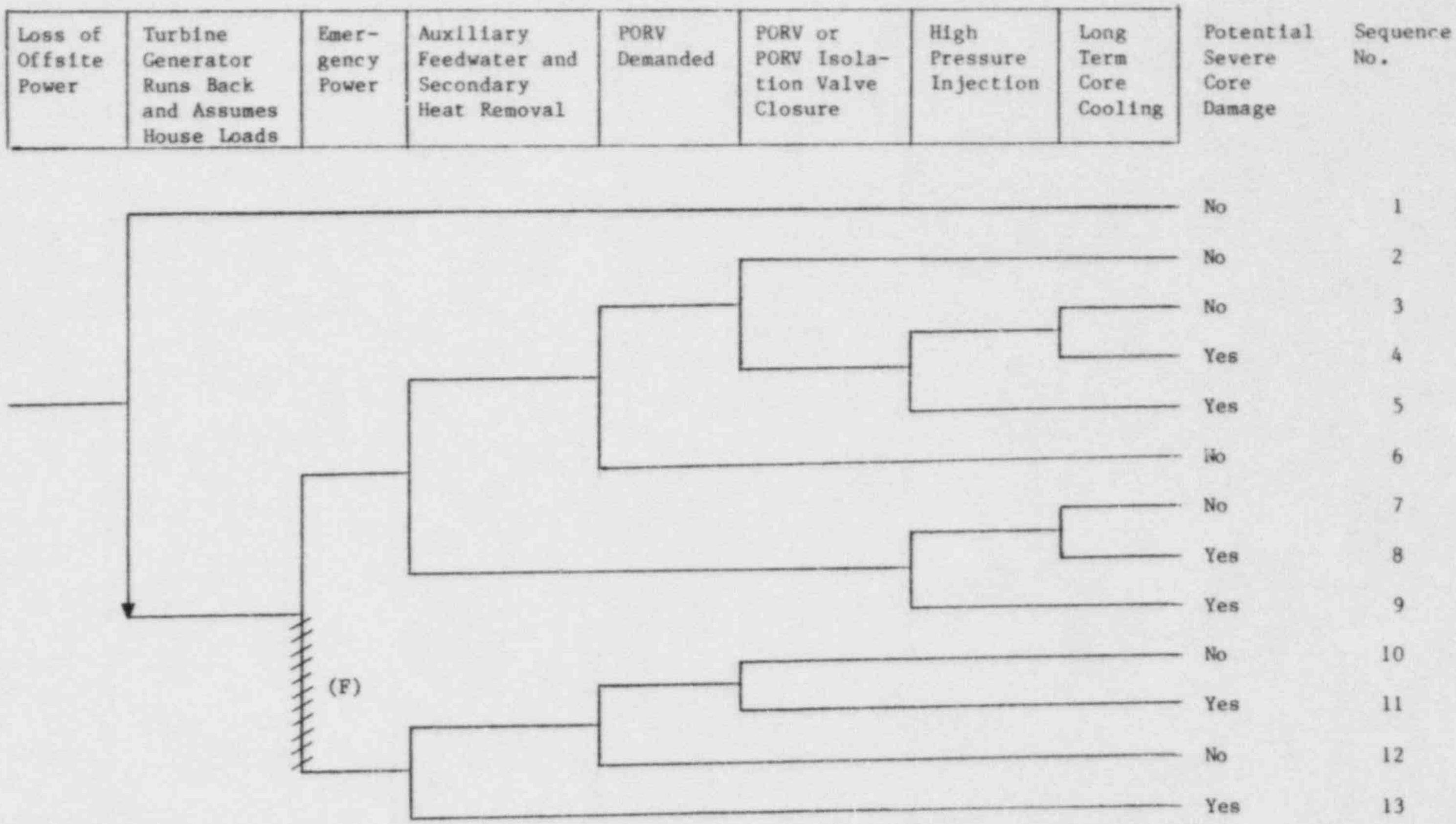
Design purpose of failed system or component:

The diesel generators provide power for safety-related loads when the unit generators and offsite power sources are unavailable.

Reactor in cold shutdown and DG "B" out of service for maintenance	DG "A" fails to run due to inability of start motor to disengage, a result of a failed starting air solenoid diaphragm	Potential Severe Core Damage



NSIC 171202 - Actual Occurrence for Unavailability of Emergency Power System at Turkey Point 3



NSIC 171202 - Sequence of Interest for Unavailability of Emergency Power System at Turkey Point 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 171202

LER NO.: 81-015

DATE OF LER: December 3, 1981

DATE OF EVENT: November 12, 1981

SYSTEM INVOLVED: Emergency power system

COMPONENT INVOLVED: Both diesel generators

CAUSE: Component failure on one diesel while other was out for maintenance

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Unavailability of two diesel generators

REACTOR NAME: Turkey Point 3

DOCKET NUMBER: 50-250

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 693 MWe

REACTOR AGE: 9.1 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Florida Power and Light

LOCATION: 25 miles south of Miami, Florida

DURATION: 30 min

PLANT OPERATING CONDITION: Cold shutdown

TYPE OF FAILURE: Failed to start;
made inoperable

DISCOVERY METHOD: Surveillance test

COMMENT: Offsite power available; DG restored in 30 min.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 171667

Date: December 15, 1981

Title: Loss of Two Instrument Buses at Davis-Besse 1

<u>The failure sequence was:</u>

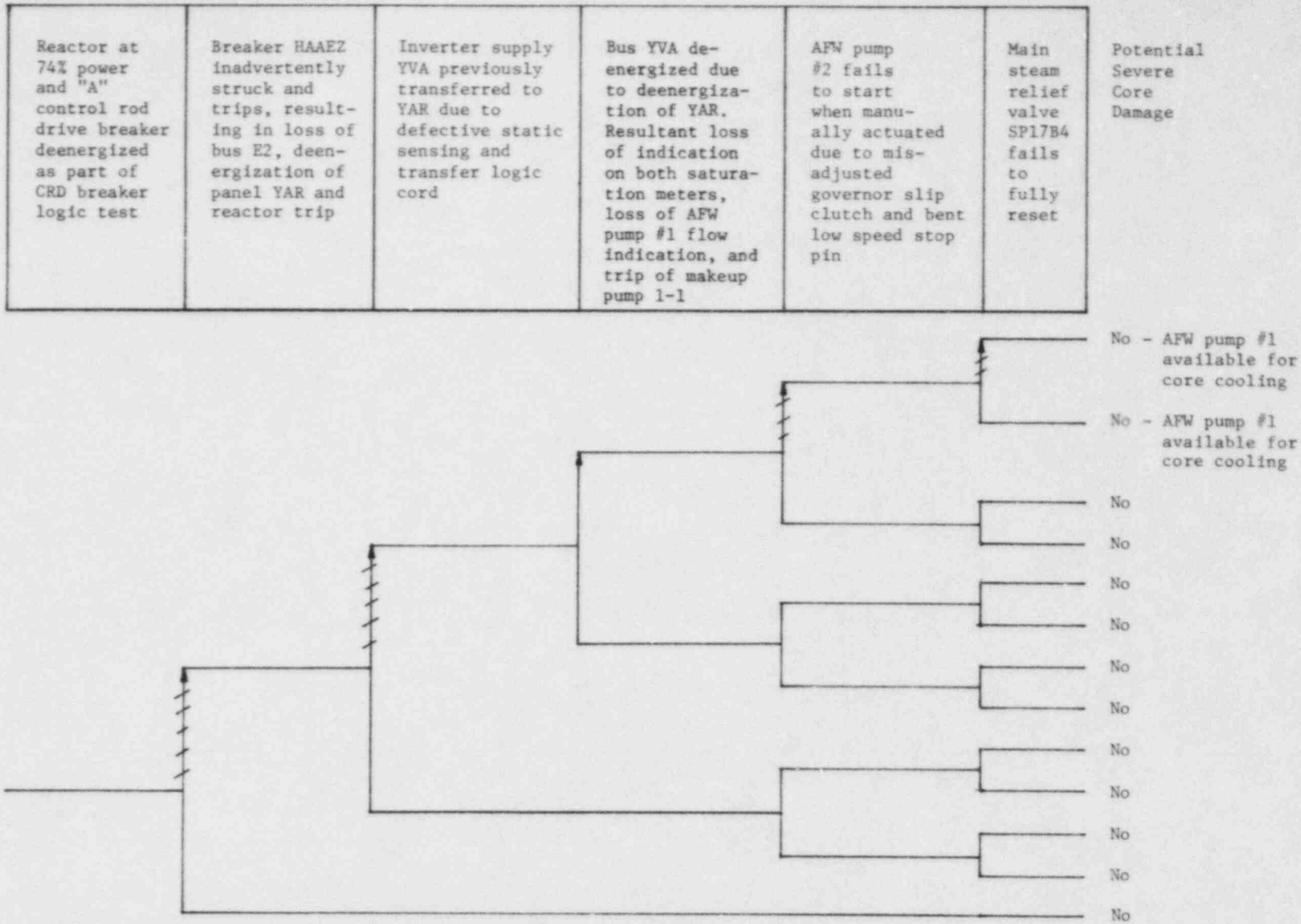
1. With the reactor at 74% power, "A" control rod drive breaker was deenergized as part of a Control Rod Drive Breaker Logic Test.
2. Construction personnel inadvertently caused a mechanical shock to nonessential breaker HAAEZ which caused a ground fault relay to activate and trip the breaker, resulting in loss of bus E2. This caused loss of the control rod drive Inductrol Power Supply which, in conjunction with the "A" CRD breaker being open for testing caused a loss of power to the control rods and a reactor trip.
3. The loss of bus E2 deenergized the power supply (E23) to the regulated instrumentation distribution panel YAR.
4. The inverter supply (YVA) for uninterruptable instrumentation distribution panel YAU had previously transferred to its alternate supply YAR due to a defective static sensing and transfer logic card in the static switch. Because of this, YAU was also deenergized when YAR deenergized. Panel YAU provides power to the following:
    - Integrated control system Y-bus
    - Nonnuclear instrumentation Y-bus
    - Control rod drive system bus 2
    - Station annunciator
    - Communication system A
    - CRD redundant position indicator
    - Computer peripherals
    - Main feed pump turbine 1-1 control
    - Reactor coolant pump 1-1-1 and 1-2-2 speed switches
5. Loss of YAU resulted in loss of indication on both saturation meters (due to a requirement for both saturation meters to be powered from both noninterruptible power supplies YAU and YBU), loss of auxiliary feed pump 1 flow indication, and trip of makeup pump 1 (when the makeup tank level time delay relay powered from YAU deenergized). Makeup pump 1-2 was in operation at the time.
6. During recovery, the auxiliary feedwater pumps were manually actuated using the steam and feedwater rupture control system and AFW pump 2 did not respond properly due to a maladjusted governor slip clutch and bent low speed stop pin.
7. Main steam safety valve SP17B4 opened and failed to properly reseat and was gagged.
8. Approximately 13 minutes after unit trip, bus E2 was reenergized from its alternate source by manually connecting bus E2 to its alternate feed on the B bus.

Corrective action:

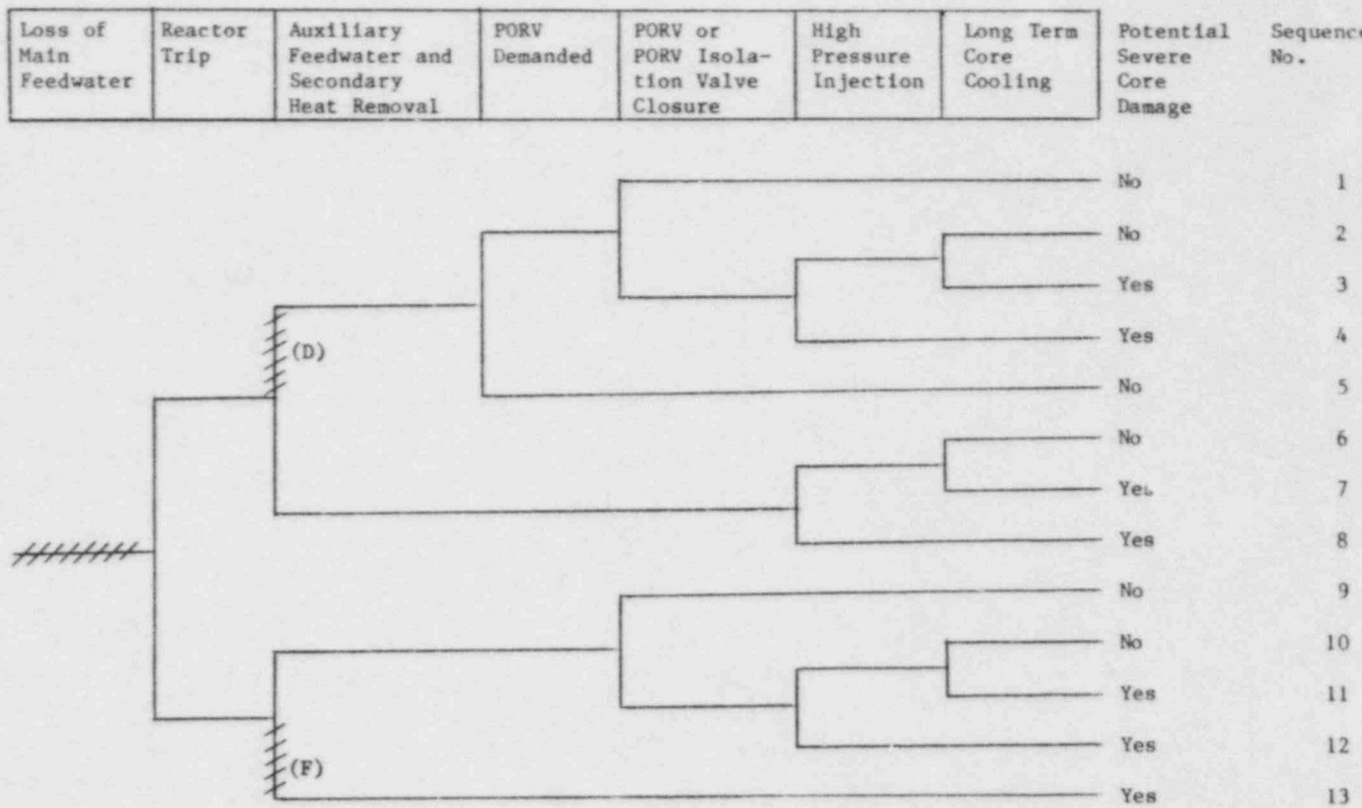
1. All type ITH relays at the unit were adjusted to as wide a gap as possible consistent with proper relay operation.
2. The defective static sensing and transfer logic ca.d in inverter YVA was replaced and YAU was returned to its normal power supply.
3. The 2 AFW pump turbine governor slip clutch was adjusted and the low speed stop pin straightened. The governor was subsequently replaced with a space.
4. Saturation margin meter wiring was modified to po‑er the meters from one power supply each.

Design purpose of failed system or component:

1. Uninterruptible buses provide a continuous source of power to instrumentation and control circuity that cannot tolerate short-term power interruptions (for example, during diesel start and loading following a LOOP).
2. The auxiliary feedwater pumps provide cooling water to the steam generators when the main feedwater system is unavailable.
3. The makeup pumps provide RCS makeup during normal operation (separate HPI pumps are provided on Davis-Besse).
4. Safety valves provide overpressure protection.



NSIC 171667 - Actual Occurrence for Loss of Two Instrument Buses at Davis-Besse 1



NSIC 171667 - Sequence of Interest for Loss of Two Instrument Buses at Davis-Besse 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 171667

LER NO.: 81-037 Rev. 1

DATE OF LER: December 15, 1981

DATE OF EVENT: June 24, 1981

SYSTEM INVOLVED: Vital power, auxiliary feedwater

COMPONENT INVOLVED: Breaker, inverter, auxiliary feedwater pumps, safety valve

CAUSE: Mechanical shock to breaker, faulted static sensing and transfer logic card

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Loss of vital bus

REACTOR NAME: Davis-Besse 1

DOCKET NUMBER: 50-346

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 3.9 years

VENDOR: Babcock & Wilcox

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Toledo Edison

LOCATION: 21 miles east of Toledo, Ohio

DURATION: N/A

PLANT OPERATING CONDITION: 74% power

TYPE OF FAILURE: Inadequate performance;
failed to start;
made inoperable

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 171700

Date: December 17, 1981

Title: Unavailability of Emergency Power System at San Onofre 1

<u>The failure sequence was:</u>

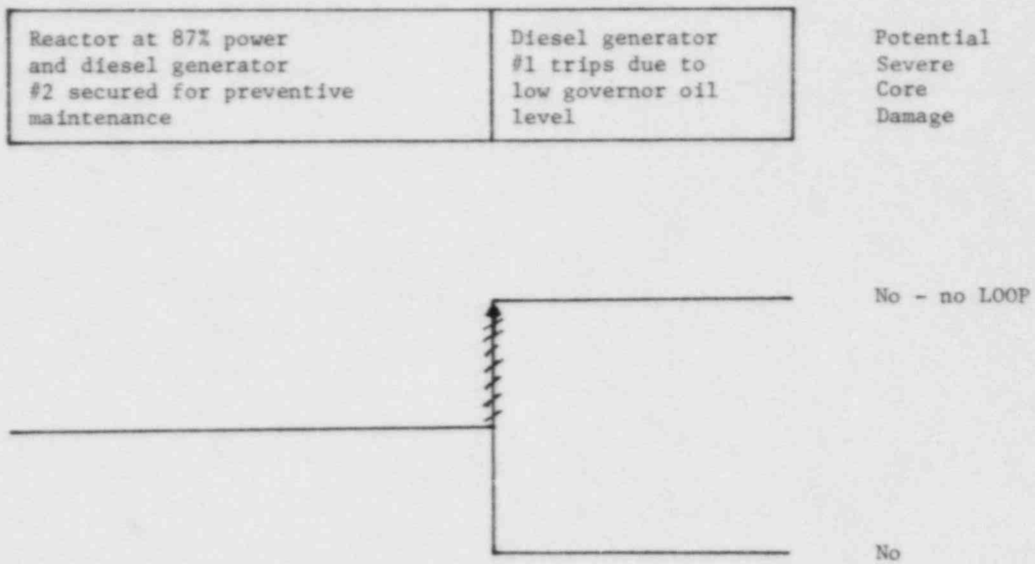
1. With the reactor at 87% and diesel generator No. 2 secured for preventive maintenance, diesel generator No. 1 tripped on overspeed because of low governor oil level. Adequate governor oil level had not been established following draining oil from the governor during the last preventive maintenance.

<u>Corrective action:</u>

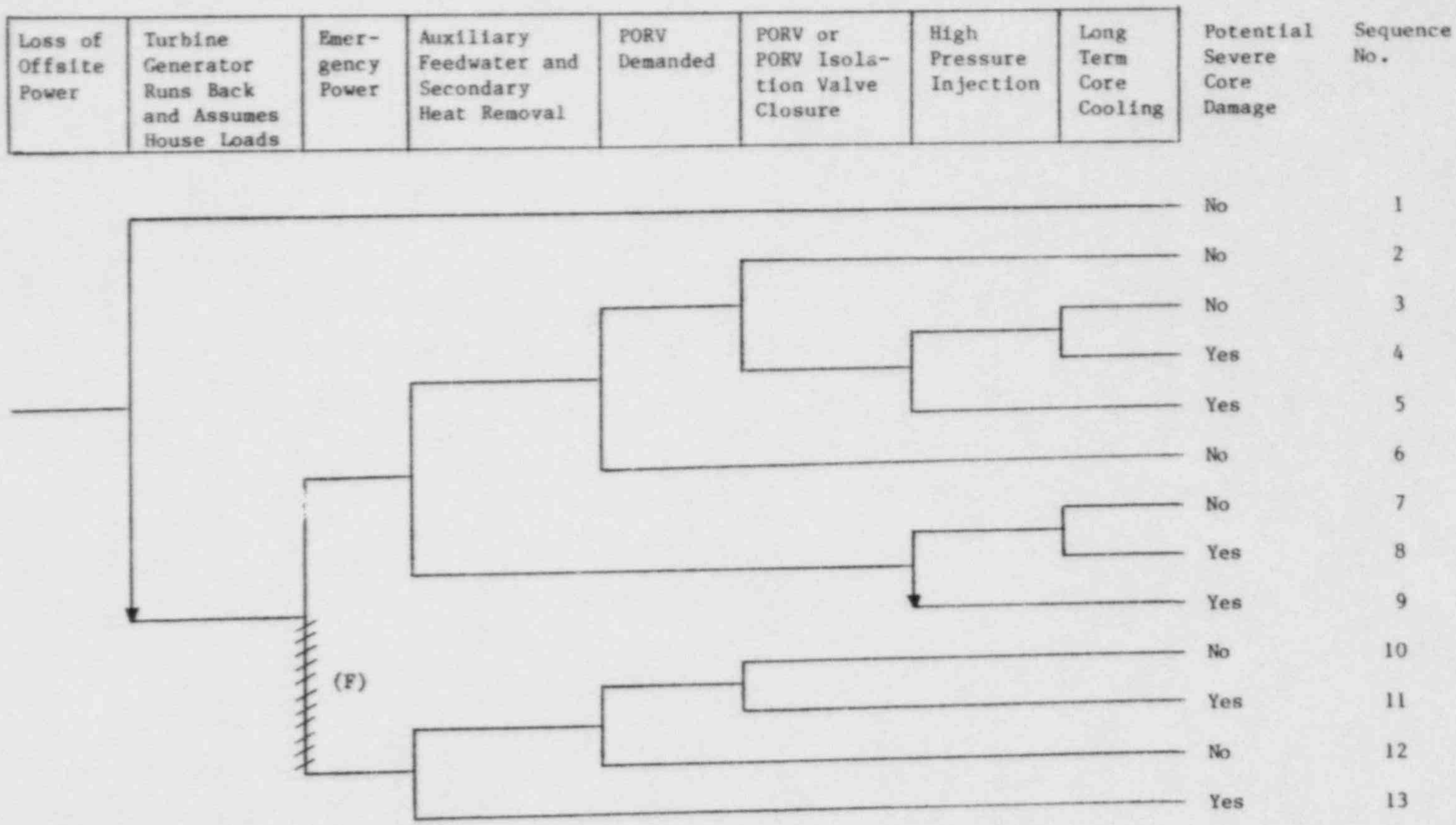
1. The No. 2 diesel generator was returned to service. Station procedures were to be revised to ensure confirmation of governor oil level prior to returning the diesels to service following maintenance.

<u>Design purpose of failed system or component:</u>

1. The diesel generators provide power to safety-related loads when the unit generator and offsite power sources are unavailable.



NSIC 171700 - Actual Occurrence for Unavailability of Emergency Power System at San Onofre 1



NSIC 171700 - Sequence of Interest for Unavailability of Emergency Power System at San Onofre 1

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 171700

LER NO.: 81-029

DATE OF LER: December 17, 1981

DATE OF EVENT: November 19, 1981

SYSTEM INVOLVED: Emergency power system

COMPONENT INVOLVED: Both diesel generators

CAUSE: Trip of one diesel while other was secured for maintenance

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Unavailability of two diesel generators

REACTOR NAME: San Onofre Unit 1

DOCKET NUMBER: 50-206

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 436 MWe

REACTOR AGE: 14.4 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Southern California Edison

LOCATION: 5 miles south of San Clemente, California

DURATION: 2 h (estimated)

PLANT OPERATING CONDITION: 87% power

TYPE OF FAILURE: Failed to start;
made inoperable

DISCOVERY METHOD: During testing

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 171733

Date: January 8, 1982

Title: Unavailability of Three Auxiliary Feedwater Pumps at Zion 2

The failure sequence was:

1. With the reactor at 90% power and 2A turbine-driven AFW pump out of service for maintenance, water leaks from two hydrogen coolers in the unit generator resulted in a generator trip and subsequent reactor trip.
2. Motor-driven auxiliary feedwater pumps 2B and 2C failed to start and run on steam generator low-low level due to improperly sensed pump suction pressure. (It was determined that, immediately after start, reverse flow existed in the pressure switch sensing line, resulting in a momentary low suction pressure indication when adequate pressure actually existed.)
3. Both pumps were manually started.

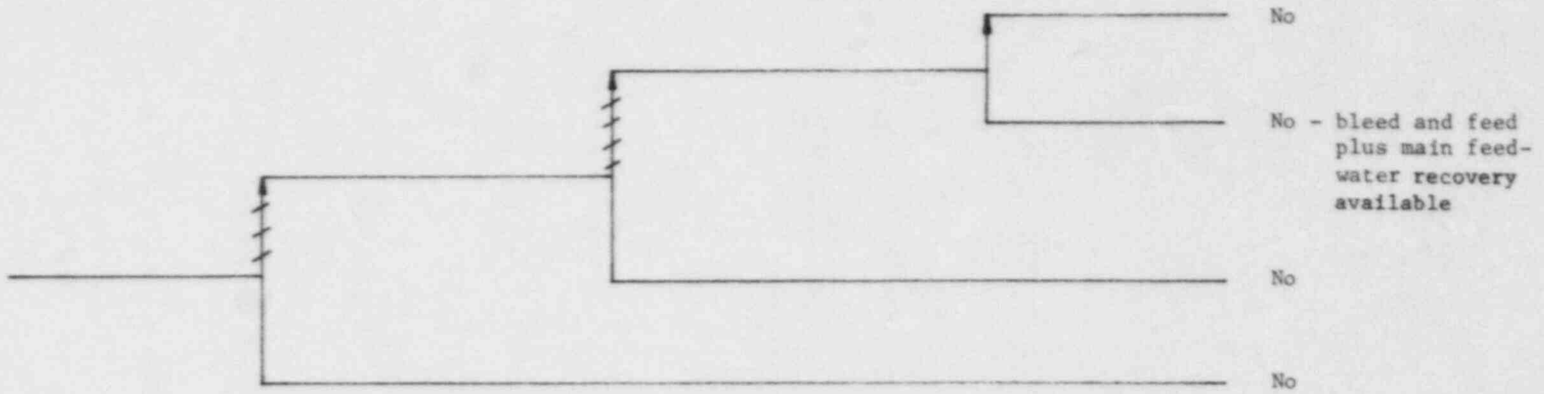
Corrective action:

A time delay relay which momentarily bypasses the low suction pressure trip on pump start was added to the pump starting circuitry.

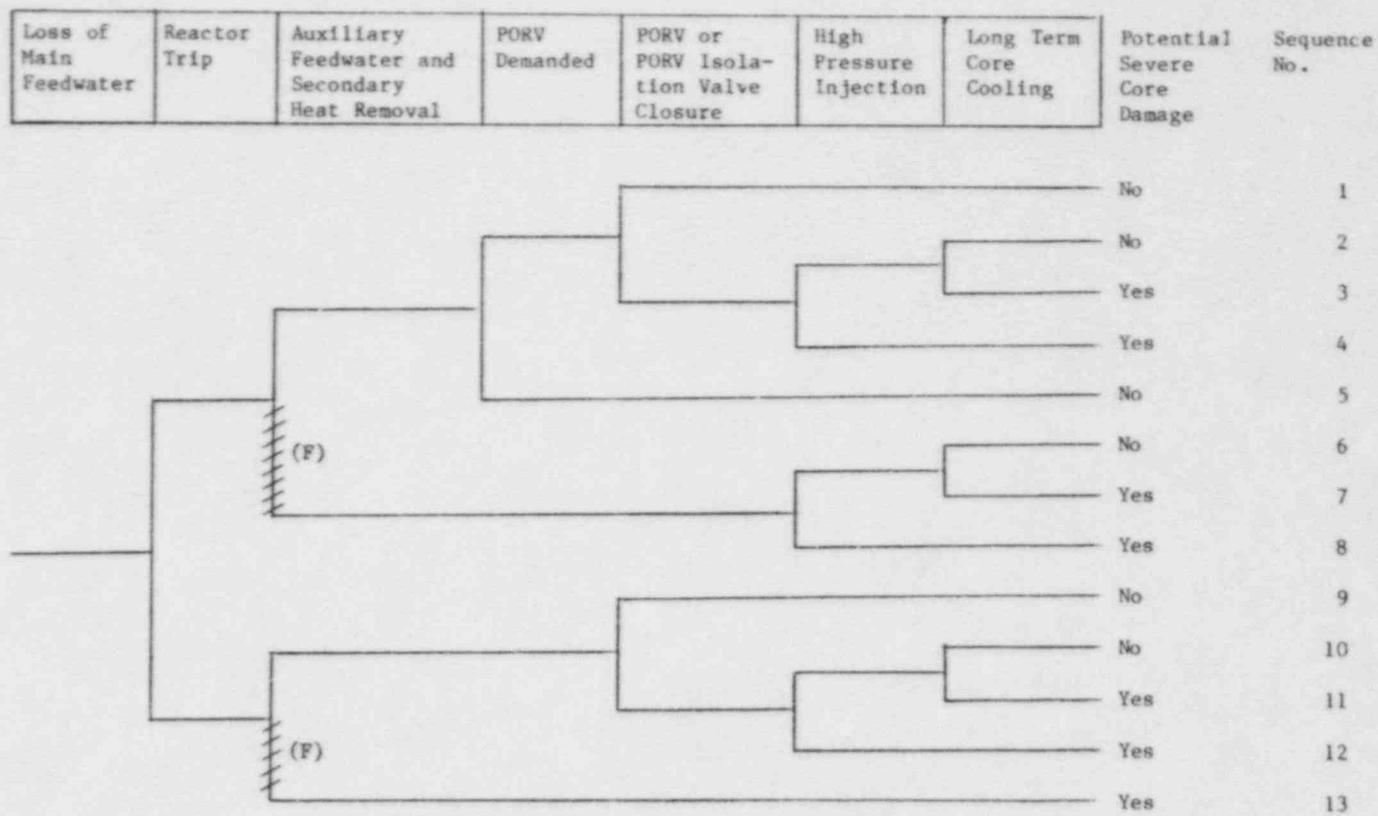
Design purpose of failed system or component:

1. The auxiliary feedwater system provides water for RCS cooling via the steam generators when the main feedwater system is unavailable.
2. An alternate safety-related source of water is service water supplied to suction of auxiliary feedwater pumps at 100 psig.

Reactor at 90% power and 2A turbine-driven AFW pump out of service for maintenance	Water leaks from two hydrogen coolers in the unit generator result in generator trip and subsequent reactor trip	Motor-driven AFW pumps fail to auto-start and run on low-low steam generator level due to low suction pressure trip caused by improperly sensed suction pressure	Motor-driven AFW pumps manually started	Potential Severe Core Damage



NSIC 171733 - Actual Occurrence for Unavailability of Three Auxiliary Feedwater Pumps at Zion 2



NSIC 171733 - Sequence of Interest for Unavailability of Three Auxiliary Feedwater Pumps at Zion 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 171733

LER NO.: 81-033

DATE OF LER: January 8, 1982

DATE OF EVENT: December 11, 1981

SYSTEM INVOLVED: Auxiliary feedwater

COMPONENT INVOLVED: Auxiliary feedwater pumps

CAUSE: One pump was unavailable due to maintenance and the other two pumps tripped due to incorrectly sensed low suction pressure on starting

SEQUENCE OF INTEREST: Loss of feedwater

ACTUAL OCCURRENCE: Unavailability of three AFW pumps following trip

REACTOR NAME: Zion 2

DOCKET NUMBER: 50-304

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 1040 MWe

REACTOR AGE: 8.0 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison

LOCATION: 40 miles north of Chicago, Illinois

DURATION: 360 h (estimated)

PLANT OPERATING CONDITION: Cooldown

TYPE OF FAILURE: Failed to start

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 171842

Date: December 18, 1981

Title: Failure of Emergency Diesel Generators at Dresden 3

The failure sequence was:

1. On October 23, 1981, with the plant at 66% power, DG 2/3 tripped due to high cooling water temperature during surveillance. Unit 3 DG was manually tripped due to high cooling water temperature indication during subsequent testing. In both cases the DG cooling water pumps had tripped prior to the DG trips. Both cooling water pumps were restarted and the diesels subsequently declared operable. The cause of the pump trips was not determined at the time.
2. On November 19, 1981, the unit 3 DG was again shut down during testing because of high coolant temperatures.
3. Subsequent testing revealed that both the Unit 2/3 and Unit 3 cooling water pump discharge check valve disks had separated from the valve operating arms and restricted cooling water flow.
4. During the event, DG U2, DG 2/3, 4-kV cross-tie, and offsite power were still available.

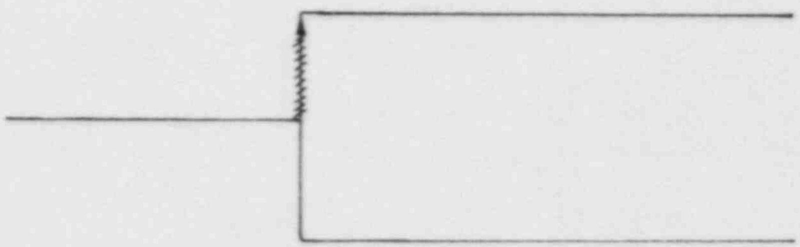
Corrective action:

1. The defective valves were replaced, the systems were satisfactorily tested and placed back in service.
2. A procedure revision was to be initiated to inspect these valves annually.

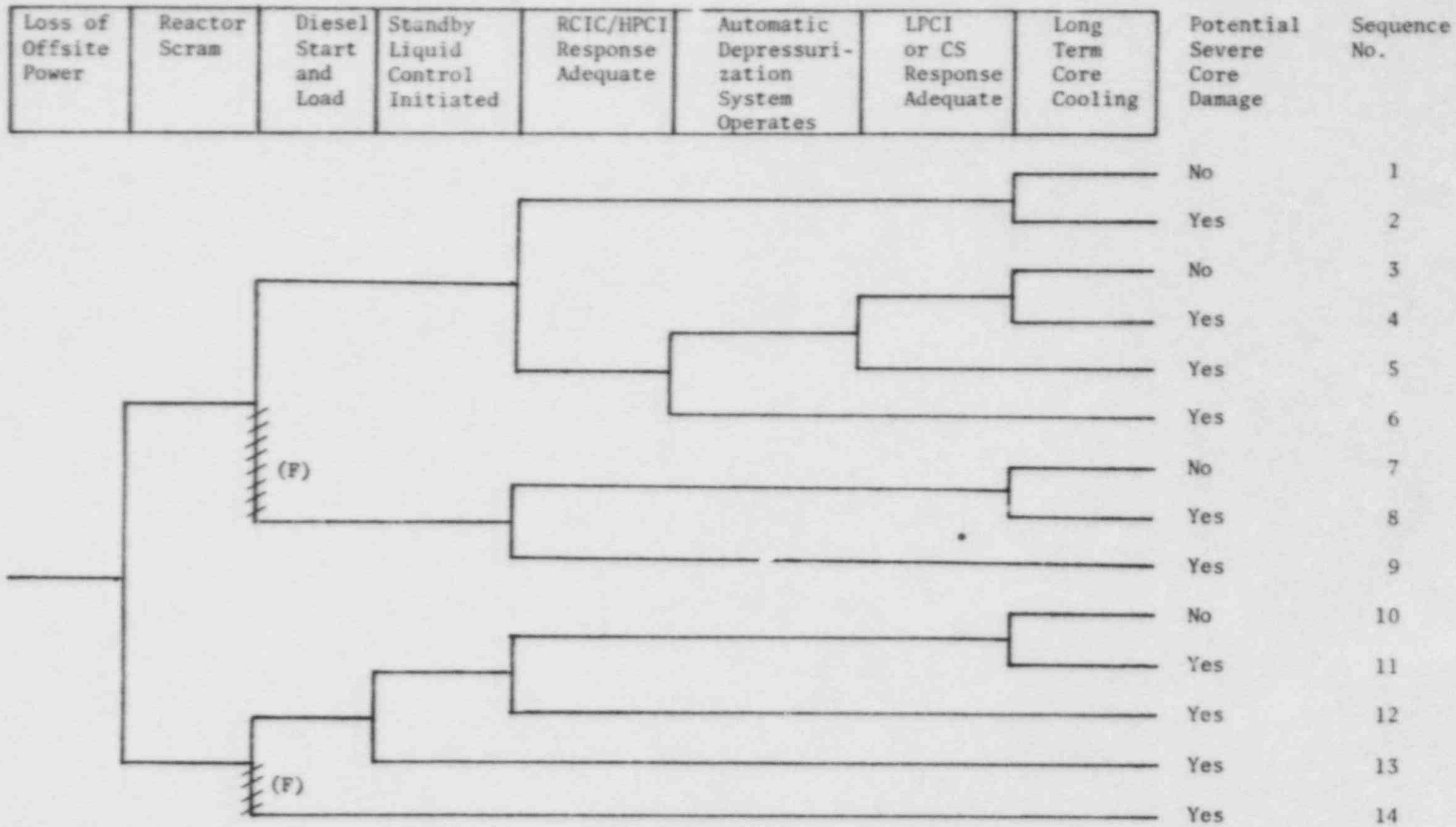
Design purpose of failed system or component:

The diesel generator system provides emergency power to safety-related loads in the event that offsite power and the unit generator are unavailable.

Reactor at power	Both diesel generators for Unit 3 declared inoperable during test due to insufficient cooling caused by defective diesel cooling water pump discharge check valves	Potential Severe Core Damage



NSIC 171842 - Actual Occurrence for Failure of Emergency Diesel Generators at Dresden 3



NSIC 171842 - Sequence of Interest for Failure of Diesel Generators at Dresden 3

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 171842

LER NO.: 81-037

DATE OF LER: December 18, 1981

DATE OF EVENT: October 23, 1981

SYSTEM INVOLVED: Emergency power

COMPONENT INVOLVED: Diesel generator cooling system check valves

CAUSE: Mechanical failure

SEQUENCE OF INTEREST: LOOP

ACTUAL OCCURRENCE: Diesel generators declared inoperable during test

REACTOR NAME: Dresden 3

DOCKET NUMBER: 50-249

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 794 MWe

REACTOR AGE: 10.7 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Sargent & Lundy

OPERATORS: Commonwealth Edison Co.

LOCATION: 7 miles east of Morris, Illinois

DURATION: .. e half test interval before October 23 plus time interval between October 23 and November 19: 1000 h (estimated)

PLANT OPERATING CONDITION: 66% power

TYPE OF FAILURE: Inadequate performance; made inoperable

DISCOVERY METHOD: Surveillance testing

COMMENT: Also see Accession 170009, Dresden 3, 50-249, LER 81-033, Nov. 6, 1981.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 171939

Date: October 28, 1981

Title: Inadvertent Steam Dump Valve Opening and Safety Injection at North Anna 2

The failure sequence was:

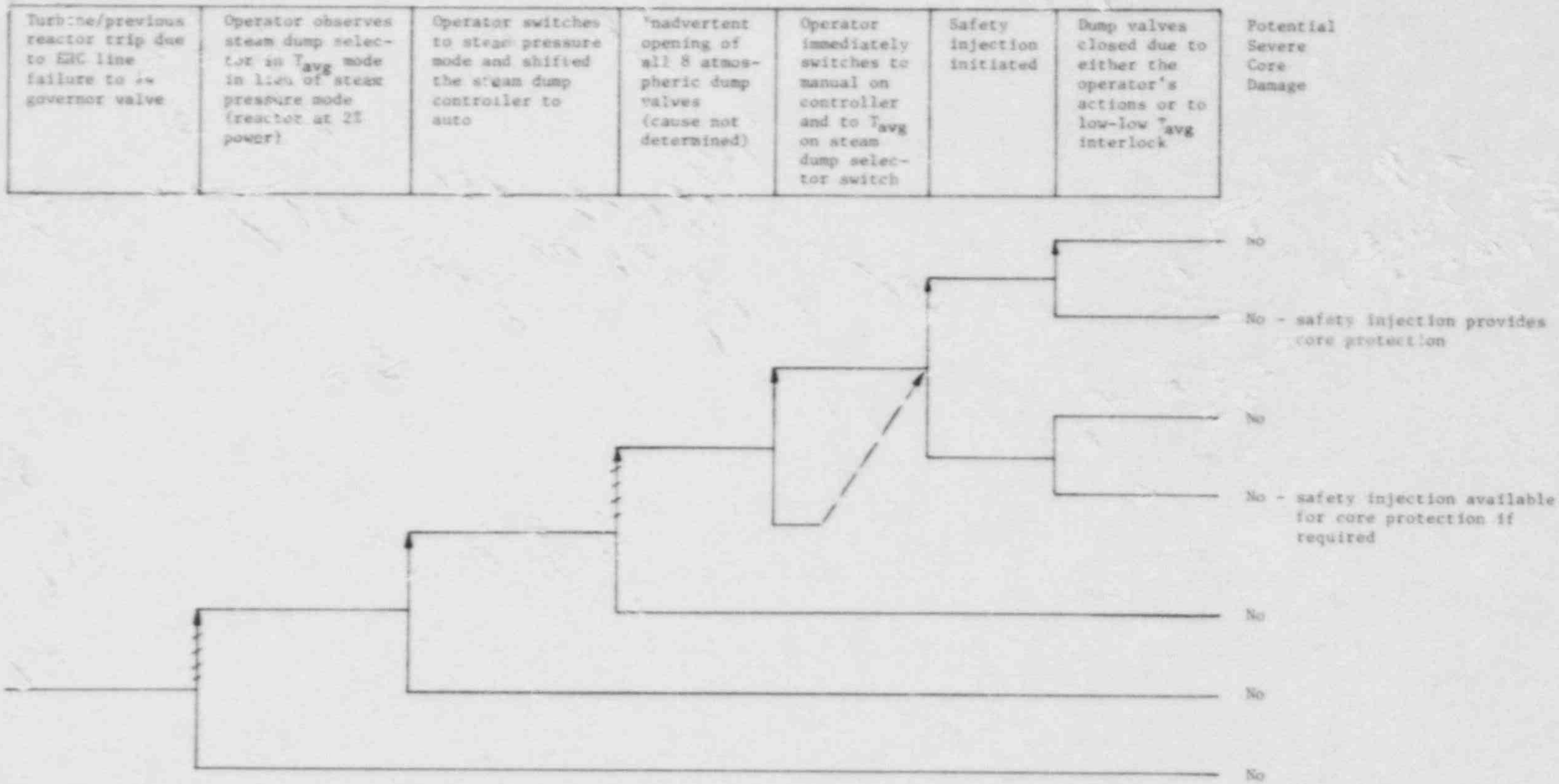
1. With the reactor at 100% power, a turbine trip and reactor trip occurred as a result of a failure in an electrohydraulic control line to No. 4 governor valve.
2. Three and one-half hours later, with the reactor at 2% power, it was observed that the steam dumps were in the T_{avg} mode in lieu of the steam pressure mode of operation.
3. The operator switched to the steam pressure mode of operation and shifted the steam dump controller to auto. All eight steam dumps opened.
4. The operator immediately shifted to manual on the controller and to the T_{avg} position on the steam dump selector switch.
5. Either this action or the low-low T_{avg} interlock closed the dump valves but not before safety injection was initiated.
6. All safety injection equipment operated as required, except the pressurizer liquid space inside isolation valve failed to close.
7. The specific cause for the high controller output was not determined, and an investigation of equipment failure and human error revealed no discrepancies.

Corrective action:

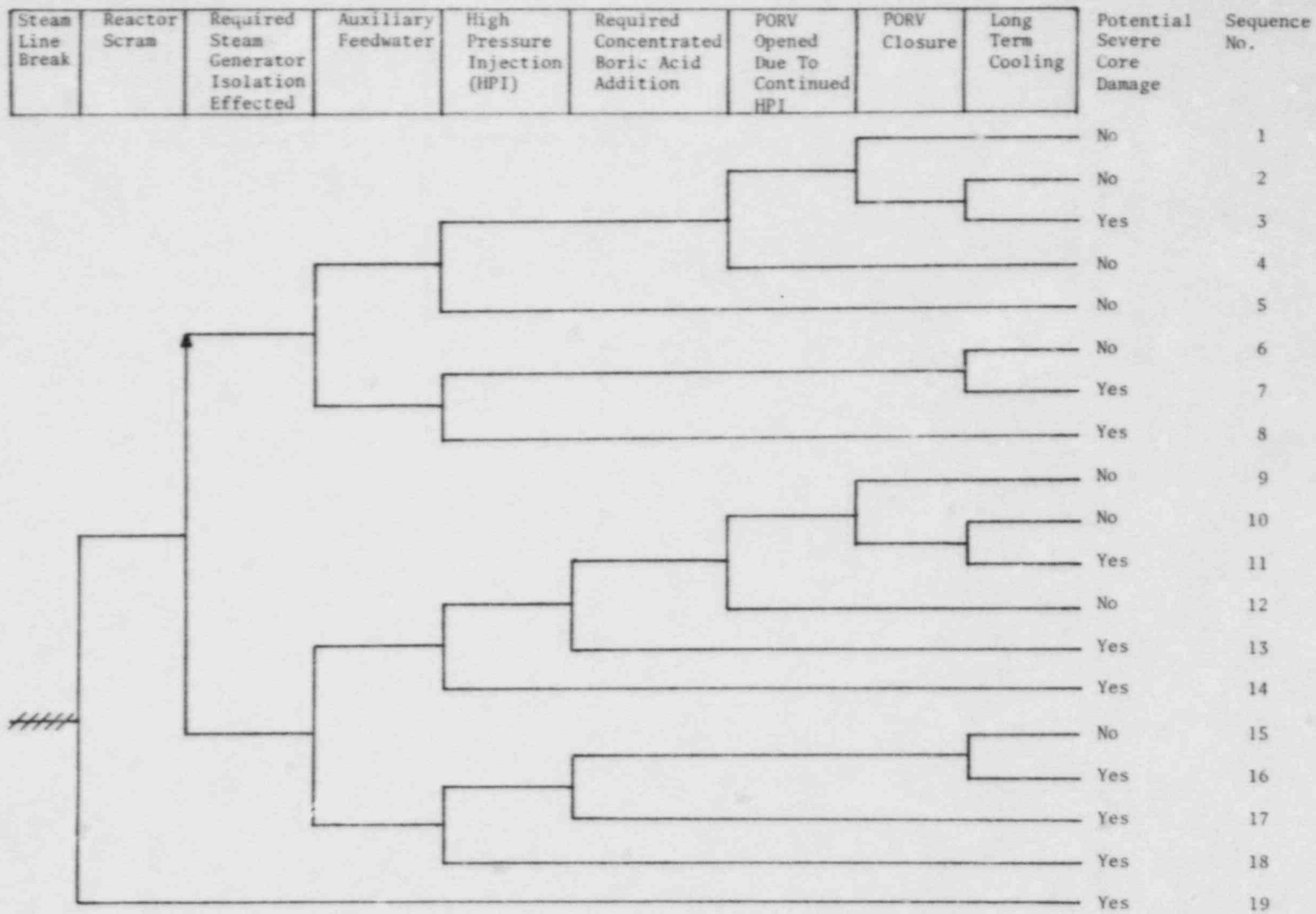
The event was reviewed for equipment failure and human error and no cause could be determined.

Design purpose of failed system or component:

The steam dump valves provide for steam generator pressure control following plant trip by venting steam to the condenser.



NSIC 171939 - Actual Occurrence for Inadvertent Steam Dump Valve Opening and Safety Injection at North Anna 2



NSIC 171939 - Sequence of Interest for Inadvertent Steam Dump Valve Opening and Safety Injection at North Anna 2

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 171939

LER NO.: 81-076

DATE OF LER: October 28, 1981

DATE OF EVENT: October 3, 1981

SYSTEM INVOLVED: Steam dump to condenser

COMPONENT INVOLVED: Dump valves

CAUSE: Inadvertent opening

SEQUENCE OF INTEREST: Main steam line break

ACTUAL OCCURRENCE: Eight opened dump valves

REACTOR NAME: North Anna 2

DOCKET NUMBER: 50-339

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 907 MWe (net)

REACTOR AGE: 1.3 years

VENDOR: Westinghouse

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Virginia Electric & Power Co.

LOCATION: 40 miles NW of Richmond, Virginia

DURATION: N/A

PLANT OPERATING CONDITION: 2% power following a trip

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD: Operational event

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 172198

Date: January 18, 1982

Title: Inadvertent MSIV Closure and Inadvertent Safety Valve Lifts at St. Lucie 1

The failure sequence was:

1. At 4:14 a.m. on December 19, 1981, a combination of low instrument air pressure to the MSIVs and a slightly higher than normal steam flow resulted in the inadvertent closure of both MSIVs.
2. The reactor pressure increased to 2410 psig (normal lift pressure is 2500 psig), causing a reactor trip on high pressurizer pressure and opening the PORVs. Pressurizer code safety valve V1200 apparently also opened, but this was not realized at the time.
3. The quench tank rupture disk ruptured because of the relief valve discharge.
4. Pressurizer safety valve V1200 disk and seat were probably damaged during the lift due to the improper installation of an adjusting ring set screw, but this was not known at the time.
5. The reactor was restarted and held below the power range during repair of the rupture disk and MSIV trouble shooting.
6. At 1:45 p.m. with the RCS at normal operating pressure, valve V1200 again spuriously opened and was verified open by acoustic flow monitor indication. A high blowdown (34% in lieu of the specified 4%) occurred due to a maladjusted nozzle ring.
7. RCS pressure decreased to 1670 psig and the reactor tripped on thermal margin/low pressure. A reactor cooldown was begun to replace V1200.
8. On December 23, 1981, after replacement of valve V1200, a turbine/reactor trip occurred during startup. The reactor was again started up and power operation was commencing when the "B" steam generator pressure relief valves started lifting. It was suspected that the "B" MSIV was closed even though it indicated open.
9. A shutdown and cooldown was initiated. The "B" MSIV was found to have one bent and one broken actuator connecting pin. The "A" MSIV was found to have two slightly bent connecting pins.

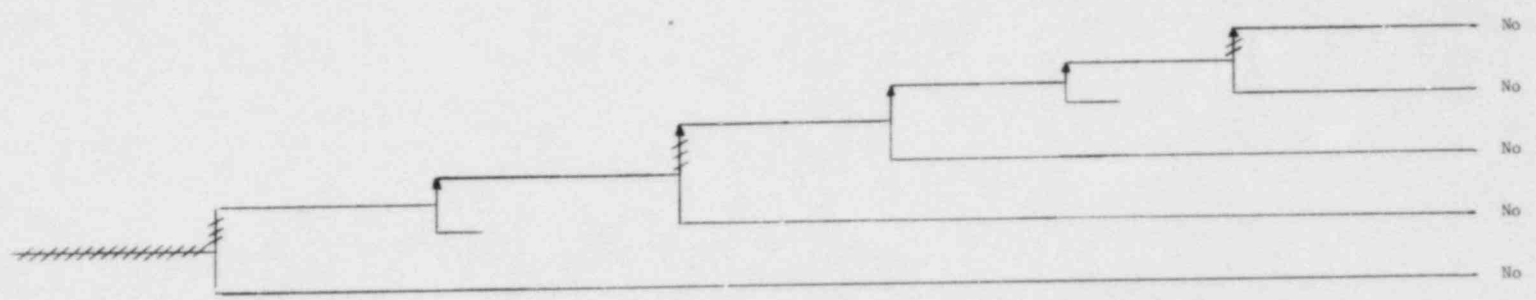
Corrective action:

Pressurizer safety valve V1200 was replaced. The MSIVs were repaired.

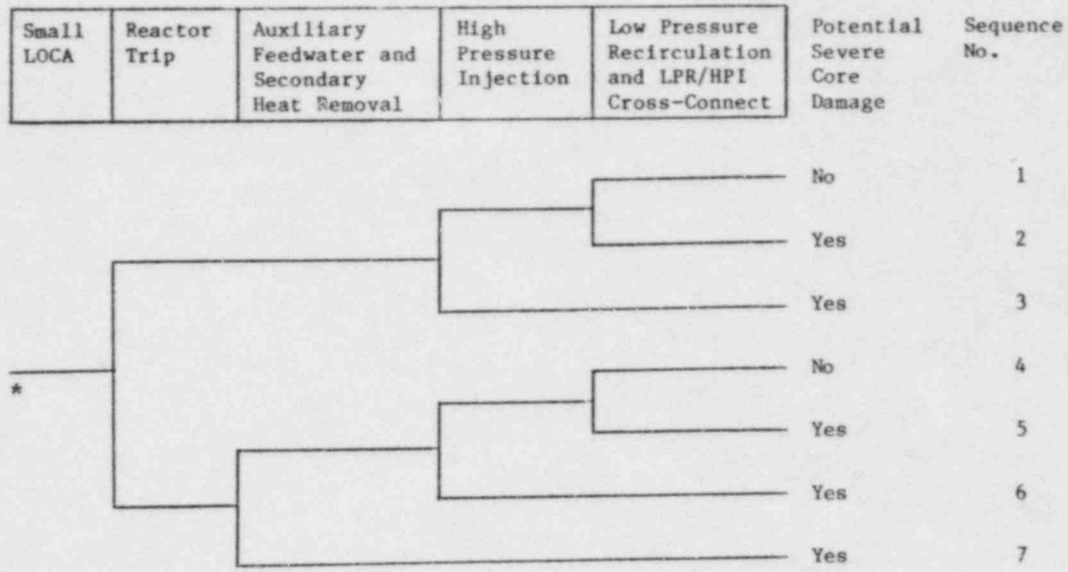
Design purpose of failed system or component:

1. The code safety valves provide relief protection to prevent vessel/piping overpressure during transient events.
2. The main steam isolation valves provide for steam generator isolation during steam line break events as well as providing for containment isolation.

Reactor at power and MSIV (both valves) closure due to low instrument air pressure and slightly high steam flow	Reactor trip on high pressurizer pressure. Quench tank rupture Disk ruptured because of relief valve discharge. Pressurizer code safety V1200 apparently lifts (lift is undetected), disk/seat damage results from improper set screw installation	Reactor restarted and held below power range for rupture disk repairs and MSIV troubleshooting	Pressurizer code safety V1200 lifts spriously; high blowdown due to maladjusted nozzle ring	RCS pressure decrease, trip on thermal margin/Low pressure	Cooldown to replace V1200	Turbine/reactor trip during next startup; reactor restarted, power operation commencing when steam generator B saftey valve started lifting due to closed MSIV B (valve was indicating open)	Potential Severe Core Damage



NSIC 172198 - Actual Occurrence Tree for Inadvertent MSIV Closure and Inadvertent Safety Valve Lifts at St. Lucie 1



NSIC 172198 - Sequence of Interest for Inadvertent Pressurizer Code Safety Valve Opening at St. Lucie 1

*reseating of safety valve V1200 would provide mitigation

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 172198

LER NO.: 81-056

DATE OF LER: January 18, 1982

DATE OF EVENT: December 19, 1981

SYSTEM INVOLVED: Main steam and pressurizer pressure relief

COMPONENT INVOLVED: MSIV, pressurizer code safety

CAUSE: MSIVs (spurious); PSRV (assembly error)

SEQUENCE OF INTEREST: LOFW

ACTUAL OCCURRENCE: LOFW

REACTOR NAME: St. Lucie 1

DOCKET NUMBER: 50-335

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 802 MWe

REACTOR AGE: 5.7 years

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Ebasco

OPERATORS: Florida Power and Light Company

LOCATION: 12 miles SE of Ft. Pierce, Florida

DURATION: N/A

PLANT OPERATING CONDITION: 98%

TYPE OF FAILURE: Inadequate performance

DISCOVERY METHOD:

COMMENT:

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 174073

Date: April 23, 1982

Title: Stuck Open Relief Valve at Duane Arnold

The failure sequence was:

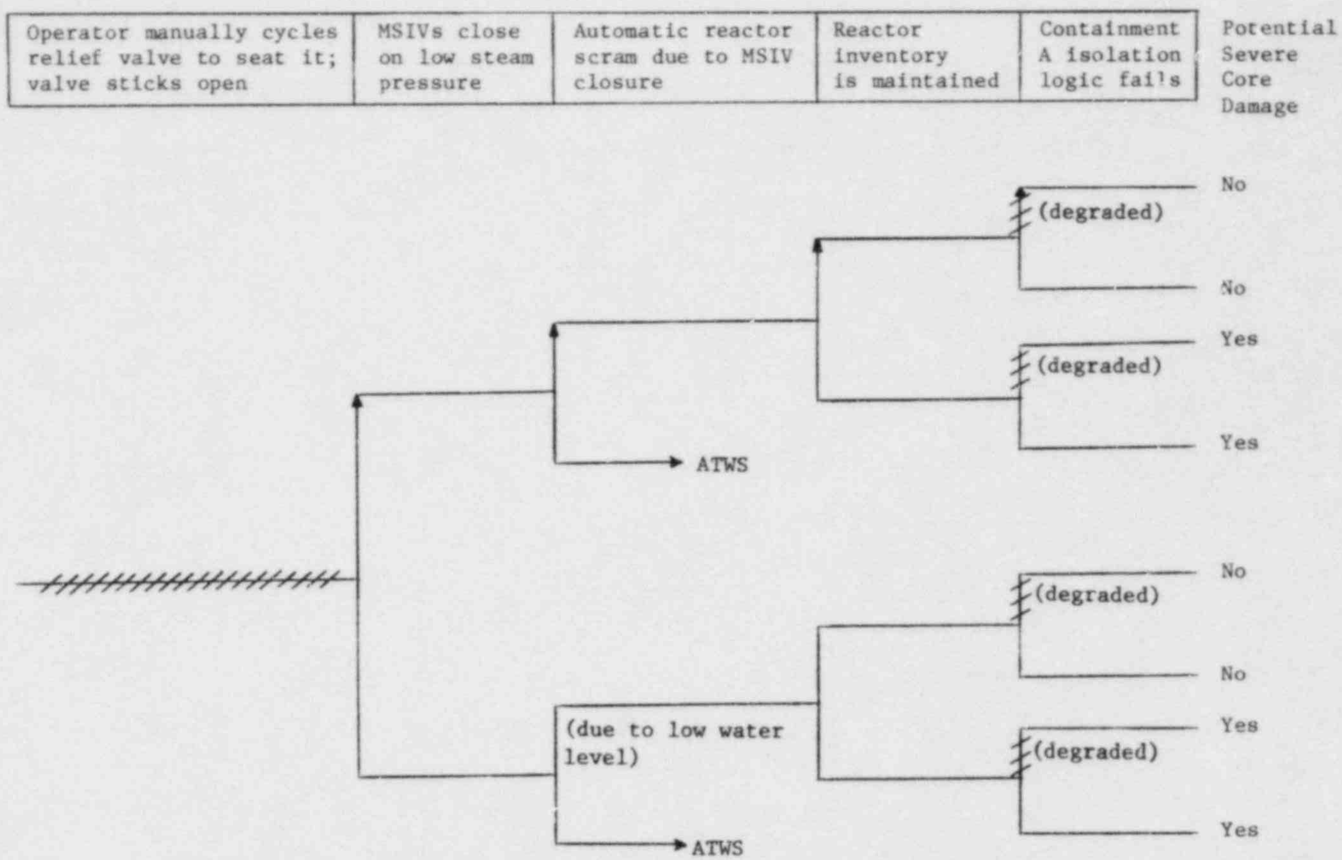
1. With the reactor at 7% power during a startup, an operator noticed a higher than normal discharge line temperature from a Target Rock Model 67F-6X10 main steam relief valve, which indicated a small steam leakage from the reactor coolant system (RCS).
2. An operator manually cycled the valve at rated pressure to attempt a better valve seating.
3. The valve stuck open and remained open for 7.5 min, resulting in RCS depressurization to <880 psig.
4. The MSIVs closed on low RCS pressure and an automatic reactor scram occurred.
5. The vessel water level fell 7.5 in. below instrument zero level (150.5 in. above the active fuel).
6. ECCS initiation was not required and ECCS was not employed in the recovery (water level and pressure were above ECCS actuation set point.
7. One train (group 3 channel A) of the containment isolation actuation logic failed to actuate on low vessel water level as required. A wiring error made during a design change and not detectable by postinstallation testing or normal surveillance tests was discovered. One valve in a number of pairs of isolation valves failed to close. Channel B did actuate and the containment was isolated.
8. The reactor was placed in cold shutdown until repairs were completed.

Corrective action:

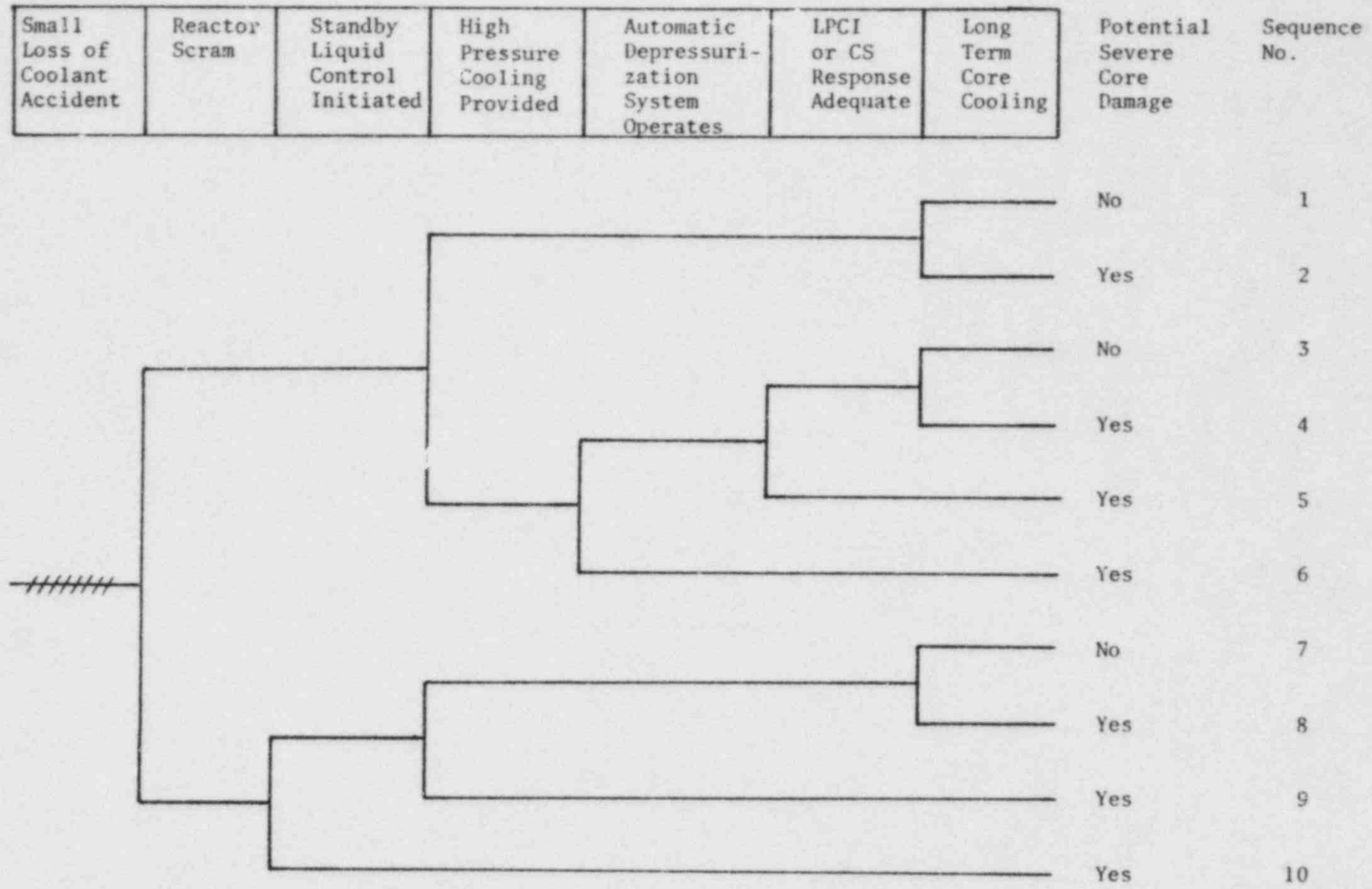
The valve was replaced. The cause of the closure failure could not be determined.

Design purpose of failed system or component:

The relief valve provides over-pressure protection for the RCS during a transient event.



NSIC 174073 - Actual Occurrence Tree for Stuck Open Relief Valve at Duane Arnold



NSIC 174073 - Sequence of Interest Tree for Stuck Open Relief Valve at Duane Arnold

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 174073

LER NO: 80-054/01X-1

DATE OF LER: April 23, 1982

DATE OF EVENT: January 11, 1980

SYSTEM INVOLVED: Over-pressure protection system

COMPONENT INVOLVED: Main steam relief valve

CAUSE: Unknown

SEQUENCE OF INTEREST: Small break LOCA

ACTUAL OCCURRENCE: Small break LOCA

REACTOR NAME: Duane Arnold

DOCKET NUMBER: 50-331

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 538 MWe

REACTOR AGE: 5.8 years

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Iowa Light and Power

LOCATION: 8 miles NW of Cedar Rapids, Iowa

DURATION: 7.5 min

PLANT OPERATING CONDITION: 7% full power (starting up)

TYPE OF FAILURE: Failed to close

DISCOVERY METHOD: Operational event

COMMENT: This event could have occurred at full power. Note: LER Rev. 0 for this event was filed as NSIC 161716.

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16. ABSTRACT (200 words or less)

Descriptions of fifty-eight operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1980-1981 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees, categorization, and subsequent analyses. This study is a continuation of the work presented in NUREG/CR-2497 which somewhat similarly evaluated the 1969-1979 events. The current study incorporates improvements which evolved from an assessment of the comments on the earlier report and applies these in the assessment of the LERs which occurred during 1980 and 1981. The report sequentially discusses (1) the general rationale for this study, (2) the program methods for LER review and documentation, (3) the calculation of function failure probabilities and initiating event frequencies based upon precursor data, (4) the use of the conditional probability of subsequent severe core damage estimates to rank precursor events and estimate an average industry-wide risk of severe core damage, and (5) the conduct of sensitivity analyses on these results. There was some apparent decrease in most initiating event frequencies and function failure probabilities in the 1980 and 1981 period, as compared to the earlier report. Although it was not possible to conclude that these decreases were statistically significant, they did result in a reduction in the industry average estimated severe core damage frequency for 1980-1981 as compared to the 1969-1979 period.

17. KEY WORDS AND DOCUMENT ANALYSIS

Nuclear Power Plant
Accident Sequence Precursors
Risk Analysis
Event Trees
Function Failures
Initiating Events

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Accident Sequences
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Demand Failures
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