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Pressure Vessel Thermal Shock at U.S. Pressurized-Water Reactors: Events and Precursors, 1963–1981

Doan L. Phung

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PRESSURE VESSEL THERMAL SHOCK AT U.S. PRESSURIZED-WATER REACTORS: EVENTS AND PRECURSORS, 1963–1981

Doan L. Phung

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FOREWORD

The Accident Sequence Precursor Program was initiated in the spring of 1979 at the Nuclear Safety Information Center at Oak Ridge National Laboratory; J. W. Minarick is the acting program manager. This program is sponsored by the Division of Risk Analysis of the Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, with Fredrick M. Manning as our technical monitor. The basic concept of this program is to analyze historical nuclear plant event data for various accident trends, as recommended by the "Risk Assessment Review Group" (The Lewis Committee Report) in 1978. While the major thrust of this program has been on "Precursors to Potential Severe Core Damage" (see ORNL/NSIC-182), in the fall of 1981 we were asked to undertake a somewhat similar study of precursors that may be involved in accident sequences leading to pressure vessel thermal shock (PVTS), so we initiated the current study.

In searching for the most competent available person to handle this task, we were most fortunate to find the author, Dr. Doan L. Phung, between assignments. Phung received his B.A. in Mathematics from Florida State University and his M.S. in Physics and M.S. and Ph.D. degrees in Nuclear Engineering from the Massachusetts Institute of Technology. He spent 10 years with United Engineers and Constructors analyzing nuclear plant safety and for the last 7 years has been at the Institute for Energy Analysis analyzing energy policies. He was a member of the WASH-1400 project and was coauthor with N. C. Rasmussen and D. D. Lanning of a 1973 paper on thermal-hydraulic consequences of a hypothetical pressurized-water reactor vessel rupture (*Nuclear Safety*, Vol. 14-4, July–August 1973). While this was his first work in the area of PVTS, his background well qualifies him for this study of the PVTS precursor data.

As in any study of this nature, the findings can be no better than the data base from which they are derived. In this instance, the data base consists of all the Licensee Event Reports (LERs) submitted during the period 1963 through 1981. However, it became apparent that not all PVTS precursors were reported as LERs (because not all violated Technical Specifications – the requirement for an LER). Furthermore, considerable judgment is involved both in identifying and categorizing PVTS precursors. Nevertheless, the author has not only been very thorough in selecting and categorizing the selected LERs, he has also attempted to establish the limitations of his data and findings at each point. This information should be of inestimable value to all those interested in events of potential concern to the PVTS issue.

Wm. B. Cottrell, Director
Nuclear Operations Analysis Center

CONTRIBUTIONS AND REVIEWS

This report supersedes its forerunner "Pressure Vessel Thermal Shock at U.S. Pressurized-Water Reactors: Events and Precursors, 1963 to Mid-1981, Interim Report," dated May 1982. The improvement is in all aspects: data, analyses, and incorporation of opinions of peer reviewers.

Although I am the only author and am solely responsible for the quality of the report, I wish to acknowledge the contributions, critiques, comments, and suggestions of the following individuals on the interim report. The listing of institutions is only for identification purposes.

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ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
A-E	Architect-engineer
AFW	Auxiliary feedwater
B&W	Babcock & Wilcox
BIT	Boron injection tank
BWR	Boiling-water reactor
BWST	Borated water storage tank
CB&I	Chicago Bridge and Iron
C-E	Combustion Engineering
CF	Core flooding
CFS	Core flooding system
DOE	Department of Energy
ECCS	Emergency core cooling system
EFPY	Effective full-power year
EFW	Emergency feedwater
EPRI	Electric Power Research Institute
ESFAS	Engineered safety feature actuation system
EX	External initiator (for event sequence)
FPC	Florida Power Corporation
FW	Feedwater
G&E	Gas and Electric
GMDC	Gross maximum dependable capacity
HP	High-pressure
HPI	High-pressure injection
HPSI	High-pressure safety injection
HSF	High steam flow
I&C	Instrumentation and control
ICS	Integrated control system
INPO	Institute for Nuclear Power Operation
LANL	Los Alamos National Laboratory
LBLOCA	Large-break loss-of-coolant accident
LER	Licensee event report
LP	Low pressure
LPSI	Low-pressure safety injection
MDC	Maximum dependable capacity
NNI	Nonnuclear instrumentation
NPRDS	Nuclear plant reliability data system
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSIC	Nuclear Safety Information Center
NSSS	Nuclear steam supply system
OF	Overfeeding (of steam generator)
ORNL	Oak Ridge National Laboratory
P&L	Power and Light
PASNY	Power Authority of the State of New York
PORV	Power-operated relief valve
PRA	Probabilistic risk analysis
PTS	Pressurized thermal shock

PVTS	Pressure vessel thermal shock
PWR	Pressurized-water reactor
R&D	Research and development
RCS	Reactor coolant system
RDM	Rotterdamsche Droogdok Mij
RHR	Residual heat removal
RT	Reactor trip
RV	Relief valve
RWST	Refueling water storage tank
RY	Reactor year
S&W	Stone and Webster
SBLOCA	Small-break loss-of-coolant accident
SFAC	Creusot Loire
SG	Steam generator
SI	Safety injection
SMUD	Sacramento Municipal Utility District
TVA	Tennessee Valley Authority
UE&C	United Engineers and Constructors, Inc.
W	Westinghouse

EXECUTIVE SUMMARY

Perspective

The U.S. nuclear industry and the Nuclear Regulatory Commission (NRC) are currently (1981–1982) conducting an intensive examination of the capability of pressurized-water reactor (PWR) pressure vessels to withstand severe thermal shocks without compromising their integrity. Many older reactor vessels were manufactured with high copper contents in welds, have accumulated large doses of fast neutrons in service, and have current nil ductility transition temperatures in the 200 to 300°F range. It is thought that operational temperature transients similar to or more severe than the Rancho Seco incident of March 20, 1978, have the potential to cause certain preexistent flaws in the wall of these vessels to propagate. The thrust of current U.S. examination is not only to examine this potential from the thermal-hydraulic and fracture mechanics viewpoint but also from the probabilistic viewpoint: What are the probability and consequences of vessel failure due to a specific accident sequence and how can such risks be reduced?

This study is intended to support this examination by drawing on the licensee event reports (LERs) of operating U.S. PWRs. The data base is ~16,000 PWR LERs stored in the computerized data bank of the Nuclear Safety Information Center (NSIC) in Oak Ridge for the period 1963–1981. This data base covers most reportable events at U.S. PWRs, subject to the changes in their reporting requirements over the years, and represents data for 329 reactor years (RYs) and 47 PWR units.

Framework

The LER search methodology includes a logic approach that narrows down the data base to specific areas of interest, the Boolean combination of keywords that turns up appropriate events from the computer data base, and the procurement of original reports for detailed analyses. The LER event analytical methodology includes constructing seven generalized event trees and defining five severity categories of pressure vessel thermal shock (PVTS) for event and precursor classification. The check of the "exhaustiveness" of the findings includes comparison against 10% of all PWR LERs and against data of both the NRC and industry.

The seven generalized event sequences leading to PVTS include: large-break loss-of-coolant accident (LBLOCA), small-break loss-of-coolant accident (SBLOCA), overfeeding of steam generators [OF/cold feedwater (FW)], high steam flow (HSF), independent high-pressure safety injection (HPSI), core flooding or independent low-pressure safety injection (CF/LPSI), and externally caused thermal shock events (EX).

The five severity categories, 0–4, are defined on the basis of thermal shock consequences, from the extremely inconsequential (severity 0) to the highly significant (severity 4). Several parameters are used to represent thermal shock consequences. They are temperature change, pressure change, duration of the transient, amount of injected water, and

duration of safety injection (SI). These parameters are related to one another in a complicated manner and are not necessarily all available for any specific event. One reason for us to differentiate thermal shock severity into five categories is to facilitate the analysis of events which could lead to PVTS. Another reason is to conceptually differentiate events from precursors. Thus, an event with a lower severity could be considered a precursor to an event of higher severity if both have the same initiating sequence.

When a more global look is warranted, events with severities 0, 1, and 2 can be put into one group called "insignificant." Events with severities 3 and 4 can be put into another group called "significant."

Results: Number of Events and Frequencies

In several dozen systematic searches and subsequent analyses, we read some 4000 LER abstracts, selected some 170 events for additional readings of original reports, and found 99 events that correspond to our definition of PVTS events or precursors.

A listing of the 34 significant events is provided in Table S.1. A more complete listing and description of all 99 events are found in Chap. 4 and Appendixes A and B.

Forty-seven PWRs were in the data base from 1963 through 1981, covering 329 RYs. These do not include Shippingport and Indian Point 1, which did not have much information in the data base and which we did not consider as a contributor to any LERs.

The number of events and their frequencies when considered in two groupings — significant and insignificant (as defined previously) — are as follows:

	Number of events	Events/RY
Significant	34	0.103
Insignificant	65	0.198
Total	99	0.301

Although numerous and sometimes dramatic, it must be noted that these actual events are really quite mild in comparison with hypothetical PVTS events postulated for safety analysis and licensing purposes. In addition, the fracture mechanics code OCA-2 was used to analyze several transient cases that bound these actual PVTS events.[1] In the most severe case, the vessel was computed to fail when the RCS temperature drops 300°F, the RCS pressure repressurized to 2500 psi, and there are preexistent flaws in the longitudinal welds with copper contents up to 0.35 wt %. The critical neutron fluence in this case was 0.43×10^{19} neutrons/cm^2.

Completeness: Correction for Frequencies

How exhaustive were our searches of PVTS events in the data bank? How complete is the data bank in covering all events that took place in operating reactors? Do the data really go back to 1963?

Table S.1. Short list of 34 significant PVTS events

Date	Reactor	Nature of event	ΔT/Δt and/or remarks
		Severity 3	
10/12/70	Connecticut Yankee	Steam dump valve stuck open	43°F/9 min
10/31/73	Ginna	Transmission disturbances; SI	85°F/10 min
10/20/73	San Onofre	FW regulating valve caused OF	78°F/10 min
12/08/76	Calvert Cliffs 2	FW control difficulties	≈40°F/10 min
12/08/76	Calvert Cliffs 2	FW control difficulties	≈50°F/10 min
05/28/78	St. Lucie 1	Bypass control valve stuck	ΔT ≈ 60°F
12/14/78	Oconee 1	Short circuit related to ICS	RCS p < 1500 psi
02/23/80	North Anna 1	False SI	42°F/15 min
04/03/80	North Anna 1	SI due to MS stop valve closure	72°F/15 min
01/29/80	Arkansas 2	Steam dump and bypass valve	RCS p ~ 1350°F
01/29/81	Robinson 2	Letdown line break; pressurizer spray leak; SI	4500–6000 gal SI
04/23/81	Salem 2	Steam dump valve used to control cooldown	5 min of SI
		Severity 4	
10/01/72	Surry 1	SG leaked	>100°F/h
11/05/72	Robinson 2	PORV stuck open	SI for 87 min
01/04/74	Oconee 2	Excessive auxiliary steam use	>100°F/h
04/17/74	Ft. Calhoun	FW regulating valve stuck open	120°F/19 min
10/07/74	Rancho Seco	Excessive auxiliary steam use	100°F/h
05/01/75	Robinson 2	RCS pump seal failure	132,500 gal lost
06/13/75	Oconee 3	PORV stuck open	RCS T ~ 480°F, p ≈ 720 psi
09/19/76	Zion 2	Loss of dc power caused RT but not trip of steam-driven FW pump	70°F/30 min, SI
03/02/77	Crystal River 3	ICS caused transient upon loss of power	164°F/15 min
04/16/77	Crystal River 3	FW valves caused OF	101°F/20 min
07/02/77	Indian Point 2	RCS pump seal failure	140°F/1.3 h
03/20/78	Rancho Seco	Loss of NNI caused transient	310°F/1 h
04/23/78	Three Mile Island 2	Steam relief valves stuck open	149°F/6 min
12/27/78	Arkansas 2	Steam dump and relief valves stuck open	107°F/52 min
01/05/79	Rancho Seco	Short in ICS caused transient	152°F/7 min
03/14/79	Millstone 2	Cold RWST flooded RCS	ΔT ~ 146°F at low pressure
03/28/79	Three Mile Island 2	Loss of FW and PORV stuck open	Core melted
09/25/79	North Anna 1	Steam dump valve stuck open	110°F/16 min
11/10/79	Oconee 3	Overfeeding due to turbine bypass valve	150°F/15 min
02/26/80	Crystal River 3	Loss of NNI caused transient	127°F/13 min; SI repressurized RCS
10/17/80	Indian Point 2	Reactor vessel exposed to river water from outside	9 ft under river water for extended period
09/14/81	McGuire 1	Overcooling during loss-of-control room test	117°F/20 min

Abbreviations:

dc = direct current
FW = feedwater
ICS = integrated control system
MS = main steam
NNI = nonnuclear instrumentation
OF = overfeeding
PORV = power-operated relief valve
RCS = reactor coolant system
RT = reactor trip
RWST = recirculating water storage tank
SG = steam generator
SI = safety injection

The first question was addressed by checking our finds against two separate samples each containing 5% of all PWR LERs. We found that our significant events may have covered 75% of all such events in the LER data bank, but the insignificant events may be only 36% representative.

The second question was addressed by checking our finds against industry's (Oconee plant and Westinghouse owners group) data. Although the data themselves are incomplete, we found that our finds may have covered only about 25% of insignificant and 67% of significant events that actually took place at PWR power plants. The overall ratio is about 50%, a value remarkable in agreement, perhaps fortuitously so, with a 1978 Advisory Committee on Reactor Safeguards (ACRS) observation of the same context.[2]

The third question was raised by realizing that data are very scarce for the few earlier commercial reactors. Indeed, the earliest data of any event in our finds is 1970. Noting that 1969 is the date when the NSIC data base started to record LERs systematically, it is reasonable to decide on a cutoff date of 1969. Consequently, the total RY value is 320 instead of 329.

Correcting for the three factors given in the preceding paragraphs, the likely number of PVTS-like events and their frequencies are as follows:

	Corrected number of events	Corrected events/RY
Significant	51	0.159
Insignificant	260	0.813
Total	311	0.972

Categories and Causes

Table S.2 shows the 99 PVTS events and precursors categorized in three groups: modes of reactor operation just before the event, initiating sequences of the event, and causes of initiation.

Table S.3 shows the frequencies per RY based on the finds per se and then corrected for three factors of completeness discussed in the preceding section. The following discussion is based upon the corrected data in Table S.3.

About half of all events occurred when the reactors were in power operation (Mode 1), and about 84% occurred when the reactor system was in hot conditions (Modes 1, 2, 3, and 4).

No LBLOCA event took place, and SBLOCA accounted for 9% of all events. About one-fourth of all events were initiated by FW-steam circuits, either by OF of steam generators (SGs) or by HSF. About half of all events were initiated by HPSI, due either to instrumentation and controls (I&C) design sensitivity or to spurious signals. In terms of severity, one-third of all SBLOCA events and ~40% of all OF/HSF events tend to be significant (Severities 3 and 4). On the other hand only 4% of HPSI-initiated PVTS events were significant.

Almost one-fourth of all events were caused by operator/maintenance/test errors, and an additional 10% can be considered as secondarily caused

Table S.2. Distribution of PVTS events and precursors regarding modes of reactor operation, initiating sequences, and causes of initiation (1963–1981)

Type of events	Events					Total
	Insignificant			Significant		
	0	1	2	3	4	
Modes of reactor operation						
1 – Power	7	11	10	8	17	53
2 – Startup	3	4	3	2	2	14
3 – Hot standby	1	6	4	2	2	15
4 – Hot shutdown	2	1			1	4
5 – Cold shutdown	6	2				8
6 – Refueling	1	4				5
Total	20	28	17	12	22	99
Initiating sequences						
LBLOCA						
SBLOCA	3	2		1	5	11
OF/Cold FW	2		6	6	7	21
HSF		2	2	2	8	14
HPSI	6	22	9	3	1	41
CF/LPSI	2	2				4
EX					1	1
Others	7					7
Total	20	28	17	12	22	99
Causes of initiation						
Operator error	3	3	5	1		12
Maintenance/test error	4	1	1	1		7
Valve failure	4	3		6	8	21
Power failure		3	5	1	2	11
I&C design error/ spurious signal	3	8		1	3	15
System/equipment failure	5	5	4	2	9	25
Water hammer	1	5				6
Fire/lightning			2			2
Total	20	28	17	12	22	99

Table S.3. Frequencies of PVTS events and precursors (1963–1981)

Type of frequency	Based on our finds[a]		Corrected for completeness[a]	
	Events/RY	Percent of total	Events/RY	Percent of total
Modes of reactor operation				
1 – Power	0.161	54	0.467	48
2 – Startup	0.043	14	0.144	15
3 – Hot standby	0.046	15	0.156	16
4 – Hot shutdown	0.012	4	0.042	>4
5 – Cold shutdown	0.024	8	0.100	10
6 – Refueling	0.015	5	0.063	>6
Total	0.301	100	0.972	100
Initiating sequences				
LBLOCA	0	0	0	0
SBLOCA	0.033	11	0.091	9
OF/Cold FW	0.064	22	0.161	17
HSF	0.043	14	0.097	10
HPSI	0.125	41	0.481	49
CF/LPSI	0.012	4	0.050	5
EX	0.003	1	0.005	<1
Others	0.021	7	0.087	9
Total	0.301	100	0.972	100
Causes of initiation				
Operator error	0.036	12	0.142	15
Maintenance/test error	0.021	7	0.080	8
Valve failure	0.064	21	0.153	16
Power failure	0.033	11	0.114	12
I&C design error/ spurious signal	0.046	15	0.156	16
System/equipment failure	0.076	25	0.227	>23
Water hammer	0.018	6	0.075	8
Fire/lightning	0.006	2	0.025	3
Total	0.301	100	0.972	100

[a]Round-off errors in event frequencies and percentages occasionally result in apparent discrepancies in the totals.

by such errors. However, when fully corrected for factors of completeness, only 1 out of every 15 such errors resulted in a significant event.

Valve failures accounted for about 20% of all events, and one-third of these were significant. These valve failures include items such as pressurizer PORVs and steam dump PORVs failing to close after a service demand, and FW regulating valves opening too wide for a certain reactor power level.

Over one-fourth of the events were caused by power failure or design inadequacies in the I&C system. Another one-fourth were caused by hardware and system failures that have not been categorized as valves and I&C failure. In fact, when all three categories — valves, I&C power, and systems and hardware — are considered together, they account for two-thirds of all PVTS events and precursors, although only 30% of these were significant.

Distribution of Events Among Vendors and Utilities

Table S.4 shows the number of reactors, RYs, and PVTS-like events for the three U.S. PWR vendors. Table S.5 shows frequencies of these events per intrinsic RY for each vendor, both uncorrected and corrected for factors of completeness. Similar data are given for architects-engineers (A-Es) and utilities in Chap. 7 but are not shown here due to lack of a trend in these data.

Table S.4. Distribution of reactors, RYs, and PVTS events and precursors among U.S. PWR vendors (1963–1981)

Vendors	Reactors		RYs		Number of PVTS-like events	
	Number	Percent	Number	Percent	Number	Percent
Babcock & Wilcox	8	17	52.68	16	20	20
Combustion Engineering	8	17	54.39	17	11	11
Westinghouse	31	66	222.19	67	68	69
Total	47	100	329.26	100	99	100

In terms of the total number of PVTS-like events, the distribution among reactor vendors is pretty much the same as the distribution of RYs or number of reactors. Combustion Engineering (C-E) reactors tend to have a lower overall frequency. In terms of severity of PVTS-like events, however, the frequencies appear to be in the ratio 3:2:1 among Babcock & Wilcox (B&W):C-E:Westinghouse (W) reactors. The only obvious difference between W and the other reactors is that the former category has three to

Table S.5. Frequencies of PVTS events and precursors among U.S. PWR vendors (events/intrinsic RY)[a] (1963–1981)

Vendors	Uncorrected			Corrected for completeness		
	Insignificant	Significant	Total	Insignificant	Significant	Total
Babcock & Wilcox	0.152	0.228	0.380	0.608	0.342	0.950
Combustion Engineering	0.074	0.129	0.203	0.296	0.194	0.490
Westinghouse	0.238	0.068	0.306	0.952	0.102	1.054

[a]Intrinsic RYs are those in Table S.4 for each vendor when uncorrected. After correction, they are the same for Babcock & Wilcox and Combustion Engineering but decrease to 213.19 RYs for Westinghouse.

four SGs, each attached to a hot and a cold reactor coolant system (RCS) leg; while C-E and B&W reactors have only two SGs, each attached to one RCS hot leg and two RCS cold legs. In addition, B&W reactors in the past appeared to have weaknesses in the nonnuclear instrumentation and integrated control systems that could initiate or aggravate PVTS-like events. These weaknesses appear to have been corrected since the Rancho Seco event of March 20, 1978, the Three Mile Island accident of March 28, 1979, and the Crystal River 3 incident of February 26, 1980.

There are 30 different utilities operating the 47 PWR reactors in our data base. Rather than studying the distribution of the PVTS-like events among them — the statistics of such study being certainly poor — we present mappings of the event on each reactor operating history both in terms of RYs and in real time. These mappings are shown in Figs. S.1 and S.2. Note that each symbol (0, 1, 2, 3, and 4) denotes an event in terms of its severity. Although the symbols can be close enough to cause confusion, no number is higher than 4 except where clearly noted (e.g., 14 are 2 events happening within a short time — a few months — apart, one was of severity 1, the other of severity 4).

There appears to be no time pattern to the occurrence of PVTS events and precursors. An exception is at the preoperational testing and during the first year of operation when both systems and personnel are presumably still trying to debug problems. Furthermore, there appears to be no learning effect between two reactors on the same site or between two sites operated by the same utility. This tentative conclusion, however, is to be subjected to further studies in light of Robert's learning theory.[3]

There were 13 reactors with no PVTS-like events, including 2 by B&W, 2 by C-E, and 9 by W (again quite proportional to the number of reactors and RYs of each vendor). At this point, we have not been able to determine whether the lack of PVTS-like events for these reactors was due to a good product, a good operating crew, a different LER reporting procedure, the deficiencies of the NSIC data bank, or all of these.

Which Was the Most Severe PVTS Event?

Most analytical work on PVTS has focused on the March 20 event at Rancho Seco as the most severe of U.S. experience. Our reading and cursory analyses of LERs indicated that the February 26, 1980, event at Crystal River 3 may be more severe. While both events were initiated by a short in the nonnuclear instrumentation system, which then fed wrong signals to the integrated control system, the Rancho Seco event was basically an OF event, whereas the Crystal River 3 event was basically an SBLOCA event followed by sustained SI. The Rancho Seco event lasted 70 min with the RCS reaching 285°F and repressurized by HPSI to 2000 psi. The Crystal River 3 event lasted 87 min, with the RCS reaching 250°F and repressurized by HPSI up to 2400 psi. While detailed analyses need to be done, it is thought that areas of the pressure vessel of Crystal River 3 could have seen even lower temperatures. Exactly how low such temperatures were would depend on the nature of the reverse flow and whether or not the vent valves opened.

ORNL–DWG 82–6061 ETD

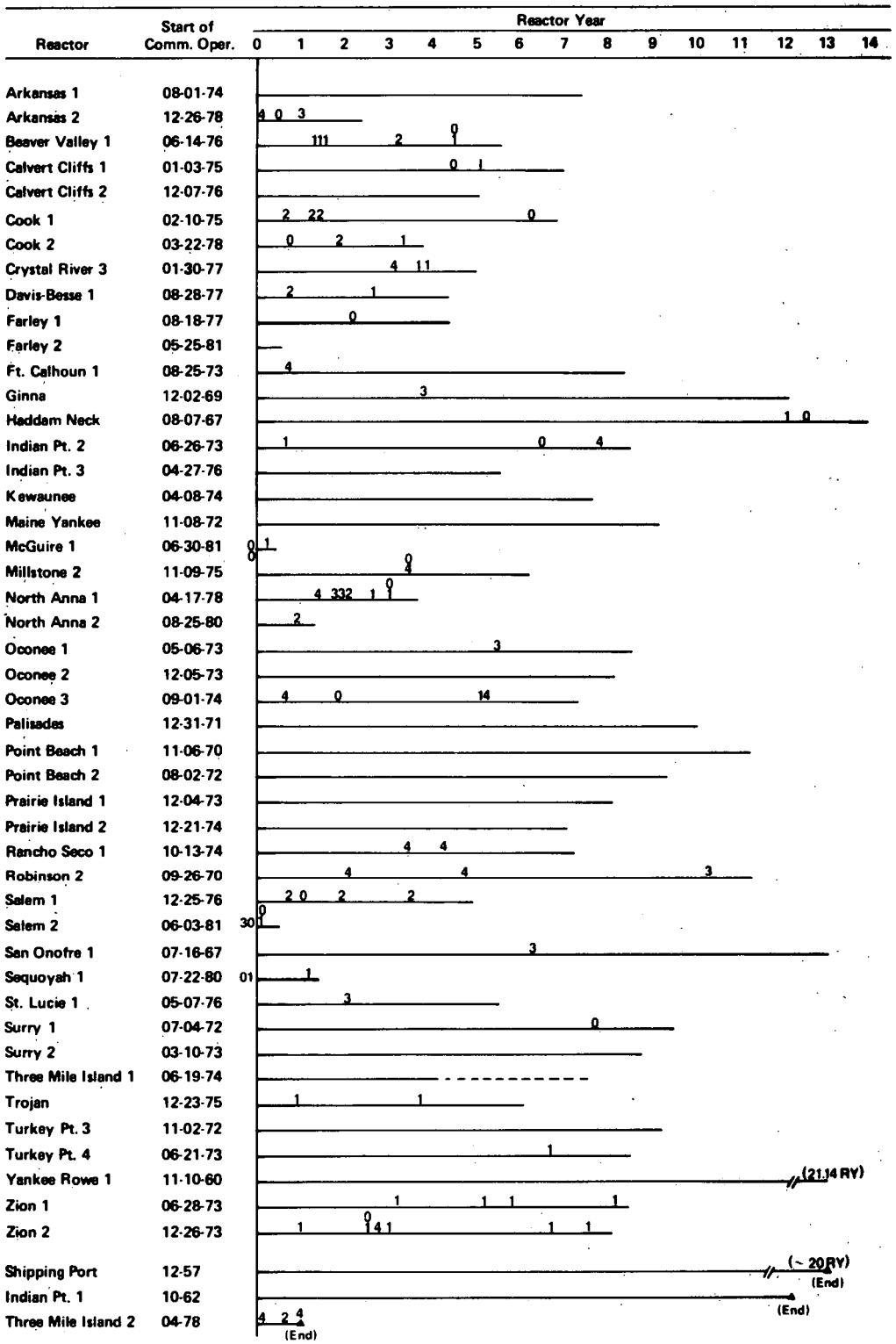

Fig. S.1. Time line display by severity symbols of U.S. PWR PVTS events vs reactor years.

ORNL–DWG 82–6967 ETD

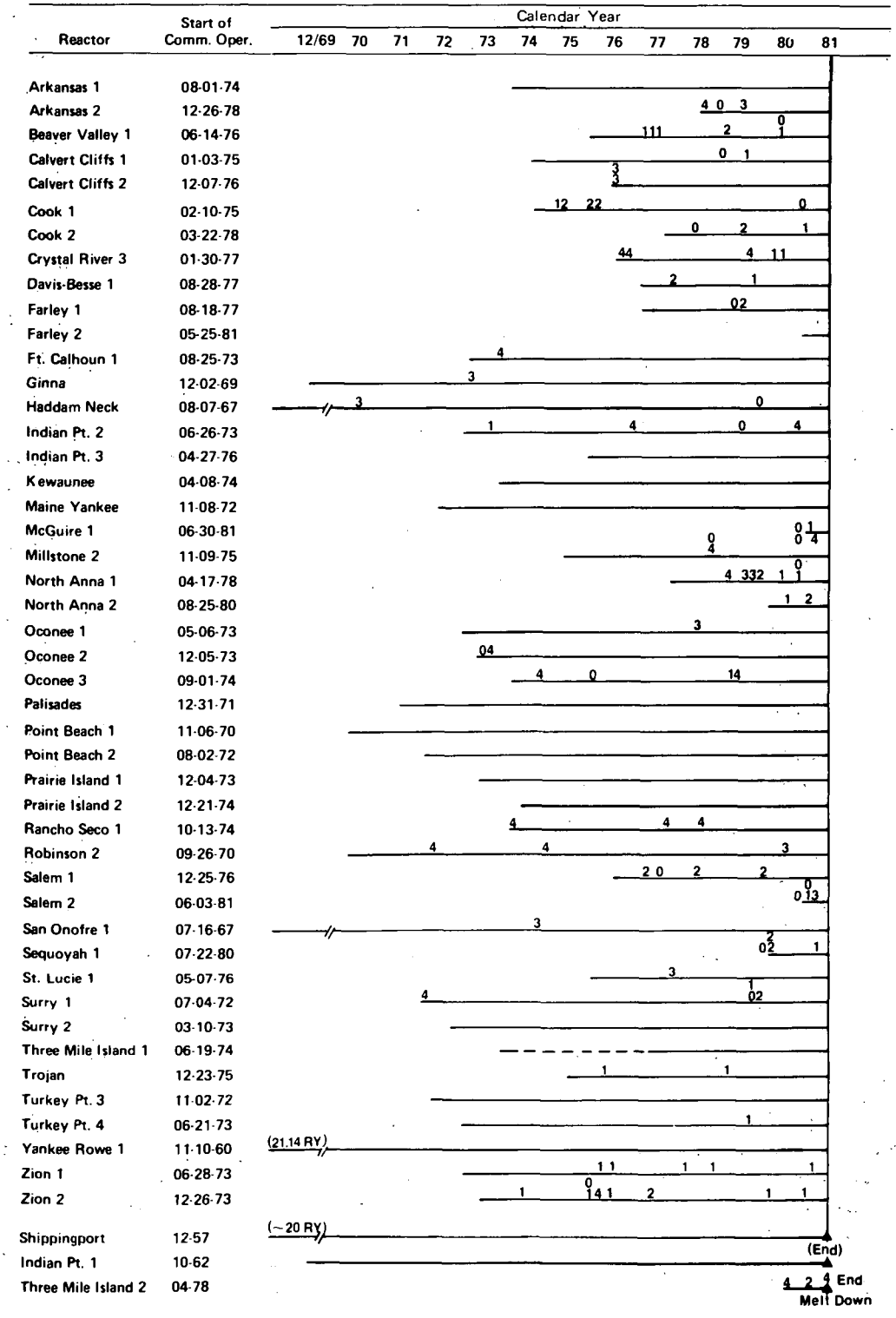

Fig. S.2. Time line display by severity symbols of U.S. PWR PVTS events vs real time.

Items of Interest to Probabilistic Risk Analyses on PVTS

The 34 significant PVTS events were examined to answer the following questions:

1. How long did it take the operator to become aware of the initiation of the event?
2. If there was a PORV stuck open, how long did it take the operator to successfully close it?
3. Did the RCS main circulation pumps shut down?
4. Was there repressurization?

Table S.6 provides a summary of our observations.

Table S.6. Summary of system/operator actions following significant event initiation

Time to operator notice/action	Number of cases
<5 min	16
Between 5 and 10 min	8
Between 10 and 15 min	1
Between 15 and 30 min	1
Not clear	7
Knew something wrong but waited until reactor shutdown	1
Total	34
Time to correct for valve failure[a]	
<5 min	3
Between 5 and 10 min	1
Between 10 and 60 min	2
Long	1
Not clear	1
No success	1
Total	9
Reactor coolant system circulation pumps condition	
Shutdown (automatic, manual, or already down)	14
Not shutdown	12
Not clear	8
Total	34
Reactor coolant system repressurization	
No	6
Yes	17
Not clear	11
Total	34

[a]Nine cases of 34 significant events involved PORV or other valve failures.

References

1. S. K. Iskander, R. D. Cheverton, and D. G. Ball, *OCA-1, A Code for Calculating the Bahavior of Flaws on the Inner Surface of a Pressure Vessel Subjected to Temperature and Pressure Transients*, NUREG/CR-2113 (ORNL/NUREG-84) (August 1981); updated version, OCA-2, September 1982.

2. U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, *Review of Licensee Event Reports (1976-1977)*, NUREG-0572 (1979).

3. P. C. Roberts and C. Burwell, *The Learning Function in Nuclear Reactor Operation and Its Implications for Siting Policy*, ORAU/IEA-81-4(M) (May 1981).

PRESSURE VESSEL THERMAL SHOCK AT U.S. PRESSURIZED-WATER REACTORS: EVENTS AND PRECURSORS, 1963–1981

Doan L. Phung*

ABSTRACT

This report searches for and analyzes events and precursors to events of pressure vessel thermal shock (PVTS) at U.S. pressurized-water reactors (PWRs). The data base includes close to 16,000 licensee event reports of 47 PWRs, covering 329 reactor years (RYs) from 1963 through 1981.

Ninety-nine events were found in half as many computer searches. Original reports and other published data on these finds were located and subsequently analyzed to extract information on severity, initiation sequences, causes, and frequencies.

Five categories of severity were defined, with categories 0, 1, and 2 being considered insignificant and categories 3 and 4 being considered significant. In addition, an event of lower severity is considered a precursor to a higher severity event of the same cause and initiation sequence. An exhaustiveness analysis indicated that the 34 significant events may have covered two-thirds of all such events, while the 65 insignificant events may represent only 25% of all such events that took place in operating plants.

About one PVTS-like event was found to occur per RY when the above factors of completeness were accounted for; 16% of all such events were significant. About 84% occurred when the reactors were in hot conditions. High-pressure safety injection, by spurious signals or by sensitivity of control setting, was the sequence for about half of all events but only 12% of the significant ones. Overcooling by the power conversion circuits was the sequence for one-fourth, and small-break loss-of-coolant accident was the sequence for 9%. When considered in terms of causes of initiation, then two-thirds of all events were caused by inadequacies in hardware and designs, one out of five such events being significant. One-fourth of all events was caused by operator/maintenance/test errors, and 1 out of 25 such errors led to a significant event.

The breakdown of events according to vendors, architect-engineers, and utilities was also provided. The frequency of PVTS-like events was found to be lower for Combustion Engineering reactors, but the frequencies of significant events from 1969 through 1981 were in the ratio 3:2:1 for Babcock & Wilcox:Combustion Engineering:Westinghouse reactors. Thirteen

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*Institute for Energy Analysis, Oak Ridge Associated Universities.

reactors of various designs and ages were found without any events. No significant learning pattern was found between units among multiunit sites and between plants among utilities with more than one plant.

Data were also provided concerning four items of interest to probabilistic risk analysis of PVTS: time to operator reaction, time to successful closure of stuck-open valves, status of main coolant recirculation pumps, and system repressurization frequency.

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1. SCOPE OF THIS REPORT

The U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry are conducting an intensive examination of potential thermal shock events that could lead to failure of a pressurized-water reactor (PWR) vessel. While such failure has always been a matter of prime concern within nuclear safety circles,[1-17] the current inquiry is new in its intensity and implication.[18-25]

Pressure vessel thermal shock (PVTS) is the event or consequence of events during which the reactor vessel is exposed to an environment (chiefly the reactor coolant) that is at a temperature quite different from that of the vessel wall. The rapid heat transfer taking place during such an event gives rise to high transient thermal stress — the thermal shock — within the vessel metal. A PVTS is colloquially called a pressurized thermal shock (PTS) when the transient thermal stresses are accompanied by high reactor coolant pressures such as could be caused by the action of the emergency core cooling system (ECCS). When these transient thermal stresses are combined with the pressure loads and with the effect of fast neutron fluences on vessel welds, it can be calculated (using fracture mechanics) that the vessel may fail under certain circumstances.[18,21,24-26]

As of 1981, there were 49 large PWRs that had produced commercial electricity in the United States, with a combined dependable capacity of ~39,000 MW(e). Some of these reactors, such as McGuire 1 and Sequoyah 1, were newly commissioned, while others such as Yankee Rowe, San Onofre, and Connecticut Yankee were already in their second or third decade of power production. Three of the reactors had been shut down permanently: Shippingport, Indian Point 1, and Three Mile Island 2 (which closed down because of its well-known core-damage accident of March 28, 1979). The cumulative reactor years (RYs) were 364 as of December 1981.

This report focuses on the PVTS events and precursors in the operating experience of 329 RYs of U.S. PWR operation (reason for the lower number given in Chap. 2). A study of operational incidents, or licensee event reports (LERs), is important not only to avoid mistakes of the past but also to identify root causes of potentially more serious accidents. Indeed, the technique of probabilistic risk analysis (PRA), which has been used increasingly since the advent of the Rasmussen Reactor Safety Study (WASH-1400),[27] would be rather hollow without realistic accident sequence

and cause frequencies that must be obtained from experience. Harold Lewis' review of WASH-1400 strongly recommended the study of LERs for accident sequence precursors.[28] This recommendation was supported by the Advisory Committee on Reactor Safeguards (ACRS) and by the NRC.[28,29]

This study is intended to assist in the ongoing examination of PVTS by NRC and industry. An attempt will be made to answer the following questions:

1. What event sequences may lead to PVTS in PWRs?
2. How many PVTS events or precursors have occurred in the operating history of U.S. PWRs?
3. What is the nature of these events regarding severity, causes, time of occurrence, and trends?
4. Can these events help answer some current controversial questions concerning PVTS [e.g., high-pressure safety injection (HPSI) function, reactor coolant pump function, warm prestressing credit, and operator actions)?

Although the study of potential or hypothetical PVTS events and their probabilities and consequences is of intense interest, such a probabilistic analysis would mislead the fact-finding nature of the present mission. Therefore, we shall refrain from calculating the cumulative probability of reactor vessel failure due to the PVTS-like events that we found.

2. DATA BASE

2.1 U.S. PWR Reactor Years and Reactor Vessel Data

Table 2.1 contains data on U.S. PWRs that are pertinent to the question of PVTS. Forty-six operating reactors are listed in Part 1 of the table, while Part 2 lists three reactors that have been permanently shut down. For each reactor, the owner, the nuclear steam supply system (NSSS) vendor, the architect-engineer (A-E), the constructor, the dependable electrical capacity, and the date of first electrical generation are listed. Also listed is information on the reactor vessel, such as manufacturer, copper content in welds, fluence per effective full-power year (EFPY), and the cumulative EFPY as of the end of 1981. These data are necessary to determine the shift in the nil ductility transition temperature (RT_{NDT}) of the reactor vessel and its capability to withstand a certain PVTS at a specific time in its operational lifetime.

A common denominator to event and cause/frequency determination — a prime objective of this study — is the number of operating RYs. There are four groupings of RYs: those covering all PWRs that produced electricity, those covering only operable PWRs, those covering the LERs under study, and those RYs corrected for certain discrepancies in the data base. An analysis of Table 2.1 yields the following results:

RY groupings	Reactor years (up to 12/81)
All 49 U.S. PWRs that have produced electricity	363.51
Only 46 operable PWRs	330.46
46 operable PWRs and Three Mile Island 2 that have LERs in the data base since 1963	329.26
RYs used for frequency calculations	319.78

Note that between 1957 (Shippingport started) and 1963 [Nuclear Safety Information Center (NSIC) started] there was little information on LERs in the public domain. Real efforts to compile LERs began only in 1969. The fourth group of RYs listed in the preceding paragraph is the third group less the period from 1963 through 1969.

2.2 Sources of Operational Events

PVTS events, precursors to PVTS events, and/or failure data could, in principle, be found from the following sources of reactor operating

Table 2.1. U.S. PWR data pertinent to PVTS

(As of December 31, 1981)

Reactor[a]	Utility/ vendor	Architect-engineers/ constructors	Gross maximum[b] dependable capacity [MW(e)]	Date of first electrical generation	Reactor pressure vessel[c]			Reactor years[d] EFPY (through 12/81)
					Manufacturer	Percent copper	Fluence per EFPY (10^{18} neutrons/cm^2)	
Arkansas 1	Ark. P&L B&W	Bechtel Bechtel	883	08/01/74	B&W	0.31 0.61	0.45	7.42 4.50
Arkansas 2	Ark. P&L C-E	Bechtel Bechtel	897	12/26/78	C-E	0.05 0.04	*e*	3.02 1.81
Beaver Valley 1	Duquesne W	S&W S&W	860	06/14/76	C-E	0.36 0.37	0.25 1.69	5.55 1.93
Calvert Cliffs 1[a]	Balt. G&E C-E	Bechtel Bechtel	860	01/03/75	C-E	0.30 0.30	1.47	7.00 5.09
Calvert Cliffs 2[a]	Balt. G&E C-E	Bechtel Bechtel	860	12/07/76	C-E	0.30 0.23	1.47	5.07 4.40
Cook 1	Ind. & Mich. Elec. W	AEP AEP	1080	02/10/75	C-E	0.13 0.40	0.34 0.63	6.89 4.73
Cook 2	Ind. & Mich. Elec. W	AEP Jones	1118	03/22/78	CB&I	*e*	*e*	3.78 2.52
Crystal River 3	Fla. Pwr. B&W	Gilbert Jones	821	01/30/77	B&W	0.31 0.35	0.55 0.58	4.92 2.66
Davis-Besse 1	Toledo Ed. B&W	Bechtel Bechtel	918	08/28/77	B&W	Forging 0.24	0.66	4.34 1.81
Farley 1	Ala. P&L W	Southern Serv. Bechtel	845	08/18/77	C-E	0.38 1.69	*e*	4.37 2.31
Farley 2	Ala. P&L W	Southern Serv. Bechtel	860	05/25/81	*e*	*e*	*e*	0.61 0.59
Ft. Calhoun[a]	Omaha Publ. C-E	Gibbs et al. Gibbs et al.	501	08/25/73	C-E	0.35 0.35	1.72	8.35 5.24

Table 2.1 (continued)

Reactor[a]	Utility/ vendor	Architect-engineers/ constructors	Gross maximum[b] dependable capacity [MW(e)]	Date of first electrical generation	Reactor pressure vessel[c]			Reactor years[d] EFPY (through 12/81)
					Manufacturer	Percent copper	Fluence per EFPY (10^{18} neutrons/cm^2)	
Ginna[a]	Rochester G&E W	Gilbert Bechtel	490	12/02/69	B&W	0.25	1.16	12.08 8.40
Haddam Neck[a] (Conn. Yankee)	Conn. Yankee Pwr. W	S&W S&W	582	08/07/67	C-E	0.22	1.09 1.31	14.40 11.88
Indian Pt. 2	Con. Ed. W	UE&C WEDCO	885	06/26/73	C-E	0.25 plate	0.50	8.52 4.71
Indian Pt. 3	PASNY W	UE&C WEDCO	1000	04/27/76	C-E	0.24 plate	0.56	5.68 2.99
Kewaunee	Wisc. Pub. W	Pioneer Pioneer	545	04/08/74	C-E	0.20	1.34	7.73 5.94
Maine Yankee[a]	Me. Yankee C-E	S&W S&W	850	11/08/72	C-E	0.36 0.85	0.85	9.15 6.22
McGuire 1	Duke W	Duke Duke	1181	06/30/81	*e*	*e*	*e*	0.51 0
Millstone 2	NE Nuclear C-E	Bechtel Bechtel	895	11/09/75	C-E	0.37	0.56	6.15 4.03
North Anna 1	Va. Elec. W	S&W S&W	918	04/17/78	RDM	Forging 0.14	1.78	3.71 2.42
North Anna 2	Va. Elec. W	S&W S&W	939	08/25/80	RDM	Forging 0.13	1.78	1.35 0.99
Oconee 1[a]	Duke B&W	Duke & Bechtel Duke	899	05/06/73	B&W	0.31	0.54	8.66 5.05
Oconee 2[a]	Duke B&W	Duke & Bechtel Duke	899	12/05/73	B&W	Forging 0.35	0.61	8.07 4.99
Oconee 3	Duke B&W	Duke & Bechtel Duke	899	09/01/74	B&W	Forging 0.24	0.61	7.33 4.86

Table 2.1 (continued)

Reactor[a]	Utility/ vendor	Architect-engineers/ constructors	Gross maximum[b] dependable capacity [MW(e)]	Date of first electrical generation	Reactor pressure vessel[c]			Reactor years[d] EFPY (through 12/81)
					Manufacturer	Percent copper	Fluence per EFPY (10^{18} neutrons/cm^2)	
Palisades	Consumers C-E	Bechtel Bechtel	675	12/31/71	C-E	0.25	1.16	10.00 4.75
Point Beach 1[a]	Wisc. Elec. W	Bechtel Bechtel	519	11/06/70	B&W	0.24	0.91 1.24	11.15 8.02
Point Beach 2[a]	Wisc. Elec. W	Bechtel Bechtel	519	08/02/72	B&W/C-E	0.25	1.24	9.42 7.53
Prairie Island 1	Northern States W	Fluor Pioneer Northern States	534	12/04/73	SFAC	Forging 0.14	1.34	8.08 5.87
Prairie Island 2	Northern States W	Fluor Pioneer Northern States	531	12/21/74	SFAC	Forging 0.19	1.34	7.03 5.57
Rancho Seco 1	Sacramento Mun. B&W	Bechtel Bechtel	917	10/13/74	B&W	0.31	0.58 0.66	7.22 3.82
Robinson 2[a]	Carolina P&L W	Ebasco Ebasco	700	09/26/70	C-E	0.34	1.59 1.72	11.27 7.79
Salem 1	Publ. Serv. E&G W	Publ. Serv. E&G UE&C	1124	12/25/76	C-E	0.24 plate	1.16	5.02 2.41
Salem 2	Publ. Serv. E&G W	Publ. Serv. E&G UE&C	1149	06/03/81	*e*	*e*	*e*	0.58 0.45
San Onofre 1[a]	South Ca. Ed. W	Bechtel Bechtel	456	07/16/67	C-E	0.19	1.74	14.46 9.50
Sequoyah 1	TVA W	TVA TVA	1163	07/22/80	RDM	*e*	*e*	1.45 0.74
St. Lucie 1	Fla. P&L C-E	Ebasco Ebasco	822	05/07/76	C-E	0.30 0.31	0.63	5.65 4.20
Surry 1	Va. Elec. Pwr. W	S&W S&W	811	07/04/72	B&W	0.18 0.25	0.34 1.56	9.49 4.96

Table 2.1 (continued)

Reactor[a]	Utility/ vendor	Architect-engineers/ constructors	Gross maximum[b] dependable capacity [MW(e)]	Date of first electrical generation	Reactor pressure vessel[c]			Reactor years[d] EFPY (through 12/81)
					Manufacturer	Percent copper	Fluence per EFPY (10^{18} neutrons/cm^2)	
Surry 2	Va. Elec. Pwr. W	S&W S&W	811	03/10/73	B&W/RDM	0.31 0.19	0.34 1.56	8.81 4.78
Three Mile Is. 1[a]	Met. Ed. B&W	Gilbert UE&C	840	06/19/74	e	e	e	4.78 3.56
Trojan	Portland GE W	Bechtel Bechtel	1122	12/23/75	CB&I	0.15 plate	0.69	6.03 3.29
Turkey Pt. 3[a]	Fla. P&L W	Bechtel Bechtel	680	11/02/72	B&W	0.31	1.97	9.16 5.85
Turkey Pt. 4[a]	Fla. P&L W	Bechtel Bechtel	680	06/21/73	B&W	0.30	1.97	8.53 5.93
Yankee-Rowe	Yankee Atomic W	S&W S&W	180	11/10/60	B&W	0.20 plate	0.78	21.14 14.67
Zion 1	Comm. Ed. W	Sargent & Lundy Comm. Ed.	1085	06/28/73	B&W	0.31 0.35	0.20 0.63	8.51 5.04
Zion 2	Comm. Ed. W	Sargent & Lundy Comm. Ed.	1085	12/26/73	B&W	0.35 0.26	0.20 0.63	8.02 4.68
		Subtotal	37,888					RY 330.46 EFPY NA
Others								
Shippingport	Duquesne/DOE W	S&W	60 net	12/57	e	e	e	~20
Indian Point 1	Con. Ed. B&W	Owner	265 net	10/62	e	e	e	12.12
Three Mile Is. 2	Met. Ed. B&W	Burns & Roe	906 net	04/78	e	e	e	0.93
		Subtotal	1,231					RY 33.05 EFPY NA
		Total	39,119					RY 363.51

Footnotes for Table 2.1

[a]Reactors having rather high values of nil ductility transition temperature (RT_{NDT}). This value is a function of original reference temperature (RT_{NDT_0}), copper content in welds, and total fluence. For TMI-1, the RYs are counted up to March 29, 1979.

[b]Gross maximum dependable capacity (GMDC) is defined as the gross electrical output at the terminals of the turbine generator during the most restrictive weather conditions. Station power need is about 30–40 MW(e) and is subtracted from GMDC to obtain the net MDC.

[c]Where two values are given, the first is for longitudinal welds, and the second is for circumferential welds. When there is no longitudinal weld at the reactor beltline area, the word "forging" is used.

[d]Reactor years are counted from the date of first electric generation through December 31, 1981. Effective full-power years (EFPYs) are obtained by multiplying reactor years to the cumulative capacity factor.

[e]Information not available.

Abbreviations:

SFAC = Creusot Loire
P&L = Power and Light
G&E = Gas and Electric
B&W = Babcock & Wilcox
CB&I = Chicago Bridge and Iron
C-E = Combustion Engineering
W = Westinghouse
S&W = Stone and Webster
AEP = American Electric Power Service Corp.
UE&C = United Engineers and Constructors, Inc.
TVA = Tennessee Valley Authority
RDM = Rotterdamsche Droogdok Mij

Sources:

1. (Graybook): *Licensed Operating Reactors*, Status Summary Report, NUREG-0020, 5(10) (October 1981 and January 1982).

2. *U.S. Central Station Nuclear Electric Generating Units: Significant Milestones*, DOE/NE-0030/3/81 (September 1981).

3. T. E. Murley (NRC), "Pressurized Thermal Shock of Pressure Vessels," material presented to the Advisory Committee on Reactor Safeguards, September 11, 1981.

experience:

1. Operating records of individual reactors.
2. LER data banks of both the NRC and NSIC. The former bank was started after the separation of NRC from the old Atomic Energy Commission but was discontinued in 1980 in favor of the NSIC data bank, which has been in continuous operation since 1963.
3. Nuclear Plant Reliability Data System (NPRDS) and related topical reports issued by the Electric Power Research Institute (EPRI), the Nuclear Safety Analysis Center (NSAC), and the Institute of Nuclear Power Operation (INPO).
4. Special topical reports on operational experience issued by the NSIC and their published versions in the bimonthly journal *Nuclear Safety*.
5. NRC Report to Congress on Abnormal Occurrences, NUREG-0090 (Ref. 30), compiled quarterly since 1975 as a charter requirement of the NRC, including topical reports related to the same subject.
6. Special PVTS reports by PWR owners' groups and/or NSSS vendors.

It should be obvious that the most complete of these sources of information are the records of individual reactors, but they are also the least accessible. On the other hand, events of safety significance, as predetermined by tech specs and/or other regulatory requirements, are selected from these records by the reactor owners and reported to the NRC. Copies of these reports are routinely sent to the NSIC in Oak Ridge, screened, coded, and entered into the NSIC data bank. Thus, the NSIC data bank represents next to the best data base available, and the most accessible.

Our principal effort in this review has been to search for events from the NSIC data bank that have the characteristics of PVTS defined previously. When a list of such events has been established and their occurrences have been analyzed, PVTS events as reported in other cited sources have also been reviewed, where possible, to compare with and complement the NSIC data base. Last but not least, PVTS analyses by the reactor vendors and by utilities are also reviewed to compare findings.

2.3 NSIC Data Bank

As of December 1981, the NSIC data bank contained ~170,000 entries. The majority of entries are related to reportable occurrences, regulatory questions and answers, safety analyses, and research and development (R&D) reports sponsored by the NRC. Some 20% of the entries are nuclear safety-related reports sponsored by the Department of Energy (DOE), proceedings of technical meetings, abstracts of journal articles, foreign documents, and other matters. There are ~46,000 entries (27%) related to PWRs, and 15,000 of those are reportable events or LERs. Also note that the number of LERs — both for PWRs and boiling-water reactors (BWRs) — increases faster with respect to other entries, as can be seen in a 5% plot of the number of LERs for PWRs vs the overall growth of the NSIC data base (Fig. 2.1). For example, between NSIC 70,000 and 80,000 (circa 1973–1974), there are about 340 LERs for PWRs (3.4% of the total), whereas between NSIC 150,000 and 160,000 (circa 1980), there are about 2,600 LERs for PWRs

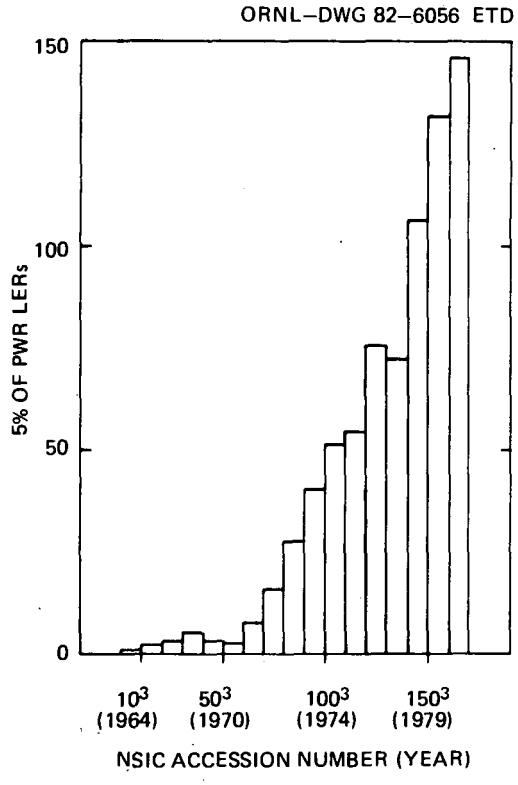

Fig. 2.1. Plot of 5% of PWR LERs against NSIC accession numbers.

(26%). Several reasons exist for this phenomenon: (1) The number of reactors under licensing is diminishing (thus licensing questions and responses are diminished), (2) the number of reactors in commercial operation is increasing, and (3) the reporting of LERs became more stringent after the Three Mile Island accident.

Entries in the NSIC data bank are recorded in the order of their receipt after preliminary logic, system, and typographical analysis by the NSIC staff. Each entry, mainly LERs, is assigned an accession number, a title describing the event, the date of occurrence, the report availability, the abstract of event sequence, the main component and system involved, and a list of keywords. The keywords are instrumental to the retrieval of the citation from the data bank when the analyst has only an idea about the subject matter but no precise information on the specific event. Over the years, the NSIC has devised, organized, and logged in over 3000 keywords designed to cover the subject matter accumulated in the data bank. Thus, each entry will contain a dozen or so keywords, which, according to the LER screener, best represent the event report as received. Understandably, these keywords cannot totally describe the events as they actually occurred or as they might be interpreted by another analyst.

3. METHODOLOGIES

The LER search methodology includes a logical approach that narrows the data base to specific areas of interest, the Boolean combination of keywords that turns up appropriate events from the computer data base, and the procurement of original reports for detailed analyses. The LER event analytical methodology includes constructing seven generalized event trees and defining five PVTS severity categories for event and precursor classification. The check of the "exhaustiveness" of the finds includes comparison of this list of events against 10% samples of all PWR LERs in the NSIC data bank and against industry's published data.

3.1 Logic for PVTS Sequences

Figure 3.1 illustrates a logical approach in defining sequences or classes of sequences in PWRs that may lead to PVTS.

By definition, PVTS is the result of changes of vessel metal temperature with respect to time, with or without coincident changes in the pressure of the reactor coolant system (RCS). Thus, PVTS in PWR vessels could only result from temperature transients due to changes in the environments adjacent to the inside and outside of the vessel wall.

The PVTS from outside of the reactor vessel could be caused by the falling off of sections of the metal insulation, resulting in the asymmetric loss of heat from the reactor vessel. Or it could be caused by the

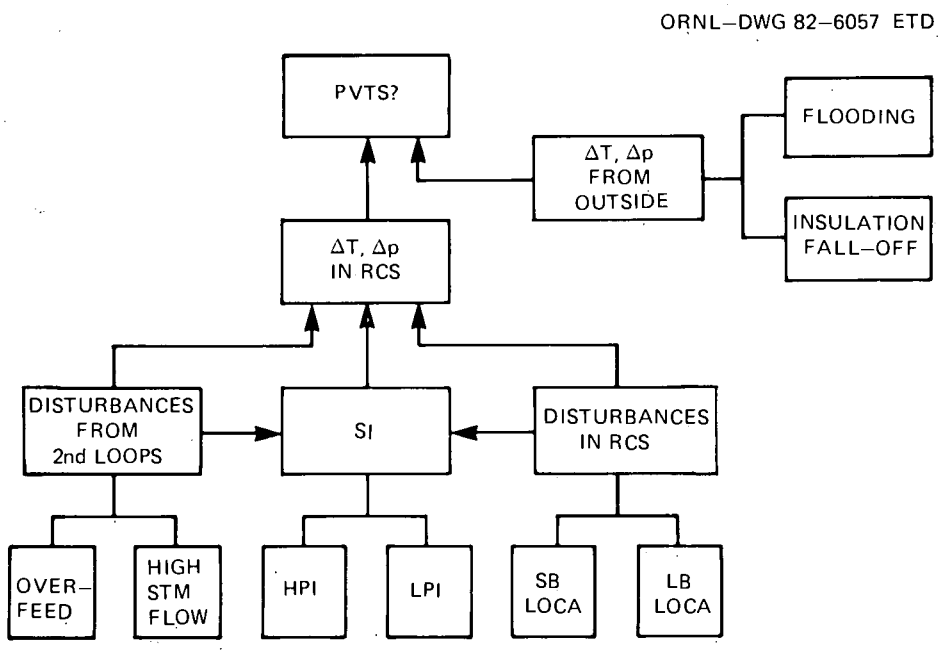

Fig. 3.1. PVTS LER search logic.

reactor cavity flooding with cold water while the reactor is under operation, resulting in asymmetric and possibly sudden changes in heat loss from the hot vessel. The latter case did indeed occur at Indian Point 2 on October 17, 1980.

Thermal shocks from inside the reactor vessel are by far more numerous, more severe, and of more intense concern. They can be the result of the loss of coolant from the RCS, of overcooling the reactor coolant by transients in the feedwater-steam system, or of the injection of ECCS water.

Loss of coolant from the RCS has been conventionally classed into two categories: large-break loss-of-coolant accidents (LBLOCA) and small-break loss-of-coolant accidents (SBLOCA). An LBLOCA occurs when a large pipe, such as the cold leg or the hot leg, suffers a break, either double-ended or longitudinally. This kind of accident has been the subject of intense study and design since the early days of commercial nuclear reactors. The SBLOCA is far more common, however. It occurs (1) when a small pipe, such as an instrumentation line or the feed-and-bleed line, fails; (2) when one of the relief valves fails to close following a service demand; (3) when the seal of the RCS pumps leaks; or (4) when some SG tubes rupture. An LOCA results in the fast drop in RCS pressure (in a matter of seconds to minutes depending on the size of the break) and the concomitant drop in temperature (after the saturation point has been reached). The drop in pressure also actuates the ECCS, specifically the HPSI when the RCS drops below ~1700 psig, the core flooding system (CFS) when the RCS pressure drops below ~600 psig, and the low-pressure safety injection/residual heat removal (LPSI/RHR) systems when the RCS pressure drops even lower. The ECCS system injects cold water into the reactor vessel to ensure that the core is covered.

Reactor coolant could also be chilled below its operating temperature ($T_h \approx 590°F$, $T_c \approx 550°F$, $T_{ave} \approx 570°F$, $p \approx 2250$ psig) in a matter of seconds to a few minutes by the high flow in the feedwater-steam circuits. By design, the heat transfer coefficient in the SGs is very high (capable of chilling the RCS by 100°F in ~20 s if full FW flow and RCS pumps continued after reactor scram). The FW flow rate and temperature and the steam flow rate and temperature are controlled by a system of finely tuned regulating, check, and bypass valves to ensure efficient power production. Thus, any disturbance in the FW or steam conditions is transmitted quickly to the RCS. If either the FW flow or the FW level in the SGs is below normal, or if the steam flow to the turbine gets disrupted, then the temperature and pressure of the RCS may increase quickly, usually resulting in a reactor trip and the occasional opening of relief valves or safety valves. If the FW flow or level is too high, or if the steam flow is too high, then the temperature and pressure of the RCS may decrease quickly, resulting in possible ECCS actuation and PVTS. This also results when FW flow is normal but its inlet temperature is too low.

While the ECCS is designed such that it is almost always actuated at abnormal conditions in the RCS or in the FW-steam circuits, it can also be actuated independently. This can be effected manually by the operator, automatically by spurious signals, or by inadvertent pump action or valve opening during testing and maintenance.

3.2 Combining Keywords for LER Searches

The RECON* computer network was used to get access to the NSIC data base, and over 40 searches were conducted for PVTS events and/or precursors at U.S. PWRs. Keywords were coined to represent major equipment, systems, events, and conditions that would be involved in the PVTS logical sequences described in the preceding section. These keywords were next compared against a master NSIC keyword list to ensure their acceptability to the computer.

Table 3.1 is a sample of LER searches for PVTS events and precursors. When one keyword is used, the computer turns up all entries in the data bank that contain the keyword. For example, when the word "failure" is used, RECON turns up 29,604 citations. Narrowing down the number of citations is therefore important if the reading is to be tractable. The RECON network accepts Boolean combinations (AND,OR,NOT) of keywords to make such focusing simple. For example, the keyword "Reactor, PWR" yields 46,577 citations; "failure," 29,604; "transient," 1,262; "thermal transient," 1,182; "pressure transient," 1,340; and "rate," 409; however, the combination (Reactor, PWR)*(failure)*(transient + thermal transient + pressure transient + flow * rate) yields only 1,031 citations, a much more manageable number.

Some of the keywords and concepts used in various ways to look for PVTS events and precursors include:

Systems or equipment:

Reactor cooling system, main cooling system, reactor coolant, pressure vessel, reactor coolant pumps, pressurizer, valves, pump seals, feed and bleed, steam generators, charging pumps, residual heat removal system.

Feedwater, steam, pumps, valves, auxiliary feedwater, emergency feedwater, dump valves, regulating valves, feedwater heaters, condenser, flow bypass, turbine.

Safety injection, refueling water storage tank, boron injection tank, high pressure safety injection, low pressure safety injection, accumulators, safety feature actuation system.

Events:

Thermal transient, pressure transient, flow transient, safety system actuation, false signal, spurious signal, loss of power, loss of instrumentation, inverter failure, fire, pressure spike, lightning strike, reactor trip, pump trip, valve failure, operator error, maintenance error, test error, blowdown.

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*RECON is a computer network maintained by the DOE in Oak Ridge, Tennessee. This network allows access through remote terminals to about 37 information data banks, including the NSIC data bank.

Table 3.1. Example of LER searches for PVTS events and precursors in U.S. PWRs

(February 28, 1982)

Description of keywords or sets of keywords	Number of citations
NSIC data bank	~170,000
Reactor, PWR	46,577
Failures	29,604
Failures at PWRs	14,856
Transients (thermal, pressure, reactor) at PWRs	682
Safety injection (ECCS, HPSI, LPCI) at PWRs	819
Main coolant system failures at PWRs (RCS, reactor coolant, leak, pressurizer, blowdown, valves, scram, spray, shutdown, pressure relief, pumps)	6,661
Feedwater-steam system failures at PWRs (steam, feedwater, auxiliary cooling, emergency cooling, valves, pressure relief, pumps, steam generator, turbine, flow, heaters)	5,899
Flooding in containment of PWRs due to pipe, valves	73
Transients in PWRs that resulted in safety injection	40
Transients in RCS of PWRs with excessive flow rate or cooling	447
Transients in RCS of PWRs with excessive flow rate or cooling and accompanied by safety injection	28
Transients in feedwater-steam circuits with excessive flow rate or cooling	529
Transients in feedwater-steam circuits with excessive flow rate or cooling and accompanied by safety injection	37
Transients in RCS that were caused by transients in the feedwater-steam circuits and accompanied by safety injection	50

Nature of events:

Overcooling, overheating, overpressurizing, depressurizing, excessive flow, excessive cooling, excessive heat-up, inadvertent.

Operating conditions:

Power, startup, hot standby, hot shutdown, cold shutdown, cold shutdown with reactor vessel head unbolted.

3.3 Procurement of Original and Other Pertinent Reports

In over 40 searches using keywords and their combinations as described in the preceeding paragraphs, some 4,000 LER titles and most of their abstracts (i.e., about 25% of the total PWR LERs as of the end of 1981) were screened and read. Approximately 170 citations of interest were found. The original licensee reports for these citations were procured for more detailed reading. About one-half of these original licensee reports were no clearer than the NSIC abstracts themselves. About 40% had more descriptions, although these descriptions did not always shed much more light on the events than did the NSIC abstracts. About 10% had quite detailed and good descriptions, and in special cases, such as the Crystal River 3 event of February 26, 1980, the Ft. Calhoun event of April 17, 1974, and the Robinson 2 event of May 1, 1975, the original descriptions and supplementary reports were very informative.

3.4 Generalized Event Trees for PVTS-Event Analysis

Seven generalized event trees (Fig. 3.2) were constructed to provide a framework on which to map the PVTS events and/or precursors found in the LER searches. These event trees correspond to the top-down logic for PVTS sequences described earlier (Fig. 3.1). In Fig. 3.2, at each decision node, the branch down signifies a "no," while the other branch signifies a "yes." No end states of the trees are marked in order to maintain visual simplicity and because additional information outside the context of this report is needed.

Each event occurred under a different circumstance and in a different way from the others. If one were to speculate on the probability of the various routes each event could have taken, then a specific tree must be constructed for that event, taking note of the various equipment functions and operator intervention. However, as mentioned earlier, this task is rather to determine the frequencies of past events. There is merit in categorizing these events under a few initiating sequences with simple, yet sufficient, details to yield useful information.

For each of the simple PVTS event trees, the initiator is represented by a circle in a similar fashion as it would be represented at the bottom of a simple fault tree. There are seven separate initiators: LBLOCA, SBLOCA, overfeeding or cold feedwater (OF/cold FW), high steam flow (HSF), HPSI, core flooding and/or LPSI (CF/LPSI), and external flooding or insulation fall-off (EX) from the external side of the reactor vessel.

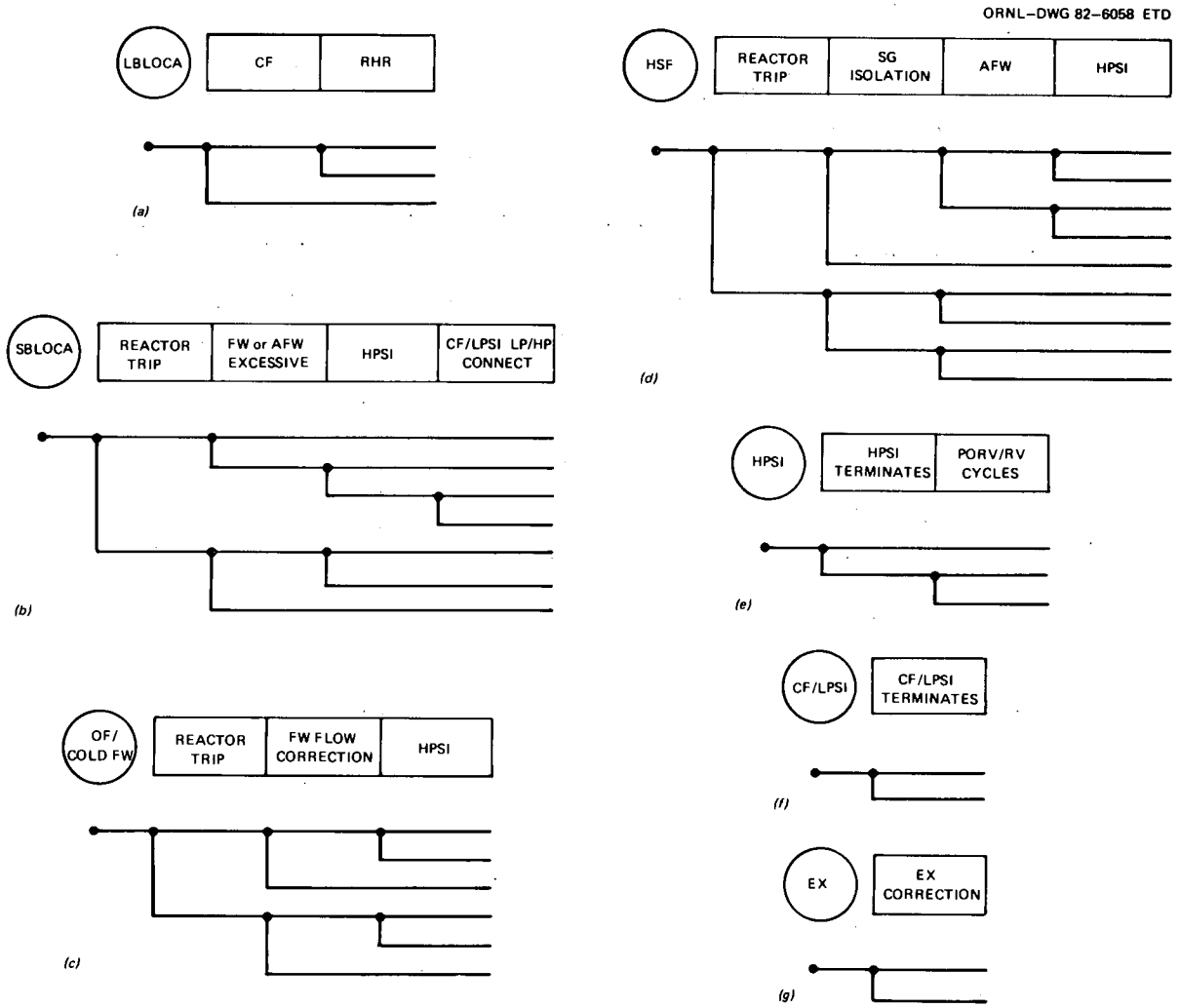

Fig. 3.2. Generalized event trees for PVTS event mapping: (a) large-break loss-of-coolant tree, (b) small-break loss-of-coolant tree, (c) overfeeding/cold FW tree, (d) high steam flow tree, (e) high-pressure safety injection tree, (f) core flooding/low-pressure safety injection tree, and (g) external flooding/external insulation fall-off tree.

Equipment and system functions following the sequence initiation, whether automatic or with operator intervention, are shown in the heading boxes following the initiator. Several functions may have been grouped to simplify the event trees. Further, these functions are displayed sequentially in the order of the time they are normally called into service. Causes for the initiator occurrence are listed at the start of the event trees, and the PVTS-related consequences are summarized at the end of each tree. In Appendix A, where each event has been mapped on one of these seven trees, a step-by-step summary of the event as analyzed from the original licensee report is also provided. A brief description of each tree follows.

The LBLOCA tree as related to PVTS is very simple, involving only two functions following the blowdown: (1) CF by water from the accumulators and (2) water recirculation by the RHR system. The consequences may be extremely drastic because the reactor vessel, still at ~550°F by the end of the blowdown, is suddenly exposed to cooling water at ~100–200°F for many minutes afterwards.

The SBLOCA tree has four system functions that determine the PVTS-related consequences: reactor trip (RT), FW or auxiliary feedwater (AFW) control, HPSI, and CF and/or high-pressure/low-pressure cross connect (HP/LP connect). Causes for SBLOCA include (1) the break of a small line (e.g., instrumentation), (2) the opening of a valve that subsequently fails to close [e.g., pressurizer power-operated relief valve (PORV)], and (3) the failure of a reactor coolant pump seal.

The OF/cold FW tree has three system functions affecting PVTS-related outcomes: RT, FW flow correction or SG isolation, and HPSI. Causes of OF may include high FW pump speed, too large an opening of FW flow regulating valves, failure of the FW valves to close on demand, and high SG level. Causes of cold FW may include sudden failure in the FW heater train.

The HSF tree has four system functions determining PVTS-related consequences: RT, SG isolation, AFW or emergency feedwater (EFW) flow, and HPSI. Causes for HSF may include the steamline break, the failure of a turbine bypass valve, and the unwanted opening of a steam dump valve or relief valve.

The HPSI tree has two system functions affecting PVTS-related consequences: termination of HPSI by the operator and the cycling of the pressurizer PORV and safety valves in case the former function fails. Any of the previous trees (LBLOCA, SBLOCA, OF/cold FW, or HSF) may involve HPSI; but HPSI can also be caused independently by spurious signals, operator errors, lightning, fire, or maintenance and test errors.

The EX tree has only one system function affecting PVTS consequences: whether the fault is corrected in time. Causes for the flooding of the reactor cavity may include a fluid line break or leak at a location such that water can be accumulated in the reactor vessel cavity. Causes for the insulation falling off from outside the reactor vessel may include faulty design and/or vibration.

3.5 PVTS Severity Categorization

So far PVTS events and PVTS precursors have been used in almost the same context. To analyze PVTS-related events, it is important to differentiate the two categories.

In the strictest sense, a PVTS event should give rise to thermal stresses large enough to challenge the integrity of the reactor vessel. However, fracture mechanics indicates that the same PVTS event may be mild to one vessel but very severe to another because of differences between the two vessels in the copper content in the welds, total fast neutron fluences in their service history, sizes of flaws, the manner that pressure and temperature change with respect to time, and last but not least, the total duration of time at low temperature.

Due to this difficulty, PVTS-like events are defined in terms of the degree of severity of the occurrences and/or consequences. Several parameters are available for differentiating the degrees of severity: change in temperature (ΔT) and pressure (Δp) of the RCS, duration over which such changes take place (Δt), the amount of safety injection water (SI_v), and the duration over which safety injection (SI) takes place (SI_t). These parameters are related to one another but certainly not in a simple manner. Further, very seldom are all of them known for a reported event.

In Table 3.2, severities 0–4 have been arbitrarily defined on the basis of these parameters. "Arbitrarily" means an educated guess – not very precise but reasonable based on the understanding of PWR thermal hydraulics and heat transfer from the vessel metal to its adjacent environment. Furthermore, judgment was used to assign an event to a specific severity category when by definition it could belong to two categories at the same time.

In sum, PVTS severity categories are defined as follows:

Severity 4: Those events that have one or several of the following characteristics:

- Change in RCS temperature $\geq$ 100°F.
- Change in RCS pressure $\geq$ 1,000 psi.
- Duration of RCS transients $\geq$ 15 min.
- Rate of RCS temperature change $\geq$ 100°F/h.
- SI $\geq$ 10,000 gal.
- Duration of SI $\geq$ 15 min.

Severity 3: Those events that have one or several of the following characteristics:

- Change in RCS temperature between 40°F (inclusive) and 100°F.
- Change in RCS pressure between 600 psi (inclusive) and 1,000 psi.
- Duration of RCS transient between 5 min (inclusive) and 15 min.
- Volume of SI water between 3,000 gal (inclusive) and 10,000 gal.
- Duration of SI between 5 min (inclusive) and 15 min.

Severity 2: Those events that have one or several of the following characteristics:

- Change in RCS temperature between 20°F (inclusive) and 40°F.
- Change in RCS pressure between 200 psi (inclusive) and 600 psi.
- Duration of RCS transient between 2 min (inclusive) and 5 min.
- Volume of SI water between 300 gal (inclusive) and 3,000 gal.
- Duration of SI between 2 min (inclusive) and 5 min.

Severity 1: Those events that have one or several of the following characteristics:

- Change in RCS temperature <20°F.
- Change in RCS pressure <200 psi.
- Duration of RCS transients <2 min.
- SI volume <300 gal.
- Duration of SI <2 min.

Table 3.2. PVTS severity definition[a]

Parameter	Severity				
	Insignificant			Significant	
	0	1	2	3	4
ΔT, change in RCS temperature, °F	Negligible	<20	$20 \leq \Delta T < 40$	$40 \leq \Delta T < 100$	≥ 100
Δt, duration of transient, min	Short	<2	$2 \leq \Delta t < 5$	$5 \leq \Delta p < 15$	≥ 15
Δp, change in RCS pressure, psi	Small	<200	$200 \leq \Delta p < 600$	$600 \leq \Delta p < 1{,}000$	$\geq 1{,}000$
SI_V, volume of safety injection, gal	Small	<300	$300 \leq SI_V < 3{,}000$	$3{,}000 \leq SI_V < 10{,}000$	$\geq 10{,}000$
SI_t, duration of safety injection, min	Short	<2	$2 \leq SI_t < 5$	$5 \leq SI_t < 15$	≥ 15

[a]1. PVTS severity is defined here to facilitate the analysis of PVTS events.
2. Lower severity can be considered a precursor of higher severity.
3. Relationship between parameters in each category often, but not always, agrees with thermal hydraulics of the system.

Severity 0: Those events that have characteristics even less severe than those of severity 1 but are noteworthy because the causes for initiators and/or the manner in which the events unfold are noteworthy.

It is noted that a drop in pressure of the RCS down to the saturation conditions (~600°F, 1200 psi) does not constitute a PVTS, although an increase in pressure over and above the operating pressure (~2250 psi) does have an adverse effect on the vessel. We used the change in pressure drop as an indicator to categorize PVTS because such drop often results in SI and/or overcooling.

In defining PVTS-like events as in the preceding paragraphs, one may loosely consider severity 0 events as precursors to severity 1 events, 1 to 2, 2 to 3, and 3 to 4. Further, if these events are to be consolidated into fewer groups, then "significant events" can be defined as those events with severities 4 and 3, and "insignificant events" can be defined as those with severities 2, 1, and 0.

Table 3.2 summarizes the severity definitions.

3.6 How Bad Are PVTS Events?

Although our finds are numerous, it must be noted that the majority of the events were very mild and should only be considered as precursors of PVTS events. Only a few events, such as occurred at Rancho Seco (3/20/78), Oconee 3 (11/10/79), Crystal River 3 (2/26/80), and Indian Point 2 (10/17/80), appear to be significant from the viewpoint of reactor vessel integrity. Such integrity would be compromised only when a series of conditions are coincident: severe PVTS event, preexistence of flaws in the vessel wall, brittleness of the vessel material at the flaws (e.g., high copper content and high exposure to fast neutrons).

In order to obtain a feel for the severity of the PVTS events, we used the fracture mechanics code OCA-II (Ref. 31) to study some PVTS cases bounding the most severe of our finds. These cases assume the existence of a longitudinal flaw up to 15% deep in the vessel weld, the existence of 0.35% copper content in the weld material, and the occurrence of a PVTS, during which the RCS temperature drops instantly by as much as 300°F and the RCS pressure could rise as high as 2500 psi. OCA-II enables us to look for the critical fast neutron fluence, which would embrittle the material to such an extent that the flaw would propagate through the vessel wall under those coincident conditions. The knowledge of such critical fast neutron fluence would allow us to determine the time in a reactor operating life when the vessel can no longer tolerate the assumed PVTS event. The results are as follows:

Assumed PVTS instantaneous RCS temperature drop (°F)	Assumed RCS pressure (psi)	Critical fast neutron fluence (neutrons/cm²)
200	1500	2.77×10^{19}
	2500	1.43×10^{19}
300	1500	0.62×10^{19}
	2500	0.43×10^{19}

The material copper content and operational fast neutron fluxes for the 47 U.S. PWRs are given in Table 2.1. As of December 1981, approximately six vessels had accumulated a fast neutron fluence higher than 1×10^{19} neutrons/cm^2. While the copper contents in the Rancho Seco and Crystal River vessel welds are as high as 0.31%, the accumulated fast fluences at the time of their respective PVTS occurrences were much smaller than 0.4×10^{19} neutrons/cm^2; therefore, the vessels would not be compromised by these occurrences even if there were preexistent flaws. PVTS events of the same magnitude, however, would need to be prevented as the reactors and vessels accumulate more service duty. Reference 32 proposes a screening criterion whereby reactors must go through a thorough evaluation of the vessel integrity 3 years before the critical fast neutron fluence is attained.

3.7 Exhaustiveness of Searches

An ad hoc approach was used to determine the thoroughness of the LER search for PVTS events and precursors. First, two samples of PWR LERs, each covering 5% of all PWR LERs, were obtained to screen for PVTS-like events and compare these against the finds. Second, the finds were compared against those made known by industry and other published documents.

4. RESULTS

4.1 Summary of Finds, Coverage Period, and RYs

In over 40 computer searches using the methodologies described in Chap. 3, some 4000 LER abstracts (i.e., ~25% of the total PWR LERs) were screened and read, and ~170 events of interest were found. After the original licensee reports of these events were studied, 99 were selected as having PVTS characteristics. They can be summarized as follows:

Severity	Events		
0	20		
1	28	Insignificant:	65 events
2	17		
3	12	Significant:	34 events
4	22	Total:	99

Tables 4.1–4.5 list these events according to their severity category. Pertinent information necessary for their location includes the NSIC accession number, the reactor name, the event date, and the letter date when the event report was sent. (The letter date is in most cases also different from the report date, the latter being the date the report was compiled by the responsible person at the utility.) Information on these events obtained from the author's reading is summarized under "Apparent Causes" and "PVTS-Related Consequences."

Appendix A provides a summary of the original licensee (and other) reports, as well as the mapping of each occurrence on one of seven generalized event trees. Appendix B provides a complete printout of the abstracts for these events, organized in the order of NSIC accession number.

As mentioned previously, these events are based on 329 RYs of PWR operation.

4.2 Insignificant Events

About two-thirds (65 out of 99 events) are of severities 0, 1, and 2. These events are considered insignificant because they caused only minor PVTS-related consequences: changes in RCS temperatures of <40°F, changes in RCS pressures of <600 psi, SI of <3000 gal, or transient duration <5 min. They were defined and sought to determine causes of their initiation and the manner of their termination. Furthermore, we believe that less severe events should be reviewed for, or could be considered as, precursors to more severe ones.

The searches for and analyses of these insignificant events consumed the majority of this project's efforts. However, as is pointed out in the exhaustiveness analysis, the 65 finds hardly constitute 36% of similar events in the LER data base and perhaps 25% of all such events that actually occurred. The information yielded by these insignificant events must therefore be considered to have large bands of uncertainties.

Table 4.1. Severity 0 PVTS events
(1963–1981)

Severity 0 events: Those events that potentially could lead to changes in temperatures and pressure in the RCS or in the environment outside the reactor pressure vessel, but due to whatever reason, did not result in consequences as severe as those defined in Severity 1.

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#170158 and #168503 07/23/81 11/06/81	Cook 1 (Mode 5)	Cold water surging into pressurizer	Heat up of pressurizer exceeding 100°F/h
#168079 06/27/81 07/02/81	Salem 2 (Mode 1/30%)	FW pump trip	SI; BIT decreased to below 20,000 ppm
#167692 05/07/81 06/04/81	McGuire 1 (Mode 6)	Operator pushed wrong button	Both SI and charging pumps were tagged out; no consequences
#167784 04/29/81 05/28/81	McGuire 1 (Mode 5)	Maintenance error	False signal to SI; SI was stopped
#166342 04/23/81 05/22/81	Salem 2 (Mode 2/0%)	Valve inadvertently opened	SI
#165876 03/29/81 04/20/81	North Anna 1 (Mode 5)	Operator error during maintenance	SI
#164791 11/17/80 03/02/81	Beaver Valley 1 (Mode 2)	Operator error	SI
#164421 10/24/79 01/29/80	Farley 1 (Mode 4)	Maintenance error (replacement of a card)	None
#159786 05/26/80 06/23/80	Sequoyah 1 (Mode 4)	System/operator error	Pressurizer heat-up rate at 151°F/30 min
#156951 04/26/80 04/30/80	Surry 1 (Mode 5)	Maintenance error	1. Gravity flow from RWST 2. Spurious SI signal 3. Accumulator water release RCS p increased 15 to 80 psig 4. Wrong jumper causing SI
#154451 02/04/80 02/12/80	Connecticut Yankee (Mode 1/100%)	Spurious signal caused PORV and MOV valves to open	RCS pressure decreased from 2016 to 1992 psig; arrest in 2 min, no SI
#153683 11/09/79 12/07/79	Indian Point 2 (Mode 1)	One SG blowdown isolation valve failed to close	No consequence because redundant valve worked
#150731 07/06/79 07/19/79	Calvert Cliffs 1 (Mode 5)	PORV failed to close during test after modification	RCS pressure dropped from 260 to 200 psig

Table 4.1 (continued)

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#150394 06/03/79 06/28/79	Arkansas 2 (Mode 3)	AFW valve would not close	No consequence because redundant valve worked
#148766 03/14/79 04/12/79	Millstone 2 (Mode 5)	LPSI suction air binding	SDC lost; flow actuated by operator from RWST; heat up to 200°F
#142387 01/30/78 02/24/78	Salem 1 (Mode 1/22%)	Operator error in matching pump speed	RCS pressure dropped to 2165 from 2220 psia
#142627 12/06/78 12/20/78	Cook 2 (Mode 1/100%)	Faulty drain valve on 5-A heater, creating unstable FW control	RCS T_{avg} increased
#117079 06/20/76 08/13/76	Zion 2 (Mode 3)	Water hammer	SI for 15 s
#116240 07/13/76 07/27/76	Oconee 3 (Mode 1/81%)	ICS design; spurious decrease in FW in 2 s	Power increased 3% during a spurious decrease in FW demand; at <0.1 min T_{avg} dropped
#091182 01/22/74 04/30/74	Oconee 2 (Mode 2)	RCS pump seal failure	No information

Table 4.2. Severity 1 PVTS events
(1963–1981)

Severity 1 events: Those events that resulted in one or several of the following:

1. change in inlet temperature of <20°F,
2. duration of the transient <2 min,
3. change in primary coolant pressure of <200 psi,
4. duration of SI <2 min, and
5. quantity of injected water <300 gal.

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#168144 07/11/81 08/10/81	Cook 2 (Mode 2)	Operator error in using excessive FW flow to raise SG level	RCS temperature dropped below 541°F
#168077 #167124 05/07/81 06/09 & 07/31/81	Zion 2 (Mode 1/92%)	False SI signals due to failed inverter	No information
#167772 #166746 05/28/81 08/06 & 07/21/81	Zion 1 (Mode 1/100%)	False SI signals due to failed inverter	No information
#166854 06/03/81 06/30/81	Salem 2 (Mode 1/20%)	Steam dump following RT due to high SG level	SI that caused BIT concentration to violate tech spec
#166686 03/30/81 04/28/81	McGuire 1 (Mode 5)	SI during fuel loading due to maintenance error	SI by RHR pump only; no significant amount of water
#166648 10/16/80 06/11/81 #162394 10/16/80 01/12/81	Crystal River 3 (Mode 1/97%)	A fuse failed while Channel 1 in "test"	SI: 250 gal; Δp: 40 psi (from 2145 to 2185 psig)
#166453 04/23 & 05/01/81 06/05/81	Sequoyah 1 (Mode 3)	First SI in Mode 3, caused by high steam flow plus low T_{avg}; second SI in Mode 2 caused by Δp between SG	First SI: 2–3 min Second SI: no information
#163551 01/01/81 01/15/81	North Anna 2 (Mode 2)	Overboration and bleed; steam dump valve failed to close	ΔT < 20°F
#162394 10/16/80 01/12/81	Crystal River 3 (Mode 1/97%)	Relay failure during channel test (fuse blown)	SI: 250 gal
#161826 09/03/80 12/03/80	Zion 2 (Mode 4)	Water hammer following RT and AFW start	SI for 1 min
#158860 04/19/80 04/30/80	Davis-Besse (Mode 6)	Loss of two buses during cold shutdown (Mode 6) for maintenance	ΔT = 30°F from 140°F SI = 3500 gal
#156952 04/26/80 05/22/80	Surry 1 (Mode 6)	1st SI: valve cycled, letting in water from RWST. 2nd SI: operator error	Accumulators pressure dropped to 120 and 300 psi, respectively

Table 4.2 (continued)

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#155451 02/15/80 03/17/80	Turkey Point 4 (Mode 3)	FW control valve stuck; MSRV lifted during hot shutdown	No drastic change of temperature in RCS
#154890 01/29/80 02/12/80	Calvert Cliffs 1 (Mode 5)	Spurious high-pressure signal on maintenance; PORV opened	Pressure dropped from 258 to 169 psia
#154747 09/06/79 12/03/79	Trojan (Mode 2)	Main steam valve open due to operator pulling wrong card from control (at 1.5% power)	SI (no information on duration)
#152983 10/19/79 11/16/79	Oconee 3 (Mode 6)	Operator opened cross-connect valve (in Mode 6)	Excessive make-up flow in cold shutdown
#149716 03/20/79 06/01/79	Zion 1 (Mode 3)	False SI signal following RT	SI for 2 min
#142480 09/14/78 12/11/78	Zion 1 (Mode 3)	False SI signals due to water hammer at hot shutdown	SI
#137296 01/24/78 03/28/78	Beaver Valley 1 (Mode 3)	Leaky pressurizer spray; cold AFW caused SI	SI for about 2–3 min; no other information
#134324 01/02/78 02/03/78	Beaver Valley 1 (Mode 1)	False SI signals due to No. 4 vital bus failure and operator error	No information
#123778 12/21/76 03/11/77	Zion 2 (Mode 1/75%)	Real T_{avg} low due to boron inleakage plus false HSF caused SI	SI for 1 min, 30 s
#122385 10/28/76 02/02/77	Trojan (Mode 3)	Operator error in adding chemicals too fast	SI, 100 gal
#120818 09/24/76 12/13/76	Zion 1 (Mode 1)	False SI signals caused by water hammer	SI for 2 min
#117634 06/20/76 08/27/76	Zion 1 (Mode 2)	False SI signals due to water hammer	SI for about 20 s
#117080 05/25/76 08/12/76	Zion 2 (Mode 4)	Water hammer	SI for 90 s
#109619 10/03/75 12/31/75	Cook 1 (Mode 1/81%)	SI following reactor and turbine trip due to spurious signal	900 gal, 3 min, $\Delta T = 16°F$
#100433 12/30/74 03/07/75	Zion 2 (Mode 1/82%)	FW control valve closed due to break of crimp connector	SI for <30 s
#089159 02/22/74 02/28/74	Indian Point 2 (Mode 6)	Mode 6; accumulators stop valves opened during testings	RCS pressure increased from 110°F, 150 psig to 100°F, 560 psig

Table 4.3. Severity 2 PVTS events
(1963–1981)

Severity 2 events: Those events that resulted in one or several of the following:

1. change in inlet temperature of 20 to 40°F,
2. duration of the transient 2 to 5 min,
3. change in primary coolant pressure between 200 and 600 psi,
4. duration of SI between 2 and 5 min, and
5. quantity of injected water between 300 and 3000 gal.

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#167550 07/03/81 07/15/81	North Anna 2 (Mode 1–Mode 3)	Fire affecting feeder cables	$\Delta T \approx 40°F$
#164865 11/27/78 03/25/81	Salem 1 (Mode 1/100%)	Recovery from RT caused by vital bus failure	SI for 3 min
#160915 10/02/80 10/31/80	Sequoyah 1 (Mode 2)	FW flow overfeed due to inadequate scaling of flow control during manual operations	$\Delta T < 40°F$ for a few minutes
#160915 10/02/80 10/31/80	Sequoyah 1 (Mode 1/10%)	Same as above	Same as above
#153713 12/02/79 01/02/80	Farley 1 (Mode 1/1%)	Atmospheric dump failed to close	RCS temperature dropped to 531°F for a few minutes; operator blocked valve
#159553 06/08/80 07/08/80	Salem 1 (Mode 1/100%)	False SI signals due to lightning	Two main steam pressure transmitters failed
#159240 06/03/80 07/02/80	Surry 1 (Mode 1/100%)	False SI signals due to loss of vital bus due to fire	Two SIs: first due to HSF and low T_{avg}; second due to faulty signal
#158645 05/23/80 06/04/80	North Anna 1 (Mode 3)	False SI signals due to loss of vital bus	SI for 5 min
#154690 01/19/80 02/06/80	Cook 2 (Mode 2)	OF due to manual change from AFW to FW	$\Delta T \simeq 20–30°F$; $\Delta p \simeq 30$ psi
#154291 09/20/79 12/19/79	Beaver Valley 1 (Mode 3)	Bus failure caused first SI; operator error caused second SI	No information
#148080 12/02/78 02/28/79	TMI-2 (Mode 1/22%)	Operator error pinned open FW valve	SI for 2 min
#145364 04/29/78 07/28/78	Davis-Besse 1 (Mode 2/20%)	FW valves stuck open due to manual to automatic juggling	RCS pressure 1620 psi
#143496 08/04/77 08/29/77	Salem 1 (Mode 1/14%)	Pump speed increasing when put from manual to automatic	$\Delta T = 36°F$; $\Delta p = 66$ psi; and $\Delta t = 2$ min

Table 4.3 (continued)

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#123778 12/21/77 03/11/77	Zion 2 (Mode 1/75%)	Power decreased due to boron inleakage	RCS temperature decreased to 540°F for 90 s
#119312 07/05/76 09/01/76	Cook 1 (Mode 3)	False signals to SI possibly due to master relay driver card	SI ≃ 600 gal, 4 min
#118247 06/03/76 09/01/76	Cook 1 (Mode 1/100%)	Loss of vital bus caused false SI signals	SI for 4 min, ΔT ≃ 20–30°F
#112149 12/13/75 03/12/76	Cook 1 (Mode 1/81%)	False signals due to maintenance	SI for 7 min, 2100 gal; ΔT ≃ 22°F

Table 4.4. Severity 3 PVTS events
(1963–1981)

Severity 3 events: Those events that resulted in one or several of the following:

1. change in inlet temperature between 40 and 100°F,
2. duration of the transient between 5 and 15 min,
3. change in primary coolant pressure between 600 and 1,000 psi,
4. duration of SI between 5 and 15 min, and
5. quantity of injected water between 3,000 and 10,000 gal.

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#167625 04/23/81 07/21/81	Salem 2 (Mode 3)	SG atmospheric relief valve was used for controlling cooldown	SI for 5 min
#164149 01/29/81 02/12/81	Robinson 2 (Mode 2)	Pressurizer spray leak; letdown line break	RCS pressure dropped below 1700 psi. 1st SI: 2–4 min. 2nd SI: duration not known; 4500–6000 gal SI, estimated
#157130 01/29/80 05/13/80	Arkansas 2 (Mode 3)	Steam dump and bypass valves; SI and pressurizer spray did not fully close	RCS pressure dropped to 1350 psia
#157093 04/03/80 04/16/80	North Anna 1 (Mode 1/100%)	Main steam stop valve closed during test causing SI	SI for 4 min, $\Delta T/\Delta t$ = 72°F/15 min
#155359 02/23/80 03/07/80	North Anna 1 (Mode 1)	False signal to SI due to noise trouble-shooting	$\Delta T/\Delta t$ = 41.2°F/15 min; SI of 6600 gal maximum
#146533 12/14/78 05/15/79	Oconee 1 (Mode 1/98%)	Short circuit related to ICS; FW/AFW control; FW valves	RCS dropped to below 1500 psi
#139563 05/28/78 06/28/78	St. Lucie 1 (Mode 1/3%)	Bypass control valve failed to close during test	$\Delta T \approx 60°F$
#120766 12/08/76 12/27/76	Calvert Cliffs 2 (Mode 2)	FW control and lack of operator experience caused overfeeding	RCS temperature dropped to 503°F and stayed low for about 10 min
#120769 12/08/76 12/27/76	Calvert Cliffs 2 (Mode 1/19%)	Xenon buildup and inability to dilute boron caused power to decrease. FW control difficulties caused RCS temperature to drop	RCS temperature dropped twice, first to 513°F, then to below 516°F; $\Delta t > 10$ min
#087016 10/20/73 10/25/73	San Onofre (Mode 1/99%)	FW regulating valves caused overfeeding	$\Delta T/\Delta t$ = 78°F/10 min; Δp = 685 psi; pressurizer empty
#085369 #085370 #087031 10/31/73	Ginna (Mode 1/100%)	Transmission disturbances due to loss of off-site power	$\Delta T/\Delta t$ = 85°F/10 min; SI
#068933 #057478 10/12/70 11/04/70 12/29/71	Connecticut Yankee (Mode 1)	Steam dump valve (1 of 10) failed to close following call to action	RCS temperature dropped 43°F in 9 min

Table 4.5. Severity 4 PVTS events
(1963–1981)

Severity 4 events: Those events that resulted in one or several of the following:

1. change in inlet temperature of >100°F,
2. duration of the transient >15 min and rate of transients >100°F/h,
3. change in primary coolant pressure of more than 1,000 psi,
4. duration of SI >15 min, and
5. quantity of injected water >10,000 gal.

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#169297 09/14/81 10/09/81	McGuire 1 (Mode 1/20%)	Overcooling during loss of control room test due to design and procedure inadequacies	ΔT ≈ 117°F over about 20 min. Repressurization by charging pumps
#162919 10/17/80 10/20/80	Indian Point 2 (Mode 1)	A leak in fan cooler unit flooded reactor vessel cavity	About 125,000 gal of river water in reactor cavity, wetting pressure vessel 9 ft
#160846 02/26/80 03/25/80	Crystal River 3 (Mode 1/99%)	Loss of NNI caused PORV and RV to open and stick open	ΔT/Δt = 127°F/13 min; RCS repressurized by SI to 2,400 psi solid; 42,800 gal coolant lost
#153772 11/10/79 11/23 & 11/30/79	Oconee 3 (Mode 1/99%)	OF due to turbine bypass valve and AFW	ΔT/Δt = 150°F/15 min; all 3 SI loops on before OF
#153164 03/28/79 10/09/79	TMI-2 (Mode 1/99%)	Loss of FW caused PORV to open and stick open; operator shut off SI	Operator stopped SI after 10–12 min; core damaged
#152751 09/25/79 10/09/79	North Anna 1 (Mode 1/78%)	Steam dump valve	ΔT/Δt = 110°F/16 min; 1st SI: 5 min; 2nd SI: 13 min
#149746 #148074 12/27/78 01/25/79	Arkansas 2 (Mode 1)	Dump valve and relief valve failed to reseat	ΔT/Δt = 107°F/52 min
#148767 03/14/79 04/13/79	Millstone 2 (Mode 5)	Cold RWST water flooded RCS when LPSI pump stopped	ΔT ≃ 146°F at very low pressure
#148112 04/17/74 04/26/74	Ft. Calhoun (Mode 1/30%)	FW regulating valve stuck open	Overfeeding 12–13 min, SI for 15 min, 4500 gal; ΔT/Δt ≃ 120°F/19 min. RCS pressure 1620 psi
#147267 01/05/79 01/26/79	Rancho Seco (Mode 1/100%)	Short in ICS due to maintenance error	ΔT/Δt ≈ 152°F/7 min
#138830 03/20/78 03/31/78	Rancho Seco (Mode 1/70%)	Loss of NNI; OF due to faulty signal and operator error	ΔT/Δt ≈ 310°F/1 h

Table 4.5 (continued)

NSIC accession # Event date Letter date	Reactor (mode/power)	Apparent cause	PVTS-related consequences
#137918 #137919 #139931 04/23/78 05/08/78	Three Mile Island 2 (Mode 1/30%)	Five relief valves failed to close for 4 min	$\Delta T/\Delta t$ = 139°F/3 min, 149°F/4–6 min; HPSI was enough to restore pressurizer level
#126585 07/02/77 07/15/77	Indian Point 2 (Mode 2/2%)	RCS pump seal failure; charging pumps used to compensate for loss of RCS coolant	Excessive cooldown (140°F) for 1.3 h; 90,000 gal in containment
#124920 04/16/77 04/21/77	Crystal River 3 (Mode 3)	Startup FW valves not fully closed due to control; SGs overfilled	RCS cooldown 101°F/20 min
#123150 03/02/77 03/10/77	Crystal River 3 (Mode 1/40%)	Loss of inverter power; ICS caused transient; atm. steam dump opened 50%	RCS cooldown 164°F in 15 min
#120819 09/19/76 12/08/76	Zion 2 (Mode 1/30%)	Loss of dc brought RT but not turbine-driven FW pump	$\Delta T/\Delta t$ = 70°F/30 min; SI ≃ 30 min, 10,000 gal maximum
#103892 #104941 #104981 06/13/75 06/27/75	Oconee 3 (Mode 1/15%)	PORVs failed to close due to boric acid crystal	Final temperature of RCS = 480°F; pressure of RCS = 720 psi
#103077 05/01/75 05/12/75	Robinson 2 (Mode 1/100%)	RCS pump seal failure	132,500 gal leaked; final condition <200°F
#097128 10/07/74 10/08,22/74	Rancho Seco (Mode 1/15%)	Excessive use of steam by plant equipment following reactor trip after loss of condenser vacuum	RCS cooldown "excessive" 100°F in 1 hr
#088077 01/04/74 02/17/74	Oconee 2 (Mode 1/75%)	Excessive steam use by plant equipment when SG at 95% filled and reactor shutdown	RCS cooldown 136 to 140°F over a period of 1 h
#076461 11/05/72	Robinson 2 (Mode 2/0%)	PORV opened and did not close	SI for 87 min, 17,400 gal (estimated)
#075613 10/01/72	Surry 1 (Mode 4)	SG failed to containment due to check valve bonnet gasket failure	RCS cooldown in excess of 100°F/h

The majority of insignificant events found and listed in Tables 4.1–4.5 involved SI (by false signals or by operator, maintenance, or test errors) and were quickly recognized by the operator, correctly analyzed, and reset. About 20% involved OF and HSF, caused mostly by a slight mismatch of FW demand and pump speed or by the opening of a steam dump or relief valve a little longer than required. Again, these problems were quickly corrected automatically by the system or by the operator. Six events took place during the preoperational testing of the reactors.

4.3 Significant Events

Thirty-four events (approximately one-third of the finds) are of severity 3 and 4, or significant events. They are called significant because either one of the following consequences resulted: changes in RCS temperature >40°F, changes in RCS pressure >600 psi, SI of >3000 gal, or transient duration of >5 min. From PVTS viewpoint, two other parameters are also significant: (1) the time over which the vessel wall is exposed to temperature change, since the vessel would not feel it if this duration is short; and (2) the rate of temperature change (e.g., 100°F/h).

Note that all 34 significant events occurred when the reactors were initially in hot conditions: 25 at power, 4 at startup, and 4 at hot standby (Modes 1–3). Furthermore, the proportion of LOCA-caused events to FW or steam-caused events is ~1:4. Safety injection as an initiator to these significant PVTS events was as common as LOCA initiators. The events were most severe when one sequence (LOCA, OF, HSF, or HPSI) led to the others because of weaknesses in system design and/or operator errors [e.g., Rancho Seco (March 20, 1978) and Crystal River 3 (February 26, 1980), to be described in detail in the following section].

The majority of causes for significant events is valve failure (14 out of 34). These include PORVs and relief valves at the pressurizer, FW regulating valves at the steam generators, and steam dump and steam relief valves in the steam lines. Also, important causes are power failure, system failure, and other inadequacies in instrumentation and control (I&C) design (a total of 18 out of 34 events). Only one event was caused primarily by operator error, and one was caused by maintenance error.

4.4 Which Was More Severe: Rancho Seco (March 20, 1978) or Crystal River 3 (February 26, 1980) Events?

Most analytical work to date on PVTS has focused on the March 20, 1978, event at Rancho Seco as the most severe of U.S. experience.[18–20] Our LER analyses indicated that the February 26, 1980, event at Crystal River 3 may be more severe than the Rancho Seco event. The recent SG tube rupture incident at Ginna (January 25, 1982) may also be quite severe from the PVTS viewpoint but does not fall within the time period of this report. Furthermore, the PVTS problem for Three Mile Island 2 is considered moot.

The cause for both Rancho Seco and Crystal River 3 events was the loss of the nonnuclear instrumentation (NNI) because of shorts and the

feeding of wrong signals to the integrated control system (ICS). Both events got the main FW shut off, control rods pulled to 103%, SGs dried out, AFW actuated, HPSI started, reactor coolant pressure increased, and pressurizer valves opened.

The difference is as follows: at Rancho Seco, the AFW was effective at 9 min (after initiation of incident); the operator maintained some FW pumps, the reactor coolant pumps, and high-pressure injection (HPI) until minute 70. The RCS temperature was as low as 285°F. At Crystal River, PORV flow did not get closed (by block valve) until about 5 to 7 min, HPI went full speed for 26 min (at ~1,100 gpm), and at ~250 gpm for the following 61 min (total injection of ~57,000 gal and about 43,000 gal of RCS water was later found in containment sump), all four RCS pumps were stopped at minute 4, and RCS pressure was maintained at relief valve level of 2,300–2,400 psi (Table 4.6).

Because the T_c sensors and the HPI nozzles were located at the opposite ends of the inoperative RCS pumps, the T_c readings were not indicative of the temperature seen by the vessel at Crystal River (Fig. 4.1, 4.2, and 4.3). Furthermore, because the system was solid for ~77 min, and

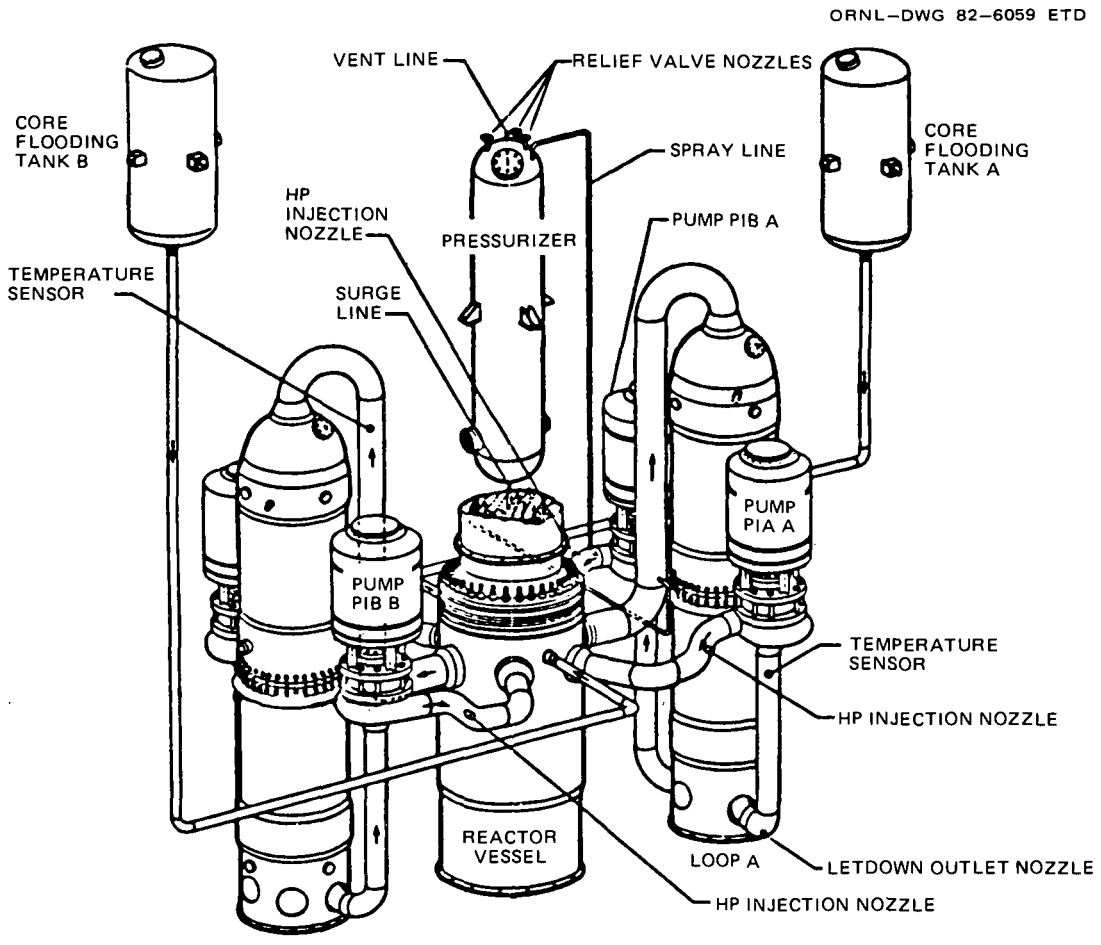

Fig. 4.1. Babcock & Wilcox typical reactor system layout. Source: Crystal River FSAR, p. 3-3.

Table 4.6. Synopsis of event sequences at Rancho Seco and Crystal River 3

Characteristic	Rancho Seco, Mar. 20, 1978	Crystal River 3, Feb. 26, 1980
Time zero	0425 h	1423:21 h
Reactor condition	70% power, T_{avg} = 582°F	98.6% power, T_{avg} = 579°F, T_c = 556.5°F
Initiating event	Loss of NNI-Y because of short (workman changing a bulb); two-thirds of NNI were lost; erroneous signals to ICS	Loss of NNI-X (may be caused by minor repair being performed in NNI cabinet Y); many indications were lost; many alarms; erroneous signals to ICS
Action of ICS	Reduced MFW to zero, withdrew rod to 103%, opened pressurizer PORV and spray, opened turbine control valve, started AFW pumps but did not open AFW valves until 9 min	Same (with AFW actuated by operator at 9 min)
RT	At 0425 h, because of high RCS pressure	At 1423:45 h, because of RCS pressure
LOCA and HPI	Time? RCS pressure 2000 psi (makeup flow less loss through relief valves)	PORV fully open, stayed open for about 5 to 7 min until operator closed block valve
	Time? RCS pump on? (At 70 min, 3 RCS pumps were kept on to "heat" the 285°F RCS water)	3 min: HPI pump started when RCS pressure <1600 psi
	Time? Both SG went dry (even though one indicated full)	4 min: RCS pumps stopped (all four) 7 min: SG A went dry, SG B almost dry 8 min: SG A rupture matrix activated
AFW	9 min: AFW valve opened; operator may also have increased FW pumps; SG filled with water	9 min: one emergency feedwater pump started 9 min: one MFW pump tripped, leaving one operating
	9 min: RCS cooled down rapidly; HPI went on (p < 1600 psi); operator used HPI to maintain RCS pressure	10 min: code safety valve opened (2400 psi) 13 min: computer lost event information until 1511 h; lost it again from 1528 to 1626 h 15 min: FW to SG B was stopped 21 min: NNI-X power returned 26 min: HPI throttled to 250 gpm 28 min: SG B rupture matrix activated 29 min: pressure 2300 psig 30 min: letdown reestablished
	70 min: NNI-Y power returned	87 min: terminated HPI; makeup and letdown control of RCS; completely solid at this time 2107 h, started RC pumps 1B and 1D; RCS pressure 2000 psig; T_{avg} = 420°F (?) pressurizer level 235 in.
Temperature in downcomer	285°F lowest	Could be very low at points in downcomer from 10 min to the end of HPI (Ref. 23 shows 250°F)

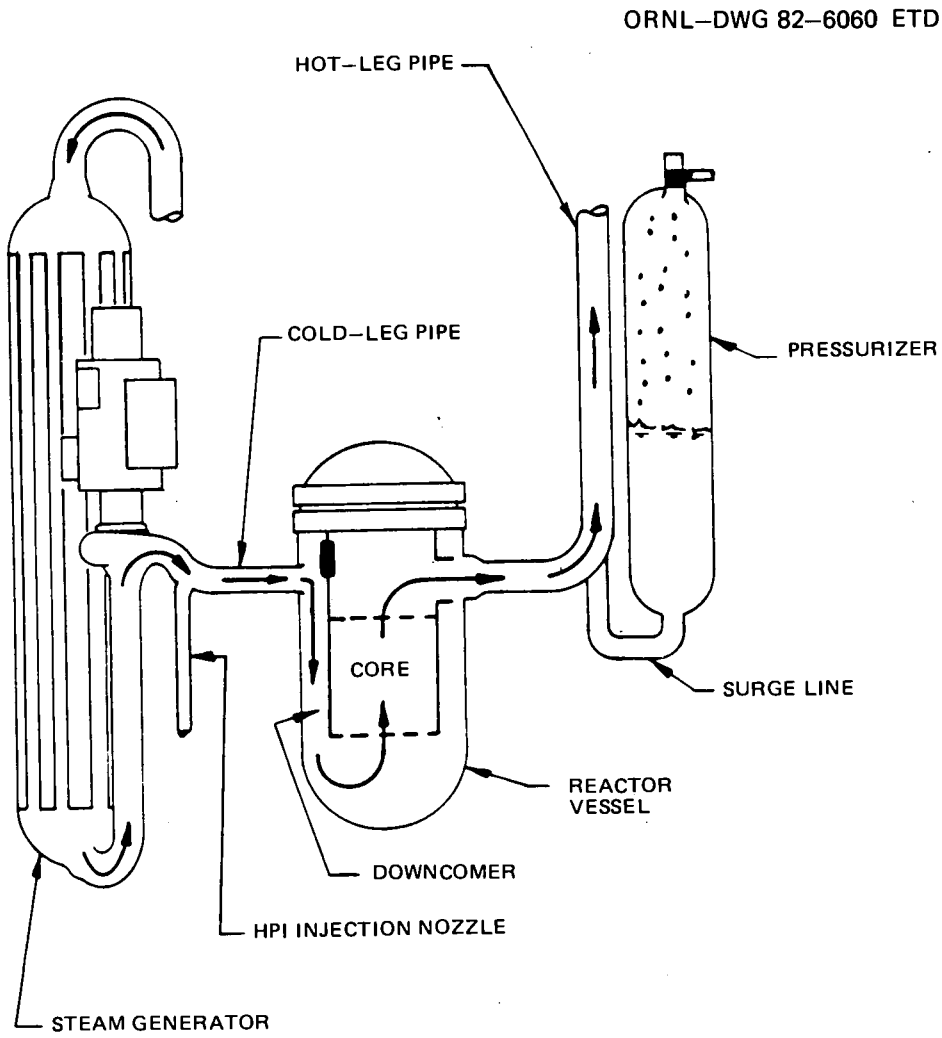

Fig. 4.2. Babcock & Wilcox design of the vent valves for hot water/cold water mixing in the recirculation mode. Source: Duke Power Company, *Oconee Nuclear Station Reactor Vessel Pressurized Thermal Shock Evaluation*, DPC-RS-1001 (January 1982).

the flow was from HPI nozzles to the downcomer, through the core, and out to the pressurizer safety valves, then the vents between the upper core shield and the downcomer may not have been open throughout the duration of the incident. As a result, the temperature seen by parts of the pressure vessel at Crystal River could have been quite low. The vessel was repressurized to 2300 psig.

The three reports on or related to the Crystal River events[21–23] did not analyze this possibility. The exception is the Babcock & Wilcox (B&W) report,[23] which contains one sentence stating that the water in the downcomer may have been as cold as 250°F (Ref. 23, p. 82).

Subsequent to this observation, which we passed on to the NRC task force on PVTS in February 1982, efforts have been made at Los Alamos National Laboratory (LANL) to reanalyze the Crystal River 3 incident using

ORNL–DWG 83–4208 ETD

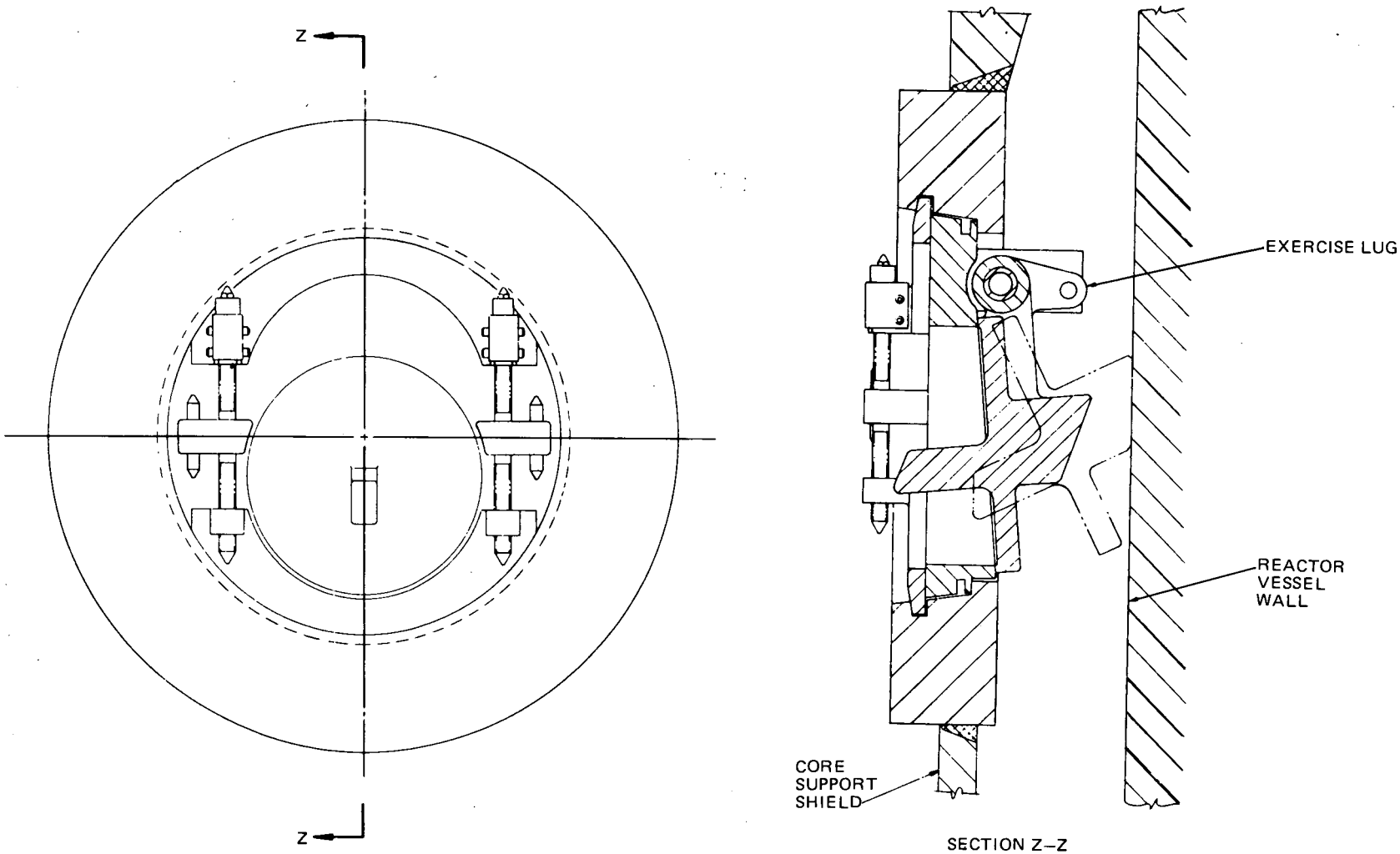

Fig. 4.3. Babcock & Wilcox design of the vent valves at Crystal River-3. Source: Crystal River-3 FSAR, Fig. 3-58.

the TRAC computer code. Some reverse flow was observed when SG A went dry. Flow through the vent valves was also observed. The analysts, however, pointed out the great uncertainty concerning several input parameters to the code.*

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*It must be noted for the records that we made extensive inquiries with people at NRC, LANL, Oak Ridge National Laboratory (ORNL), Science Application, Inc., B&W, Sacramento Municipal Utility District (SMUD), and Florida Power Corporation (FPC). Except for SMUD, which ignored our telephone and letter inquiries, and FPC, which disagreed with some details of the events, all other analysts we contacted expressed the opinion that the Crystal River 3 event might have been more severe.

5. EXHAUSTIVENESS ANALYSIS

5.1 Check of Finds Against the NSIC Data Bank

Table 5.1 shows the checks of finds against those PVTS-like events obtainable through the reading of 757 LER abstracts. These abstracts constitute 5% of all PWR LERs, obtained by picking the 20th abstract starting from a random abstract numbered 1–10. This reading yielded 4 out of 757 events with PVTS characteristics. Two of these events were

Table 5.1. PVTS events in random LER reading – check on exhaustiveness of finds

NSIC accession # of LER related to PVTS	Description	Tentative severity assignment	Was it found?
	First check: 757 LERs (5%)		
168722	SG FW valve fails to close at Oconee 3	0	
159553	SI actuations at Salem 1	2	Yes
147267	Reactor cooldown rate excessive at Rancho Seco	4	Yes
130050	RCS T_{avg} <525°F at Davis-Besse 1	2	
4 of 757			2 of 4
	Second check: 756 LERs (5%)		
166228	Vital bus loses power at Crystal River 3	1	
160915	RCS temperature below limit at Sequoyah 1	2	Yes
160449	Pressurizer relief valve stuck open at Palisades	0	
153713	Atm relief valve stuck open at Farley 1	2	Yes
152983	RCS pressure exceeded limit at Oconee 3	1	Yes
143418	RCS pump seal fails at Connecticut Yankee	2	
143274	RCS pump seal fails at Indian Point 2	3	
141115	RCS temperature below limit at Farley 1	1	
140871	Lightning causes SI at Farley 1	2	
137918	RCS cooldown excessive at TMI-2	4	Yes
103892	RCS cooldown rate excessive at Oconee 3	4	Yes
11 of 756			5 of 11

Combining the two checks:
- 15 PVTS-like events found in 1513 random PWR LERs, 1%
- 7 PVTS-like events out of 15 events found in the logical searches, 47%
- Insignificant events, 4 out of 11, 36%
- Significant events, 3 out of 4, 75%

found by systematic searches, the results of which are listed in Tables 4.1–4.5.

Because the agreement of findings against the "5% check" is so low, another 5% of the LER abstracts (756 abstracts) were ordered for a second check. This second reading yielded 11 PVTS-like events. Five of these events were found in Tables 4.1–4.5.

Combining the two checks yields 15 PVTS-like events in 1,513 random PWR LERs, or 1%. The total 99 finds among ~16,000 PWR LERs amount to only about 0.6%. Three out of the 4 significant events (75%) but only 4 out of the 11 insignificant events were found by systematic searches. Thus, the conclusion is that the 65 insignificant events (severity 0, 1, and 2) barely are representative, and any frequencies obtained for them must be used with utmost care. On the other hand, the 75% exhaustiveness of the significant findings (34 events) could be considered adequate. Because these events are of higher interest to current PVTS investigation, frequencies obtained from this data base may be relevant after correcting for this 75% factor.

5.2 Check of Finds Against Industry's Records

Duke Power Company published in 1982 a report evaluating the PVTS situation for the Oconee Nuclear Power Station (DPC-RS-1001, January 1982).[24] This report lists 16 operational PVTS-like events in the operating history of Oconee 1, 2, and 3 up to May 1981 (Table 5.2). Three out of seven significant events were found in the NSIC data base.

The Westinghouse (W) Owners Group made a presentation to the NRC on June 22, 1982, during which a list of eight W reactor PVTS events was made public. Table 5.3 shows these events. As the Ginna event of January 25, 1982, is outside the scope of this study (only up to December 1981), the real number of events to be matched against our finds is 7. Three of these were found, and two were significant (out of five).

Combining both Duke and W data bases, it is possible to arrive at the following correction factors for completeness.

	Industry's data	This search	This search corrected for LER data bank	Possible ratio of actual events to our finds
Significant events	12	6	8	1.5
Insignificant events	11	1	2.8	4.0
Total	23	7	10.8	2.1

Thus, it is possible that the actual number of PVTS-like events and precursors that occurred in operating reactors from 1963 through 1981 is more than twice the number we located in the NSIC data bank. Curiously, the ACRS stated in 1978 that licensees appearing before it reported that at least twice as many events are judged worthy of noting in their plant files as are reported to the NRC as LERs.[29] The agreement between these data and such observation is probably fortuitous.

Table 5.2. PVTS events reported by Duke Power Company — check on exhaustiveness of finds

Reported event date	Description	Tentative severity assignment	Was it found?
05/05/73	ICS upset resulted in OF of Oconee 1 (precommercial)	3	
01/04/74	HSF demand following RT at Oconee 2	4	
07/11/74	ICS power loss, OF, and PORV opening at Oconee 3	3	
09/10/74	Wrong ICS response to false turbine pressure signal at Oconee 2	1	
09/17/74	ICS fails to run back FW following RT at Oconee 2	1	
03/07/75	MFW control valve stuck open at Oconee 2	1	
04/30/75	ICS slow in running back FW following loss-of-load test at Oconee 3	2	
05/25/75	ICS response to a blown fuse caused overcooling at Oconee 3	3	
06/13/75	PORV failed open at Oconee 3	4	Yes
07/13/75	Wrong ICS response to turbine valve transistor failure at Oconee 3	0	
08/14/76	OF caused by manual control at Oconee 1	0	
12/14/78	ICS short caused OF of SG at Oconee 1	3	Yes
11/10/79	Loss of ICS and instrumentation caused overcooling at Oconee 3	4	Yes
01/30/80	ICS overfeeding SG following RT at Oconee 2	2	
03/14/80	ICS failed to run back FW following RT at Oconee 3	2	
05/04/81	Signal drift causing ICS to withdraw rods at Oconee 1	0	

Summary: 3 of 16 PVTS-like events found
3 significant of 7 significant events found
0 insignificant of 9 insignificant events found

5.3 Check of Finds Against Events of Pressure Transients at Low Temperatures

Although searches were directed toward changes in temperatures or pressures that could lead to temperature changes, pressure transient events would be expected to occur while the reactors were at low temperatures (cold shutdown mode). Several of these incidents took place in the first half of the 1970s, mostly caused by inadequacies in design to relieve the RCS pressure should an inadvertent error (by operator or test and maintenance personnel) result in the start of a pump or the closure of a bleed valve while the reactor was at low temperatures and/or solid condition.

Since 1977, several design fixes have been incorporated to operating reactors (such as larger relief valves that can also pass water as well as steam). Incidents of pressure shocks while RCS is at low temperatures are expected to drastically be reduced in number.

Table 5.4 lists 29 such transients before 1977 as reported in NUREG-0258 (Ref. 13). When a simple search for "pressure transient in PWRs"

Table 5.3. PVTS events reported by the Westinghouse Owners Group

Reported event date	Reactor and event description	Severity assignment	Did we find it?
04/28/72	Robinson 2: preop small steam line break; T_i = 530°F, T_F = 320°F, p_F = 2050 psi, Δt = 1 h	4	
11/05/72	Robinson 2: stuck SG valve; T_i = 550°F, T_F = 389°F, p_F = 1700 psi, Δt = 2 h	4	Yes
02/26/75	Point Beach 1: SG tube rupture; $T_i \approx$ 550°F, $T_F \approx$ 430°F, $p_F \approx$ 1000 psi, $\Delta t \approx$ 2 h	2-4	
05/01/78	Robinson 2: RC pump seal failure; T_i = 450°F, T_F = 310°F, p_F = 1000 psi, Δt = 30 min	4	
09/25/79	North Anna 1: stuck steam dump valve; T_i = 542°F, T_F = 472°F, p_F = 2350 psi, Δt = 15 min	4	Yes
10/02/79	Prairie Island 1: SG tube rupture; T_i = 528°F, T_F = 350°F, $p_F \approx$ 1000 psi, Δt = 1 h	3-4	
05/23/80	North Anna 1: spurious SI actuation; T_i = 543°F, T_F = 525°F, p_F = 2360 psi, Δt = 15 min	2	Yes
01/25/82	Ginna: SG tube rupture and stuck PORV; T_i = 550°F, T_F = 265°F, p_F = 1400 psi, Δt = 45 min	4	1982 not included in our study scope

Summary: For events to 1981 inclusive, we found three out of seven events and two out of four significant events.

Table 5.4. Pressure shock events in the early 1970s

Reactor	Date	Pressure transients (psi)		Severity assignment	Was it found?[a]
		From	To		
Beaver Valley 1	02/24/76	400	1000	2	Yes
	03/05/76	400	1150	3	
	03/13/76	425	495	0	
Cook 1	04/14/76	N.A.[b]	1040	N.A.[b]	Yes
Ginna	1969	95	1300	4	
Indian Point 2	02/16/72	420	670	2	Yes
	02/17/72	420	650	2	Yes
	03/08/72	400	640	2	
	04/06/72	422	680	2	Yes
	05/18/73	440	575	1	Yes
	01/23/74	425	525	1	Yes
	02/22/74	150	560	2	Yes
	09/12/76	400	515	1	Yes
Indian Point 3	09/30/76	50	2250	4	Yes
Oconee 2	11/15/73	800	1860	4	
Palisades	09/01/74	N.A.[b]	960	N.A.[b]	
Point Beach 2	12/10/74	345	1400	3	Yes
	02/28/76	400	830	2	Yes
Prairie Island 1	10/31/73	420	1100	3	Yes
	01/16/74	395	840	2	
Prairie Island 2	11/27/74	N.A.[b]	900	N.A.[b]	Yes
Surry 1	01/28/73	450	590	1	Yes
St. Lucie 1	08/12/75	210	600	2	Yes
	06/17/76	435	815	2	Yes
Trojan	07/22/75	400	3326	4	Yes
Turkey Point 3	12/03/74	50	800	3	Yes
Zion 1	06/13/73	110	1290	4	
	06/03/75	100	1100	4	Yes
Zion 2	09/18/75	95	1300	4	Yes

[a]Our original PVTS searches did not locate any of these overpressurization-at-low-temperature incidents. A simple search for "pressure transient" incidents located all of those marked yes.

[b]N.A. = not available.

Sources:

1. Gary Zech, *Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors*, Final Report; NUREG-0224 (1978).

2. W. E. Vesely et al., *The Octavia Computer Code: PWR Reactor Pressure Vessels Failure Probabilities Due to Operationally Caused Pressure Transients*, NUREG-0258 (1978).

was made, 21 events were easily found (a find ratio of about 70%). We further found five additional incidents, all quite minor, between 1977 and 1981. This confirms our earlier belief that the problem is practically solved. Table 5.5 lists these additional events.

Table 5.5. Additional pressure shock events, 1977–1981

NSIC #	Reactor	Date	Description
167707	Surry 1	07/02/81	PORV lifted due to pressure transient
165876	North Anna	02/29/81	SI actuated at North Anna 1
154890	Calvert Cliffs 1	01/29/80	PORV inadvertently opened
152983	Oconee 3	10/19/79	RCS pressure exceeds operating mode limit
137299	North Anna 1	03/14/78	RCS overpressurized

6. FREQUENCIES ANALYSIS

Realistic frequencies of occurrence and causes are important to improving operation and to probabilistic analyses of events that may occur as an extension of known precursors. The 99 PVTS-like events and precursors have been analyzed in terms of sequence initiators, primary causes (and sometimes secondary causes), and modes of operation before the occurrence of the events.

Although our finds include only 34 significant events and 65 insignificant events, frequency determination must account for the factors discussed earlier in the exhaustiveness analysis of the finds. These include a factor of 1.5 for significant finds, 4.0 for insignificant finds, and 320 RY instead of 329 RY as common denominator of experience.

6.1 Modes of Reactor Operation

Inasmuch as the core is a heat source during the PVTS event, its mode of operation just before the incident is important to determine the function of several operating parameters: reactor coolant pump operation, FW flow, SI, and the degree of mixing in the annulus area of the reactor vessel. In PWR operation, six modes of operation are distinguished:

Mode	Description
1	Steady-state power
2	Startup
3	Hot standby
4	Hot shutdown
5	Cold shutdown
6	Cold shutdown with pressure vessel head untensioned (e.g., for refueling)

The breakdown into modes of operation of the 99 events is shown in Table 6.1. Uncorrected and corrected frequencies are shown in Table 6.2.

These data indicate that about one-half of the events took place when the reactors were at mode 1, and about 85% of the events took place when the RCS was hot. Furthermore, most of the events that occurred when the RCS was cold were insignificant PVTS events.

6.2 Initiating Sequences

While it is realized that each event occurred differently from the rest, the 99 events have been broadly categorized into 8 initiator categories: 7 with event trees previously described (LBLOCA, SBLOCA, OF/cold FW, HSF, EPSI, CF/LPSI, and EX) and "others." The "others" category contains events with miscellaneous initiators. The frequencies of these

Table 6.1. Distribution of finds with respect to mode of reactor operation

Mode	Insignificant events			Significant events		Total number of events
	0	1	2	3	4	
Hot						
Mode 1	7	11	10	8	17	53
Mode 2	3	4	3	2	2	14
Mode 3	1	6	4	2	2	15
Mode 4	2	1			1	4
Cold						
Mode 5	6	2				8
Mode 6	1	4				5
	20	28	17	12	22	99

Table 6.2. Frequency distribution of PVTS events and precursors with respect to mode of reactor operation

Mode	Based on finds		Corrected for completeness	
	Events/RY	Percent of total	Events/RY	Percent of total
Hot				
Mode 1	0.161	54	0.467	48
Mode 2	0.043	14	0.144	15
Mode 3	0.046	15	0.156	16
Mode 4	0.012	4	0.042	>4
Cold				
Mode 5	0.024	8	0.100	10
Mode 6	0.015	5	0.063	>6
	0.301	100	0.972	100

broad categories of initiating sequences are important because U.S. PWR vendors have at times identified different sequences as more limiting to the reactor vessels from the viewpoint of thermal shock.

The breakdown of 99 events into initiating sequences is shown in Table 6.3. The frequencies are shown in Table 6.4.

These data indicate that about one-half of all events were initiated by SI that was caused by false or spurious signals; operator, test, or maintenance errors; or causes not directly having roots in the pressure and/or temperature transients of the heat transfer circuits. It appears

Table 6.3. Distribution of finds[a] with respect to initiating sequences

Initiating sequence	Insignificant events			Significant events		Total number of events
	0	1	2	3	4	
LBLOCA						
SBLOCA	3	2		1	5	11
OF/Cold FW	2	0	6	6	7	21
HSF		2	2	2	8	14
HPSI	6	22	9	3	1	41
CF/LPSI	2	2				4
EX					1	1
Others	7	(1)	(1)	(1)		7+(3)
	20	28	17	12	22	99

[a]Where a secondary cause is observed, it is noted in parentheses but not counted in the tally.

Table 6.4. Frequencies of initiating sequences

Initiating sequence	Based on finds		Corrected for completeness	
	Events/RY	Percent of total	Events/RY	Percent of total
LBLOCA				
SBLOCA	0.033	11	0.091	9
OF/Cold FW	0.064	22	0.161	17
HSF	0.043	14	0.097	10
HPSI	0.125	41	0.481	49
CF/LPSI	0.012	4	0.050	5
EX	0.003	1	0.005	<1
Others	0.021	7	0.087	9
	0.301	100	0.972	100

that rationales and I&C for such SIs should be rethought so that these dedicated safety systems can be spared unwarranted challenges.

No LBLOCA was found, but there were 11 SBLOCAs, or ~10% of the events. These were mostly caused by PORVs or relief valves that failed to close in the pressurizers or by failures of the RCS pump seals. Approximately one-third of the SBLOCAs (after correction) constitute significant PVTS events.

About 27% of the events were transients in the RCS caused by thermal-hydraulic transients in the FW-steam circuits. Four out of ten of these (after correction) are significant and break down approximately half and half between OF of the SGs and HSF.

One event, at Indian Point 2 on or before December 17, 1980, was a result of extensive flooding of the reactor vessel cavity caused by a leak in the containment fan cooler waterline. This enabled cold river water to immerse the reactor vessel from the outside up to 9 ft. The consequences of this flooding are still under investigation by the NRC.[33]

Seven events could not be mapped on a generalized event tree. They all belong to severity 0 and are related to matters quite insignificant to PVTS such as the heat-up of RCS above a temperature limit, the heat-up of the pressurizer faster than 100°F/h, or the failure of one SG blowdown valve (but the redundant valve operated as designed).

6.3 Causes for Initiators

Several causes have been identified that were responsible for initiating the sequence of events. Table 6.5 shows a breakdown of the primary causes and number of events associated with them. Table 6.6 lists the

Table 6.5. Distribution of apparent causes[a] to the PVTS events and precursors

	Insignificant events			Significant events		Total number of events
	0	1	2	3	4	
Operator error	3(1)	3(3)	5(3)	1(2)	(3)	12(12)
Maintenance/test error	4	1	1	1	(1)	7(1)
Valve failure	4	3		6	8	21
Power failure		3	5	1	2(2)	11(2)
I&C error/spurious signal	3	8(5)	(5)	1(1)	3	15(11)
System/equipment failure	5	5(1)	4	2(2)	9(1)	25(4)
Water hammer	1	5				6
Fire/lightning			2			2
	20	28	17	12	22	99

[a]Where a secondary cause is observed, it is noted in parentheses but not counted in the tally.

Table 6.6. Frequencies of apparent causes[a] to PVTS events and precursors

	Based on finds		Corrected for completeness	
	Events/RY	Percent of total	Events/RY	Percent of total
Operator error	0.036	12(12)	0.142	15(10)
Maintenance/test error	0.021	7(1)	0.080	8
Valve failure	0.064	21	0.153	16
Power failure	0.033	11(2)	0.114	12(1)
I&C error/spurious signal	0.046	15(11)	0.156	16(13)
System/equipment failure	0.076	25(4)	0.227	23(3)
Water hammer	0.018	6	0.075	8
Fire/lightning	0.006	2	0.025	3
	0.300	99	0.972	101

[a]Where a secondary cause is observed, it is noted in parentheses but not counted in the tally.

frequencies. Where there were secondary causes, these are noted in parentheses but not counted in the tally.

Several notes must be mentioned in the definition of causes.

Operator errors include those committed by the control room operators, senior control room operators, shift supervisors, and/or area operating staff in direct communication with the control room. These errors include inadvertent errors, wrong implementation of procedures, and negligence. They do not include, however, reactions to instrumentation signals that may turn out to be wrong signals because of faulty and/or inadequate design, or reactions to unforeseen or un-thought-out events. These latter actions, should they be wrong, would be assigned only a secondary operator error status (in parentheses), whereas the primary causes are either control errors or system/equipment failures.

Maintenance and/or test errors involve personnel other than or in addition to the operators cited previously. Ordinarily, a cause is assigned into this category when certain routine maintenances and/or tests were in progress, or when such maintenances and/or tests were being carried out while the system was in cold shutdown, and when the error could not be clearly attributed to the control room operators and supervisors.

Valves cause the initiation of a sequence when they do not carry out their functions as designed (e.g., open on demand but fail to close when the demand is over) or do not respond to control room direction. Valve failure also occurs when a valve does not reseat properly, when excessive boron crystallization deteriorates its function, or when an instrumentation line fails (e.g., break of a crimp connector).

Power failure is a primary cause when a loss of power causes abnormal system behavior, leading to the initiation of a sequence of events. Ordinarily, with good I&C design, such a loss of power should not adversely affect the function of the systems without additional operator errors. In this case power failure should be given a secondary status as initiating cause.

The I&C design is a primary cause when spurious and faulty signals can initiate an event sequence, or when an upset (e.g., loss of power, a short, lightning, or fire) can lead the plant control system to act abnormally, thus initiating an event sequence or misleading the operators into initiating such a sequence.

All other hardware failures that could not be assigned to valves, I&C, and power failure are called system/equipment failures. These include pump failure to reduce speed on demand or as designed, FW heater train malfunction, break of a feed-and-bleed line, SG tube leaks, or a leak in the nonnuclear line that could flood the containment with river water.

Water hammer is listed as a separate cause because it occurred frequently in some reactors, leading mostly to SI. Fire and lightning are also listed because two cases were found where these external (common) causes initiated abnormal behavior of vital plant equipment.

Tables 6.5 and 6.6 give rise to several interesting observations.

1. About 15 of all events were caused by operator/maintenance/test errors, and an additional 10% can be considered as secondarily caused by those errors. However, when fully corrected for, only 1 out of 15 such errors resulted in a significant PVTS-like event. Thus, the common trend to blame operators and/or maintenance/test personnel for possible severe PVTS events appears to be without merit.

2. Valve failures accounted for 16 to 22% of all PVTS-like events, and 40% of these resulted in significant overcooling events. This can be verified by examining Tables 4.4 and 4.5 where many severity 3 and 4 events were caused by failures of valves in the RCS or valves in the feedwater-steam circuits.

3. One-fourth of all PVTS-like events were caused by power failure or inadequacies in the I&C systems. Another 14% can be assigned as secondarily caused by such failures and inadequacies. One out of eight such failures (when fully corrected) resulted in a significant PVTS event.

4. One-fourth of all PVTS events were caused by system and equipment failure (e.g., those hardware failures that could not be classed as valve, power, and I&C failures). Two out of ten such failures resulted in significant PVTS events.

5. If valve failures, power failures, system failures, hardware failures, and design inadequacies are considered together, then they account for two-thirds of all PVTS-like events. One out of four such events is significant. It is thus obvious that more attention must be given to system and hardware if serious future PVTS events are to be avoided. Four classes of hardware/system weaknesses are

- stuck-open PORVs both on the primary and secondary side,
- RCS pump seal failures,
- SG tube failure (will be more frequent as SGs age), and
- difficulties in adjusting FW control values during transition from manual to automatic operation or vice versa.

7. TREND ANALYSIS

With correction factors to adjust for the incompleteness of our finds as discussed earlier, we can gain some preliminary insights by investigating the distribution of PVTS events and precursors with respect to time, owners, nuclear system vendors, and balance-of-plant A-Es.

7.1 Data Base by NSSS Vendors, A-Es, and Utilities

Tables 7.1 and 7.2 show the 47 PWRs in the data base distributed according to nuclear system vendors, balance-of-plant A-Es, and owners.

Table 7.1. Reactors and RYs by vendors and A-Es[a]

Vendor or A-E	Number of reactors	Percent	RYs (through 1981)	Percent
NSSS vendors				
Babcock & Wilcox	8	17	52.68	16
Combustion Engineering	8	17	54.39	17
Westinghouse	31	66	222.19	67
Total	47	100	329.26	100
Architect-Engineers				
Bechtel	14	30	109.97	33
Burns & Roe	1	2	0.93	~0
Ebasco	2	4	16.92	5
Gibbs	1	2	8.35	3
Gilbert	3	6	21.29	7
Pioneer (Fluor Poineer)	3	6	22.84	7
Sargent & Lundy	2	4	16.53	5
Stone & Webster	8	17	73.60	22
United Engineers	2	4	14.20	4
Utilities	11	24	44.63	14
Total	47	99	329.26	100

[a]Round-off errors in the percentages occasionally result in apparent discrepancies in the totals.

We note that W manufactured 30 of 47 operable PWRs, with 222 RYs of experience by December 1981. B&W had 9 reactors with 53 RYs of experience, and C-E had 8 reactors with 54 RYs of experience.

Bechtel was A-E for 14 reactors with 110 RYs and Stone & Webster was A-E for 8 reactors with 74 RYs of experience. The remaining seven A-Es had less than three reactors each. Utilities or their subsidiaries were A-Es for 11 reactors with 45 RYs of experience.

Table 7.2. Reactors and RYs by owners

Owner-operator	Number of reactors	RYs (1981)
Alabama Power & Light	2	4.98
Arkansas Power & Light	2	10.44
Baltimore Gas & Electric	2	12.07
Carolina Power & Light	1	11.57
Commonwealth Edison	2	16.53
Connecticut Yankee	1	14.40
Consolidated Edison	1	8.52
Consumers	1	10.00
Duke	4	24.57
Duquesne	1	5.55
Florida Power & Light	3	23.34
Florida Power Corporation	1	4.92
Indiana & Michigan Electric	2	10.67
Maine Yankee	1	9.15
Metropolitan Edison	2	5.71
New England Nuclear	1	6.15
Northern States	2	15.11
Omaha Public Power	1	8.35
PASNY	1	5.68
Public Service & Gas	2	5.60
Portland Electric	1	6.03
Rochester Gas & Electric	1	12.08
Sacramento Municipal	1	7.22
Southern California Edison	1	14.46
Toledo Electric	1	4.34
TVA	1	1.45
Virginia Electric Power	4	23.36
Wisconsin Public	1	7.73
Wisconsin Electric	2	20.57
Yankee Atomic	1	19.00
Total	47	329.55[a]

[a]Round-off errors result in a slightly different total from 329.26 RYs.

As far as utilities or owners/operators are concerned, Duke Power, Florida Power and Light, and Virginia Electric Power had over 23 RYs of experience; both Wisconsin Electric and Yankee Atomic had 19-20 RYs. The next 9 utilities had between 10 and 17 RYs of PWR experience each.

7.2 Distribution of PVTS Events and Precursors Among PWR Vendors

Table 7.3 shows the distribution of PVTS events and precursors among PWR vendors. Table 7.4 shows the frequencies of these events per intrinsic RY. Intrinsic RY is defined as the RYs accumulated by all reactors of the same vendor.

Table 7.3. Distribution of PVTS events and precursors among PWR vendors

Vendors	Insignificant events			Significant events		Total
	0	1	2	3	4	
Babcock & Wilcox	2	4	2	1	11	20
Combustion Engineering	3	1		4	3	11
Westinghouse	15	23	15	7	8	68
Total	20	28	17	12	22	99

Table 7.4. Frequencies of PVTS events and precursors among U.S. PWR vendors (Events/intrinsic RY)[a]

	Babcock & Wilcox	Combustion Engineering	Westinghouse
Based on finds			
Significant	0.228	0.129	0.068
Insignificant	0.152	0.074	0.238
Total	0.380	0.203	0.306
Corrected for completeness			
Significant	0.342	0.194	0.102
Insignificant	0.607	0.296	0.952
Total	0.949	0.490	1.054

[a]Intrinsic reactor years are (1) uncorrected: B&W = 52.7 RYs, C-E = 54.4 RYs, W = 222.2 RYs; (2) corrected: B&W = 52.7 RYs, C-E = 54.4 RYs, and W = 213.2 RYs.

In terms of events found, the relative numbers by vendors are not too far from their contributions in the number of reactors and RYs. The exception is that C-E reactors had relatively fewer events. In term of events for intrinsic RY after correction for completeness of finds, the overall frequency of C-E PVTS events and precursors is half as high as those of B&W and W reactors. When only significant events are looked at, however, the frequency ranks highest with B&W reactors, followed by C-E's, and then by W's, in the ratio 3:2:1.

While these results need to be examined in more detail, we note that the most distinguishing difference between C-E/B&W reactors and W reactors is that the former has two SGs whereas the latter has three to four SGs. Furthermore, B&W reactors had well-known weaknesses in nonnuclear instrumentation and integrated control systems, which may initiate severe transients due to power failure and/or electrical shorts. It must be noted, however, that since the Three Mile Island 2 accident and the Crystal River 3 incident, these weaknesses have been identified and much improved. For example, no B&W reactor underwent a significant PVTS event between the Crystal River 3 incident (February 26, 1980) and the end of 1981.

7.3 Distribution of PVTS Events and Precursors Among Utilities

As the 47 operable PWRs belong to 30 different utilities, we found it more useful to plot the PVTS events and precursors on the operating history of each reactor. Thus, Figs. 7.1 and 7.2 display the time line drawn for each reactor starting from the date of first commercial operation (reactor year = 0) and extending to the end of 1981. On each time line we plotted the PVTS-like events (denoted by the degree of severity, namely, 0, 1, 2, 3, and 4) in order to gain a visual perspective of the time distribution of the events for each reactor.

Three observations can be made.

1. There appears to be no time pattern to the occurrence of PVTS events and precursors. An exception appears to be at the preoperational testing and during the first year of operation when both system and personnel are presumably still trying to debug problems.
2. There were 13 reactors with no PVTS events (Tables 7.5 and 7.6). These included two by B&W, two by C-E, and nine by W. This mix is roughly proportional to the contribution of each vendor to the total number of reactors and RYs.

The further study of these PVTS-free reactors would reveal technical and institutional aspects that heretofore have only been speculated on. For example, why did Three Mile Island 1 have no PVTS events, while Three Mile Island 2 had at least two PVTS events before the meltdown accident of March 28, 1979? Why did Arkansas 1, a B&W reactor, have no PVTS events, while other B&W reactors had many events and Arkansas 2, a C-E reactor, had at least two significant events? Was it because of the high caliber of operating crews at those reactors, or of different reporting requirements, or of different ways to treat reports by plant authorities?

ORNL–DWG 82–6061 ETD

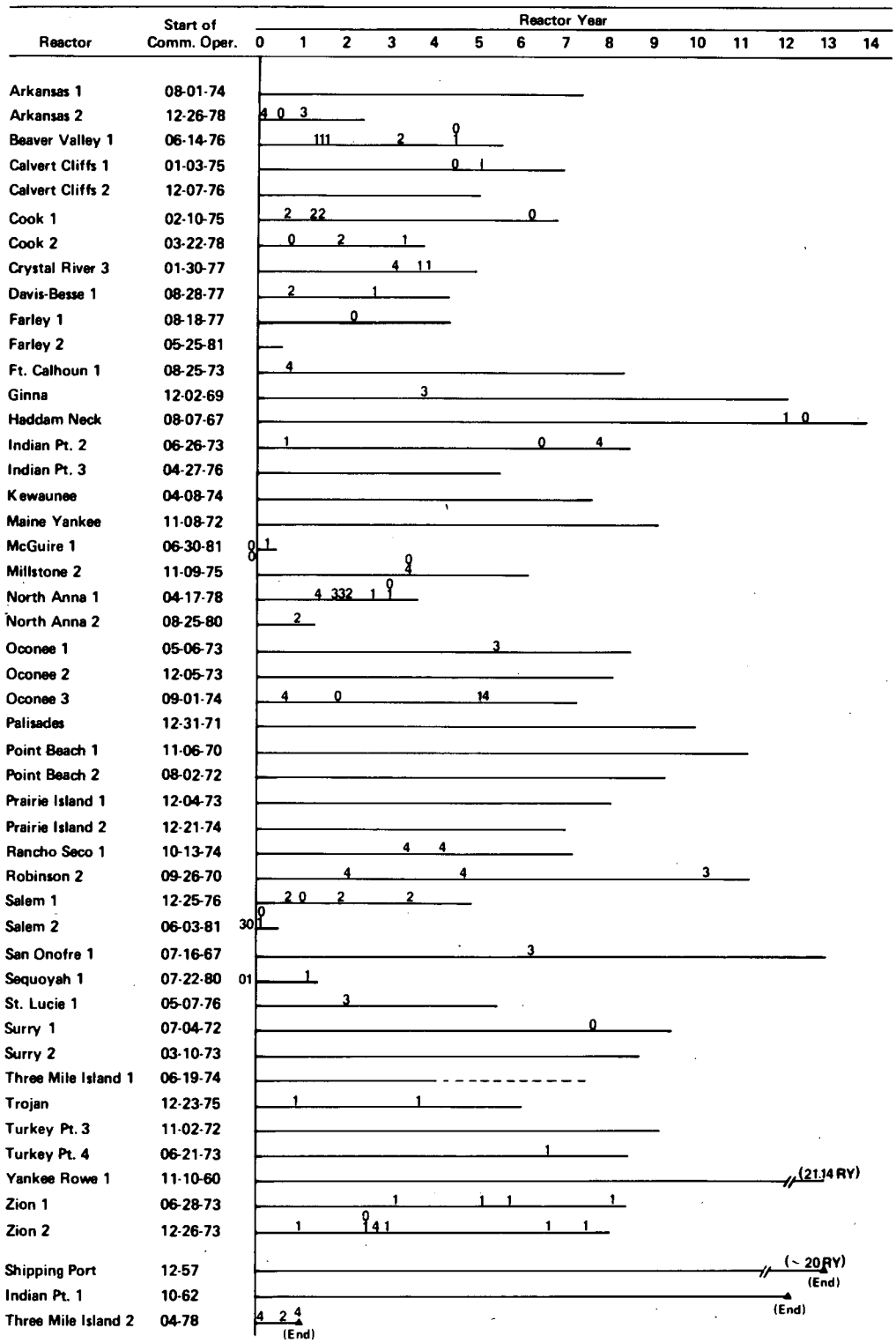

Fig. 7.1. Time line display by severity symbols of U.S. PWR PVTS events vs reactor years.

ORNL–DWG 82–6967 ETD

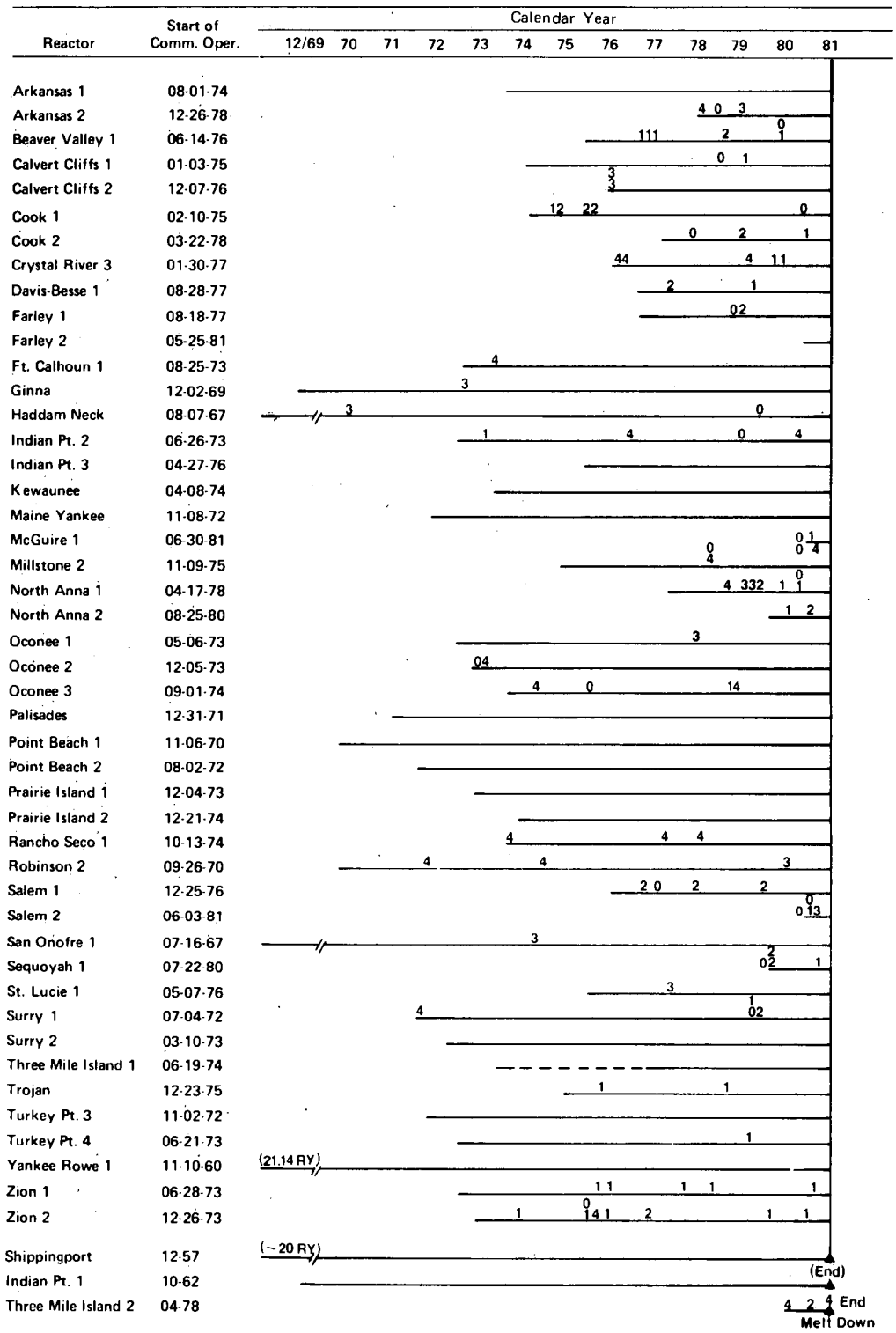

Fig. 7.2. Time line display by severity symbols of U.S. PWR PVTS events vs real time.

Table 7.5. Reactors with no PVTS events and precursors

Vendors and reactors	Owners	Architects-Engineers	RYs
Babcock & Wilcox			
Arkansas 1	Arkansas P&L	Bechtel	7.42
Three Mile Island 1	Metropolitan Edison	Gilbert	4.78
Subtotal			12.20
Combustion Engineering			
Maine Yankee	Maine Yankee	S&W	9.15
Palisades	Consumers	Bechtel	10.00
Subtotal			19.15
Westinghouse			
Farley 2	Alabama P&L	Southern	0.61
Indian Point 3	PASNY	UE&C	5.68
Kewaunee	Wisconsin Public	Pioneer	7.73
Point Beach 1	Wisconsin Electric	Bechtel	11.15
Point Beach 2	Wisconsin Electric	Bechtel	9.42
Prairie Island 1	Northern States	Fluor Pioneer	8.08
Prairie Island 2	Northern States	Fluor Pioneer	7.03
Surry 2	Virginia Electric	S&W	8.81
Turkey Point 3	Florida P&L	Bechtel	9.16
Subtotal			67.67
Total			99.02

Table 7.6. Significant PVTS events and precursors by owners, vendors, and A-Es[a]

Reactor	Owner	Vendor	A-E	Number of significant PVTS events
Arkansas 2	Arkansas P&L	C-E	Bechtel	2
Calvert Cliffs 2	Baltimore G&E	C-E	Bechtel	2
Crystal River 3	Florida Pwr.	B&W	Gilbert	3
Ft. Calhoun 1	Omaha Public	C-E	Gibbs	1
Ginna	Rochester G&E	W	Gilbert	1
Haddam Neck	Conn. Yankee Pwr.	W	S&W	1
Indian Point 2	Con. Ed.	W	UE&C	2
McGuire	Duke	W	Duke	1
Millstone 2	NE Nuclear	C-E	Bechtel	1
North Anna 1	Virginia Elec.	W	S&W	3
Oconee 1	Duke	B&W	Duke	1
Oconee 2	Duke	B&W	Duke	1
Oconee 3	Duke	B&W	Duke	2
Rancho Seco 1	Sacramento	B&W	Bechtel	3
Robinson 2	Carolina P&L	W	Ebasco	3
Salem 2	Public Serv. & Gas	W	Public Serv. & Gas	1
San Onofre 1	Southern Cal. Ed.	W	Bechtel	1
Surry 1	Virginia Elec.	W	S&W	1
St. Lucie 1	Fla. P&L	C-E	Ebasco	1
Zion 2	Comm. Ed.	W	S&L	1
Three Mile Island 2	Met. Ed.	B&W	Burns & Roe	2
			Total	34

[a]B&W reactors: 12 events
C-E reactors: 7 events
W reactors: 15 events

Similar questions can be raised for the other reactors.

3. The hypothesis that utility gains more operating experience by having many reactors on a site does not appear to be supported by our data base. We have cited above the cases with Arkansas 1 and 2 and Three Mile Island 1 and 2 where the younger reactor on the same site appeared to have more trouble in operation than the older one. Other examples include Calvert Cliffs 1 and 2; Cook 1 and 2; Oconee 1, 2, and 3; Salem 1 and 2; and Zion 1 and 2. Learning experience appears to exist at Indian Point 2 and 3 (although they now belong to different utilities), North Anna 1 and 2, and Surry 1 and 2.

A utility with more than one site for PWR plants did not necessarily learn from the experience of other sites either. Thus, after over 20 years of experience with the three Oconee reactors (B&W design), Duke Power could not avoid many startup problems with McGuire (a W design), which recorded 4 PVTS-like events during its first year, one of which was of severity 4. Similarly, Virginia Electric Power had many PVTS-like events at North Anna 1 and 2 despite its many years of experience at Surry 1 and 2.

Our preliminary hypothesis is that a well-trained crew with certain degrees of independence from management pressure may be responsible more than anything else for the good construction and operation of a nuclear plant. We probably can point to the well-known records of Point Beach 1 and 2, Prairie Island 1 and 2, and Yankee Rowe.

7.4 Action of Operator and Systems Following Initiation of PVTS-Like Events

A most important consideration following the initiation of a PVTS-like event is system or operator actions. The degree of event severity depends a great deal on such actions. There are so many possibilities and yet so little experience that most PRA work on PVTS inevitably has large bands of uncertainties. The policymaker is faced with a difficult decision: Should one rely on PRA calculations, allowing credits for detailed operator/system actions or should one specify an upper bound for those actions and subsequently make sure that all actual possibilities would result in consequences less severe than those of the upper bound? Here we should note that industry relies a great deal on the versatility of the operator, while regulatory bodies tend to assume no operator action until some time after event initiation.

With the perspective described in the preceding paragraph, efforts have been made to extract information from the 34 significant PVTS events and precursors on the following four questions:

1. How long did it take the operator to become aware of the initiation of a PVTS-like event?
2. If there was a PORV stuck open, how long did it take the operator to successfully close or block it?
3. Did the RCS main circulation pumps shut down?
4. Was there repressurization?

Table 7.7 provides a summary of information regarding these four questions when our 34 significant events were examined. Table 7.8 is a detailed listing of the information.

Regarding the utility of the reactor operators in system upsets, it is noted that the operators were aware of an upset condition and took corrective actions within 5 min in 16 out of 34 cases. In 24 cases, it took them <10 min; in 25 cases, it took <15 min; and in 26 cases, it took <30 min. The information was not clear on seven cases, and in one case,

Table 7.7. Summary of system/operator actions following significant event initiation

Time to operator notice/action	Number of cases
<5 min	16
Between 5 and 10 min	8
Between 10 and 15 min	1
Between 15 and 30 min	1
Not clear	7
Knew something wrong but waited until reactor shutdown	1
Total	34
Time to correct for valve failure[a]	
<5 min	3
Between 5 and 10 min	1
Between 10 and 60 min	2
Long	1
Not clear	1
No success	1
Total	9
Reactor coolant system circulation pumps condition	
Shutdown (automatic, manual, or already down)	14
Not shutdown	12
Not clear	8
Total	34
Reactor coolant system repressurization	
No	6
Yes	17
Not clear	11
Total	34

[a]Nine cases of 34 significant events involved PORVs.

Table 7.8. Listing of system/operator actions following initiation of significant PVTS events

Events	Sequence	Time to operator first action	Time to isolation of failed PORV	Shutdown RCS pumps	Repressurization (by what)
			Severity 4		
169297	OF	<7 min	N/A	No	Yes
162919	EX	Observed abnormality but waited till reactor shutdown to inspect	N/A	No	Maintain RCS pressure
160846	SBLOCA	<4 min	<7 min	Yes	Yes
153772	OF	<10 min	N/A	No	Yes
153164	SBLOCA	<8 min but wrong actions	Long	Yes (due to vibration)	No
152751	HSF	<10 min	N/A	Yes	Yes
149746 (148074, 146587)	HSF	Less than a few minutes	30 min to 1 h	Not clear	Yes
148767	CF/LPSI	Not clear	N/A	N/A	No
148112	OF	<8 min	N/A	No	Yes
147267	OF	<7 min	N/A	No	Not clear
138830	OF	Less than a few minutes	N/A	No	Yes
137918 (137919, 139931)	HSF	Less than a few minutes	<4 min	Not clear	Yes (HPSI)
126585	SBLOCA	Less than a few minutes	N/A	Yes (due to pump seal failure)	Yes (charging pump)
124920	OF	Less than a few minutes	N/A	Yes, auto.	Not clear
123150	HSF	Not clear	N/A	Not clear	Yes (after a long time)
120819	OF	<30 min	N/A	Yes, auto.	Not clear
103892 (104941, 104981)	SBLOCA	Less than a few minutes	Less than a few minutes	Not clear	No
103077	SBLOCA	<13 min	N/A	Yes, due to pump trouble	No
097128 (096427)	HSF	<5 min	Less than a few minutes	No	Yes
088077	HSF	Not clear	N/A	Yes, auto.	Not clear
076461	HSF	Less than a few minutes	Could not	No	No
075613	HSF	Not clear	Not clear	Yes, mode 4	Not clear
			Severity 3		
167625	HPSI	Less than a few minutes	N/A	Yes, mode 3	Not clear
164149	SBLOCA	<3 min	<41 min (pressurizer spray leak)	Yes (to stop pressurizer spray leak)	Yes (stopping pressurizer spray leak)
157130	HSF	Not clear	N/A	Not clear	Not clear
157093	HPSI	<4 min	N/A	No	Yes
155359	HPSI	<3 min	N/A	No	Yes
146533	OTHER	Not clear	N/A	Not clear	Not clear
139563	OF	Less than a few minutes	N/A	Not clear	Not clear
120769	OTHER/OF	<10 min	N/A	No	Yes
120766	OF	<10 min	N/A	No	Yes, at pressure
087016	OF	<1 min	N/A	Yes, reactor down	No
085369 (085370, 087031)	OF	Less than a few minutes	N/A	Yes, auto	Not clear
068933 (057478)	HSF	Not clear	N/A	Not clear	Not clear

the operators noticed abnormal behavior of the instrumentation lines below the reactor vessel but decided that the system was safe to operate until the scheduled reactor shutdown. It was not discovered until then that the reactor vessel was flooded up to 9 ft by river water from the outside.

Of the 9 applicable cases when valves, mostly PORVs, failed, the operators were able to correct for them within 5 min for three cases, 10 min for four cases (cumulative), and 60 min for six cases (cumulative). One failure could not be corrected for, one failure took a long time, and the information for another is not clear.

Normally the control system is such that the RCS recirculating pumps automatically coast down upon reactor trip and onset of AFW. These pumps can also be manually overridden by the operator (i.e., the operator can maintain all or a few of the RCS pumps in operation or can shut all of them down). Of the 34 significant PVTS events, we observed (1) 14 cases with RCS circulating pumps automatically or manually shutdown upon reactor trip or remaining in shutdown condition at the initiation of the event; (2) 12 cases with RCS pumps continuing operation; and (3) 8 cases that were unclear.

The repressurization of RCS following cooldown and pressure drop is important in whether or not a credit for warm prestress can be taken in the fracture mechanics analysis of the vessel. Although the time history of such repressurization is essential, we were able to obtain from the data base only the information on whether or not repressurization existed. The RCS did get repressurized in 17 cases and did not get repressurized in 6 cases, while the information was not clear in the 11 remaining cases. With the importance placed on restoring system pressure and temperature following a relatively minor system upset (almost all PVTS events and precursors found in this study are considered minor system upsets), we must assume that the RCS did get repressurized in the majority of events selected.

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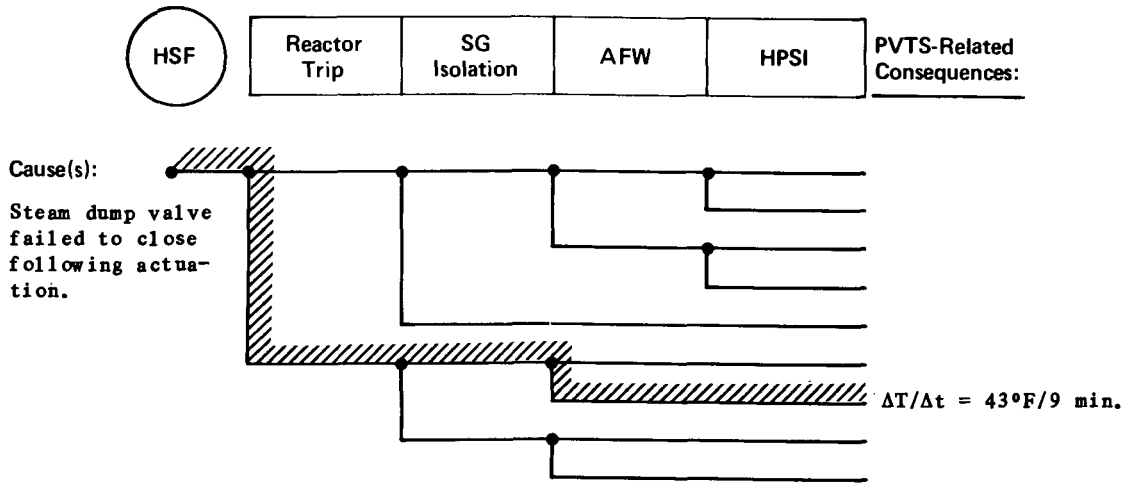
Appendix A

MAPPING OF PVTS EVENTS AND PRECURSORS ON GENERALIZED EVENT TREES

This appendix contains summaries of PVTS events and their mapping on generalized event trees arranged by increasing NSIC accession numbers. The abbreviated NSIC description of these events is given in Appendix B. The results of this analysis are listed in the text, Tables 4.1–4.5. The seven generalized event trees are described in Sect. 3.4.

Reactor:	Connecticut Yankee		
NSIC No.:	057478; 068933	Severity Category:	3
Event Date:	10/12/70	Operating Mode:	1
Letter Date:	11/04/70 12/29/71	Cause Category:	Steam dump valve failed to close following actuation.

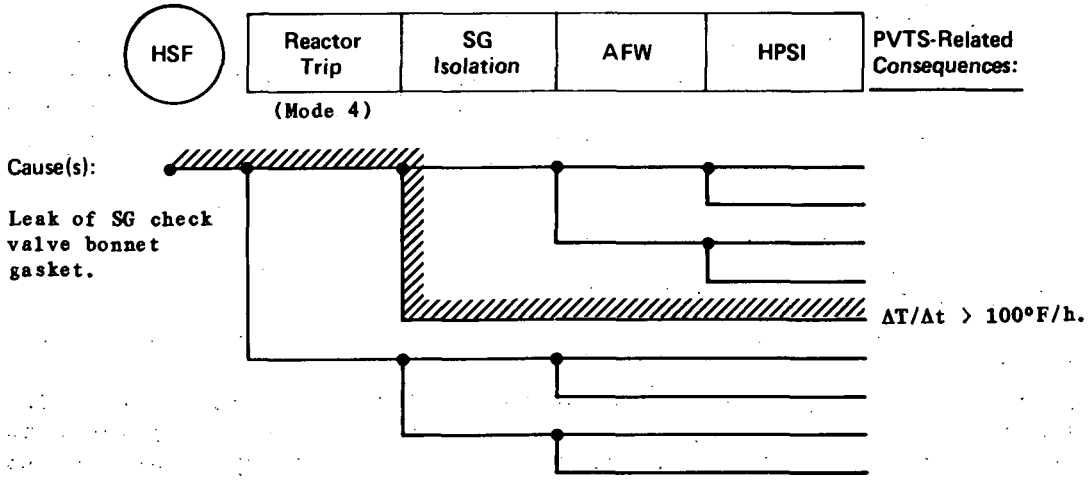


Summary of Events:

1. Mode 1, 100% power.
2. Spurious low-flow signal caused reactor trip. Steam dump valves opened; one out of ten failed to close due to maladjustment of the positioner coil stop.
3. RCS temperature dropped 43°F in 9 min.
4. Operator actuated a manual valve to block the failed dump valve.

Reactor: Surry 1
NSIC No.: 075613
Event Date: 10/01/72
Letter Date: 10/11/72

Severity Category: 4
Operating Mode: 4
Cause Category: SG lost water and steam to containment.

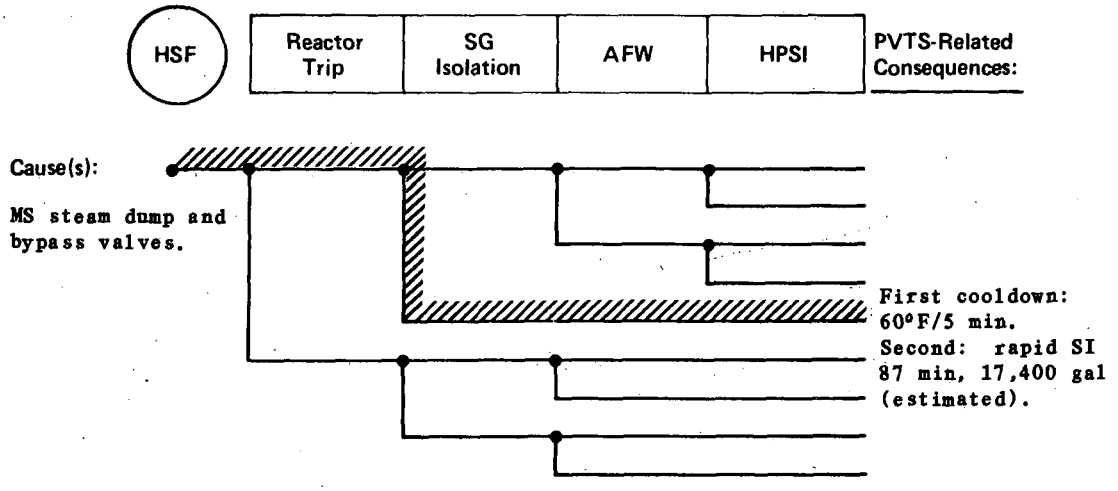


Summary of Events:

1. Mode 4, hot shutdown.
2. A leak in the SG check valve bonnet gasket, causing a gradual decrease in the SG level. The leak caused a flow path from the SG ring back to the check valve. This caused a water hammer.
3. Water hammer damaged the bonnet gasket further, causing SG water to blow down to containment.
4. High steam flow caused RCS to cool down in excess of 100°F/h.

Reactor: Robinson 2
NSIC No.: 076461
Event Date: 11/05/72
Letter Date: 11/30/72

Severity Category: 4
Operating Mode: 3
Cause Category: Steam bypass and dump valves.

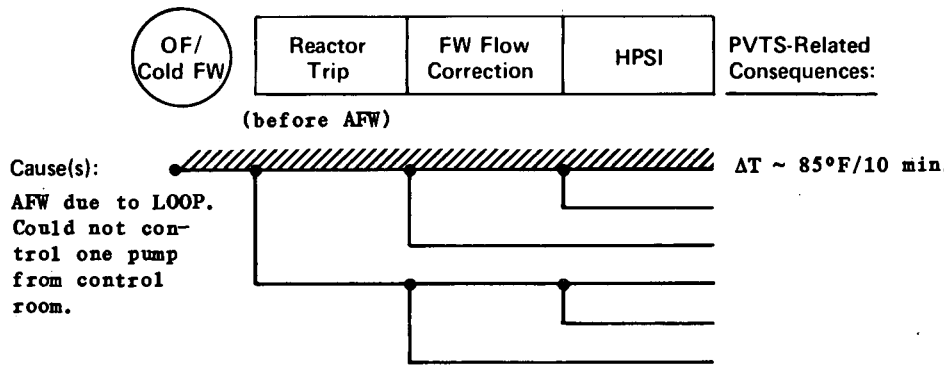


Summary of Events:

1. Reactor was brought down to 70% power the previous day for a weekly turbine valve test.
2. 0103 h, Nov. 5, 1972, difficulty was experienced in the testing. Operator tripped the reactor manually. Main steam PORV failed open for 5 min, reducing RCS temperature to 509°F. (ΔT = 38°F). Boration took place for 2 min.
3. 0326 h, startup began; T_{avg} = 547°F and was maintained by the main steam PORV and dump valve.
4. 0333 h, reactor was subcritical; PORV opened but did not close. After 15 min, there was emergency boration and HPSI went on for 1 h 17 min (0506 h).
5. Total ΔT and total amount of HPSI were not reported. Decision at 0430 h to bring the plant to cold shutdown may reflect the fact that RCS temperature dropped substantially.
6. 0900 h, RCS temperature was 200°F.

Reactor: Ginna
NSIC No.: 085369/085370/087031
Event Date: 10/31/73
Letter Date:

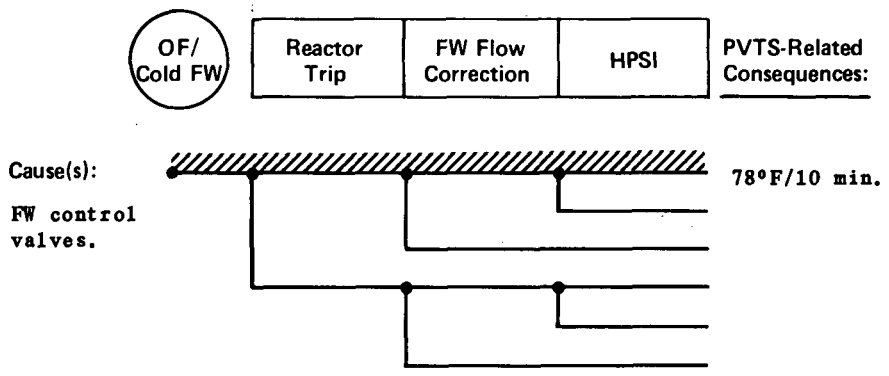
Severity Category: 3
Operating Mode: Mode 1/100%
Cause Category: Inadequate transmission capability. Inadequate FW shutdown capacity.



Summary of Events:

1. Mode 1, 100% power; 1/4 115 kV out of service due to construction; auxiliary power provided from station switchyard and an external 34.5-kV line.
2. Second 15 kV tripped due to a ground fault.
3. Power swings caused remaining 2 transmission lines to trip; loss of station switchyard, hence 1 auxiliary power source. The remaining auxiliary power source was overloaded leading to total LOOP.
4. Turbine trip/reactor trip. Diesel generator started.
5. AFW started on low SG level and LOOP. RCS temperature decreased.
6. Operator secured motor-driven AFW pump from control room and turbine-driven pump locally (due to circuitry).
7. SI started due to low RCS pressure.
8. ΔT ~ 85°F/10 min.

Reactor:	San Onofre 1		
NSIC No.:	087016	Severity Category:	3
Event Date:	10/20/73	Operating Mode:	Mode 1
Letter Date:	10/25/73	Cause Category:	FW control valves.

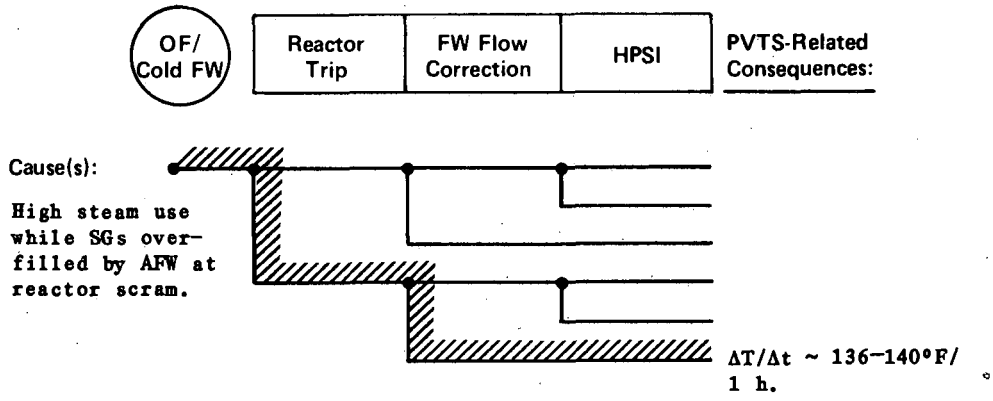


Summary of Events:

1. Unit was being removed from service to investigate turbine problems indicated by excessive vibration; condenser salt water in-leakage; SGs were blown down, trisodium phosphate was added to mitigate salt water, but to no effect.
2. 1:19 h, turbine stop valve closed; FW control valves automatically went to 80% (T_{avg} > 540°F).
3. 1:20 h, unit off line; T_{avg} = 535°F and decreasing rapidly; FW on automatic; reactor on automatic; excitation field breaker open.
4. Reactor control placed on manual; CB-2 pulled to increase T_{avg}, FW control placed in manual, and flow control valve closed. SG level increased rapidly; pressurizer level decreased to 10%.
5. Operator closed FW block valves.
6. 1:21 h, SI initiated; reactor tripped; level high in all SGs.
7. 1:29 h, FW pumps secured; valve closed.
8. 1:35 h, charging and letdown flow established.
9. Cold leg temperature dropped from 548 to 470°F; p ~ 1400 psig. Charging water temperature 94°F. Pressurizer level dropped from 20% to below surge line.

Reactor: Oconee 2
NSIC No.: 088077
Event Date: 01/04/74
Letter Date: 01/17/74

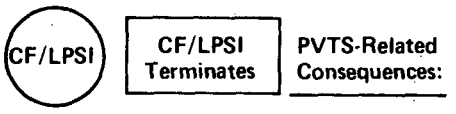
Severity Category: 4
Operating Mode: 1
Cause Category: High steam use when SG overfilled and reactor shut down.



Summary of Events:

1. Mode 1, 75% power.
2. Spurious signal in the 230-kV switchyard breaker caused isolation of switchyard and RCS and FW pumps trip. Reactor scrammed.
3. AFW pumps started and filled SGs to 95% as designed to induce natural circulation in RCS.
4. Need of steam by auxiliary equipment and high SG level caused excessive cooldown of RCS ΔT/Δt = 80°F/15 min. Overall ΔT of RCS was 136–140°F over a period of 1 h.

Reactor:	Indian Point 2		
NSIC No.:	089159	Severity Category:	1
Event Date:	02/22/74	Operating Mode:	6
Letter Date:	02/28/74	Cause Category:	Inadvertent signal during maintenance.



Cause(s):

Inadvertent signal.

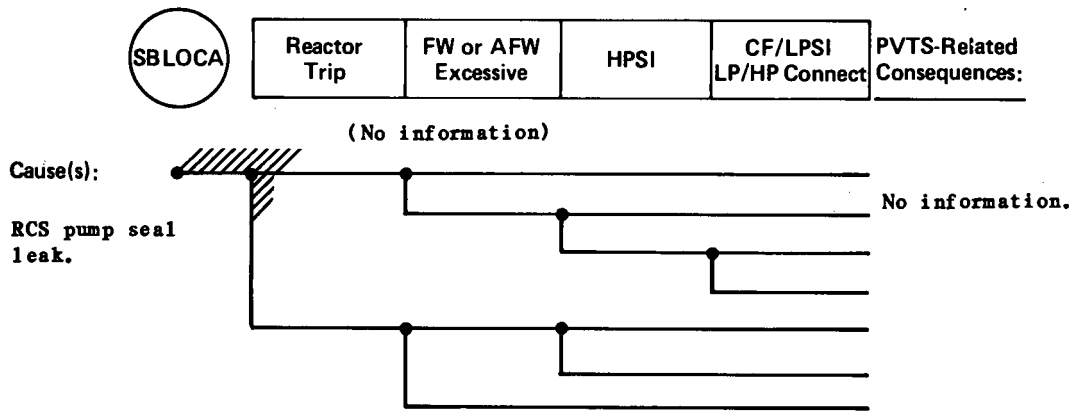
Δp from 150 to 560 psig.

Summary of Events:

1. Mode 6, RHR in operation, RCS at 150 psig, 115°F.
2. Test was being performed in PT-M11 (Steamline Pressure Analog Channel Function).
3. An inadvertent SL signal was generated, causing accumulator stop valves to open. Water flowed into RCS, causing pressure to increase from 150 to 560 psig.

Reactor: Oconee 2
NSIC No.: 091182
Event Date: 01/22/74
Letter Date: 04/30/74

Severity Category: 0
Operating Mode: 2
Cause Category: RCS pump seal failure.

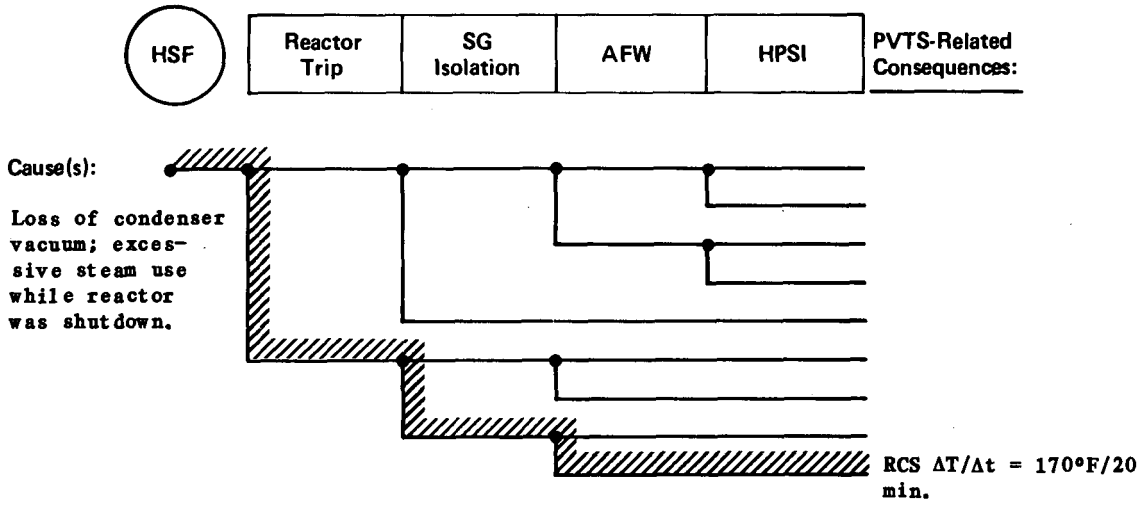


Summary of Events:

1. Mode 2, routine startup operation.
2. RCS pump 2B2 developed an upper seal leakage.
3. No information on LOCA and thermal hydraulics.

Reactor: Rancho Seco
NSIC No.: 096427/097128
Event Date: 10/07/74
Letter Date: 10/08,22/74

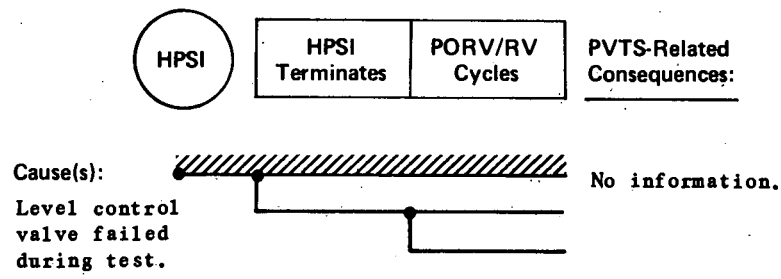
Severity Category: 4
Operating Mode: 1
Cause Category: Excessive use of steam while reactor was shut down.



Summary of Events:

1. Mode 1, 15% power.
2. Condenser vacuum decreased, and efforts to restore it failed.
3. Condenser bypass valves were closed to protect condenser. All six atmospheric steam dump valves were isolated previously for positioner adjustments. Only code relief valves were available. Operator tripped reactor.
4. High RCS pressure and temperature (Δp = 80 psi, ΔT = 7°F), high steam pressure (1010 psig) were stemmed by the automatic lifting of a code steam relief valve for 1 min.
5. Continued demand of steam by auxiliary loads caused RCS to drop further to 1840 psig, 533°F at 4.5 min.
6. HPSI 1A and 1B were started to maintain RCS pressure and pressurizer level. An auxiliary boiler was started to provide auxiliary steam. Feeding of SG was stopped.
7. At 26 min, RCS temperature was 408°F (ΔT = 168°F). At 40 min, stable condition was achieved, RCS temperature was 460°F, pressure was 2155 psig.
8. B&W analysis says total ΔT is 170°F in 20 min.

Reactor:	Zion 2		
NSIC No.:	100433	Severity Category:	1
Event Date:	12/30/74	Operating Mode:	1/82%
Letter Date:	03/07/75	Cause Category:	Control valve failed during test.

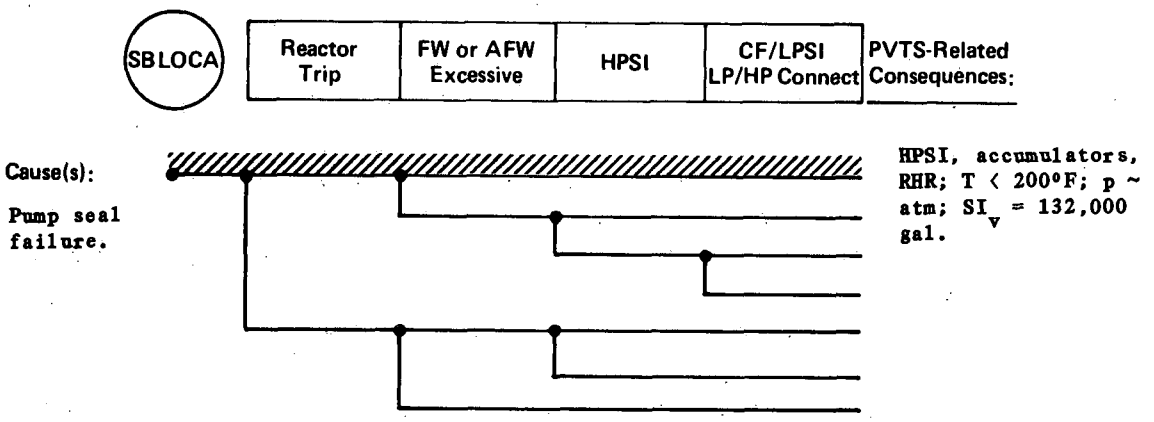


Summary of Events:

1. 1357 h, Mode 1, 82% power. Level control valve for 2C SG went closed while safeguards testing was in progress.
2. Steam flow mismatch and low SG level caused reactor trip (3% drop in level in 3C SG, 8% increase in level in other three SGs).
3. Δp between SGs caused SI. It was terminated in <30 s.

Reactor: Robinson 2
NSIC No.: 103077
Event Date: 05/12/75
Letter Date:

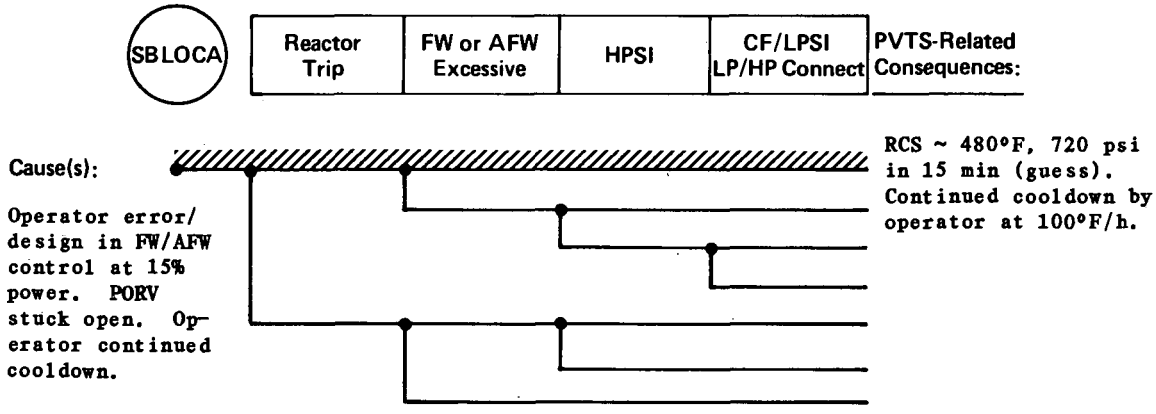
Severity Category: 4
Operating Mode: Mode 1/100%
Cause Category: RCS pump seal failure.



Summary of Events:

1. Mode 1/100% power. RCS pump C had seal problems. Seal flow was >6 gpm and fluctuated. Power was reduced to 36% power; Loop C was idled.
2. Rapid shedding of load caused SG high level, which led to turbine trip and reactor trip. Steam dump valve was used to cool down RCS also.
3. Cooling to 3 RCS pumps was lost due to flashing in pump C seal coolant line. All pumps were stopped.
4. Loss of coolant through RCS pump C seal caused high pressure in reactor cooldown drain tank (RCDT).
5. Cooling to RCS pumps was restored thanks to block valves. Pump C was started with increased seal flow.
6. RCS cooldown was started using condenser dump.
7. Pump C seals #2 and #3 failed. RCS pressure dropped fast. Pump C was stopped. HPSI pumps A, B, and C were started. Pressurizer level stabilized; HPSI pump C stopped.
8. Pressurizer spray actuated to bring RCS pressure down to RHR. Accumulators discharged and were stopped by isolation valves.
9. RHR took over. RCS pressure at atm; T < 200°F; containment temperature 100°F; 132,500 gal in containment.

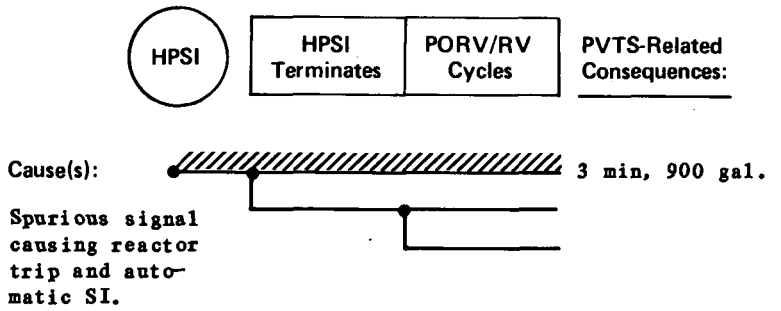
Reactor:	Oconee 3		
NSIC No.:	103892/104941/104981	Severity Category:	4
Event Date:	06/13/75	Operating Mode:	Mode 1/15%
Letter Date:	06/27/75	Cause Category:	Operator/design error. PORV stuck open. Continued operator error.



Summary of Events:

1. Mode 1/15% power, on the way to shutdown maintenance.
2. Not stated, but most likely operator switched from auto FW control to manual, or to AFW manual; regulating valves may have been closed too narrowly, causing lack of FW and RCS pressure increase.
3. PORV opened but got stuck open due to boric acid crystals. A solenoid operator plunger was stuck in the open position.
4. RCS blew down until block valve was actuated to isolate the PORV. At that time (<15 min?) RCS T ~ 480°F, p ~ 720 psi.
5. Shutdown continued with cooldown rate at 100°F/h. Assuming that first cooldown taking place over 15 min, the cooling rate was ~400°F/h.

Reactor:	Cook 1		
NSIC No.:	109619	Severity Category:	1
Event Date:	10/03/75	Operating Mode:	1
Letter Date:	12/31/75	Cause Category:	Spurious signal causing reactor trip.

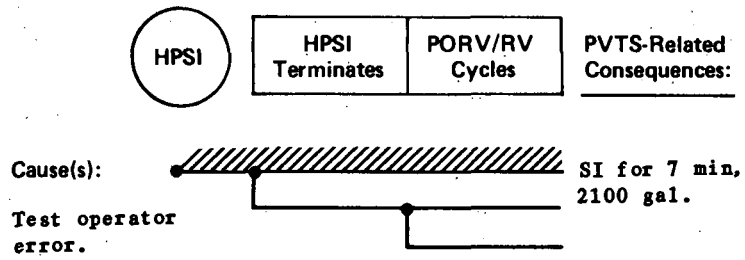


Summary of Events:

1. Mode 1, 81% power.
2. Reactor protection lights flashed; no cause was later found.
3. Reactor and turbine tripped, steam line isolated, SI on.
4. After assuring that all critical parameters were normal, SI was terminated.
5. 900 gal, 3 min of SI; $\Delta T = 16°F$.

Reactor: Cook 1
NSIC No.: 112149
Event Date: 12/13/75
Letter Date: 03/12/76

Severity Category: 2
Operating Mode: 1, 81%
Cause Category: Test operator error.

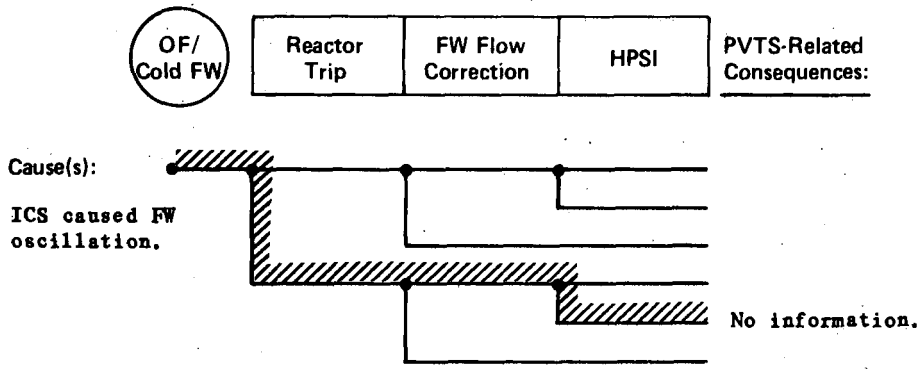


Summary of Events:

1. Mode 1, 81% steady-state power.
2. Technician tested a channel but failed to inform control room to switch control to standby channel.
3. Test procedure caused the steam flow signal to be removed, hence causing a false indication that steam flow was reduced. This caused the #2 SG FW regulating valve to close, resulting in rapid decrease in SG level.
4. Reactor trip occurred when SG level was at 25% while steam flow was still high. Turbine trip also caused high steam flow via steam dump.
5. SI was actuated due to high steam flow, low T_{avg}.
6. Control operator did not know the cause for SI and allowed it to go on for 7 min before terminating it. Total injection 2100 gal.

Reactor: Oconee 3
NSIC No.: 116240
Event Date: 07/13/76
Letter Date: 07/27/76

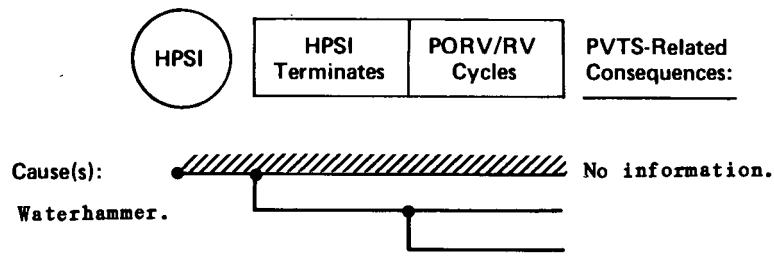
Severity Category: 0
Operating Mode: 1
Cause Category: ICS-caused FW oscillation and rod withdrawal.



Summary of Events:

1. Mode 1, 81% power. FW transient caused reactor power to increase to 103%.
2. Operator put FW in manual and corrected the incident.

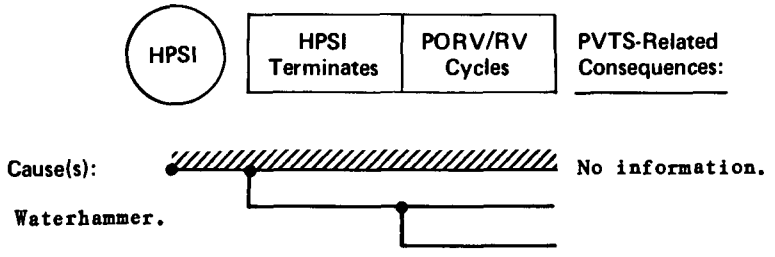
Reactor:	Zion 2		
NSIC No.:	117079	Severity Category:	0
Event Date:	06/20/76	Operating Mode:	3
Letter Date:	08/13/76	Cause Category:	Waterhammer.



Summary of Events:

1. 01:03 h, Mode 1, 100% power. Turbine trip caused reactor trip. RCS at hot shutdown condition. AFW on.
2. Waterhammer occurred in 2C SG, causing high Δp between SGs.
3. SI cccurred; operator reset SI in 15 s.

Reactor:	Zion 2		
NSIC No.:	117080	Severity Category:	1
Event Date:	05/25/76	Operating Mode:	1
Letter Date:	08/12/76	Cause Category:	Waterhammer.

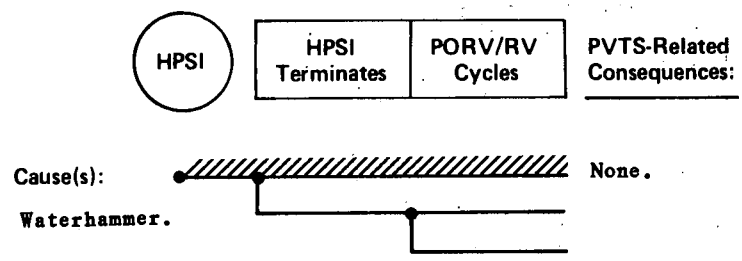


Summary of Events:

1. 1055 h, Mode 1, 875 MW(e) (81%). 2B FW pump tripped, causing reactor trip. AFW went on automatically, feeding about 100 gpm to each SG.
2. 1129 h, reactor in Mode 4.
3. Waterhammer in 2C SG, causing Δp between SGs resulting in SI.
4. SI was reset in 90 s.

Reactor: Zion 1
NSIC No.: 117634
Event Date: 06/20/76
Letter Date: 08/27/76

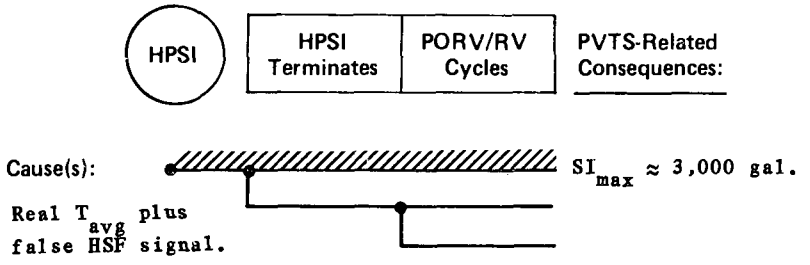
Severity Category: 1
Operating Mode: 2/20%
Cause Category: Waterhammer.



Summary of Events:

1. 1427 h, Mode 2, 20% power.
2. Operator attempted to place feedwater flow controller from manual to automatic.
3. Rapid FW increase in 1D SG caused a waterhammer.
4. SI occurred due to Δp between SGs. It was reset in 20 s.

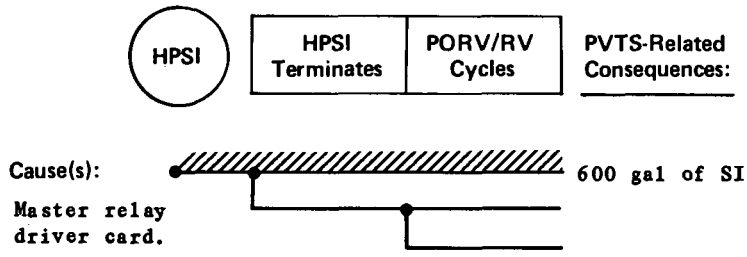
Reactor:	Cook 1		
NSIC No.:	118247	Severity Category:	2
Event Date:	06/03/76	Operating Mode:	1
Letter Date:	09/01/76	Cause Category:	Inverter failure.



Summary of Events:

1. Mode 1, 100% power; all automatic; all control rods in except bank D at 215 step.
2. 2045 h, inverter CRID-I for vital bus channel 1 failed. Reactor trip (each vital bus supplies power to the position logic for one of the RCS pump breakers).
3. T_{avg} dropped, 567 to 547°F.
4. False high steam flow (due to loss of vital bus) plus low T_{avg} caused SI for 4 min, using 165°F water.

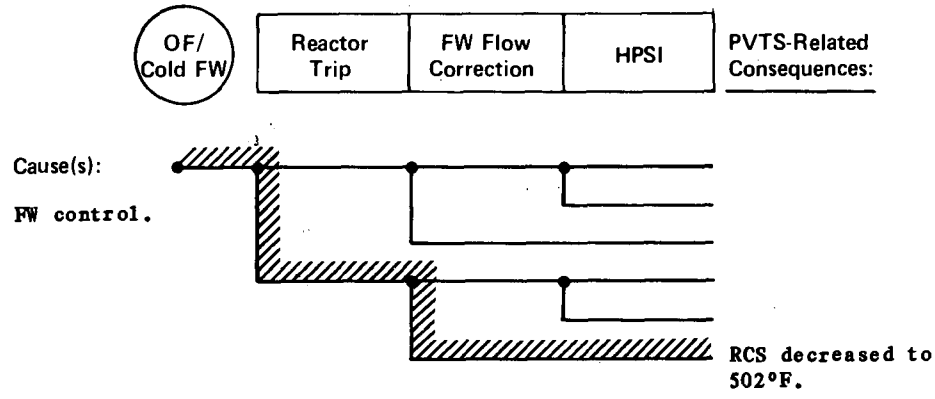
Reactor:	Cook 1		
NSIC No.:	119312	Severity Category:	2
Event Date:	07/05/76	Operating Mode:	3
Letter Date:	09/30/76	Cause Category:	Faulty master relay driver card.



Summary of Events:

1. Mode 3, hot standby. T_{avg} = 548°F, p = 2235 psig; shutdown margin 7.9% Δk/k. Prepared for power after 4 d of outage.
2. 18:34. SI on safeguard train A; no cause found except possibly by a faulty master relay driver card. 1 centrifugal charging pump ran about 4 min, injecting 600 gal of 165°F water.
3. No noticeable ΔT in hot or cold legs.

Reactor:	Calvert Cliffs 2		
NSIC No.:	120766	Severity Category:	3
Event Date:	12/08/76	Operating Mode:	2
Letter Date:	12/27/76	Cause Category:	FW control.

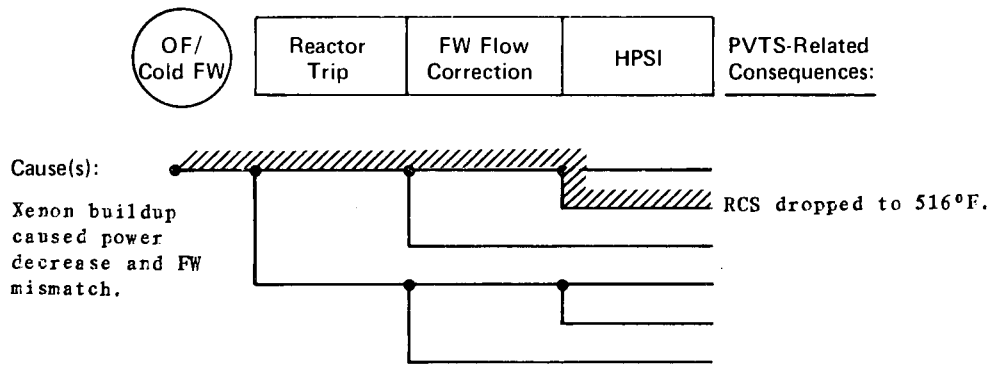


Summary of Events:

1. Mode 2, being escalated for power testing.
2. Main turbine generator was paralleled.
3. Load was reduced at 27 min to respond to decreasing T_{avg} of RCS.
4. Rapid feeding of FW to increased SG level caused RCS T_{avg} to decrease to 503°F.
5. T_{avg} was restored to >515°F within 10 min.
6. It was determined that this event was due to (a) "single element" mode of SG level control, (b) slight positive temperature coefficient, and (c) lack of operator experience.

Reactor: Calvert Cliffs 2
NSIC No.: 120769
Event Date: 12/08/76
Letter Date: 12/27/76

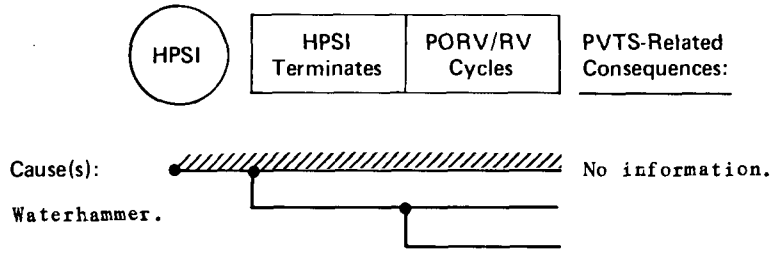
Severity Category: 3
Operating Mode: 1
Cause Category: Power decreased which FW did not match.



Summary of Events:

1. Mode 1, 19% power. Power decreased due to xenon buildup, but dilution of boron was not possible due to positive moderator coefficient.
2. T_{avg} of RCS decreased to below lo-lo limit. Steam generator level also decreased.
3. Turbine load was decreased and T_{avg} increased to within limit.
4. Feedwater overfeeding caused T_{avg} to drop further to 513°F, returning to 515°F within 10 min after turbine load was decreased.
5. A second heavy cycle of overfeeding caused the T_{avg} to again decrease from 516°F.
6. Operator manually tripped the reactor.

Reactor:	Zion 1		
NSIC No.:	120818	Severity Category:	1
Event Date:	09/25/76	Operating Mode:	3
Letter Date:	12/13/76	Cause Category:	Waterhammer.

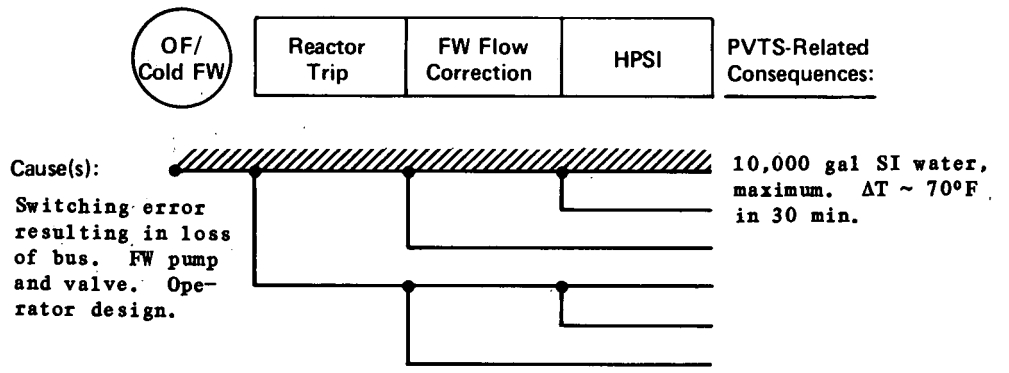


Summary of Events:

1. Reactor was shut down for repairing hanger on ID line.
2. 1801 h, feedwater was introduced into line 1D. A waterhammer resulted, causing high Δp between SGs.
3. SI resulted. It was reset in 2 min.

Reactor: Zion 2
NSIC No.: 120819
Event Date: 09/19/76
Letter Date: 12/08/76

Severity Category: 4
Operating Mode: Mode 1/30%.
Cause Category: Operator error. FW pump and valves.

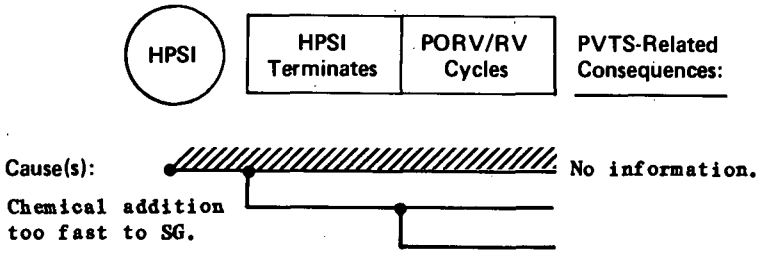


Summary of Events:

1. 0032 h, 30% power and increasing.
2. A switching error occurred while taking bus 211 off equalizing charge. Bus 211 de-energized.
3. Two RCS pumps tripped, causing reactor/turbine trip.
4. Turbine-driven FW pump did not trip; bypass valve did not close; overfeeding resulted; Δp between SG resulted in HPSI.
5. Due to loss of bus 211, HPSI was allowed to continue for 30 min until power was restored and safety conditions verified.

Note: It could be estimated that up to 10,000 gal of SI water was injected; ΔT ~ 70°F in 30 min. This case is a borderline between Categories 3 and 4.

Reactor:	Trojan		
NSIC No.:	122385	Severity Category:	1
Event Date:	10/28/76	Operating Mode:	3
Letter Date:	02/02/77	Cause Category:	Chemical addition to SG too fast.

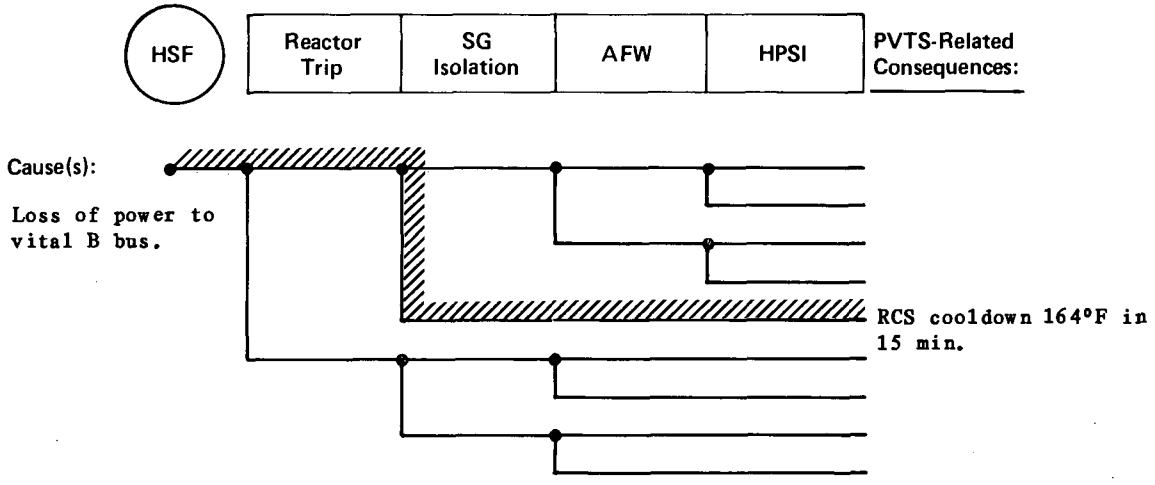


Summary of Events:

1. Reactor in Mode 3, T_{avg} ~ 538°F, p ~ 2235 psig. One RCS pump was in operation.
2. Chemical was being added too fast to #4 SG, causing Δp between SGs because only one pump was working.
3. SI went on adding 100 gal of water at 12% boric acid.

Reactor: Crystal River 3
NSIC No.: 123150
Event Date: 03/02/77
Letter Date: 03/10/77

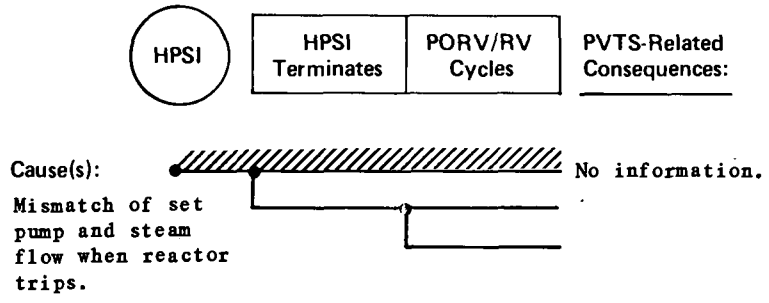
Severity Category: 4
Operating Mode: 1
Cause Category: Loss of power to vital B bus.



Summary of Events:

1. Mode 1, 40% power.
2. Loss of vital ac power bus B (due to inverter failure) caused loss of inverter B vital bus.
3. ICS caused reactor trip, turbine trip, and atmospheric steam dump valve to open 50%.
4. MFW pumps tripped due to loss of vacuum. AFW pumps fed SGs.
5. RCS cooldown rate was 164°F in 15 min.

Reactor:	Zion 2		
NSIC No.:	123778	Severity Category:	1
Event Date:	12/21/76	Operating Mode:	1/75%
Letter Date:	03/11/77	Cause Category:	Control.

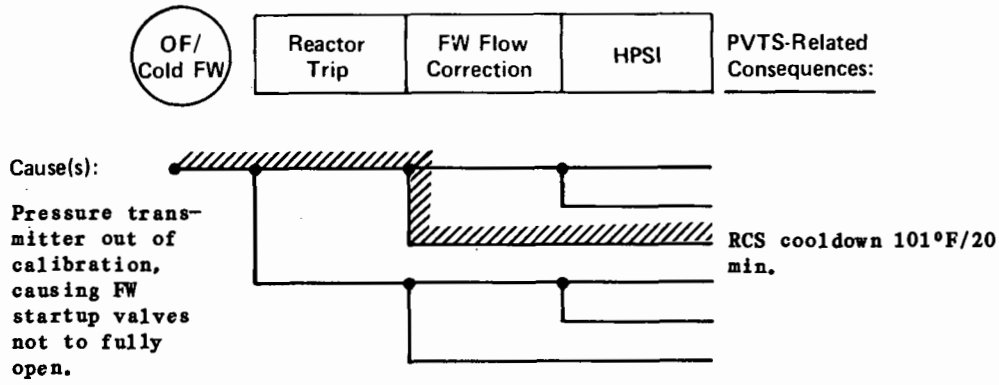


Summary of Events:

1. Mode 1, 75% power. Monthly test was performed on the BIT isolation valves. The BIT was stroked but failed to close; outlet valve also leaked. Hence borated water entered RCS. T_{avg} decreased rapidly to 540°F due to reactor power decrease.
2. 0002 h, the reactor was tripped manually. Setpoint of steam flow dropped automatically to 40%, causing normal steam flow to be high. This in conjunction with low T_{avg} caused SI.
3. SI was reset in 90 s.

Reactor: Crystal River 3
NSIC No.: 124920
Event Date: 04/16/77
Letter Date: 04/21/77

Severity Category: 4
Operating Mode: 3
Cause Category: Control of FW startup valves.

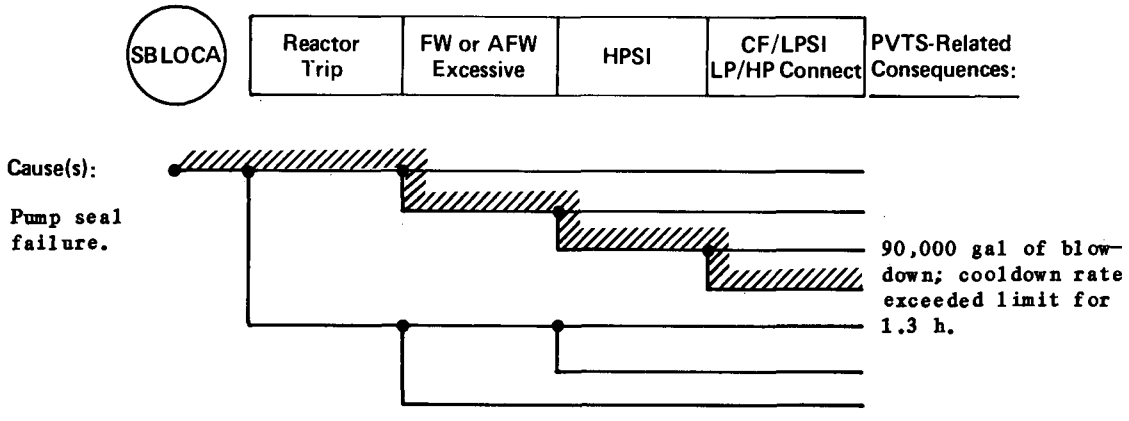


Summary of Events:

1. Mode 3, power 0%, shutdown test from outside control room.
2. Pressure transmitter was out of calibration, causing startup FW valves (FWV-39 and 40) not to close fully. FW pump discharges could not be reduced.
3. SGs were overfilled and drained; atmospheric steam dump valves were opened.
4. RCS cooldown was at 101°F/20 min.
5. Procedures were revised to trip the feedwater pump when appropriate.

Reactor: Indian Point 2
NSIC No.: 126585
Event Date: 07/02/77
Letter Date: 07/22/77

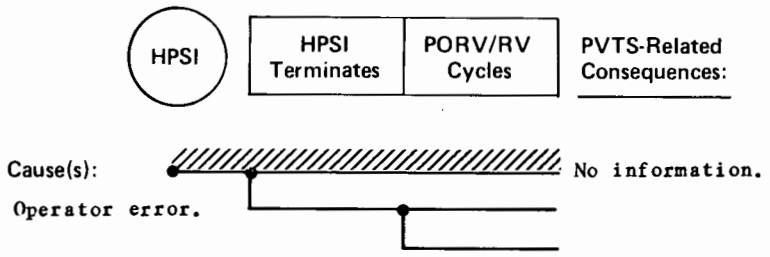
Severity Category: 4
Operating Mode: 2
Cause Category: RCS pump seal failure.



Summary of Events:

1. Mode 2, 2% power, being started up.
2. RCS pump 23 indicated seal failure; pressurizer level decreased.
3. A second charging pump was actuated to compensate for the level decrease.
4. RCS pump 23 and reactor were tripped.
5. Cooldown limits were exceeded by 5°F for 1.3 h (e.g., 140°F/80 min) in expeditious transfer to mode 3.
6. Total leakage to containment 90,000 gal; maximum leak rate was 75 gpm.

Reactor:	Beaver Valley 1		
NSIC No.:	134324	Severity Category:	1
Event Date:	01/02/78	Operating Mode:	1
Letter Date:	02/03/78	Cause Category:	Operator error.

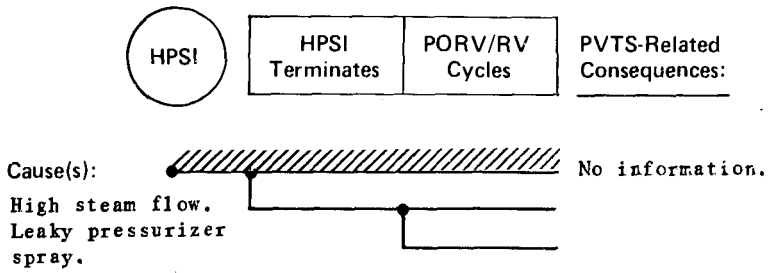


Summary of Events:

1. 23:45 h, Mode 1. Output frequency of No. 4 inverter was oscillating. This vital bus was transferred to auxiliary power supply. But the transfer was done out of phase and the supply breaker trip. No. 4 vital bus was therefore deenergized.
2. 23:46 h, No. 4 vital bus was transferred back to No. 4 inverter. This caused reactor trip.
3. False low-low T_{avg} and false high steam flow caused SI.
4. Operator reset HPSI.

Reactor: Beaver Valley 1
NSIC No.: 137296
Event Date: 01/24/78
Letter Date: 03/28/78

Severity Category: 1
Operating Mode: 3
Cause Category: High steam flow. Cold AFW. Leaky pressurizer spray. Operator error (?).

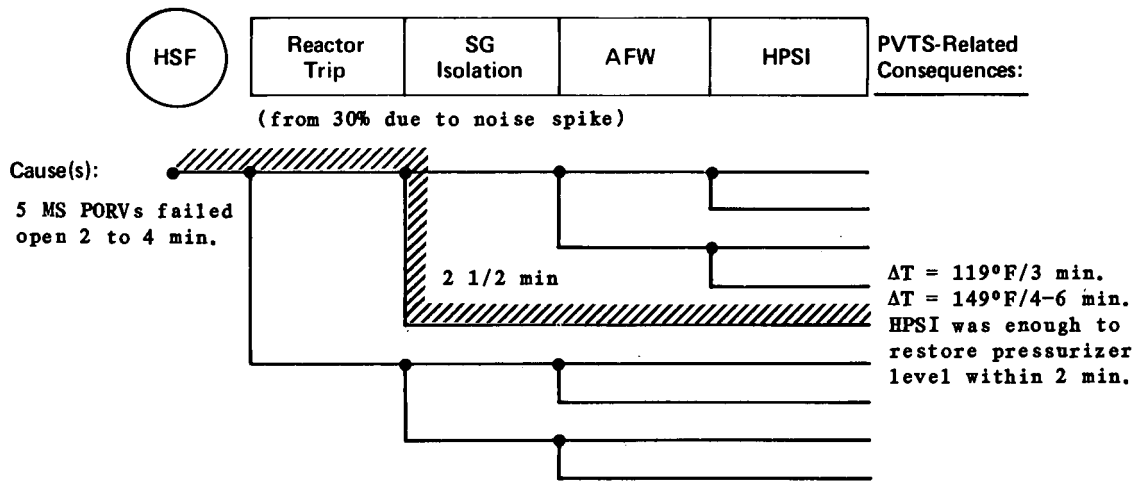


Summary of Events:

1. 0348 h, Mode 1. Reactor trip due to turbine trip. AFW started on low-low SG levels.
2. Operator did not isolate steam to the MSRs immediately after reactor trip. Cold AFW and a leaky pressurizer spray valve also caused RCS temperature to drop.
3. SI resulted.

(Why do we expect the operator to isolate steam to the MSRs right after reactor trip?)

Reactor:	Three Mile Island 2		
NSIC No.:	137918/137919/139931	Severity Category:	4
Event Date:	04/23/78	Operating Mode:	Mode 1/30%
Letter Date:	05/08/78 (report date)	Cause Category:	Main steam relief valves.

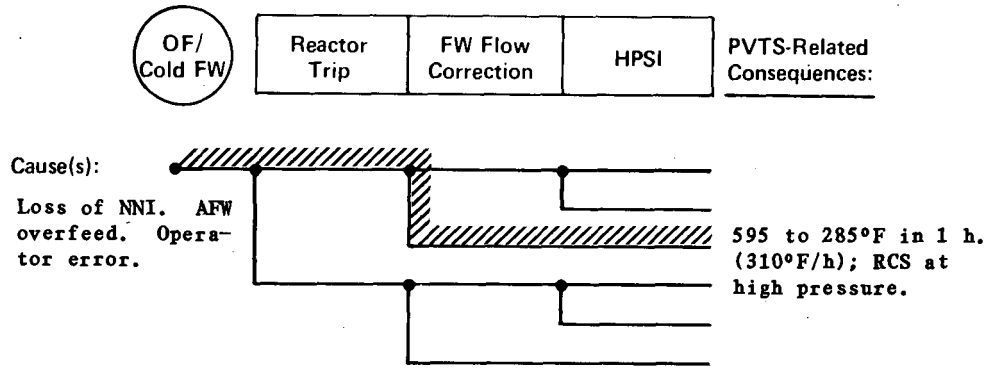


Summary of Events:

1. 1651 h, Mode 1, 30% power, operating on 3 RCS pumps (RCP-2A not working).
2. Noise spiked in N18 power range detector resulting in reactor trip and turbine trip.
3. SI actuated at 1652 h.
4. Very rapid pressure increase in B SG; rather rapid pressure increase in A SG.
5. 4 relief valves on B SG and 1 on A SG lifted. They did not reseat until 2 to 4 min. When all valves were reseated, the SG pressure was between 550 and 600 psig.
6. Operator cut back on FW, shut down the letdown isolation valve, started a second makeup pump, opened high-pressure injection valves on the side of the operating makeup pumps. Feed pump speed was in manual and was not run back until 1 min 20 s.
7. RCS $\Delta T/\Delta t$ = 119°F/3 min.
8. FW isolated at 2.5 min.; some relief valves on B SG reseated. Turbine bypass valve closed. Total $\Delta T/\Delta t$ = 134°F.
9. 1653 h, RCS pressure recovered thanks to SI.

Reactor: Rancho Seco
NSIC No.: 138830
Event Date: 03/20/78
Letter Date: 03/31/78

Severity Category: 4
Operating Mode: Mode 1/70%
Cause Category: Control, operator error.

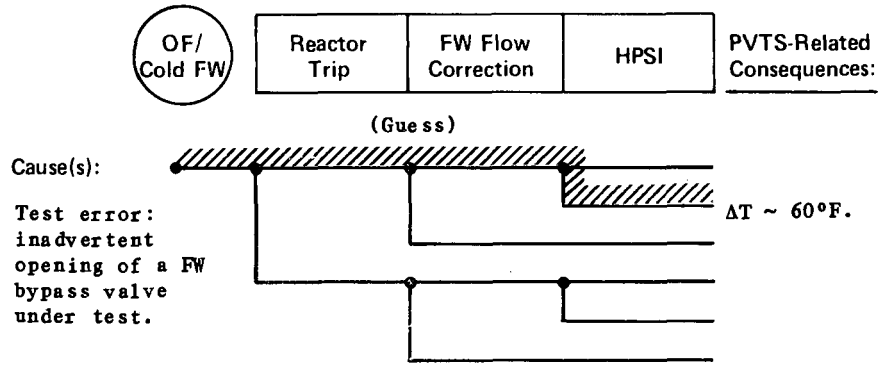


Summary of Events:

1. Mode 1, 70% power, T_{avg} = 582°F.
2. At about 0425 h, while changing a light bulb on the panel, operator dropped the bulb onto the light assembly. Short circuit.
3. Protection system for dc power got actuated, cutting off ac power to all NNI-Y dc power supplies. Two-thirds of NNI signals were lost (p, T, level, flow, etc.). Erroneous signals were provided to ICS.
4. Due to faulty signals, ICS reduced FW to zero, withdrew reactor control rods to 103%.
5. RCS pressure increased; reactor trip at 0425 h; pressurizer code safety valve lifted (2500 psig); RCS pressure decayed to 2000 psi in 9 min.
6. AFW started on loss of FW but AFW valve remained closed due to erroneous signals from SG levels.
7. Both SG went dry (even SG B indicated full). AFW valve opened thanks to level of SG A. Rapid cooldown started. Operator may have also increased the flow of main FW.
8. HPSI on when p < 1600 psi.
9. Both SG were filled. Operator did not believe T_{avg} indicator. He maintained RCS pressure with HPSI, using pressurizer level and pressure indications.
10. At 0535 h (70 min), T_{avg} = 285°F.
11. When power was restored to NNI-Y, operator used pressurizer spray to reduce RCS pressure, drained SG, kept RCS running to increase RCS temperature.

Reactor: St. Lucie 1
NSIC No.: 139563
Event Date: 05/30/78
Letter Date: 06/28/78

Severity Category: 3
Operating Mode: Mode 1/3%
Cause Category: Operator/test error. Valve data inadequate.



Summary of Events:

1. Mode 1, low power (3%); feedwater control system under test.
2. A 15% FW bypass control valve was inadvertently opened fully, causing excessive flow in 1A SG.
3. RCS T_{avg} dropped to 514°F, which was 1F below action by specification.
4. Reactor was tripped manually.

Reactor:	Cook 2		
NSIC No.:	142027	Severity Category:	0
Event Date:	12/06/78	Operating Mode:	1
Letter Date:	12/20/78	Cause Category:	Feedwater heater and control.

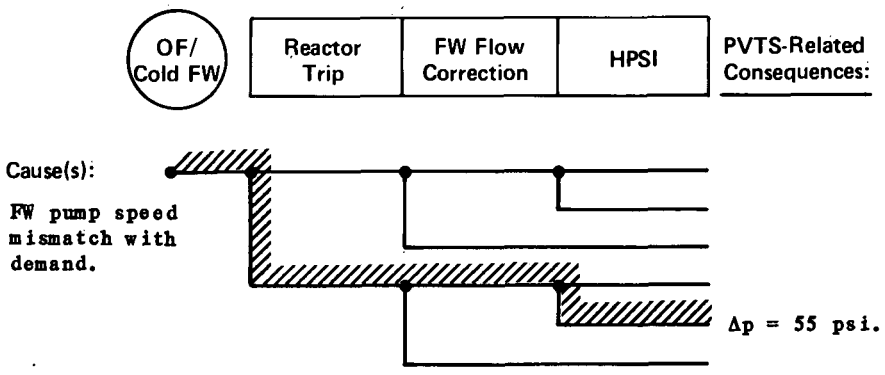
(No event tree)

Summary of Events:

1. Mode 1, 100% power.
2. Trouble at 5A high-pressure heater level control. FW control became unstable.
3. 10% power reduction was performed. While decreasing turbine load, RCS temperature exceeded 576°F for 20 min.

Reactor: Salem 1
NSIC No.: 142387
Event Date: 01/30/78
Letter Date: 02/24/78

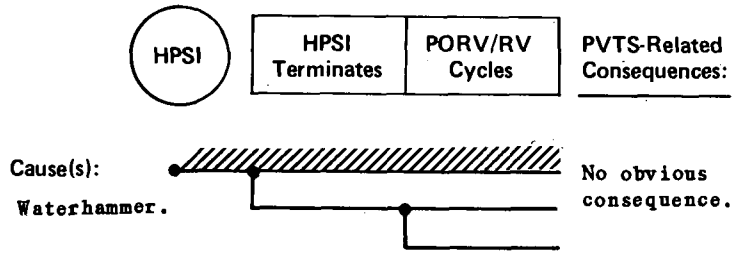
Severity Category: 0
Operating Mode: 1
Cause Category: Manual to auto transfer caused mismatch to pump speed with demand.



Summary of Events:

1. Mode 1, 22% power.
2. 0045 h, No. 11 SG feed pump was transferred from manual to auto, but operator failed to match pump speed to master demand.
3. SG level fluctuated; RCS pressure dropped to 2165 from 2220 psia, violating tech spec.
4. Recovery within 4 min.

Reactor:	Zion 1		
NSIC No.:	142480	Severity Category:	1
Event Date:	09/14/78	Operating Mode:	3
Letter Date:	12/11/78	Cause Category:	Waterhammer.



Summary of Events:

1. 0817 h, Mode 1, 63% power. Turbine trip causing reactor trip.
2. Three AFW pumps started, feeding 150 gpm to each SG.
3. 0829 h, waterhammer happened; the origin was not known to be at the feedwater rings or in the turbine building.
4. SI was actuated due to high Δp between SGs.

Reactor:	Cook 2		
NSIC No.:	142627	Severity Category:	0
Event Date:	12/06/78	Operating Mode:	1
Letter Date:	12/20/78	Cause Category:	FW level control.

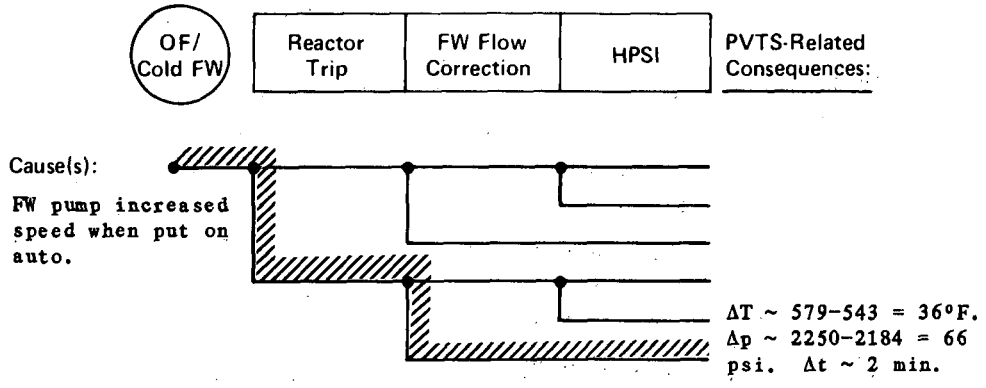
(No event tree)

Summary of Events:

1. Mode 1, 100% power.
2. Trouble at 5-A high-pressure heater level control. FW control became unstable.
3. 10% power reduction was performed. While decreasing turbine load, RCS temperature exceeded 576°F for 20 min.

Reactor: Salem 1
NSIC No.: 143496
Event Date: 08/04/78
Letter Date: 08/29/78

Severity Category: 2
Operating Mode: Mode 1/14%
Cause Category: FW pump increased speed when put on automatic.

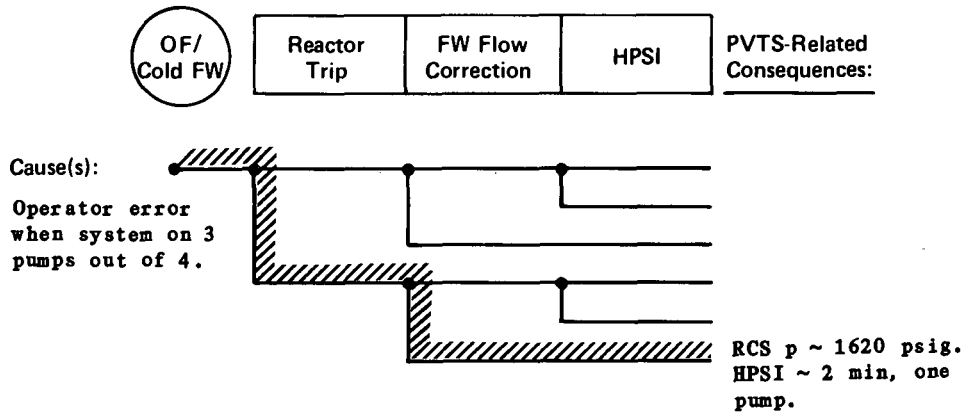


Summary of Events:

1. 0205 h, Mode 1, 14% power, operator put No. 12 SG feed pump into automatic.
2. Pump speed increased rapidly, decreasing RCS T_{avg} to 543°F.
3. ΔT ~ 579°F − 543°F = 36°F. This is the fourth occurrence of this type. (See 77-36/03L, 77-57/03L, 77-60/03L.)

Reactor: Davis-Besse 1
NSIC No.: 145364
Event Date: 04/29/78
Letter Date: 07/28/78

Severity Category: 2
Operating Mode: Mode 1/20%
Cause Category: Operator error. FW valve opened too wide.

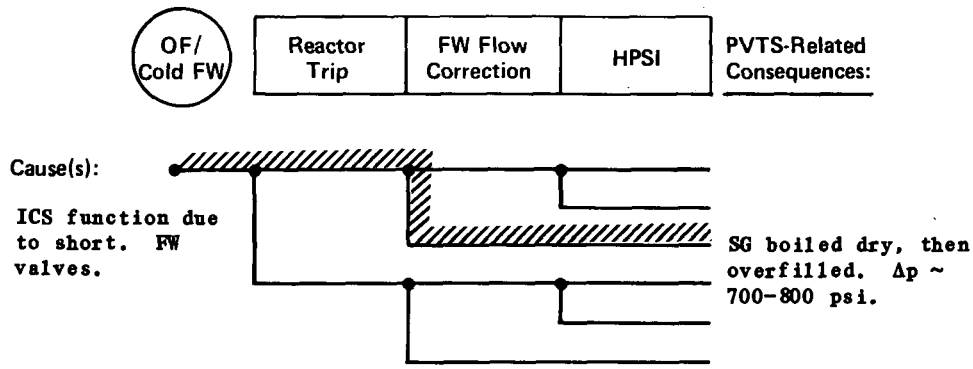


Summary of Events:

1. Unit in process of shutting down for maintenance; 20% power.
2. Operator transferred the sensitive FW control from flow control to level control for SG #1, causing startup valve to open too wide.
3. RCS pressure dropped to 1620 psig; HPSI was started, but only one pump worked (the other was taken out of service for maintenance. HPSI was shut off in 9 min; pumped only 2 min.

Reactor: Oconee 1
NSIC No.: 146533
Event Date: 12/14/78
Letter Date: 01/15/79

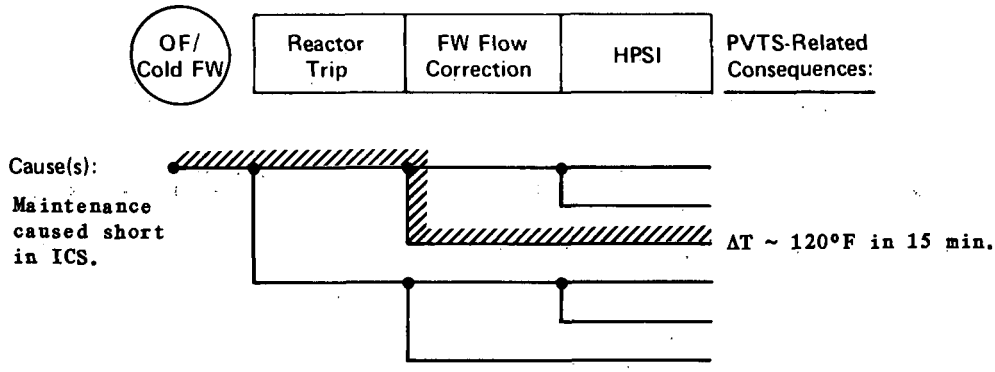
Severity Category: 3
Operating Mode: 1
Cause Category: ICS function upon short in a line; FW valves.



Summary of Events:

1. Mode 1, 98% power; T_{avg} statalarm was erratic.
2. During an investigation, a short occurred on the power cord feeding ICS T_{avg} recorder. An erroneous T_{avg} drop of 13°F resulted.
3. ICS withdrew control rod to maintain T_{avg}. Reactor trip on high p, high T_{avg} at 16:36 h.
4. Both main FW pumps tripped on high discharge pressure. AFW pumps started then stopped when main FW pumps reset and restarted.
5. 16:38 h, SG B level dropped from 110 to 6 in. SG A level dropped from 110 to 0 in.
6. AFW pumps started to feed SG B.
7. 16:41 h, level in SG A returned to normal, SG B to 35 in., then to 0 again. This was possibly due to malfunction of feedwater valves.
8. 16:49 h, SG B was refilled up to AFW header. RCS dropped to 1500 psi.
9. HPSI occurred.
10. Postincident analysis indicated this was due to FW valves inability to maintain correct FW flow.

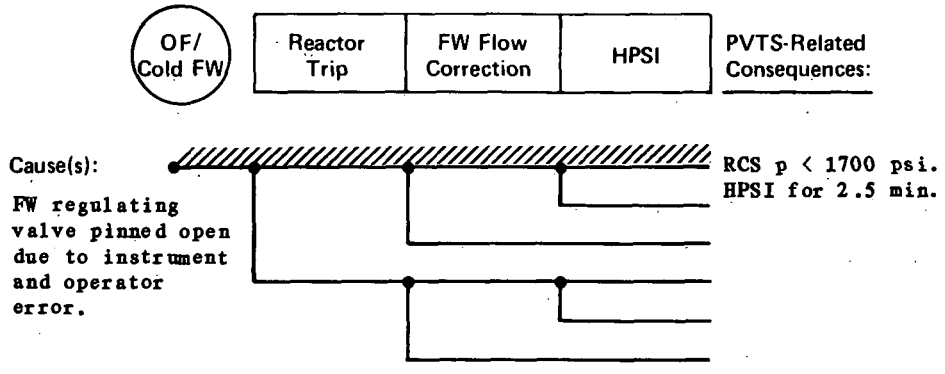
Reactor:	Rancho Seco		
NSIC No.:	147267	Severity Category:	4
Event Date:	01/05/79	Operating Mode:	1
Letter Date:	01/19,26/79	Cause Category:	Maintenance error.



Summary of Events:

1. Mode 1, 100% power, T_{avg} = 582°F.
2. A technician performing a modification in the ICS allowed a wire connected to a terminal in the ICS panel to swing free; ±25 V CD power supply was shorted.
3. ICS AC supply breakers were tripped automatically as designed.
4. Loss of ICS logic ran feedwater valves back to 50% position; RCS pressure increased to 2355 psig.
5. Reactor trip at 0.16 s. FW now is too large for the power; RCS pressure dropped to 1900 psig in 70 s.
6. 2 min 53 s, RCS pressure dropped to 1600 psig. HPSI was actuated.
7. 5 min, FW continued, augmented further by AFW. Total flow 2.2 x 10^6 lb/h.
8. 7 min, both main FW pumps were tripped by operator; only AFW remained.
9. RCS temperature dropped to 430°F. (ΔT = 152°F.)

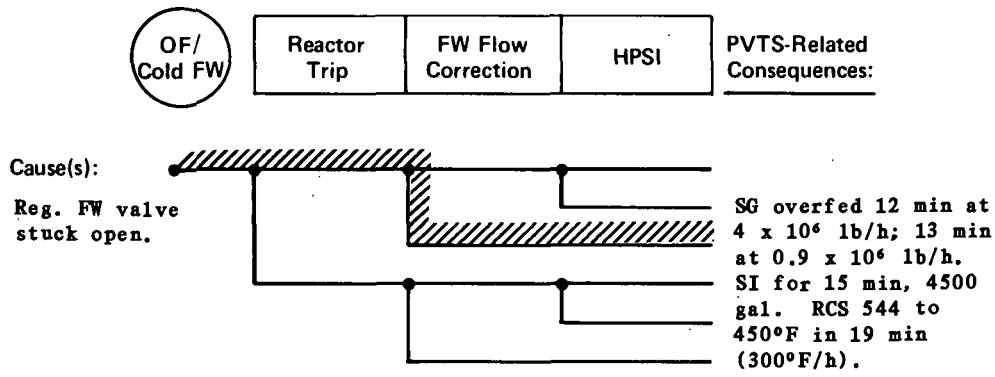
Reactor:	Three Mile Island 2		
NSIC No.:	148080	Severity Category:	2
Event Date:	12/02/78	Operating Mode:	Mode 1/22%
Letter Date:	02/28/79	Cause Category:	Operator error.



Summary of Events:

1. Mode 1, 22% power, T_{avg} = 584°F (higher than normal due to FW heaters being placed in service).
2. Operator switched from startup to regulating FW valves.
3. Startup valve at 80% open; regulating valve fully open but Δp indicated 0. Operator increased FW pump speed.
4. Rapid RCS cooldown. Reactor trip on low RCS pressure; HPSI went on (2.5 min).
5. Operator stopped letdown flow, started a second makeup pump, reduced FW pump speed, and closed main FW block valves.

Reactor:	Fort Calhoun 1		
NSIC No.:	148112	Severity Category:	4
Event Date:	04/17/74	Operating Mode:	Mode 1/90%
Letter Date:	06/26/75	Cause Category:	FW valve stuck open.

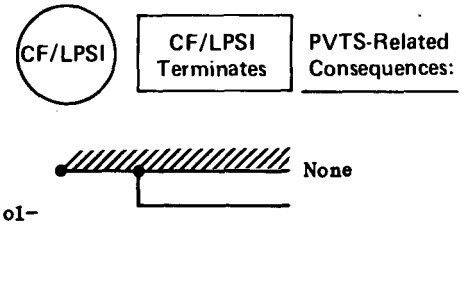


Summary of Events:

1. At 18:50 h, an FW regulating control valve failed open (later found due to a fractured air line to the upper side of valve actuator); level in same SG was high (RA SG).
2. Operator cannot close the valve from control room. Local operator closed it manually, but only partially.
3. RC-2A feedflow dropped to zero (believed due to closure of check valve and $\Delta p \sim 200$ psi). No pressure drop across RC-2B, but feedflow fluctuated $\pm 2 \times 10^6$ lb/h.
4. T_c increased to 544 from 534°F; power dropped to 75%. Operator opened turbine control valve to control T_c (18:58 h).
5. Seeing RC-2A level decreased, operator opened FW valve, but this was not enough; reactor trip on low SG level.
6. Pressure in pressurizer dropped; FW to RC-2B reduced automatically by valve.
7. FW to RC-2A continued. Operator couldn't close isolation valve from control room; on-site operator closed it part way.
8. 19:17 h, RCS temperature ~ 450°F, pressure ~ 1550 psi; CIAS actuated; FW flow reduced to 0.9×10^6 lb/h 12 min after reactor trip.
9. 25 min after reactor trip, operator shut off FW pump.
10. 19:25 h (35 min), SI was stopped. It can be estimated that SI water was 4500 gal maximum.
11. Inspection indicated leakage in RC-2A valve gaskets and seals of RCS pumps. Component cooling water was lost for 45 min.

Reactor: Millstone 2
NSIC No.: 148766
Event Date: 03/14/79
Letter Date: 04/12/79

Severity Category: 0
Operating Mode: 5
Cause Category: RCS flooding actuation.



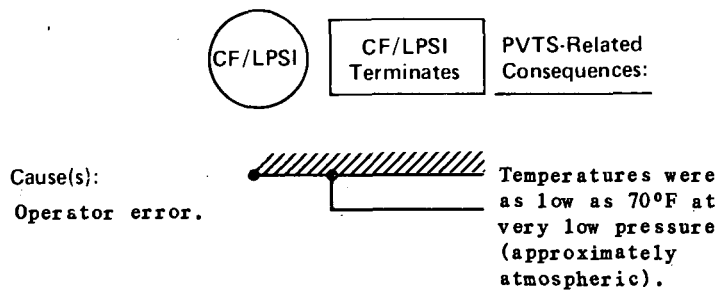
Cause(s):

Flooding RCS following list of LPSI.

Summary of Events:

1. Mode 5. SG primary side manways open. RCS on shutdown cooling with RCS level at hot leg center line.
2. Due to LPSI suction air binding, shutdown cooling was lost. Partial flow was accomplished by flow drip of the RCS from RWST. RCS heating to 208°F, resulting in entry to Mode 4.
3. Compliance with all Mode 4 tech specs was verified, and containment purge valves were shut.

Reactor:	Millstone 2		
NSIC No.:	148767	Severity Category:	4
Event Date:	03/14/79	Operating Mode:	5
Letter Date:	04/13/79	Cause Category:	Operator error.

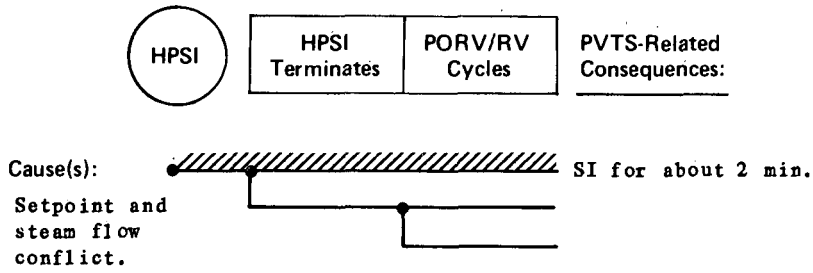


Summary of Events:

1. The reactor was in cold shutdown (Mode 5), at atmospheric pressure. An LPSI pump tripped.
2. Operator added water to the RCS from the RWST, making the RCS temperature swing from 200 to 70°F and then to 176°F. This exceeded the tech spec cooldown limit.

Reactor: Zion 1
NSIC No.: 149716
Event Date: 03/02/79
Letter Date: 06/01/79

Severity Category: 1
Operating Mode: 1/100%
Cause Category: Setpoint and steam flow conflicts.

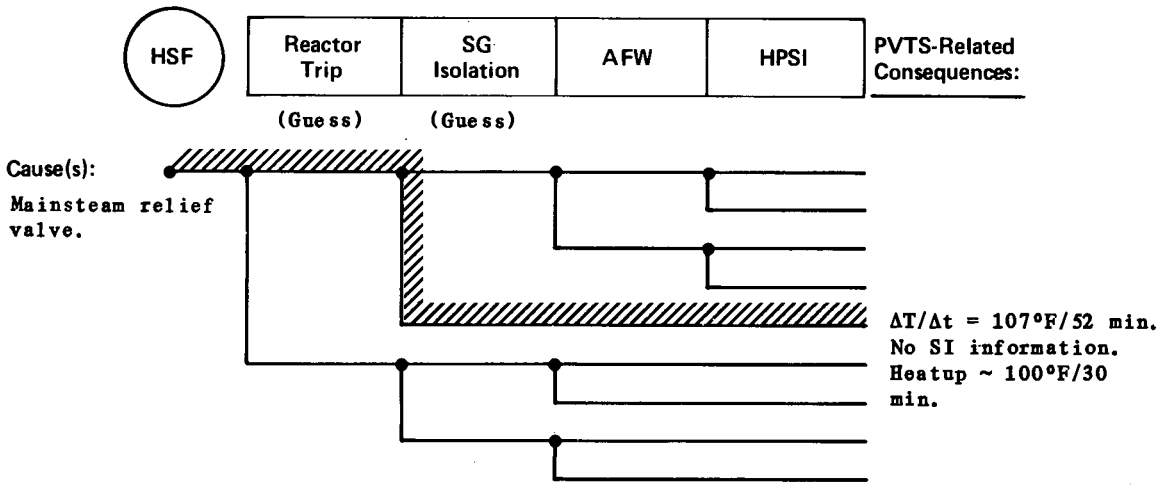


Summary of Events:

1. 0820 h, Mode 1, full power.
2. Low steam pressure bistables tripped for IM surveillance. 1B FW pump dropped off due to circuit malfunction.
3. 0821 h, 1D SG indicated low level, high FW flow. These signals caused reactor trip.
4. Steam flow setpoint automatically dropped to 40%, but actual flow remained high for a while. This high steam flow in the face of low setpoint caused HPSI.
5. 0823 h, SI stopped by operator.
6. 0853 h, 5 waterhammers due to AFW inflow. Some containment isolation valves did not close.

Reactor: Arkansas 2
NSIC No.: 149746/148074/146587
Event Date: 12/27/78
Letter Date: 01/25/79

Severity Category: 4
Operating Mode: Mode 1
Cause Category: Stuck open steam relief valve.



Summary of Events:

1. Mode 1 (no information on power).
2. During a main turbine roll, the main steam dump valve failed to open. Main steam relief valve lifted, but failed to reseat.
3. RCS ΔT/Δt = 107°F/52 min.
4. After more than 1 h, the relief valve was reseated and RCS T_{avg} was reported to 545°F within 30 min.
5. No information on SI.

Reactor:	Arkansas 2		
NSIC No.:	150394	Severity Category:	0
Event Date:	06/03/79	Operating Mode:	3
Letter Date:	06/20/79	Cause Category:	

(No event tree)

Summary of Events:

1. Mode 3. EFW control valve from an electric EFW pump to the B SG would not respond to MS isolation valve signal and would not stroke shut. A redundant valve operated as required.
2. It was found out later that the cost and tripper bar was out of adjustment.

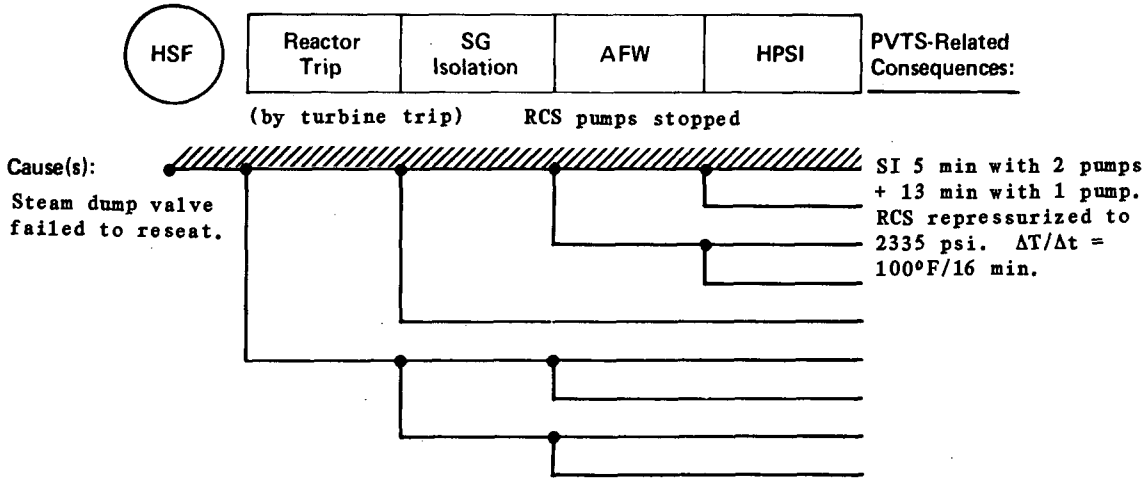
Reactor:	Calvert Cliffs 1		
NSIC No.:	150731	Severity Category:	0
Event Date:	07/06/79	Operating Mode:	5
Letter Date:	07/19/79	Cause Category:	Valve failure.

(No event tree)

Summary of Events:

1. Reactor in Mode 5, RCS at 260 psi.
2. 0920 h, operator tested PORV after modification during outage. RC-402-ERC failed to fully close after lifting. RCS pressure decreased from 260 to 220 psia. The redundant valve operated as required.
3. 15:40 h, the ERV was adjusted, tested, and put back to service.

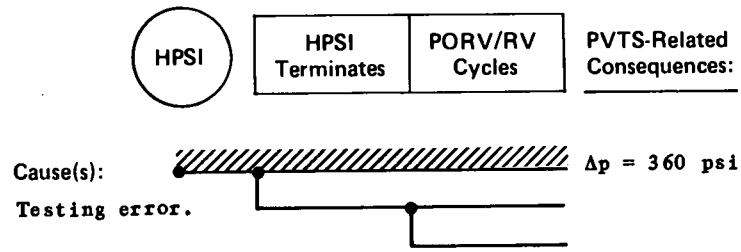
Reactor:	North Anna 1		
NSIC No.:	152751	Severity Category:	4
Event Date:	09/25/79	Operating Mode:	Mode 1/78%
Letter Date:	10/09/79	Cause Category:	Steam dump valve failed to open.



Summary of Events:

1. 0544 h, Mode 1, 78% power. A tube rupture inside the 5B drain cooler caused the cooler dump valve to cycle.
2. 0609 h, turbine trip due to high level in the 5B feedwater heater; reactor tripped.
3. MS dump valve opened to reduce RCS temperature to 547°F but failed to close when that point was reached. Operator later isolated it by closing the MS trip valves.
4. Due to high steam flow, RCS temperature and pressure decreased (later found to be 100°F in 16 min). SI was actuated at 0614 h.
5. RCS pumps were manually tripped by operator. RCS pressure began to increase. One of two SI pumps was secured at 0619 h. At 0600 h, PORV of the pressurizer began to cycle at maintained pressure at 2335 psig until normal feed and bleed was established.
6. Following the above event, continued operation of an SI pump and the routing of letdown coolant and the missing of a flange in the noble gas vent line caused radioactive gas to be released to the auxiliary building.

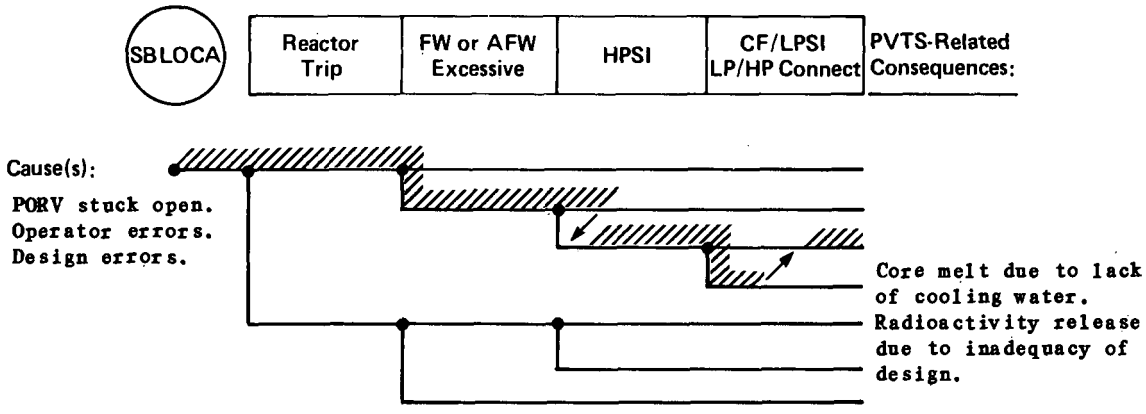
Reactor:	Oconee 3		
NSIC No.:	152983	Severity Category:	1
Event Date:	10/19/79	Operating Mode:	6
Letter Date:	11/16/79	Cause Category:	Testing error.



Summary of Events:

1. 1800 h, Mode 5, 308 psig, 200°F.
2. During test of motor of the HPI cross-connect isolation valve (to RCS loop A), the valve was inadvertently opened.
3. Makeup flow was injected to RCS, increasing the pressure to 360 psig in <1 min.
4. Onsite operator reclosed the valve manually.

Reactor:	Three Mile Island 2		
NSIC No.:	153164	Severity Category:	4
Event Date:	03/28/79	Operating Mode:	Mode 1/99%
Letter Date:	03/28/79	Cause Category:	Valves/design/operator.



Summary of Events:

1. Mode 1, 99% power.
2. A condensate pump trip led to FW pump trips, turbine trip, and reactor trip.
3. RCS pressure caused PORV to open. It was stuck open.
4. AFW was blocked by three improperly closed AFW valves. They were manually opened 8 min after turbine trip.
5. HPSI went on at low RCS pressure. But at 10-12 min, operator throttled HPSI due to erroneous level indications in pressurizer.
6. Core melt and radioactivity leaked to the auxiliary building due to inadequacy of design.

Reactor:	Indian Point 2		
NSIC No.:	153683	Severity Category:	0
Event Date:	11/09/79	Operating Mode:	1
Letter Date:	12/07/79	Cause Category:	Valve failure.

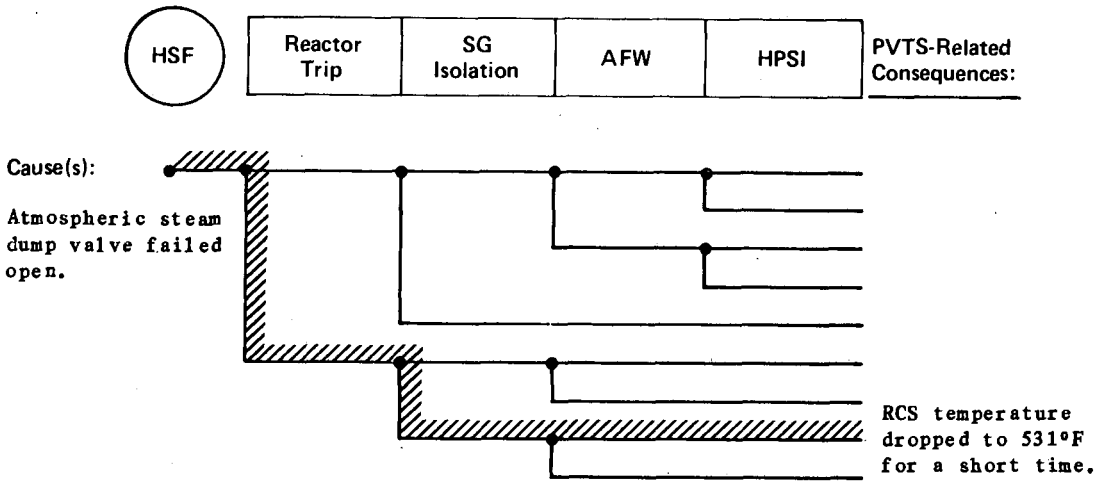
(No event tree)

Summary of Events:

1. Mode 1.
2. Blowdown from No. 24 SG was being terminated, but the isolation valve PCV-1217 would not close.
3. The redundant valve operated as required.

Reactor: Farley 1
NSIC No.: 153713
Event Date: 12/02/79
Letter Date: 01/02/80

Severity Category: 2
Operating Mode: 1/2, 1% power
Cause Category: Atmospheric steam dump valve.

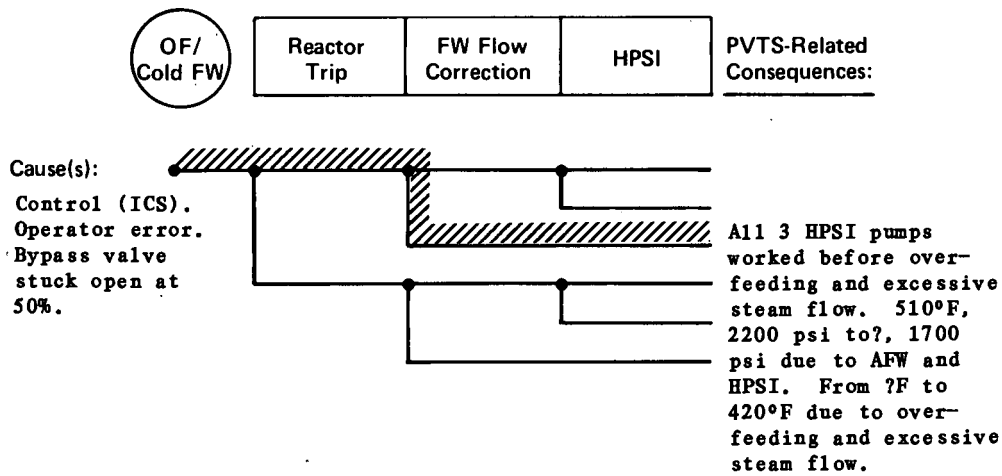


Summary of Events:

1. Mode 1, 1% power, transition to Mode 2 with secondary side steam load being secured.
2. Atmospheric steam dump valve failed to close due to a broken positioning arm.
3. RCS temperature dropped to 531°F (below the lo-lo level of 541°F).
4. Operator immediately isolated the dump valve and RCS temperature rose again to above 541°F.

Reactor: Oconee 3
NSIC No.: 153772
Event Date: 11/10/79
Letter Date: 11/23,30/79

Severity Category: 4
Operating Mode: Mode 1/99%
Cause Category: ICS; operator error; stuck bypass valve.

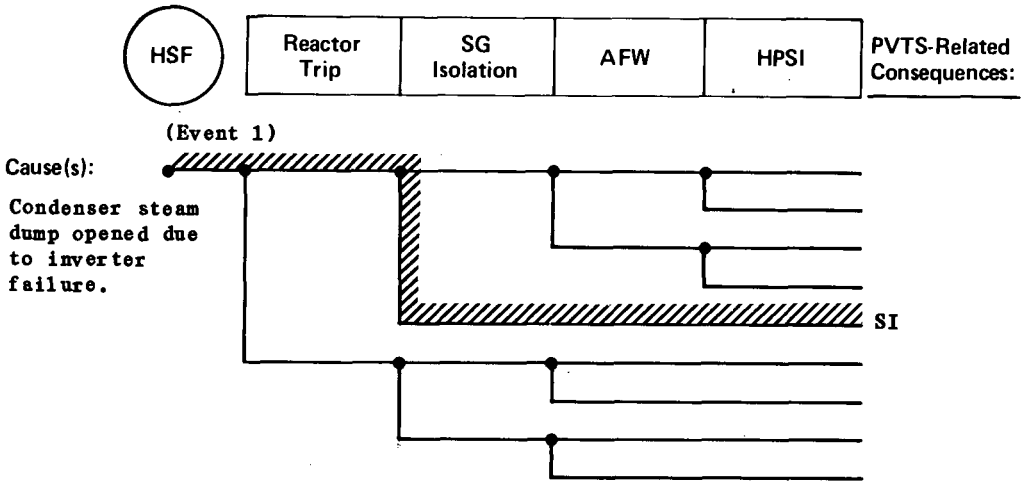


Summary of Events:

1. At 15:16 h, reactor at 99% power, spurious low condenser hotwell level tripped hotwell pumps, condensate booster pumps, resulting in low FW flow and reactor runback at 75%.
2. Reactor trip due to high RCS pressure.
3. 15:16 h + 20 s, power to ICS lost due to blown fuses in KI inverter. Auto transfer to regulated AC supply failed.
4. 15:16 h + 2.5 min, manual transfer to regulated AC supply.
5. Due to ICS power loss, both FW pump tripped, and a number of indicators of primary and secondary conditions were lost. AFW started by steam-driven and motor-driven pumps; RCS wide-range pressure indicator worked.
6. HPSI C was started manually; HPSI A had been operating before transient. HPSI B had been started automatically due to low RCS pump seal flow.
7. When ICS power was restored, RCS pressure was 1700 psi; pressurizer level 20 in. and increasing.
8. 15:16 h + 10 min HWP and CBP were restarted; level in SG B increased abruptly; RCS cooldown from 532 to 420°F.
9. Turbine bypass valve opened at 50%, was closed manually but remained open for the next 20 min.
10. 18:00 h, hot shutdown condition established; normal cooldown followed.

Reactor: Beaver Valley 1, First SI
NSIC No.: 154291
Event Date: 09/20/79
Letter Date: 12/19/79

Severity Category: 2
Operating Mode: ?
Cause Category: Inverter failure.

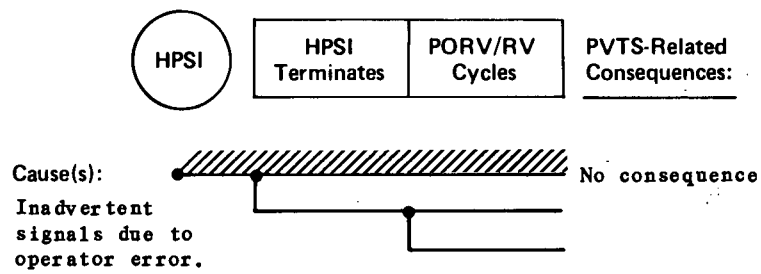


Summary of Events:

<u>Event 1</u> High Steam Flow

1. 0502 h, No. 4 vital bus inverter failed. This bus was automatically transferred to auxiliary power supply.
2. Condenser steam dump valve opened. This led to high steam flow, low steam pressure.
3. Reactor tripped. SI was actuated.
4. Operator verified that SI was inadvertent and reset it. (This was twelfth operational SI, second preoperational SI).

Reactor:	Beaver Valley 1, Second SI		
NSIC No.:	154291	Severity Category:	2
Event Date:	09/20/79	Operating Mode:	3
Letter Date:	12/19/79	Cause Category:	Operator error.



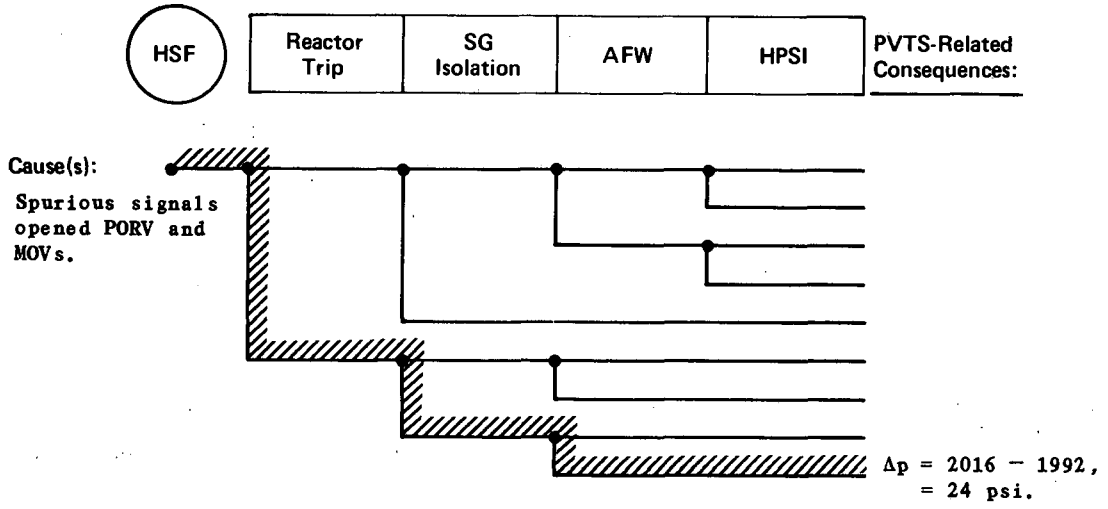
Summary of Events:

<u>Event 2</u> Spurious SI

1. Plant was in Mode 3, RCS temperature ~ 543°F.
2. 1108 h, vital bus #4 inverter was transferred from its auxiliary power to the inverter. But this inverter was not operable (operator error!).
3. High steam flow signal plus low-low temperature signal caused SI.

Reactor: Connecticut Yankee
NSIC No.: 154451
Event Date: 02/04/80
Letter Date: 02/12/80

Severity Category: 0
Operating Mode: 1
Cause Category: Spurious signal.

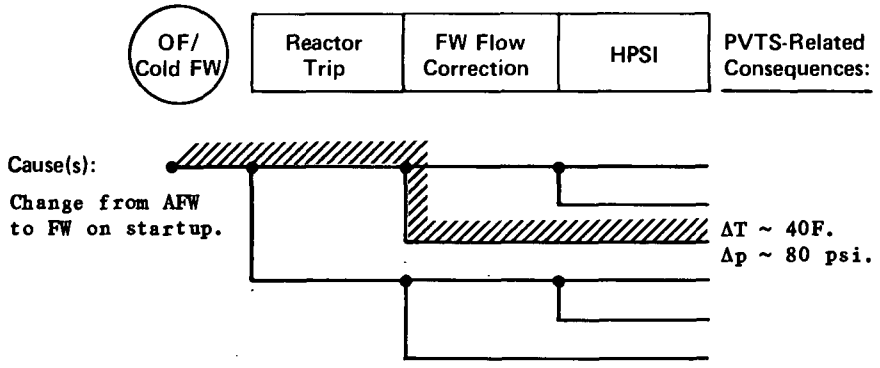


Summary of Events:

1. Reactor at Mode 1, 100% power.
2. Spurious signals opened PORV and MOVs.
3. Operator arrested incident in 2 min. RCS pressure dropped from 2016 to 1992 psig.

Reactor: Cook 2
NSIC No.: 154690
Event Date: 01/19/80
Letter Date: 02/06/80

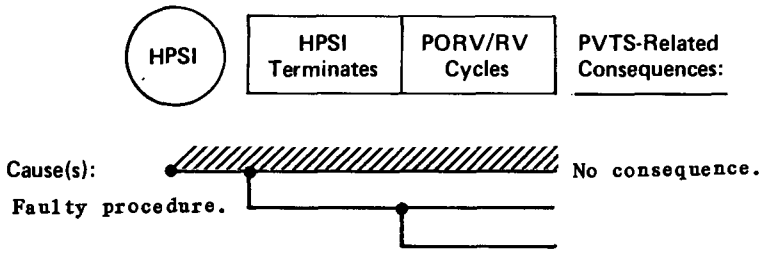
Severity Category: 2
Operating Mode: Mode 1/10%
Cause Category: FW pump control manually upon changeover from AFW.



Summary of Events:

1. Reactor at 10% power and increasing.
2. Change AFW to FW manually; to recover an SG level, too much feedwater was fed inadvertently.
3. RCS T decreased to 539°F (perhaps from 570°F), to 2170 psig (perhaps from 2250 psig).
4. Reactor trip and turbine trip.

Reactor:	Trojan		
NSIC No.:	154747	Severity Category:	1
Event Date:	09/06/79	Operating Mode:	2/1.5%
Letter Date:	12/03/79	Cause Category:	Faulty procedures.

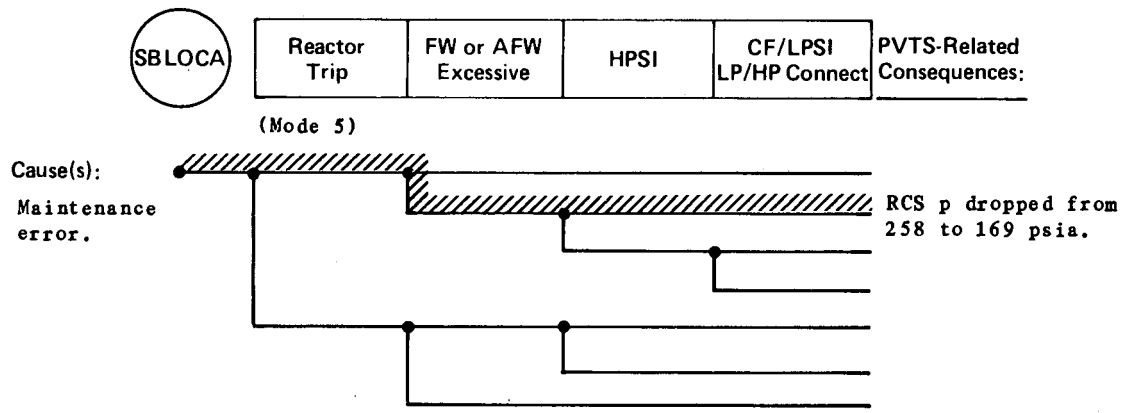


Summary of Events:

1. Reactor being started up (Mode 2), 1.5% power.
2. While troubleshooting a turbine generator control circuit, the backup acceleration amplifier card was pulled out of the rack due to faulty G.E. procedure.
3. Main steam control valve went wide open.
4. High steam flow, low steam pressure caused SI.

Reactor: Calvert Cliffs 1
NSIC No.: 154890
Event Date: 01/29/80
Letter Date: 02/12/80

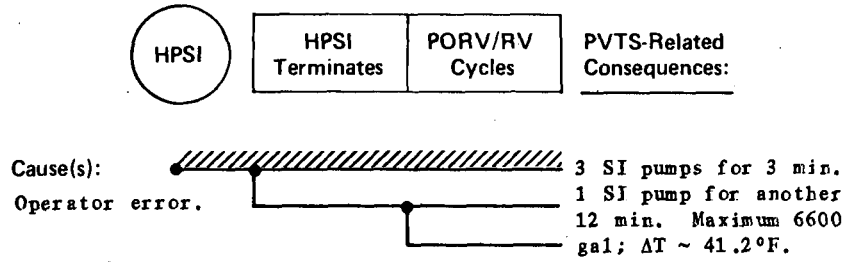
Severity Category: 1
Operating Mode: 5
Cause Category: Maintenance error.



Summary of Events:

1. 11:45 h, Mode 5, 258 psia. A technician conducted maintenance of pressure transmitter PT-103-1 when the bonnet was unscrewed. A spurious high-pressure signal resulted.
2. The RCS was at that time being used to flush the core; PORV opened.
3. Operator started all charging pumps, turned on all pressurizer heaters and stopped the two operating RCS pumps.
4. RCS pressure dropped to 169 psia at 11:48 h when PORV was shut by going to override.
5. Throughout the incident, the shutdown cooling of 6000 gpm remained in operation.

Reactor:	North Anna 1		
NSIC No.:	155359	Severity Category:	3
Event Date:	02/23/80	Operating Mode:	1
Letter Date:	03/07/80	Cause Category:	Operator error.

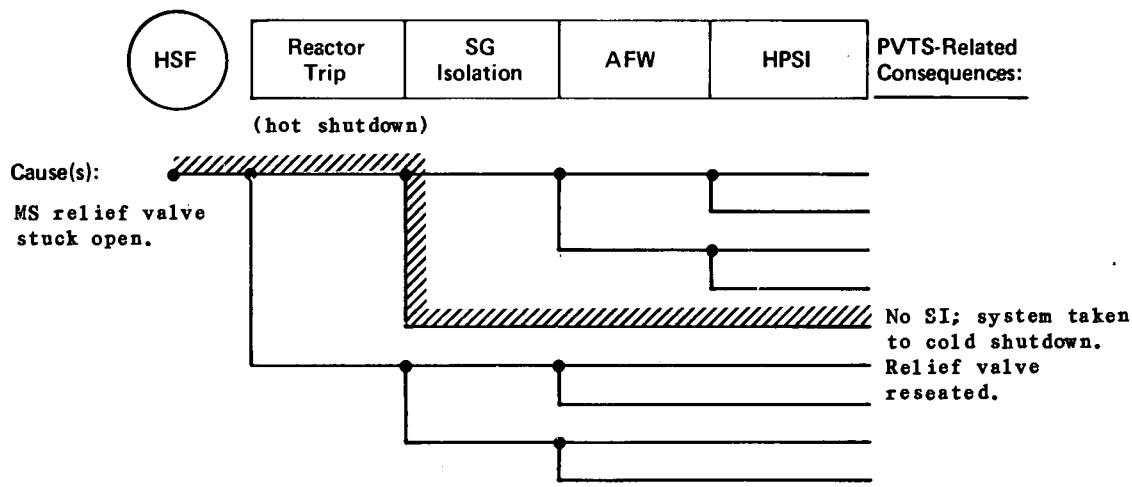


Summary of Events:

1. 19:12 h, Mode 1. Metal noise was detected in steam generators.
2. Operator opened nonreturn valve NRC-MS101C to allow Westinghouse to collect noise data.
3. As the same valves in other SGs were closed, Δp existed between SGs, and this caused SI as designed.
4. 19:15 h, SI signals were reset, 2 SI pumps were secured. The remaining SI pump continued for 15 min. ΔT in cold leg B was 41.2°F. Pressurizer level increased from 16 to 59%.
5. Our analysis indicated that up to 6600 gal of SI water was injected.

Reactor: Turkey Point 4
NSIC No.: 155451
Event Date: 02/15/80
Letter Date: 03/17/80

Severity Category: 1
Operating Mode: Mode 4
Cause Category: MS relief valve (1) stuck open for a short time.

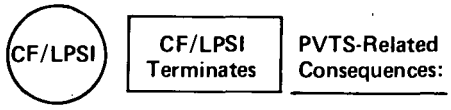


Summary of Events:

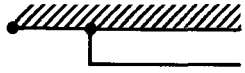
1. Reactor at hot shutdown. The 4C steam generator feedwater control valve was being tested.
2. A rapid change in steam generator level occurred.
3. Main steam relief valve RV-1410 lifted and did not fully reseat due to binding between the lifting gear and the release nut.
4. Steps were taken to bring the reactor to cold shutdown. During this period, maintenance personnel reseated the valve.

Reactor: Surry 1
NSIC No.: 156951
Event Date: 04/26/80
Letter Date: 04/30/80

Severity Category: 0
Operating Mode: 5
Cause Category: Operator error and faulty relay.



Cause(s):
SI locked in due to operator error and faulty relay.



4 SI when RCS at atmospheric pressure.

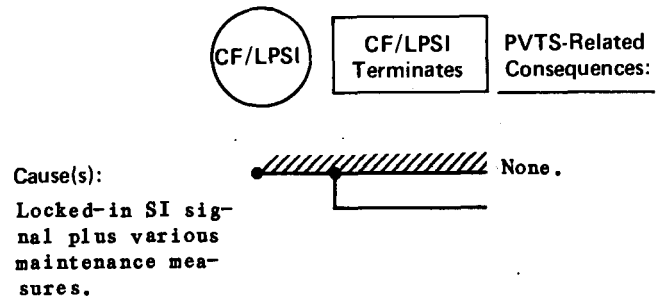
Summary of Events:

1. Cold shutdown with water in RCS partially drained. Containment not isolated.
2. Valve was cycled.
3. Gravity flow from RWST happened.
4. On Apr. 30, 1980, at 0218 h, electrician worked on relay racks; spurious SI in train A happened and could not be stopped.
5. 0511 h, water from RWST was gravity-flowing into RCS through BIT because of the opening of a MOV due to "locked-in" safety injection signal. Stopped by cutting off power.
6. 0920 h, control room operator restored power to the MOV. Rapid flooding by accumulators.

   B accumulator pressure 600 psi → 120 psi.
   A accumulator pressure 600 psi → 380 psi.
   RCS pressure increased atmosphere → 80 psig.

7. 1007 h, efforts to close MOV on A accumulator caused valve to partially open and SI again into RCS.

Reactor:	Surry 1		
NSIC No.:	156952	Severity Category:	1
Event Date:	04/26/80 - 04/30/80	Operating Mode:	6
Letter Date:	05/22/80	Cause Category:	SI signal got "locked in."



Summary of Events:

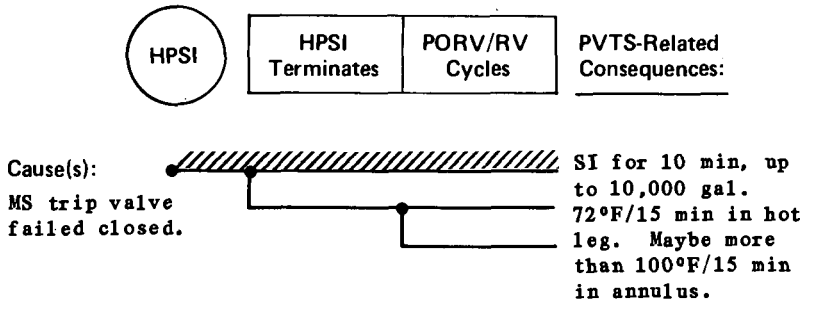
April 26
1. Mode 5 with RCS system partially drained.
2. Valve was cycled, allowing water from RWST to flow by gravity into RCS, increasing RCS boron concentration from 2567 to 3217 ppm.

April 30
1. 0218 h, an electrician working in the relay racks caused a train A SI signal. This signal would not reset.
2. 0511 h, it was discovered that the RWST was gravity-draining to the RCS via the BIT due to a valve locked open.
3. RCS boron concentration was diluted from 3090 to 2926 ppm.

April 20
1. 0920 h, accumulator motor operated valves were checked.
2. Power was actuated to the valves while SI signal was still locked in.
3. Water from A&B accumulators entered RCS. RCS pressure increased from atmosphere to 80 psi.

April 30
1. 10:07 h, an electrician installed a jumper in the MOV power supply breaker of the A accumulator MOV.
2. MOV opened, causing another discharge of the accumulator water to the RCS.

Reactor:	North Anna 1		
NSIC No.:	157093	Severity Category:	3
Event Date:	04/03/80	Operating Mode:	1
Letter Date:	04/16/80	Cause Category:	Main steam trip valve.

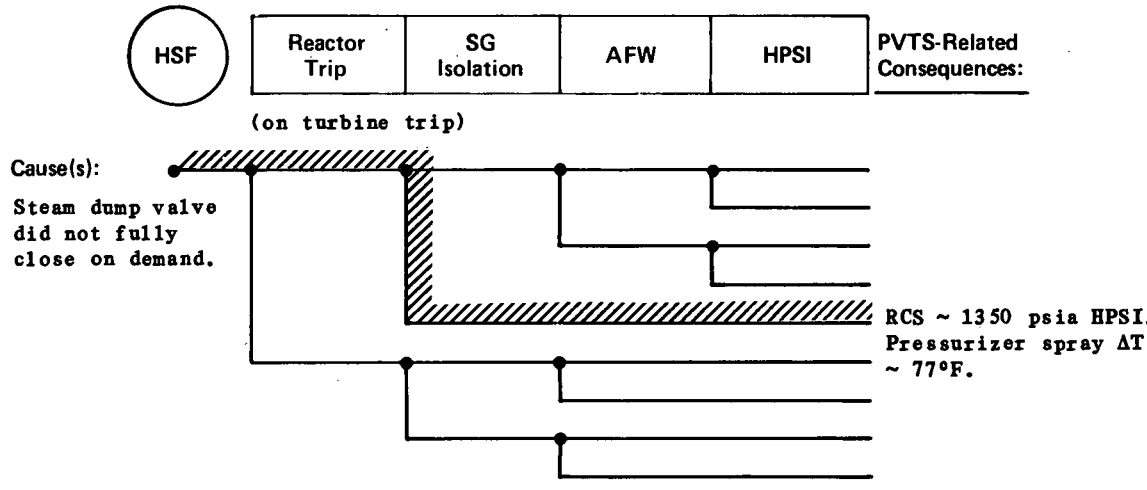


Summary of Events:

1. 23:14 h, Mode 1, 100% power. A periodic part-stroke testing of main steam trip valve TV-MS-101A was performed.
2. High steam flow pulled down valve disc then tripped valve closed due to reduction in air cylinder pressure. A partial design error in the valve control circuit was believed responsible.
3. High steam flow and low steam line pressure caused SI. This went on for 10 min. Temperature dropped 72°F in hot leg during the first 15 min.
4. We analyzed that up to 10,000 gal of SI water may have been injected, and that temperature in annulus may have dropped more than 100°F during the first 15 min. Reactor trip was assumed.

Reactor: Arkansas 2
NSIC No.: 157130
Event Date: 01/29/80
Letter Date: 05/13/80

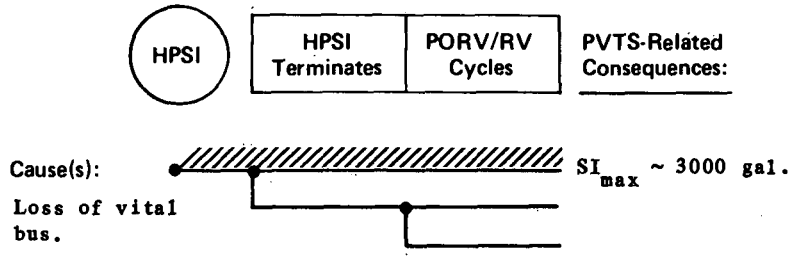
Severity Category: 3
Operating Mode: 3
Cause Category: MS dump valve failed open. Pressurizer spray failed open.



Summary of Events:

1. During a 100% turbine trip test, reactor tripped as designed; pressurizer spray (2CV-4651) opened but did not fully close on demand; steam dump and bypass valve (2CV-0301) opened but did not fully close on demand.
2. RCS pressure dropped to 1350 psia (Δp = 900 psi); there was only 57°F saturation margin (ΔT ~ 77°F).

Reactor:	North Anna 1		
NSIC No.:	158645	Severity Category:	2
Event Date:	05/23/80	Operating Mode:	3
Letter Date:	06/04/80	Cause Category:	Loss of vital bus.

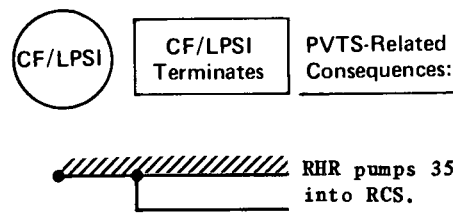


Summary of Events:

1. 0415 h, hot standby. Loss of vital bus 1-III due to start of RCS pump when others were also running, causing a loss of voltage and rejection of 1J diesel; 2 fuses were blown.
2. All three RCS pumps were tripped due to automatic loss of cooling water; a false low-low T_{avg} signal was also present.
3. High steam flow (due to loss of power to FW flow/steam flow comparator) + low-low T_{avg} resulted in SI (sixth occurrence).
4. SI was terminated at 0151 h (after 5 min).

Reactor: Davis-Besse 1
NSIC No.: 158860
Event Date: 04/19/80
Letter Date: 04/30/80

Severity Category: 1
Operating Mode: 6
Cause Category: Spurious essential bus opening while another bus was in maintenance.



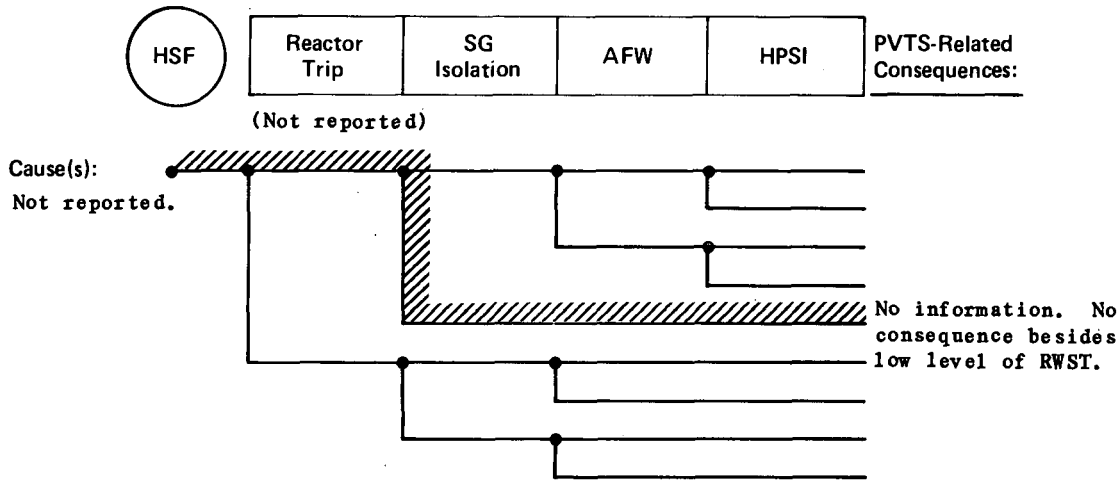
Cause(s):
Two essential buses opened, 1 by maintenance and 1 spuriously.

Summary of Events:

1. Reactor in Mode 6 (cold shutdown, low pressure), RCS partly drained.
2. 1400 h, two essential buses were opened, one due to maintenance, and one possibly due to construction vibration.
3. SI was actuated. HPSI pumps were racked out by procedures at this mode. RHR pump #1 was out for maintenance. RHR pump #2 was operating. SFAS caused RHR pump #2 to be transferred to the emergency sump, which was fed by BWST.
4. Approximately 3500 gal were injected into RCS.
5. 1402 h, RHR pump #2 was shut down.
6. RCS temperature rose from 140 to 170°F during the 2 h 30 min shutdown of RHR #2 loop to drain out air and refill it with water.

Reactor: Sequoyah 1
NSIC No.: 159219/160405
Event Date: 06/23/80
Letter Date: 07/31/80

Severity Category: 1
Operating Mode: Not reported.
Cause Category: "Inadvertent."

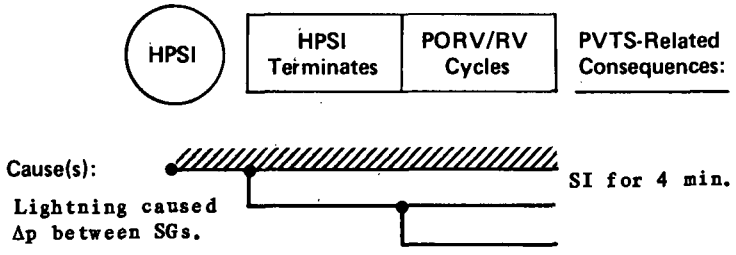


Summary of Events:

1. Very sketchy report.
2. Low-low T_{avg} and high steam flow in two loops brought about "inadvertent" SI. (Why inadvertent?)
3. Level in RWST was low before the event. Because of SI, the level of RWST was below tech spec. Level was reestablished in about 50 min.

Reactor: Salem 1
NSIC No.: 159553
Event Date: 06/08/80
Letter Date: 07/08/80

Severity Category: 2
Operating Mode: 1
Cause Category: Lightning.



Summary of Events:

1. 0844 h, 100% power.
2. Lightning struck south penetration area.
3. 7 MS pressure transmitter transient; 2 failed.
4. Reactor trip.
5. Δp > 100 psi between SG (false signals) SI.
6. 0848 h, SI terminated (4 min).

Reactor:	Sequoyah 1		
NSIC No.:	159786	Severity Category:	0
Event Date:	05/26/80	Operating Mode:	4
Letter Date:	06/23/80	Cause Category:	Control logic and system.

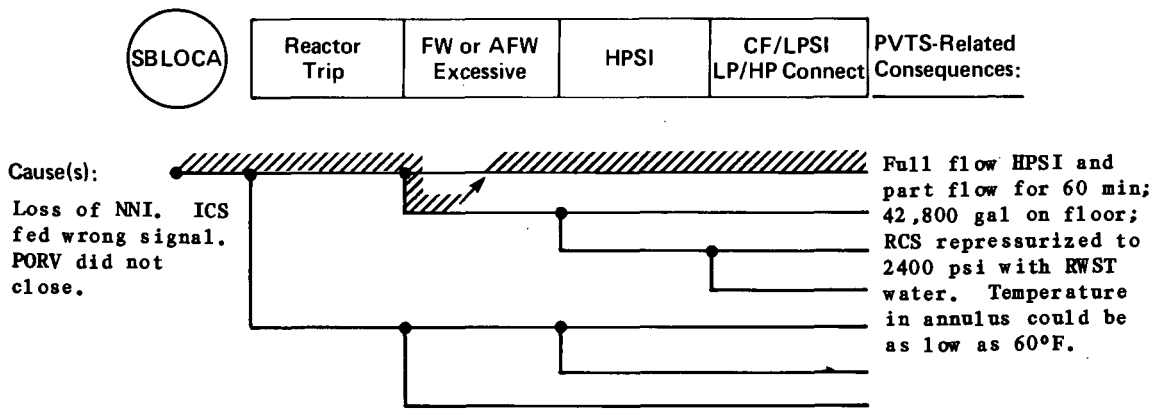
(No event tree)

Summary of Events:

1. Mode 4, in recovery from an SI.
2. Pressurizer temperature heatup 151°F in 30 min because cold SI surged into pressurizer.

Reactor: Crystal River 3
NSIC No.: 160846
Event Date: 02/26/80
Letter Date: 03/25/80

Severity Category: 4
Operating Mode: Mode 1/98.6%
Cause Category: Loss of NNI; wrong signals to ICS; PORV failed open; operator error.

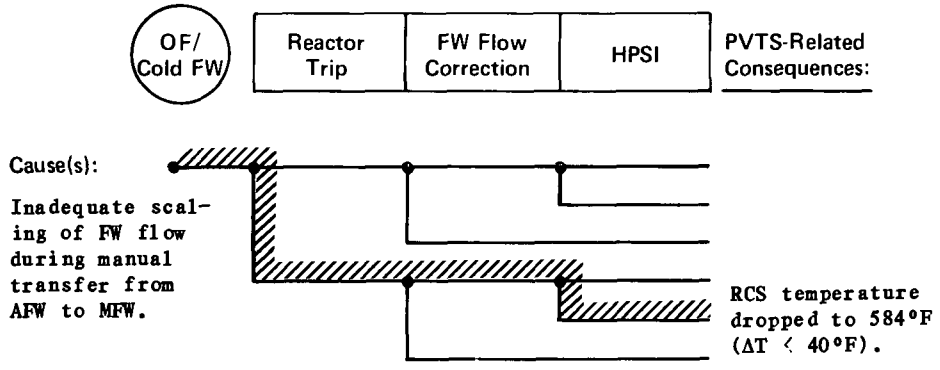


Summary of Events:

1. Initial conditions: Mode 1/98.6%; p = 2157 psi, pressurizer level 202 in.; T_h = 599–600°F; T_c = 556–557°F; T_{avg} = 579–580°F. RC flow = 73 x 10^6#/h each loop; letdown flow 48 gpm. SG A level 67%; p = 911 psig; flow = 5 x 10^6#/h. SG B level 65%; p = 909 psig; flow = 5 x 10^6#/h.
2. 1423 h, loss of 24 V DC NNI-X power supply. False signals to ICS; FW valves closed, rod withdrawn to 103%, pressurizer spray went on.
3. Reactor trip on high RC pressure; PORV opened and failed to close 1427 h. SG A boiled dry; so did B but signal did not show. AFW went on but valve did not open until 9 min.
4. 1426 h, HPSI repressurized system solid; code safety valves opened and cycled below set pressure.
5. 1523 h, T_c = 470°F(?). Temperature in annulus was around 250°F, but our analysis indicated that it could be as low as 60°F. Full HPSI flow for 32 min. Partial HPSI flow until minute 84 when letdown flow established.
6. 2107 h, RC pump resumed.
7. 42,800 gal in containment sump.

Reactor: Sequoyah 1
NSIC No.: 160915, #1
Event Date: 10/02/80
Letter Date: 10/31/80

Severity Category: 2
Operating Mode: 2
Cause Category: Inadequate scaling of FW flow during manual operation.

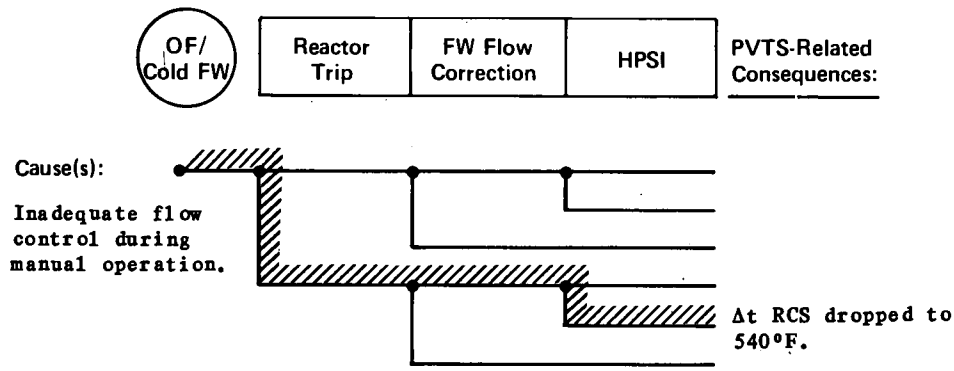


Summary of Events:

1. Mode 2, 5% power.
2. While manually transferring from AFW to MFW during power ascension test, overfeeding occurred due to inadequate scaling and improper venting of FW flow transmitter.

Reactor: Sequoyah 1
NSIC No.: 160915, #2
Event Date: 10/05/80
Letter Date: 10/31/80

Severity Category: 2
Operating Mode: 1/10%
Cause Category: Inadequate scaling of FW flow during manual operation.

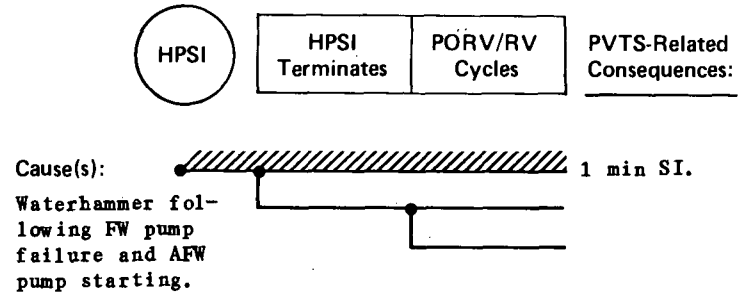


Summary of Events:

1. Mode 1, 10% power.
2. Overfeeding while on manual level control and testing the steam driven AFW pump. RCS temperature decreased to 540.3°F (ΔT ≈ 34°F).

Reactor: Zion 2
NSIC No.: 161826
Event Date: 09/03/80
Letter Date: 12/03/80

Severity Category: 1
Operating Mode: 4 (hot shutdown).
Cause Category: Waterhammer.

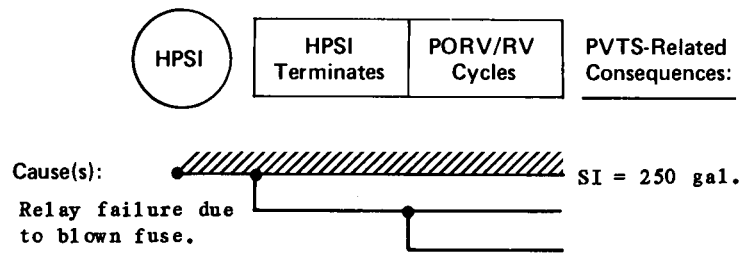


Summary of Events:

1. 0519 h, 2B FW pump failed, causing low level in 2C SG, resulting in reactor trip.
2. All three AFW pumps started and fed water to SG at about 260 gpm per SG.
3. 0535 h, waterhammer occurred in 2B feedwater line, causing pressure spikes in 2B and 2D loops.
4. SI was actuated due to Δp between SGs. SI for about 1 min.

Reactor: Crystal River 3
NSIC No.: 162394
Event Date: 10/16/80
Letter Date: 01/12/81

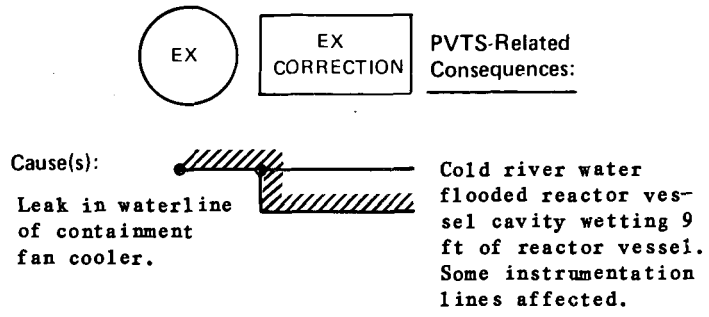
Severity Category: 1
Operating Mode: 1
Cause Category: Relay failure during channel test.



Summary of Events:

1. Fuse blown. Relay failure during channel test.
2. Reactor was at Mode 1, 97% power.
3. SI of 250 gal.

Reactor:	Indian Point 2		
NSIC No.:	162919	Severity Category:	4
Event Date:	Before 10/17/80	Operating Mode:	1
Letter Date:	10/20/80	Cause Category:	Flood from outside due to a fan waterline leak.

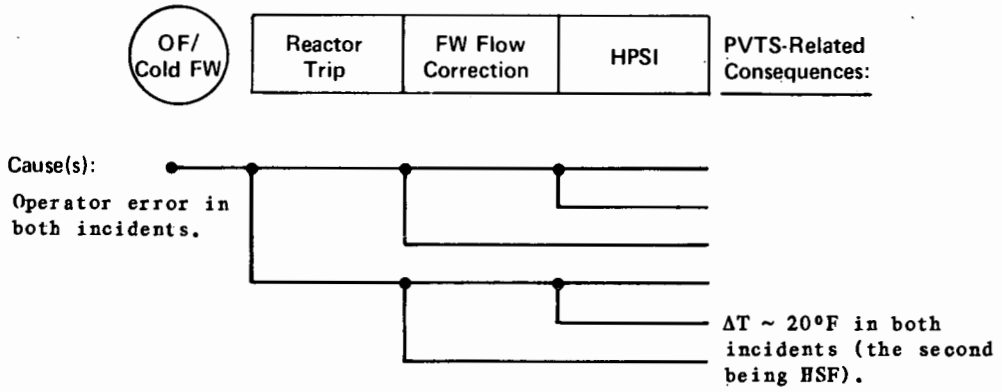


Summary of Events:

1. On Oct. 17, 1980, the reactor was shut down to repair some reactor instrumentation lines. Workers entering containment discovered that the reactor cavity had been flooded.
2. It was determined later that the waterline of a fan cooler unit (used to cool the air inside reactor containment) had resulted in the undetected flood. About 125,000 gal was in the containment, and about 9 ft of the reactor vessel was under cold river water.
3. This incident is still under investigation (as of April 1982).

Reactor: North Anna 2
NSIC No.: 163551
Event Date: 01/01/81
Letter Date: 06/05/81

Severity Category: 1
Operating Mode: 2
Cause Category: Operator error.



Summary of Events:

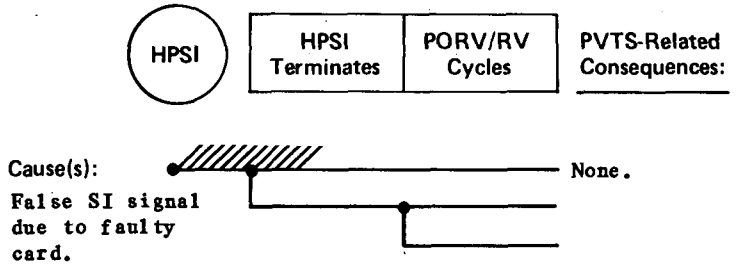
First incident

1. Mode 2, 2% power.
2. CVCS control tank was overborated and letdown was in excess. T_{avg} dropped below 541°F. This went on for about 10 min and was recovered by (a) securing SG blowdown, (b) withdrawing rods, and (c) diluting RCS boron.

Second incident

1. Operator opened a condenser steam dump. When over, he should close one turn. Instead, he opened one turn. T_{avg} dropped again to below 541°F for about 3 min.
2. Recovery was done by withdrawing rods.

Reactor:	Farley 1		
NSIC No.:	163421	Severity Category:	0
Event Date:	10/24/79	Operating Mode:	4
Letter Date:	01/21/80	Cause Category:	Faulty control card.

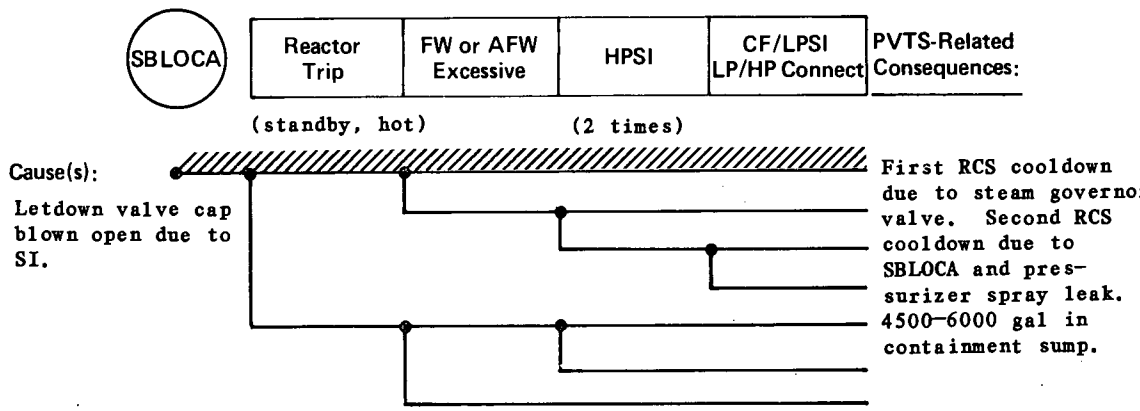


Summary of Events:

1. 14:27 h, Mode 4.
2. SI was actuated on the A train. Trouble shooting pointed to the replacement of Universal Card A416.

Reactor: Robinson 2
NSIC No.: 164149
Event Date: 01/29/81
Letter Date: 02/12/81

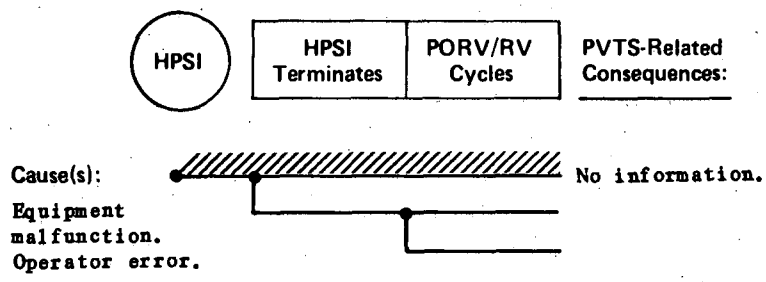
Severity Category: 3
Operating Mode: 2
Cause Category: SG governor valve/pressurizer spray.



Summary of Events:

1. Reactor being brought to hot standby (Mode 3) for turbine E-H system repair.
2. 0624 h, generator output breakers were opened. A governor valve spiked open, apparently due to the E-H oil leak problem.
3. High steam flow and RCS T_{avg} < 543°F caused the B safeguard train to accurate SI. Train A was subsequently engaged at 0625 h.
4. SI was reset at 0627 h (total 2 min of SI).
5. A letdown drain valve appeared to have vibrated open during previous operation, and the pipe cap, which serves as a pressure boundary, was blown open following SI. A leak in the letdown line in the order of 100 gpm was apparent (found out later).
6. 0635 h, RCS pressure decreased as containment pressure and dewpoint found to be still leaking at 5–7 gpm. Containment sump indicated 3000 gal.
7. Pressure continued to decrease due to leaky pressurizer spray. SI was automatically actuated. RCS pumps were stopped to stop the leak. RCS pressure turned around and increased.
8. "Normal Control" of RCS pressure established.
9. A total of 4500–6000 gal was found in containment sump.

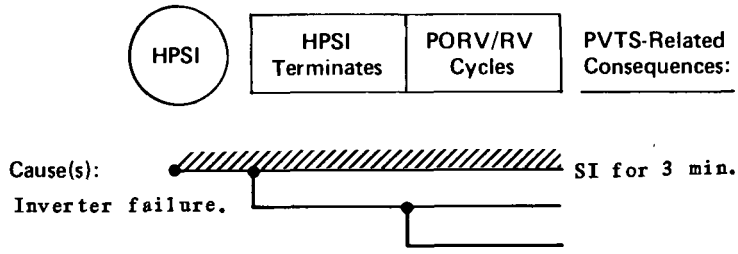
Reactor:	Beaver Valley 1		
NSIC No.:	164791	Severity Category:	0
Event Date:	11/17/80	Operating Mode:	2
Letter Date:	03/02/81	Cause Category:	Equipment malfunction; operator error.



Summary of Events:

1. Normal startup. A low pressure steam line bistable intermittently tripped and reset (later found out as due to a malfunctioning signal converter card).
2. Station supervisor and operator tried to trip the bistable at the process racks, but they operated the wrong channel. (BS-MS-484 instead of BS-MS-485).
3. SI occurred.

Reactor:	Salem 1		
NSIC No.:	164865	Severity Category:	2
Event Date:	11/27/78	Operating Mode:	1
Letter Date:	03/25/78	Cause Category:	Inverter failure.

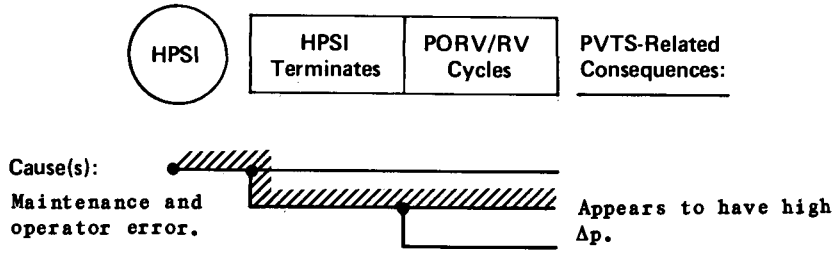


Summary of Events:

1. Mode 1, 100% power.
2. 0445 h, loss of 1B vital bus due to output transformer and two regulating reactor failures in supply inverter.
3. RCS low flow; reactor trip; $T_{avg} \rightarrow 543°F$.
4. Nos. 12 and 13, AF pumps did not start; SG level low.
5. 0502 h, recovery from reactor trip. High steam flow + low T_{avg} caused SI for 3 min.
6. 1B diesel, No. 11 charging pump and No. 12 RHR pumps failed to start.
7. SI reset. 0630 h, No. 11 RHR pump could not be restarted because breaker would not close: inverter had failed. 1100 h, 1B diesel operational.
8. 1315 h, started cooldown.
9. 2100 h, hot shutdown, Mode 4. 2240 h, RHR was initiated.

Reactor: North Anna 1
NSIC No.: 165876
Event Date: 03/29/81
Letter Date: 04/20/81

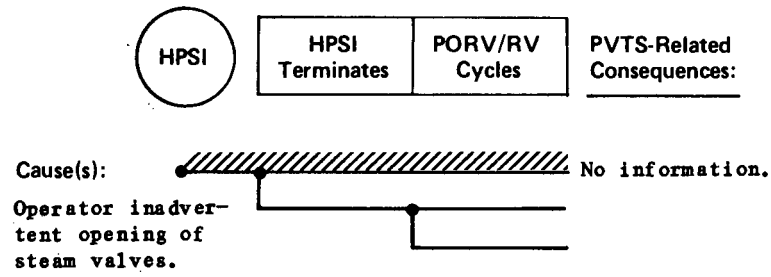
Severity Category: 0
Operating Mode: 5
Cause Category: Maintenance and operator error.



Summary of Events:

1. Mode 5.
2. During testing of train B solid state protection system, operator unblocked low pressurizer signal.
3. This plus low T_{avg} actuated SI.
4. RCS PORV lifted to relieve RCS pressure.
5. SI was stopped.

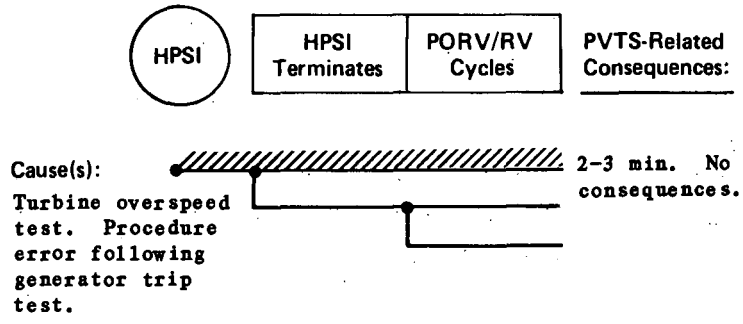
Reactor:	Salem 2		
NSIC No.:	166342	Severity Category:	0
Event Date:	04/23/81	Operating Mode:	2
Letter Date:	05/22/81	Cause Category:	Operator error.



Summary of Events:

1. 0330 h, Mode 2 and cooling down.
2. Inadvertent opening of valve 24 MS 10 causing alarm "24 SG steam pressure 100 psi low."
3. This plus low T_{avg} caused SI.

Reactor:	Sequoyah 1		
NSIC No.:	166453	Severity Category:	1
Event Date:	04/23; 05/01/81	Operating Mode:	1st SI, Mode 3.
Letter Date:	06/05/81	Cause Category:	2nd SI, Mode 2. Test and operator error.



Summary of Events:

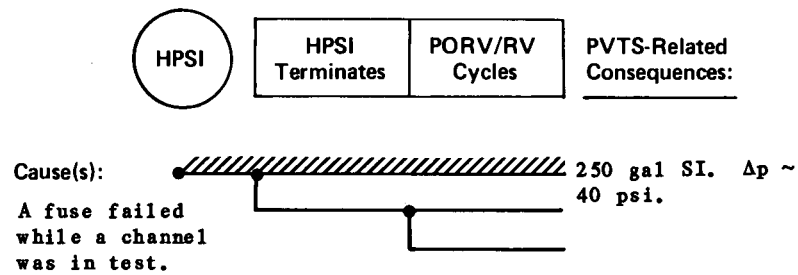
April 23, 1981

1. Mode 3, p = 2234 psig, T_{avg} = 539.8°F.
2. During turbine overspeed test, high steam flow, low T_{avg} caused SI.
3. SI for 2 to 3 min.

May 1, 1981

1. Mode 2, 2220 psig, T_{avg} = 500°F.
2. After generator trip test, two reactor coolant pumps were returned to service simultaneously, causing Δp between the steam generators.
3. SI was actuated by 2 Δp signals between SGs; it was terminated quickly.

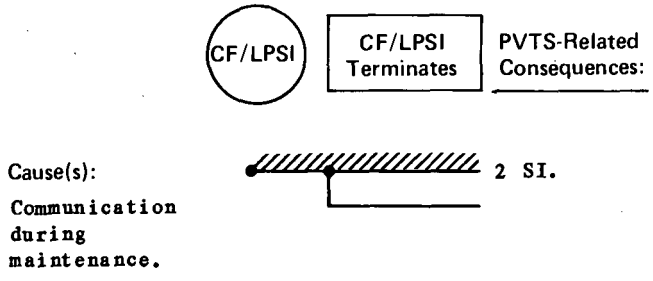
Reactor:	Crystal River 3		
NSIC No.:	166648	Severity Category:	1
Event Date:	10/16/80	Operating Mode:	Mode 1/97%
Letter Date:	06/11/81	Cause Category:	Fuse failed while one channel was under test.



Summary of Events:

1. Mode 1, 97% power.
2. Channel 1 was in test; a fuse in channel 2 failed due to failed relay.
3. Two out of three signals caused SI.
4. 250 gal of SI; RCS pressure increased from 2145 to 2185 psig.

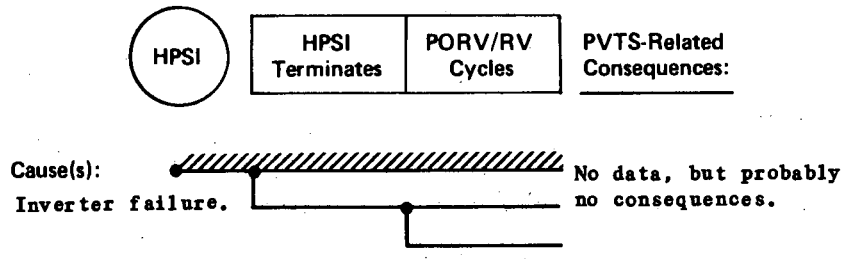
Reactor:	McGuire 1		
NSIC No.:	166686	Severity Category:	1
Event Date:	03/30/81	Operating Mode:	5
Letter Date:	04/28/81	Cause Category:	Maintenance and communication error.



Summary of Events:

1. 1500 h, Mode 5, initial fuel loading.
2. Technicians working in logic cabinet, left work area without proper notes; supervisor came to area, reset switches without inquiring; ESF signal was actuated; some RHR water was injected.
3. A second SI was actuated when a technician and an operator attempted to unlock the feedwater isolation valves.

Reactor:	Zion 1		
NSIC No.:	166746/167772	Severity Category:	1
Event Date:	05/28/81	Operating Mode:	1/99%.
Letter Date:	08/06/81, 07/31/81	Cause Category:	Inverter failure.

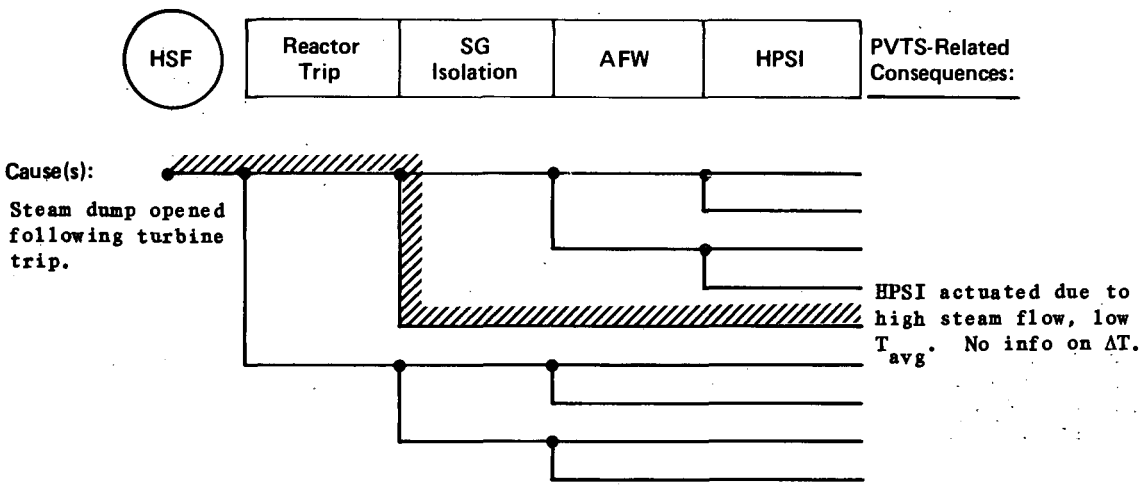


Summary of Events:

1. Mode 1, 99% power.
2. Inverter 114 failed (due to SOLA transformer shorted to ground).
3. Reactor trip.
4. SI due to high steam flow and Δp.

Reactor: Salem 2
NSIC No.: 166854
Event Date: 06/03/81
Letter Date: 06/30/81

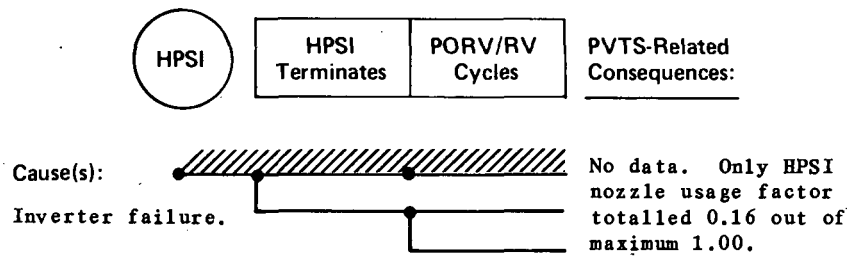
Severity Category: 1
Operating Mode: 1/20%
Cause Category: Steam dump valve opened following turbine trip.



Summary of Events:

1. 0440 h, Mode 1, 20% power.
2. A high level in No. 22 SG caused a turbine trip and reactor trip.
3. The steam dump opened, causing high steam flow, low RCS T_{avg}; HPSI was actuated.
4. No information on duration, quantity of SI, nor ΔT, Δp of RCS. The only information was that the boron concentration in boron injection tank dropped to below 20,000 ppm (from the normal 20,000–22,500 ppm).

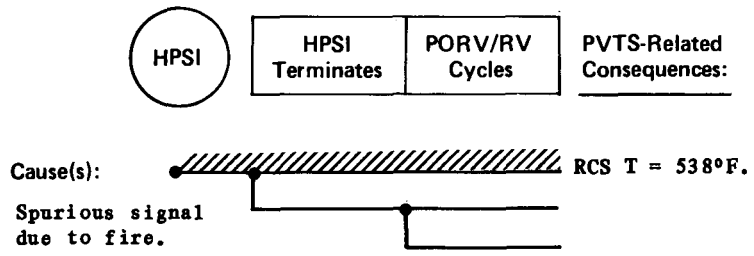
Reactor:	Zion 2		
NSIC No.:	167124, 168077	Severity Category:	1
Event Date:	05/07/81	Operating Mode:	1/92%
Letter Date:	06/29/81 07/31/81	Cause Category:	Inverter failure



Summary of Events:

1. Mode 1, 92% power.
2. Inverter 213 failed and slave transformer burned up due to master inverter SOLA transformer shorted.
3. One FW pump tripped due to speed control. Reactor tripped.
4. Loss of inverter also caused PI-526A and 536A to fail low. FW flow now became high. The two coincident signals caused HPSI.

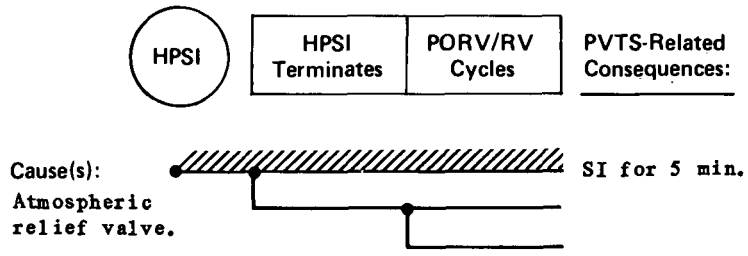
Reactor:	North Anna 2		
NSIC No.:	167550	Severity Category:	2
Event Date:	07/03/81	Operating Mode:	3
Letter Date:	07/15/81	Cause Category:	Fire causing spurious signal.



Summary of Events:

1. 0723 h, Mode 3; a spurious signal occurred (low T_{avg} and high steam flow) due to voltage transient.
2. The spurious signal was due to a fire that affected feeder cables from 1C reserve station service transformer.
3. 0748 h, RCS temperature = 538°F.

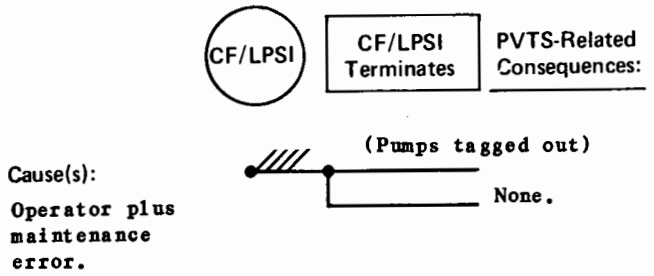
Reactor:	Salem 2		
NSIC No.:	167625	Severity Category:	3
Event Date:	04/23/81	Operating Mode:	3
Letter Date:	07/21/81	Cause Category:	Operator/system error.



Summary of Events:

1. Mode 3, 0% power, cooling down to Mode 5 for repairing an unisolable leak in the CVCS charging line.
2. SG atmosphere relief valve was being used to control cooldown rate.
3. Interaction between control and setpoints caused #24 SG to have pressure different from other steam generators by more than 100 psi.
4. SI occurred for about 5 min.

Reactor:	McGuire 1		
NSIC No.:	167692	Severity Category:	0
Event Date:	05/07/81	Operating Mode:	6
Letter Date:	06/04/81	Cause Category:	Operator plus maintenance error.

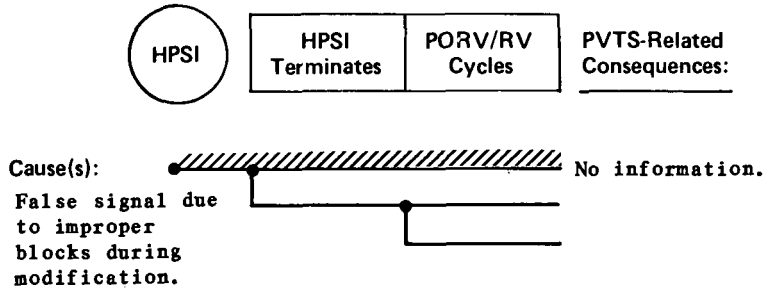


Summary of Events:

1. Mode 6, prior to initial criticality.
2. Inadvertent SI was initiated due to operator pushing wrong button during test.
3. Both SI and charging pumps were tagged out, hence no volume of water was transferred.

Reactor: McGuire 1
NSIC No.: 167784
Event Date: 04/29/81
Letter Date: 05/28/81

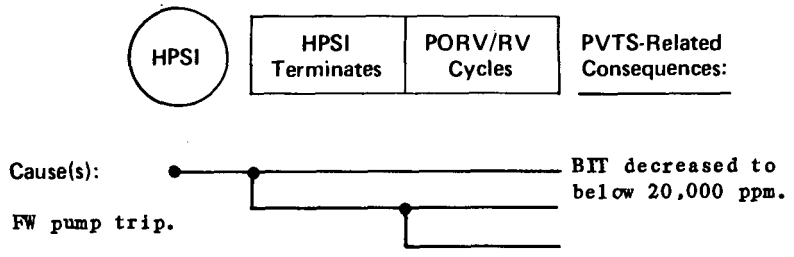
Severity Category: 0
Operating Mode: 5
Cause Category: Maintenance error.



Summary of Events:

1. Mode 5, cold shutdown.
2. During modification work on train A logic cabinets, some blocks were cleared without reblock before returning to service. This was in violation of procedure.
3. SI occurred but immediately secured.

Reactor:	Salem 2		
NSIC No.:	168079	Severity Category:	0
Event Date:	06/27/81	Operating Mode:	1
Letter Date:	07/27/81	Cause Category:	FW pump trip.



Summary of Events:

1. Mode 1, 30% power, 260 MW(e).
2. 0403 h, FW pump on No. 22 SG tripped, causing low-low level.
3. 0408 h, reactor tripped.
4. Low T_{avg} and high steam flow caused SI.
5. Boron injection tank concentration decreased to below 20,000 gpm.

Reactor:	Cook 1		
NSIC No.:	168503; 170158	Severity Category:	0
Event Date:	07/23/81	Operating Mode:	5
Letter Date:	11/06/81	Cause Category:	Pressurizer heatup excessive.

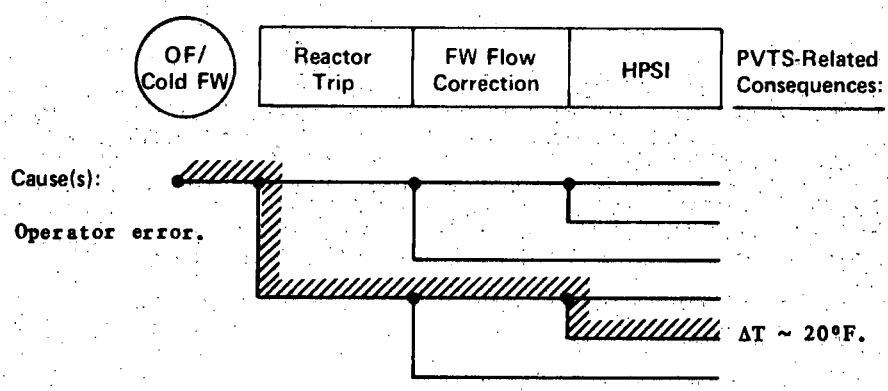
(No event tree)

Summary of Events:

1. Mode 5, RCS in process of filling and venting; RCS temperature ≃ 175°F, p ≃ 100 psi increasing to 325 psi.
2. Pressurizer level was low due to water surging into the air space in the SG tubes, then cold water (175°F) surged back into SG; contacting 400°F metal.
3. Pressurizer heatup exceeded 100°F/h, against tech spec. Report stated that this was only an apparent heatup, because during preoperating testing, the maximum heatup rate achieved with all heaters in service was 67.5°F/h.

Reactor: Cook 2
NSIC No.: 168144
Event Date: 07/11/81
Letter Date: 08/10/81

Severity Category: 1
Operating Mode: 2
Cause Category: Operator error.

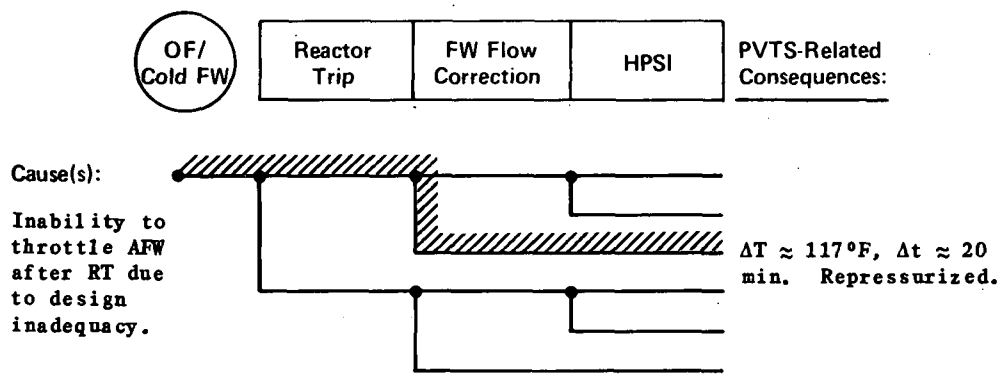


Summary of Events:

1. Mode 2 (startup).
2. An excessive feed rate was used to raise the level of the SG. RCS temperature dropped below tech spec minimum of 541°F despite operator's attempt to reduce feed rate and to withdraw control rods.
3. ΔT ~ 20°F.

Reactor: McGuire 1
NSIC No.: 169297
Event Date: 09/14/81
Letter Date: 10/09/81

Severity Category: 4
Operating Mode: 1/20%
Cause Category: Design and procedure inadequacy.



Summary of Events:

1. Mode 1, 20% power.
2. 01:12 h, performance of the loss of control room test. One group of operators left the control room to man the reactor trip circuit breaker (in room next to control room); another group left to man the auxiliary shutdown panel and the AFW pump control panels (located in the pump room several floors below the control room).
3. 01:13:19 h, operator manually tripped the reactor from the RI breakers. Almost immediately afterwards, T_{avg} < 564°F, MFW isolation valves closed, SG level setpoint dropped to 1o-1o level, and all three AFW pumps started.
4. AFW pumped at the maximum rate because throttle valves had been fully opened and cannot be throttled by the operators. During the next several minutes, SG filled rapidly, T_{avg} fell, pressure level fell, and RCS pressure dropped.
5. 01:20:26 h, operator started the second centrifugal charging pump to recover pressurizer level and pressure. When RCS pressure was below 1955 psig, operators blocked SI. Both charging pumps were running. Minimum RCS pressure was about 1900 psig.
6. 01:21:06 h, pressure level decreased to zero. Operators tripped the four RCS pumps.
7. 01:30:14 h, pressure level recovered; when RCS pressure was above 1955 psig, SI block was cleared. Low SG pressure actuated SI. The MS isolation closed due to SI to help steam heat removal.
8. 01:33:40 h, SI was reset on both trains.
9. Recovery proceeded smoothly with a minor SI reinitiation on Train A. ΔT = 116.8°F "during the hour following reactor trip." If 01:33:40 h is taken as the "final time," then ΔT ≈ 20 min.

Appendix B

NSIC ABSTRACTS OF PVTS EVENTS AND PRECURSORS

This appendix contains printouts of citations obtained from the NSIC data bank in searches for PVTS events and precursors. The abstracts are arranged by increasing accession numbers. Each citation has a title, the involved reactor, an abstract, the place of original report availability, and a listing of keywords.

It is noted that an event may be reported in several citations.

✓ ACCESSION NUMBER 57478 10-01 11/18/82

PLANT TRIP AND REACTOR COOLDOWN
CONNECTICUT YANKEE ATOMIC POWER COMPANY
DOCKET 50213-82 +. 3 PAGES, LETTER - CONNECTICUT YANKEE ATOMIC POWER COMPANY TO DIVISION OF REACTOR LICENSING (AEC) - NOVEMBER 4, 1970, TYPE--PWR, MFG--WEST., AE--STONE + WEBSTER
A SPURIOUS LOW-FLOW SIGNAL CAUSED A PLANT TRIP WHICH OPENED STEAM-DUMP VALVES. WHEN COOLANT TEMPERATURE STARTED TO DECREASE, ONE OF TEN DUMP VALVES FAILED TO CLOSE, AND SYSTEM TEMPERATURE DECREASED 43 F IN 9 MIN BEFORE A MANUAL VALVE TO THE DUMP VALVE HEADER WAS CLOSED. ANALYSIS OF THE COOLDOWN INDICATED NO HARM TO THE REACTOR VESSEL ON SYSTEM BOUNDARIES. COMPONENTS WHICH CAUSED THE SPURIOUS FLOW SIGNAL WERE REPLACED TO PREVENT A RECCURANCE. THE VALVE FAILURE WAS TRACED TO POSITION COIL STOP, WHICH WAS READJUSTED AND LOCKED IN POSITION.
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FAILURE, EQUIPMENT + REACTOR, PWR + CONNECTICUT YANKEE (PWR) + CONTAINMENT INTEGRITY + STRESS ANALYSIS + FAILURE, INSTRUMENT + INSTRUMENT, FLOW + SCRAM, SPURIOUS + OPERATING EXPERIENCE + THERMAL TRANSIENT + VALVES + FAILURE + PSCDR,

✓ ACCESSION NUMBER 75613 10-01 11/18/82

GASKET FAILURE IN STEAM LINE AT SURRY 1
VIRGINIA ELECTRIC AND POWER COMPANY
DOCKET 50280-87 +. 6 PAGES, LETTER - VIRGINIA ELECTRIC AND POWER COMPANY TO DIRECTORATE OF LICENSING (AEC) - OCTOBER 11, 1972, DOCKET 50-280, TYPE--PWR, MFG--WEST., AE--STONE + WEBSTER
ON OCT. 1, 1972, THE GASKET ON A 14-IN. SWING CHECK VALVE BONNET FAILED FROM A WATER HAMMER. THE FAILURE CAUSED A HIGH COOLDOWN RATE (100 F/HR) AND A POSITIVE CONTAINMENT PRESSURE. THE REACTOR WAS AT HOT SHUTDOWN CONDITIONS. SAFETY INJECTION WAS INITIATED. PRIOR TO THE INCIDENT, THERE WAS A LEAK IN THE CHECK-VALVE-BONNET GASKET, CAUSING A GRADUAL DECREASE IN THE STEAM-GENERATOR LEVEL. THE LEAK CAUSED A FLOW PATH FROM THE STEAM GENERATOR RING BACK TO THE CHECK VALVE. THIS IS BELIEVED TO HAVE CAUSED A SEVERE HAMMER. THE FLANGE WAS RAISED 3/8 INCH; PART OF GASKET WAS MISSING.
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FAILURE + HYDRAULIC EFFECT + REACTOR, PWR + INCIDENT, EQUIPMENT + SEAL + DAMAGE + MAIN COOLING SYSTEM + SURRY 1 (PWR) + LEAK + PRESSURE TRANSIENT + VALVE, CHECK + THERMAL TRANSIENT + TRANSIENT + PIPES AND PIPE FITTINGS + PSCDR

✓ ACCESSION NUMBER 68933 10-01 11/18/82

STRESS DURING COOL DOWN AT CONNECTICUT YANKEE
CONNECTICUT YANKEE ATOMIC POWER COMPANY, WETHERSFIELD, CONNECTICUT
DOCKET-50213-126 +. 4 PAGES, LETTER - CONNECTICUT YANKEE ATOMIC POWER COMPANY TO DIVISION OF REACTOR LICENSING (AEC) - DECEMBER 29, 1971, DOCKET 50-213; TYPE--PWR, MFG--WEST., AE--STONE + WEBSTER
STRESSES GENERATED DURING THE REACTOR COOLANT SYSTEM COOLDOWN ON OCT. 12, 1970, WERE ANALYZED. THE TRANSIENT PRODUCED NO THERMAL STRESS WHICH EXCEEDED LIMITS FOR PRIMARY OR PRIMARY-AND-SECONDARY STRESS. THE TRANSIENT IS A POSSIBLE SOURCE OF A FATIGUE CRACK. THEREFORE, 12 CRITICAL AREAS WERE ANALYZED. THE PRESSURIZER COOLDOWN WAS EQUIVALENT TO 68.5 F/HR (WAS DESIGNED FOR 70 F/HR).
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + THERMAL MECHANICAL EFFECT + CONNECTICUT YANKEE (PWR) + STRESS ANALYSIS + PRESSURIZER + STRUCTURAL INTEGRITY + MAIN COOLING SYSTEM + THERMAL TRANSIENT + PRESSURE VESSELS + FAILURE

✓ ACCESSION NUMBER 76461 10-01 11/18/82

TWO RELIEF VALVES MALFUNCTION AT ROBINSON 2
CAROLINA POWER AND LIGHT COMPANY
3 PAGES, LETTER TO DIRECTORATE OF LICENSING (AEC), NOV. 30, 1972, DOCKET 50-261, TYPE--PWR, MFG--WEST., AE--EBASCO
POWER LEVEL WAS 70%, AND WEEKLY TURBINE VALVE TESTING WAS BEING PERFORMED. DIFFICULTIES WITH TESTING OCCURRED AND A FAST REDUCTION IN ELECTRICAL LOAD FROM 520 MWE TO 150 OCCURRED. OPERATOR TRIPPED REACTOR BECAUSE OF TRANSIENT. STEAM DUMPING WAS IN EFFECT BUT PRESSURE RELIEF VALVE RV-1 MALFUNCTIONED AND STAYED OPEN 5 MIN. PRIMARY TEMP. REDUCED TO 509 F. UPON RESTART AT 0333 HR, RELIEF VALVE RV-2 OPENED AND WOULD NOT CLOSE. SAFETY INJECTION LATER ACTUATED AND S1 VALVE-867A DID NOT OPEN. TORQUE SWITCH WAS RESET. RV-2 INNER VALVE AND GUIDE BUSHING WERE BADLY SCARRED. RV-1, -2, AND -3 WERE THEN POLISHED. VENDOR CONSULTED FOR MODIFICATIONS TO PREVENT SCARRING.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + REACTOR TRANSIENT + STEAM + SAFETY INJECTION + FAILURE, INSTRUMENT + INSTRUMENT, SWITCH + PRESSURE RELIEF + REACTOR POWER + ROBINSON 2 (PWR) + DAMAGE + PRESSURE TRANSIENT + OPERATING EXPERIENCE + TRANSIENT + VALVES + ACTUATION + PSCDR

ACCESSION NUMBER 85369 10-01 11/18/82

EXCESSIVE COOLDOWN RATE AT GINNA FOLLOWING LOSS OF OFF-SITE POWER (AO 73-9)
ROCHESTER GAS AND ELECTRIC CORPORATION
4 PAGES, LETTER WITH ATTACHMENT TO DIRECTORATE OF LICENSING (AEC), OCTOBER 31, 1973, DOCKET 50-244, TYPE--PWR, MFG--WEST., AE--GILBERT ASSOC.
DURING A PERIOD WHEN 2 OF THE 4 115-KV TRANSMISSION LINES FROM THE STATION WERE OUT OF SERVICE AND THE SYSTEM DEMAND WAS LIGHT COMPARED WITH THE GENERATOR OUTPUT, INSTABILITY OF THE TRANSMISSION SYSTEM DEVELOPED CAUSING LOSS OF ALL OFF-SITE POWER. A REACTOR TRIP THEN OCCURRED. A LO-LO STEAM-GENERATOR-LEVEL SIGNAL STARTED BOTH FEEDWATER PUMPS. THE RAPID ADDITION OF AMBIENT TEMPERATURE CAUSED AN 85 F DROP IN THE REACTOR-COOLANT COLD LEG IN 10 MIN. SAFETY INJECTION WAS INITIATED. NO DAMAGE OCCURRED. PROCEDURES WERE CHANGED TO PREVENT FULL-POWER OPERATION UNLESS ALL 4 TRANSMISSION LINES ARE FUNCTIONING.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + PROCEDURES AND MANUALS + SCRAM, REAL + SAFETY INJECTION + INSTABILITY + FAILURE, ADMINISTRATIVE CONTROL + MAIN COOLING SYSTEM + OFF SITE + GINNA (PWR) + THERMAL TRANSIENT + POWER TRANSMISSION + PSCDR

ACCESSION NUMBER 85370 10-01 11/18/82

MALFUNCTION OF SI PUMP SUCTION TRANSFER FROM LOSS OF POWER AT GINNA (AO 73-10)
ROCHESTER GAS AND ELECTRIC COMPANY
4 PAGES, LETTER WITH ATTACHMENT TO DIRECTORATE OF LICENSING (AEC), OCTOBER 31, 1973, DOCKET 50-244, TYPE--PWR, MFG--WEST., AE--GILBERT ASSOC.
SAFETY INJECTION (SI) OCCURRED FROM LOW PRESSURE AND LOW LEVEL IN THE PRESSURE VESSEL FOLLOWING A REACTOR TRIP FROM LOSS OF OFFSITE POWER (AO-73-9). THE SI PUMP SUCTION TRANSFERRED FROM THE BORIC ACID STORAGE TANKS TO THE REFUELING WATER STORAGE TANK PREMATURELY. FAILURE RESULTED FROM LOSS OF POWER TO INSTRUMENT BUS 1A. THIS LOSS GENERATED A FALSE SIGNAL. THE LOSS OF POWER FOR 38 SECONDS IS THOUGHT TO BE FROM AN OPERATOR ERROR SINCE NO MALFUNCTION WAS FOUND. ALL INSTRUMENT BUS PANEL DOORS WILL BE LOCKED TO PREVENT RECURRENCE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

CONTROL SYSTEM + FAILURE + FAILURE, OPERATOR ERROR + REACTOR, PWR + SCRAM, REAL + ELECTRIC POWER + SAFETY INJECTION + SHUTDOWN SYSTEM, SECONDARY + EMERGENCY COOLING SYSTEM + GINNA (PWR) + PSCDA + PSCDR

ACCESSION NUMBER 87016 1111-01 1 03/22/83

SAFETY INJECTION INITIATION AT SAN ONOFRE 1 FROM RAPID COOLDOWN OF THE SYSTEM
SOUTHERN CALIFORNIA EDISON COMPANY, CALIFORNIA
7 PAGES, LETTER WITH ATTACHMENT TO DIRECTORATE OF REGULATORY REGULATION (AEC), OCTOBER 22, 1973, DOCKET 50-206, TYPE--PWR, MFG--WEST., AE--BECHTEL
THE UNIT WAS BEING REMOVED FROM SERVICE TO INVESTIGATE TURBINE PROBLEMS WHICH WERE INDICATED BY EXCESSIVE VIBRATION OF A BEARING AND BY CONDENSER SALT WATER LEAKAGE. A SAFETY INJECTION REACTOR TRIP OCCURRED SHORTLY AFTER TRIPPING OF THE TURBINE. UNDER THE OPERATING CONDITIONS WITH THE FEEDWATER VALVES ON AUTOMATIC, THE VALVES OPENED WHEN THE TURBINE WAS TRIPPED CAUSING RAPID FILLING OF THE STEAM GENERATORS. THIS CAUSED A RAPID COOLDOWN OF THE REACTOR COOLANT SYSTEM AND ACTUATION OF SAFETY INJECTION. ALL SYSTEMS FUNCTIONED PROPERLY DURING THE EVENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

CONTROL SYSTEM + FAILURE + REACTOR, PWR + SCRAM, REAL + SAFETY INJECTION + SAN ONOFRE 1 (PWR) + FAILURE, DESIGN ERROR + VIBRATION + MAIN COOLING SYSTEM + STEAM GENERATOR + TURBINE + THERMAL TRANSIENT

ACCESSION NUMBER 87031 10-01 11/18/82

EXCESSIVE COOLDOWN RATE AND MALFUNCTION OF SAFETY INJECTION AT GINNA (AO 73-9 AND -10)
ROCHESTER GAS AND ELECTRIC CORPORATIONS
4 PAGES, LETTER TO DIRECTORATE OF REGULATORY OPERATIONS, REGION 1 (AEC), OCTOBER 23, 1973, DOCKET 50-244, TYPE--PWR, MFG--WEST., AE--GILBERT ASSOC.
LOSS OF OUTSIDE POWER CAUSED A REACTOR TRIP. DISTURBANCES ON THE INSTRUMENT BUSES GENERATED A LOW-LOW STEAM GENERATOR LEVEL CONDITION IN BOTH STEAM GENERATORS WHICH AUTOMATICALLY STARTED THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP. THIS INJECTED COLD WATER INTO THE STEAM GENERATORS AND CAUSED A RAPID COOLDOWN OF THE REACTOR COOLANT SYSTEM WHICH EXCEEDED THE LIMIT. THE RAPID COOLDOWN RESULTED IN A SAFETY INJECTION SIGNAL. THE SAFETY INJECTION PUMPS SUCTION SWITCHED FROM THE BORIC ACID STORAGE TANK TO THE REFUELING WATER STORAGE TANK PREMATURELY. AN INVESTIGATION OF ALL THESE PROBLEMS IS IN PROGRESS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ELECTRIC POWER + SAFETY INJECTION + INSTRUMENT CALIBRATION + RATE + OFF SITE + COOLING + GINNA (PWR) + COOLANT LEVEL TRANSIENT + PSCDR

ACCESSION NUMBER 88077 10-01 11/18/82

REACTOR COOLANT SYSTEM COOLDOWN RATE EXCEEDS LIMITS AT OCONEE 2 (AO-270/74-1)
DUKE POWER COMPANY
4 PAGES, LETTER WITH ATTACHMENT TO DIRECTORATE OF LICENSING (AEC), JANUARY 17, 1974, DOCKET 50-270, TYPE--PWR, MFG--B+W, AE--DUKE
WHEN A SPURIOUS SIGNAL IN THE 230KV SWITCHYARD CIRCUIT BREAKER FAILURE RELAY CIRCUITRY RESULTED IN THE ISOLATION OF THE SWITCHYARD, THE REACTOR SCRAMMED FROM 75% POWER. THE SCRAM TRIPPED THE FEEDWATER PUMPS. THE EMERGENCY FEEDWATER PUMPS STARTED AND FILLED THE STEAM GENERATORS TO THE 95% LEVEL AS DESIGNED. THIS HIGH WATER LEVEL PLUS NORMAL REQUIRED STEAM RESULTED IN A COOLDOWN RATE OF 140 F PER HOUR IN ONE LOOP AND 135.5 F PER HOUR IN ANOTHER WHICH EXCEEDS THE 100 F PER HOUR LIMIT. REDUCTION IN WATER LEVEL SET POINT IS BEING STUDIED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SCRAM, REAL + FAILURE, DESIGN ERROR + MAIN COOLING SYSTEM + STEAM GENERATOR + SET POINT + THERMAL TRANSIENT + OCONEE 2 (PWR)

√ ACCESSION NUMBER 89159 10-01 11/18/82

INADVERTENT SAFETY INJECTION SIGNAL AT INDIAN POINT 2
CONSOLIDATED EDISON COMPANY
2 PAGES, LETTER TO DIRECTORATE OF LICENSING (AEC), MARCH 5, 1974, DOCKET 50-247, TYPE--PWR, MFG--WEST., AE--UNITD ENGR
CAUSE - MAINTENANCE ERROR. DURING STEAM LINE PRESSURE ANALOG TESTING, AN INADVERTENT SAFETY INJECTION SIGNAL WAS GENERATED WHICH CAUSED THE ACCUMULATOR DISCHARGE STOP VALVES TO OPEN. THE REACTOR WAS IN THE COLD SHUTDOWN CONDITION. WHEN THE VALVES OPENED, THE SYSTEM WAS IN THE SOLID MODE AND PRESSURE ROSE TO 560 PSIG WHICH EXCEEDED THE 500 PSI LIMIT FOR THE SYSTEM TEMPERATURE. THE TRIP OCCURRED WHEN A BISTABLE WAS PLACED IN THE TRIP POSITION ACCORDING TO TEST PROCEDURES. AT THIS TIME 2 OTHER BISTABLES WERE TRIPPED, ONE OF WHICH HAD NOT BEEN RESET AFTER TESTING.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SAFETY INJECTION + INSTRUMENT, SWITCH + TEST, SYSTEM OPERABILITY + FAILURE, MAINTENANCE ERROR + INDIAN POINT 2 (PWR) + MAIN COOLING SYSTEM + PRESSURE TRANSIENT + ELECTRONIC FUNCTION UNITS

√ ACCESSION NUMBER 91182 10-01 11/18/82

ADDITIONAL INFORMATION ON COOLANT PUMP SEAL FAILURE AT OCONEE 2
DUKE POWER COMPANY
4 PAGES, LETTER TO DIRECTORATE OF LICENSING (AEC), APRIL 30, 1974, DOCKET 50-270, TYPE--PWR, MFG--B+W, AE--DUKE
CAUSE - EXCESS UPPER SEAL LEAKAGE. FAILURE OF A REACTOR COOLANT PUMP SEAL, REPORTED EARLIER, RESULTED WHEN THE UPPER SEAL LEAKAGE, COMBINED WITH THE SEAL STAGING FLOW, EXCEEDED THE CAPACITY OF THE PUMP HEAT EXCHANGERS. THE RESULTING THERMAL SHOCK FRACTURED THE ROTATING SEALRINGS AND OTHER COMPONENTS. MODIFICATIONS WERE MADE TO INCREASE THE CAPACITY OF THE HEAT EXCHANGER SYSTEM AND TO MONITOR SEAL LEAKAGE AND THE SEAL TEMPERATURE. ALSO, MODIFICATIONS WERE MADE TO ASSURE THAT THE IMPELLER CAP SCREW IS HELD IN POSITION BY THE ADDITION OF ANOTHER SPIRAL PIN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + THERMAL MECHANICAL EFFECT + MODIFICATION + FAILURE, DESIGN ERROR + SEAL + MAIN COOLING SYSTEM + LEAK RATE + SYSTEM CAPACITY + THERMAL TRANSIENT + OCONEE 2 (PWR) + PUMPS + HEAT EXCHANGERS

ACCESSION NUMBER 96427 10-01 11/18/82

MAXIMUM COOLDOWN RATE EXCEEDED AT RANCHO SECO
SACRAMENTO MUNICIPAL UTILITY DISTRICT, CALIFORNIA
5 PAGES, LETTER WITH ABNORMAL OCCURRENCE REPORT 74-6 TO DIRECTORATE OF LICENSING (AEC), OCT. 22, 1974, DOCKET 50-312, TYPE--PWR, MFG--B+W, AE--BECHTEL
CAUSE - PROCEDURE DEFICIENCY. DURING OPERATION AT 15% POWER, THE OPERATOR NOTICED THAT THE MAIN CONDENSER VACUUM WAS DECREASING. WHEN EFFORTS TO RESTORE THE VACUUM FAILED, THE CONDENSER BYPASS VALVE WAS CLOSED AND THE REACTOR WAS TRIPPED. AT THIS TIME THE ATMOSPHERIC RELIEF VALVES WERE ISOLATED. THE TRANSIENT WAS CONTROLLED BY THE SECONDARY SIDE PRESSURE RELIEF VALVES. AUXILIARY LOADS BEING SUPPLIED BY THE MAIN STEAM CAUSED EXCESSIVE COOLDOWN SINCE THERE WAS SO LITTLE DECAY HEAT AVAILABLE BECAUSE OF LOW OPERATING-TIME. THE AUXILIARY BOILER WAS STARTED TO RESTORE STABLE CONDITIONS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + PROCEDURES AND MANUALS + SCRAM, REAL + REACTOR STARTUP EXPERIENCE + FAILURE, ADMINISTRATIVE CONTROL + LEAK + CONDENSER + PRESSURE TRANSIENT + RANCHO SECO (PWR) + THERMAL TRANSIENT + PRESSURE RELIEF + VALVES + COOLING SYSTEM, SECONDARY + PSCDR

ACCESSION NUMBER 97128 10-01 11/18/82

COOLDOWN RATE EXCEEDS LIMITS AT RANCHO SECO
SACRAMENTO MUNICIPAL UTILITY DISTRICT, CALIFORNIA
1 PAGE, LETTER TO DIRECTORATE OF REGULATORY OPERATIONS (AEC), REGION V, OCTOBER 8, 1974, DOCKET 50-312, TYPE--PWR, MFG--B+W, AE--BECHTEL
CAUSE - LIFTING OF RELIEF VALVE. FOLLOWING A LOSS OF CONDENSER VACUUM AND THE SHUTTING OF THE TURBINE BYPASS VALVES, THE OPERATOR TRIPPED THE REACTOR FROM 15% POWER. IN THE ENSUING TRANSIENT, A MAIN STEAM RELIEF VALVE LIFTED AND BLEW DOWN FOR 5 MINUTES, RESULTING IN A RAPID COOLDOWN AND A DECREASE IN PRESSURIZER LEVEL. HIGH PRESSURE INJECTION WAS INITIATED MANUALLY WHICH CONTRIBUTED ADDITIONALLY TO THE COOLDOWN. OVER A 1-HR PERIOD, THE ACTUAL SYSTEM TEMPERATURE CHANGED ABOUT 110 F. THIS EXCEEDS THE 100 F LIMIT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE, EQUIPMENT + FAILURE + REACTOR, PWR + PRESSURE RELIEF + BLOWDOWN + RANCHO SECO (PWR) + THERMAL TRANSIENT + VALVES + COOLING SYSTEM, SECONDARY + ACTUATION + PSCDR

ACCESSION NUMBER 100433 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 2
COMMONWEALTH EDISON COMPANY, ILLINOIS
1 PAGE, LETTER TO AEC DIRECTORATE OF REGULATORY OPERATIONS, REGION 3, MARCH 7, 1975, DOCKET 50-304, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
CAUSE - BROKEN CONNECTOR ON RELAY. THE LEVEL CONTROL VALVE FOR A STEAM GENERATOR CLOSED DURING SAFEGUARDS TESTING. THE REACTOR TRIPPED FROM STEAM FLOW/FEED FLOW MISMATCH WITH LOW LEVEL. THE VALVE CLOSED AS A RESULT OF A BROKEN CRIMP CONNECTOR ON A RELAY. THE PRESSURE OSCILLATIONS THAT OCCURRED IN THE STEAM GENERATOR AS THE FEED RING WAS BEING COVERED ACTUATED SAFETY INJECTION. THE SAFETY INJECTION WAS TERMINATED WITHIN 30 SECONDS WHEN ANALYZED AS INADVERTANT. DURING NORMAL TRIPS, THE FEED RINGS ARE NOT UNCOVERED. VIBRATION CAUSED THE BROKEN CONNECTOR TO SEPARATE FROM THE RELAY. AVAL: PDR
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SCRAM, REAL + SAFETY INJECTION + FAILURE, INSTRUMENT + VIBRATION + STEAM GENERATOR + PRESSURE TRANSIENT + ZION 2 (PWR) + CABLES AND CONNECTORS + RELAYS + FEEDWATER + TRANSIENT

ACCESSION NUMBER 103077 10-01 11/18/82

REACTOR COOLANT PUMP SEAL SYSTEM FAILS AT ROBINSON 2
CAROLINA POWER & LIGHT COMPANY
8 PAGES, LETTER WITH AO 75-9 TO NRC DIR. OF REGULATORY OPERATIONS, REGION 2, MAY 12, 1975, DOCKET 50-261, TYPE--PWR, MFG--WEST., AE--EBASCO
CAUSE - NOT STATED. THE PLANT HAD JUST REACHED FULL POWER EARLY IN THE MORNING FOLLOWING A MAINTENANCE OUTAGE. THAT EVENING REACTOR COOLANT PUMP C SEAL LEAKOFF OSCILLATED FULL RANGE SEVERAL TIMES AND STABILIZED AT A FLOW ABOVE 6 GPM. A LOAD REDUCTION OF 10%/MIN WAS STARTED TO IDLE THE PUMP. A REACTOR TRIP OCCURRED. THE PUMP WAS LATER RESTARTED TO AID IN COOLDOWN. BOTH OTHER SEALS FAILED AT THIS TIME. FAILURE CAUSED OVERHEATING OF THE PUMP SHAFT AND SEAL HOUSING. THESE WILL BE REPLACED. PROPER OPERATION OF SAFEGUARDS EQUIPMENT AND ACTIONS TAKEN BY THE OPERATORS PREVENTED MORE EXTENSIVE DAMAGE AFTER THE SEAL FAILURE. ALL SYSTEMS FUNCTIONED AS DESIGNED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE, COMPONENT + FAILURE + REACTOR, PWR + REACTOR COOLANT + SAFETY INJECTION + INCIDENT + SEAL + ROBINSON 2 (PWR) + MAIN COOLING SYSTEM + LEAK + PRESSURE TRANSIENT + OPERATOR ACTION + REACTOR SHUTDOWN + PUMPS + PSCDA + PSCDR

ACCESSION NUMBER 103892 10-01 11/18/82

REACTOR COOLANT SYSTEM COOLDOWN RATE EXCESSIVE AT OCONEE 3
DUKE POWER COMPANY, N.C.
2 PAGES, LETTER WITH AOR 75-7 TO U.S. NRC, REGION 2, JUNE 27, 1975, DOCKET 50-287, TYPE--PWR, MFG--B&W, AE--DUKE
CAUSE - OPERATOR ERROR. DURING A ROUTINE SHUTDOWN FOR MAINTENANCE, A MINOR SYSTEM TRANSIENT OCCURRED WHICH RESULTED IN OPENING A POWER-ACTUATED PRESSURIZER RELIEF VALVE WITH REACTOR POWER AT 15%. THE VALVE REMAINED OPEN AND THE RC SYSTEM DEPRESSURIZATION CONTINUED UNTIL THE ISOLATION VALVE WAS CLOSED. THE SHUTDOWN CONTINUED WITH A COOLDOWN RATE OF 100 F/HR. HOWEVER, WHEN THE INITIAL DROP IN TEMPERATURE FROM DEPRESSURIZATION WAS INCLUDED THE RATE EXCEEDED THE 100 F/HR LIMIT BY 1 F/HR. BORIC ACID CRYSTAL BUILDUP ON THE CONNECTING PIN OF THE LEVER ARM OF THE PILOT VALVE CAUSED THE VALVE TO REMAIN OPEN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + FAILURE, OPERATOR ERROR + REACTOR, PWR + PRESSURE RELIEF + BLOWDOWN + MAIN COOLING SYSTEM + RATE + PRESSURE TRANSIENT + COOLING + OCONEE 3 (PWR) + VALVES + ACTUATION + PSCDA + PSCDR

ACCESSION NUMBER 104941 10-01 11/18/82

ADDITIONAL INFORMATION ON EXCESSIVE COOLDOWN RATE AT OCONEE 3
DUKE POWER CO., CHARLOTTE, N.C.
3 PAGES, LETTER TO NRC, REGION 2, AUG. 8, 1975, DOCKET 50-287, TYPE--PWR, MFG--B&W, AE--DUKE
CAUSE - RELIEF VALVE MALFUNCTION. REACTOR POWER WAS BEING REDUCED FROM 100% TO 15% BY THE INTEGRATED CONTROL SYSTEM FOR A MAINTENANCE SHUTDOWN. WHEN 15% WAS REACHED, UNIT LOAD DEMAND WAS 65 MWE AND POWER GENERATION WAS 115 MWE. THIS DIFFERENCE EXISTED BECAUSE THE REACTOR WAS OPERATING AT ITS LOWER LIMIT OF 15% AND COULD NOT FOLLOW LOAD DEMAND. A TRANSIENT OCCURRED THAT TRIPPED THE REACTOR. DURING THE TRANSIENT, A RELIEF VALVE OPENED AND FAILED TO CLOSE. THIS TRANSIENT WAS TERMINATED BY CLOSING THE ISOLATION VALVE. COOLDOWN RATE WAS 101 F/HR DURING THE FIRST HOUR. THE RELIEF VALVE FAILED BECAUSE OF HEAT EXPANSION, BORIC ACID CRYTAL BUILDUP ON THE VALVE LEVER, AND BENDING OF THE SOLENOID SPRING BRACKET.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + REACTOR TRANSIENT + SCRAM, REAL + STEAM + REACTOR CONTROL + PRESSURE RELIEF + FAILURE, MAINTENANCE ERROR + BLOWDOWN + THERMAL TRANSIENT + COOLING SYSTEM, SECONDARY + OCONEE 3 (PWR) + VALVES + ACTUATION + PSCDR

ACCESSION NUMBER 104981 10-01 11/18/82

HIGH PRESSURE SERVICE WATER PUMP FAILS TO START AT PEACH BOTTOM 2
PHILADELPHIA ELECTRIC COMPANY
2 PGS, LTR TO NRC DIV. OF REACTOR LICENSING, AUG. 8, 1975, DOCKET 50-277, TYPE--BWR, MFG--G.E., AE--BECHTEL
CAUSE - BREAKER CELL SWITCH SUPPORT BENT. WHILE TESTING THE HIGH PRESSURE SERVICE WATER SYSTEM WITH THE REACTOR AT 55% POWER, PUMP 2A FAILED TO START. THE BREAKER CELL SWITCH SUPPORT WAS BENT. THIS APPARENTLY OCCURRED DURING OPERATOR QUALIFICATION TESTS IN WHICH THE BREAKER WAS RACKED OUT AND IN. THE PUMP WAS NOT RUN AFTER THE BREAKER WAS RETURNED TO SERVICE BECAUSE NO WORK WAS ASSOCIATED WITH THIS TEST. THE SUPPORT WAS STRAIGHTENED AND THE PUMP TESTED SUCCESSFULLY.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

REACTOR, BWR + FAILURE + FAILURE, OPERATOR ERROR + TESTING + TEST, SYSTEM OPERABILITY + DAMAGE + AUXILIARY COOLING + PEACH BOTTOM 2 (BWR) + BREAKER + PUMPS

ACCESSION NUMBER 109619 10-01 11/18/82

SAFETY INJECTION ACTUATION OCCURS AT COOK 1
INDIANA & MICHIGAN POWER CO., BRIDGMAN, MI
3 PGS, LTR W/SPECIAL RPT SI-03 TO NRC OFFICE OF I & E, REGION III, DEC. 31, 1975, DOCKET 50-315, TYPE--PWR, MFG--WEST., AE--UTILITY
CAUSE - UNKNOWN. WHILE OPERATING AT 81% POWER WITH SYSTEM CONTROLS IN AUTOMATIC AND FUNCTIONING PROPERLY, ALL REACTOR PROTECTION SYSTEM STATUS LIGHTS FLASHED ON AND OFF SEVERAL TIMES. ON THE SECOND OR THIRD CYCLE REACTOR/TURBINE TRIP AND SAFETY INJECTION WITH STEAM LINE ISOLATION WAS VERIFIED. AFTER ASSURING THAT ALL CRITICAL PARAMETERS WERE NORMAL, SAFETY INJECTION WAS TERMINATED AFTER 900 GALLONS WERE INJECTED IN 3 MINUTES. THOROUGH CHECKING OF POWER SUPPLIES AND TESTING OF THE RPS DID NOT REVEAL ANY ABNORMALITIES. NO EVIDENCE OF DAMAGE WAS FOUND. THIS IS THE 3RD INADVERTENT ACTUATION OF THE SAFETY INJECTION SYSTEM.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + ACCIDENT, LOSS OF POWER + REACTOR, PWR + SAFETY INJECTION + SCRAM, SPURIOUS + COOK 1 (PWR) + THERMAL TRANSIENT + PSCDR

ACCESSION NUMBER 112149 10-01 11/18/82

SPECIAL REPORT ON SAFETY INJECTION AT COOK 1
INDIANA & MICHIGAN POWER CO., BRIDGMAN, MI
3 PGS, LTR W/RI-04 TO NRC OFFICE OF I & E, REGION III, MARCH 12, 1976, DOCKET 50-315, TYPE--PWR, MFG--WEST., AE--UTILITY
CAUSE - FAILURE TO FOLLOW PROCEDURES. DURING SURVEILLANCE TESTING WITH THE REACTOR AT 81% POWER, THE TECHNICIAN FAILED TO NOTIFY THE CONTROL ROOM OPERATOR TO HAVE THE CONTROLLING CHANNEL CHANGED FROM THE ONE TO BE TESTED TO THE STANDBY CHANNEL. TESTING OF THE STEAM FLOW SIGNAL THEN RESULTED IN A REACTOR-TURBINE TRIP AND INITIATION OF SAFETY INJECTION. THE CAUSE OF SAFETY INJECTION ACTUATION WAS NOT IMMEDIATELY APPARENT TO THE OPERATOR AND THE SYSTEM WAS ALLOWED TO OPERATE FOR 7 MINUTES PRIOR TO TERMINATION. THERE WAS NO EVIDENCE OF ANY MECHANICAL DAMAGE OR ABNORMAL PIPING MOVEMENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + PROCEDURES AND MANUALS + SCRAM, REAL + SAFETY INJECTION + TEST, SYSTEM OPERABILITY + FAILURE, MAINTENANCE ERROR + COOK 1 (PWR) + REACTOR SHUTDOWN + THERMAL TRANSIENT + COMMUNICATION

ACCESSION NUMBER 116240 10-01 11/18/82

POWER INCREASES ABOVE POWER LEVEL CUTOFF AT OCONEE 3
DUKE POWER CO., CHARLOTTE, NC
3 PGS, LTR W/RO-287/76-9 TO U.S. NRC, REGION II, JULY 27, 1976, DOCKET 50-287, TYPE--PWR, MFG--B&W, AE--DUKE
CAUSE - FEEDWATER TRANSIENT. THE REACTOR WAS OPERATING AT 81% FULL POWER UNTIL XENON CONDITIONS COULD ALLOW OPERATION ABOVE THE 82.5% POWER LEVEL CUTOFF. DURING THIS TIME, A FEEDWATER TRANSIENT CAUSED A POWER INCREASE OF 3%, THUS EXCEEDING THE CUTOFF LIMIT BY 1.5%. THE FEEDWATER TRANSIENT RESULTED FROM A SPURIOUS 80% DECREASE IN FEEDWATER DEMAND FOR 2 SECONDS. THIS INDUCED AN OSCILLATION IN THE FEEDWATER SYSTEM RESULTING IN AN INCREASE IN FEEDWATER FLOW AND LOWERING OF THE AVERAGE REACTOR TEMPERATURE. THE OPERATOR PROMPTLY PLACED FEEDWATER CONTROLS IN MANUAL AND DECREASED FLOW TO LOWER POWER BELOW THE CUTOFF LEVEL.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + XENON + REACTOR POWER + OPERATOR ACTION + TRANSIENT + FAILURE, INHERENT + COOLING SYSTEM, SECONDARY + COOLANT LEVEL TRANSIENT + FEEDWATER + OCONEE 3 (PWR)

ACCESSION NUMBER 117079 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 2
COMMONWEALTH EDISON CO., ZION, IL
2 PGS, LTR W/RPT TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, AUG. 13, 1976, DOCKET 50-304, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
CAUSE - STEAM GENERATOR WATER HAMMER. ON 6-10-76, UNIT 2 EXPERIENCED A PLANNED TURBINE TRIP FROM 100% POWER. THEN, WITH THE UNIT IN HOT SHUTDOWN, A SAFETY INJECTION OCCURRED DUE TO HIGH STEAMLINE DELTA P ON 2 LOOPS. THE SHOCK OF A WATER HAMMER IN A THIRD LOOP CAUSED THE 2 STEAMLINE TRANSMITTERS TO SPIKE HIGH. EXPERIMENTS HAVE SHOWN THAT A SHOCK WILL CAUSE THIS. SAFETY INJECTION PROCEEDED FOR ABOUT 15 SECONDS. NO DAMAGE OCCURRED. AUXILIARY FEEDWATER FLOW WILL BE LIMITED TO 50 GPM PER STEAM GENERATOR WHERE POSSIBLE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + HYDRAULIC EFFECT + REACTOR, PWR + SAFETY INJECTION + INSTRUMENT, SWITCH + INSTRUMENT, ABNORMAL INDICATION + STEAM GENERATOR + FLOW + ZION 2 (PWR) + AUXILIARY + COOLING SYSTEM, SECONDARY + FEEDWATER + SENSORS, PRESSURE + TRANSIENT

ACCESSION NUMBER 117080 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 2
COMMONWEALTH EDISON CO., ZION, IL
2 PGS, LTR W/RPT TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, AUG. 12, 1976, DOCKET 50-304, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
CAUSE - WATER HAMMER IN STEAM GENERATOR. ON 5-25-76, UNIT 2 WAS OPERATING AT 2760 MWT WHEN A REACTOR TRIP OCCURRED DUE TO FEEDWATER PUMP 2B TRIP. FOLLOWING THE TRIP, AUXILIARY FEEDWATER WAS BEING ADDED AT 100 GPM PER STEAM GENERATOR. ABOUT 30 MINUTES LATER WITH THE UNIT IN HOT SHUTDOWN, A SAFETY INJECTION OCCURRED DUE TO HIGH STEAM LINE DELTA P ON 2 LOOPS. THE SHOCK OF A WATER HAMMER ON A THIRD LOOP CAUSED THE 2 STEAMLINE TRANSMITTERS TO GIVE A FALSE READING BY SPIKING HIGH. SAFETY INJECTION PROCEEDED FOR 90 SECONDS. AUXILIARY FEEDWATER FLOW WILL BE LIMITED TO 50 GPM WHEN POSSIBLE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + HYDRAULIC EFFECT + REACTOR, PWR + SAFETY INJECTION + INSTRUMENT, SWITCH + INSTRUMENT, ABNORMAL INDICATION + STEAM GENERATOR + FLOW + ZION 2 (PWR) + AUXILIARY + COOLING SYSTEM, SECONDARY + FEEDWATER + SENSORS, PRESSURE

ACCESSION NUMBER 117634 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 1
COMMONWEALTH EDISON CO., ZION, IL
2 PGS, LTR W/RPT TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, AUG. 27, 1976, DOCKET 50-295, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
CAUSE - HYDRAULIC TRANSIENT. DURING A STARTUP WITH THE REACTOR AT 20% POWER, WHILE ATTEMPTING TO PLACE THE FEEDWATER FLOW CONTROLLERS IN AUTOMATIC, A SAFETY INJECTION OCCURRED DUE TO HIGH STEAMLINE DELTA P ON 2 LOOPS. A RAPID FEEDWATER FLOW INCREASE ON STEAM GENERATOR 1D EITHER OF ITSELF CAUSED A MECHANICAL SHOCK TO STEAMLINE PRESSURE TRANSMITTERS 1B AND 1D OR CAUSED A HYDRAULIC TRANSIENT IN SG 1D WITH SUBSEQUENT MECHANICAL SHOCK TO THE SAME TRANSMITTERS. THE SHOCK CAUSED THE TRANSMITTERS TO SPIKE HIGH RESULTING IN AN INDICATED HIGH DELTA P ON THE OTHER 2 TRANSMITTERS INITIATING SAFETY INJECTION.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + HYDRAULIC EFFECT + REACTOR, PWR + SAFETY INJECTION + STEAM GENERATOR + PRESSURE TRANSIENT + ZION 1 (PWR) + REACTOR SHUTDOWN + FEEDWATER + SENSORS, PRESSURE + TRANSIENT

ACCESSION NUMBER 118247 10-01 11/18/82

REACTOR SCRAMS AND SAFETY INJECTION OCCURS AT COOK I
INDIANA & MICHIGAN POWER CO., BRIDGMAN, MI
3 PGS, LTR W/SPECIAL RPT. SI-05 TO NRC OFFICE OF I & E, REGION III, SEPT. 1, 1976, DOCKET 50-315, TYPE--PWR, MFG--WEST., AE--UTILITY
CAUSE - LOSS OF A VITAL BUS CHANNEL. WHILE OPERATING AT 100% POWER IN THE AUTOMATIC MODE OF CONTROL, A REACTOR TRIP OCCURRED DUE TO FAILURE OF THE INVERTER FOR VITAL BUS CHANNEL I. WHEN THE CHANNEL WAS LOST, THE RPS RECEIVED A REACTOR COOLANT PUMP BREAKER OPEN INDICATION WHICH RESULTED IN THE REACTOR TRIP. A SAFETY INJECTION ACTUATION OF SAFEGUARDS TRAIN B AND STEAM LINE ISOLATION RESULTED FROM A FALSE INDICATION OF HIGH STEAM LINE FLOW ON 1 CHANNEL WITH A VERY MOMENTARY DROP IN TAVG TO THE LOW-LOW VALUE. SAFETY INJECTION WAS TERMINATED IN ABOUT 4 MINUTES. NO DAMAGE OCCURRED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE, EQUIPMENT + FAILURE + REACTOR, PWR + SCRAM, REAL + SAFETY INJECTION + ELECTRIC POWER, VITAL + COOK 1 (PWR) + REACTOR SHUTDOWN + TRANSIENT + GENERATORS

ACCESSION NUMBER 119312 10-01 11/18/82

SPECIAL REPORT ON SAFETY INJECTION ACTUATION AT COOK 1
INDIANA & MICHIGAN POWER CO., BIRDGMAN, MI
3 PGS, LTR W/SPECIAL RPT. SI-06 TO NRC OFFICE OF I & E, REGION III, SEPT. 30, 1976, DOCKET 50-315, TYPE--PWR, MFG--WEST., AE--UTILITY
CAUSE - UNKNOWN. WHILE PREPARING FOR A STARTUP AND WITH THE REACTOR IN HOT STANDBY, A SAFETY INJECTION ACTUATION OCCURRED ON SAFEGUARDS TRAIN A FOR NO APPARENT REASON. NO STATUS LIGHTS WERE RECEIVED AND NO REACTOR TRIP OCCURRED. IT WAS FELT THAT THE PROBLEM WAS ASSOCIATED WITH THE MASTER RELAY DRIVER CARD. THIS CARD WAS REPLACED. ABOUT 600 GALLONS OF BORATED WATER AT 165 F INITIALLY WERE INJECTED INTO THE SYSTEM. THIS IS THE SIXTH INADVERTANT ACTUATION OF THE SAFETY INJECTION SYSTEM. CONSERVATIVELY IT CONSTITUTES 1/50 OF ALLOWABLE THERMAL CYCLES.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SAFETY INJECTION + MAIN COOLING SYSTEM + COOK 1 (PWR) + THERMAL TRANSIENT + ELECTRONIC FUNCTION UNITS + PSCDR

ACCESSION NUMBER 120766 10-01 11/18/82

REACTOR COOLANT SYSTEM TEMPERATURE DECREASES AT CALVERT CLIFFS 2
BALTIMORE GAS & ELECTRIC CO., BALTIMORE, MD
3 PGS, LTR W/LER 76-10/3L TO NRC OFFICE OF I & E, REGION I, DEC. 27, 1976, DOCKET 50-318, TYPE--PWR, MFG--COMB., AE--BECHTEL
CAUSE - LACK OF OPERATOR EXPERIENCE. DURING ESCALATION TO POWER TESTING, THE MAIN TURBINE GENERATOR WAS PARALLELED. TWENTY-SEVEN MINUTES LATER LOAD WAS REDUCED IN RESPONSE TO DECREASING REACTOR COOLANT SYSTEM TEMPERATURE. RAPID FEEDING OF THE STEAM GENERATORS TO COMPENSATE FOR LEVEL SHRINK CAUSED T AVG TO DECREASE TO 503 F. T AVG WAS RESTORED TO GREATER THAN 515 F WITHIN 10 MINUTES. LACK OF OPERATOR EXPERIENCE WITH A SLIGHTLY POSITIVE TEMPERATURE COEFFICIENT AND THE SINGLE ELEMENT MODE OF STEAM GENERATOR CONTROL CONTRIBUTED TO THIS EVENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

CALVERT CLIFFS 2 (PWR) + REACTOR, PWR + REACTOR STARTUP EXPERIENCE + FAILURE + THERMAL TRANSIENT + MAIN COOLING SYSTEM + FAILURE, OPERATOR ERROR + CONTROL + STEAM GENERATOR

ACCESSION NUMBER 120769 10-01 11/18/82

REACTOR COOLANT SYSTEM DROPS BELOW LIMIT AT CALVERT CLIFFS 2
BALTIMORE GAS & ELECTRIC CO., BALTIMORE, MD
3 PGS, LTR W/LER 76-11/3L TO NRC OFFICE OF I & E, REGION I, DEC. 27, 1976, DOCKET 50-318, TYPE--PWR, MFG--COMB., AE--BECHTEL
CAUSE - PLANT CONTROL AT LOW POWER. WITH THE REACTOR AT 19% POWER, CORE POWER WAS DECREASING, DUE TO XENON BUILDIN AND COMPENSATING BORON DILUTIONS WERE BEING MADE TO STABILIZE POWER. AT THIS TIME A POSITIVE MODERATOR TEMPERATURE COEFFICIENT EXISTED CAUSING CONTROL BY DILUTION TO BE DIFFICULT TO PREDICT. OVER A PERIOD OF TIME THE DILUTIONS WERE NOT SUFFICIENT AND T AVG AND STEAM GENERATOR LEVEL DECREASED. STEAM GENERATOR FEEDING WAS MADE TOO FAST DECREASING T AVG BELOW THE LIMIT. TURBINE LOAD WAS DECREASED AND T AVG RETURNED TO WITHIN LIMITS. A SECOND HEAVY FEEDING CAUSED T AVG TO DROP AGAIN. THE OPERATOR SCRAMMED THE REACTOR.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

CALVERT CLIFFS 2 (PWR) + REACTOR, PWR + REACTOR STARTUP EXPERIENCE + FAILURE + THERMAL TRANSIENT + MAIN COOLING SYSTEM + CONTROL + STEAM GENERATOR + FAILURE, OPERATOR ERROR + REACTOR SHUTDOWN + SCRAM, REAL + REACTOR CONTROL

ACCESSION NUMBER 120818 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 1
COMMONWEALTH EDISON CO., ZION, IL
2 PGS, LTR W/RPT TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, DEC. 13, 1976, DOCKET 50-295, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
CAUSE - WATER HAMMER. ON 9/25/76 WHEN 8 HANGERS ON FEEDWATER PIPE 1D WERE FOUND BROKEN, THE UNIT WAS BROUGHT TO HOT SHUTDOWN. THE NEXT DAY WHILE ATTEMPTING TO UNISOLATE THE FEEDWATER LINE TO BRING THE UNIT TO COLD SHUTDOWN, A WATER HAMMER OCCURRED WHEN COOL WATER ENTERED THE HOT FEEDWATER LINE. THE WATER HAMMER CAUSED 2 PRESSURE TRANSMITTERS TO SPIKE HIGH RESULTING IN A SAFETY INJECTION INITIATION. AFTER 2 MINUTES, WHEN ASSURED OF STABLE PLANT PARAMETERS, THE OPERATOR TERMINATED THE INJECTION.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

ZION 1 (PWR) + REACTOR, PWR + FAILURE + SAFETY INJECTION + HYDRAULIC EFFECT + PRESSURE TRANSIENT + FEEDWATER + TRANSIENT

ACCESSION NUMBER 120819 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 2
COMMONWEALTH EDISON CO., ZION, IL
2 PGS, LTR W/RPT TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, DEC. 8, 1976, DOCKET 50-304, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
CAUSE - SWITCHING ERROR. ON 9/19/76 WHILE OPERATING AT 30% POWER, A SWITCHING ERROR OCCURRED WHILE TAKING BATTERY 211 OFF EQUALIZING CHARGE RESULTING IN BUS 211 BEING DE-ENERGIZED. THIS CAUSED 2 REACTOR COOLANT PUMPS TO TRIP RESULTING IN A REACTOR TRIP/TURBINE TRIP. DUE TO THE LOSS OF BUS 211, THE TURBINE DRIVEN FEEDWATER PUMP DID NOT TRIP. THE RAPID ADDITION OF FEEDWATER CAUSED A PRESSURE DROP IN 2 STEAM GENERATORS AND A SAFETY INJECTION. THIS COULD NOT BE VERIFIED FROM FIRST OUT ANNUNCIATORS SINCE THEY ARE FED FROM BUS 211. WHEN POWER WAS RESTORED TO BUS 211 30 MINUTES LATER, IT WAS DETERMINED THAT AN SI INCIDENT HAD NOT OCCURRED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

ZION 2 (PWR) + REACTOR, PWR + FAILURE + SAFETY INJECTION + PRESSURE TRANSIENT + SCRAM, REAL + FAILURE, OPERATOR ERROR + FEEDWATER + TRANSIENT + PSCDR

ACCESSION NUMBER 122385 10-01 11/18/82

SPECIAL REPORT ON SAFETY INJECTION ACTUATION AT TROJAN
PORTLAND GENERAL ELECTRIC, PORTLAND, OR
2 PGS, LTR W/RPT TO NRC REGION V, FEB. 2, 1977, DOCKET 50-344, TYPE--PWR, MFG--WEST., AE--BECHTEL
DATE OF EVENT - 102876. POWER LEVEL - 0%. CAUSE - OPERATOR ERROR. SAFETY INJECTION OCCURRED AS A RESULT OF ADDING CHEMICAL TOO FAST TO STEAM GENERATOR 4 WITH ONLY ONE REACTOR COOLANT PUMP OPERATING. THE RAPID ADDITION CAUSED A LOW STEAM LINE PRESSURE IN SG4 WITH RESPECT TO SGS 2 AND 3. SAFETY INJECTION IS INITIATED AT A LOW STEAM LINE PRESSURE SET POINT OF 100 PSI IN ONE SG WITH RESPECT TO ANY 2 OF THE REMAINING 3 SGS. ALL PLANT ECCS FEATURES PERFORMED THEIR SAFETY FUNCTIONS AS DESIGNED EXCEPT FOR 2 VALVES (REPORTED ON 11/11/76). THE OPERATOR CONTROLLING FEEDWATER FLOW DURING THE ADDITION MADE AN ERROR. HE WAS REPRIMANDED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

TROJAN (PWR) + REACTOR, PWR + FAILURE + SAFETY INJECTION + PRESSURE TRANSIENT + STEAM GENERATOR + RATE + ADDITIVE + FAILURE, OPERATOR ERROR + FEEDWATER + TRANSIENT + PSCDR

ACCESSION NUMBER 123150 10-01 11/18/82

EXCESS COOLDOWN RATE RESULTS FROM LOSS OF ICS POWER AT CRYSTAL RIVER 3
FLORIDA POWER CORP., ST. PETERSBURG, FL
3 PGS, LTR W/LER 77-20 TO NRC OFFICE OF I & E, REGION II, MARCH 10, 1977, DOCKET 50-302, TYPE--PWR, MFG--B&W, AE--GILBERT ASSOC.
DATE OF EVENT - 030277. POWER LEVEL - 40%. CAUSE - INVERTER OUTPUT DIODE FAILS. DURING NORMAL OPERATION, LOSS OF AC INVERTER B WAS CAUSED BY LOSS OF VITAL AC BUS B. THIS CAUSED A REACTOR TRIP, TURBINE TRIP, AND THE ATMOSPHERIC STEAM DUMP VALVES TO OPEN 50%. THE COOLDOWN RATE WAS ABOUT 164 F IN 15 MINUTES. AN OUTPUT DIODE FAILED IN AC INVERTER B CAUSING THE CHAIN OF EVENTS. THE DIODE WAS RENEWED AND THE INVERTER RESTORED TO OPERATING CONDITION. THE RCS REMAINED ACCEPTABLE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CRYSTAL RIVER 3 (PWR) + THERMAL TRANSIENT + ELECTRIC POWER + SCRAM, REAL + REACTOR SHUTDOWN + MAIN COOLING SYSTEM + FAILURE, COMPONENT + GENERATORS + SOLID STATE DEVICE + FAILURE, INHERENT + PRESSURE RELIEF + REACTOR STARTUP EXPERIENCE + PSCDA + PSCDR

ACCESSION NUMBER 12377B 10-01 11/18/82

SAFETY INJECTION OCCURS AT ZION 2
COMMONWEALTH EDISON CO., ZION, IL
2 PGS, LTR W/RPT TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, MARCH 11, 1977, DOCKET 50-304, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
DATE OF EVENT - 122177. POWER LEVEL - 75%. CAUSE - LEAKING VALVES. WHILE OPERATING AT 75% POWER DURING TESTING THE BIT ISOLATION VALVES, BIT INLET VALVE 2MOV-SI8803A WAS STROKED OPEN BUT FAILED TO CLOSE. THIS FAILURE ALONG WITH THE BIT OUTLET VALVES 2MOV-SI8801A AND/OR B LEAKING THROUGH CAUSED BORATION OF THE REACTOR COOLANT SYSTEM CAUSING POWER TO DECREASE AND HENCE TAVG TO DECREASE. THE REACTOR WAS TRIPPED MANUALLY DUE TO TAVG DECREASING RAPIDLY. A SAFETY INJECTION FOLLOWED DUE TO HIGH STEAM FLOW AND LO-LO TAVG. THE CAUSE WAS DUE TO THE HIGH STEAM FLOW SET POINT BEING AUTOMATICALLY RESET TO 40% ON A REACTOR TRIP.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 2 (PWR) + SAFETY INJECTION + FAILURE, EQUIPMENT + VALVES + LEAK + CHEMICAL SHIM + TEST, SYSTEM OPERABILITY + SCRAM, REAL + REACTOR SHUTDOWN + THERMAL TRANSIENT + FAILURE, INHERENT + CONTAINMENT ISOLATION

ACCESSION NUMBER 124920 10-01 11/18/82

EXCESSIVE COOLDOWN RATE EXPERIENCED DURING TESTING AT CRYSTAL RIVER 3
FLORIDA POWER CORP., ST. PETERSBURG
4 PPS, LTR W/LER 77-29 TO NRC OFFICE OF I & E, REGION II, APRIL 21, 1977,
DOCKET 50-302, TYPE--PWR, MFG--B&W, AE--GILBERT ASSOC.
DATE OF EVENT - 041677. POWER LEVEL - 0%. CAUSE - HIGH LEVEL IN STEAM GENERATORS. DURING A TEST OF SHUTDOWN FROM OUTSIDE THE CONTROL ROOM, AN EXCESSIVE COOLDOWN RATE OF 101 F IN 20 MINUTES WAS EXPERIENCED. THIS WAS PRIMARILY CAUSED BY OVERFILLING OF THE STEAM GENERATORS. THE OVERFILLING WAS DUE TO AN OUT-OF-CALIBRATION PRESSURE TRANSMITTER NOT CAUSING STARTUP FEEDWATER VALVES FWV-39 AND -40 TO FULLY CLOSE. THE TRANSMITTER WAS CALIBRATED. PROCEDURES WERE REVISED TO TRIP THE FEED PUMP WHEN APPROPRIATE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CRYSTAL RIVER 3 (PWR) + THERMAL TRANSIENT + MAIN COOLING SYSTEM + TESTING + FAILURE, INSTRUMENT + INSTRUMENT CALIBRATION + SYSTEM CAPACITY + FAILURE, INHERENT + STEAM GENERATOR + PSCDA + PSCDR

ACCESSION NUMBER 126585 10-01 11/18/82

REACTOR COOLANT PUMP SEAL PACKAGE FAILS AT INDIAN POINT 2
CONSOLIDATED EDISON CO. OF NEW YORK, INC., NEW YORK, NY
3 PGS, LTR W/RO-77-2-14(A) TO NRC OFFICE OF I & E, REGION I, JULY 15, 1977, DOCKET 50-247, TYPE--PWR, MFG--WEST., AE--UE&C
DATE OF EVENT - 070277. POWER LEVEL - 2%. CAUSE - NOT YET DETERMINED. DURING A REACTOR STARTUP, ALARMS INDICATED A FAILURE OF THE SEAL PACKAGE OF REACTOR COOLANT PUMP 23. A SECOND CHARGING PUMP WAS PLACED IN SERVICE TO COMPENSATE FOR THE RESULTANT DECREASE IN PRESSURIZER LEVEL. THE PUMP WAS TRIPPED AND THE REACTOR WAS SHUT DOWN. COOLDOWN LIMITS WERE EXCEEDED BY 5 F FOR 1.3 HOURS. TOTAL LEAKAGE TO CONTAINMENT ABOUT 90000 GALLONS WITH A MAXIMUM LEAK RATE OF 75 GPM. REPAIRS ARE IN PROGRESS. THE EXTENT OF DAMAGE TO THE PUMP IS NOT KNOWN AT THIS TIME.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + INDIAN POINT 2 (PWR) + SEAL + PUMPS + MAIN COOLING SYSTEM + LEAK + FAILURE, COMPONENT + REACTOR SHUTDOWN + THERMAL TRANSIENT + REACTOR, PWR + PSCDR

ACCESSION NUMBER 134324 10-01 11/18/82

VITAL BUS DE-ENERGIZED RESULTING IN REACTOR TRIP AND SAFETY INJECTION AT BEAVER VALLEY 1
DUQUESNE LIGHT CO., PITTSBURGH, PA
3 PGS, LTR W/LER 78-2/03L TO U.S. NRC, REGION I, FEB. 3, 1978, DOCKET 50-334, TYPE--PWR, MFG--WEST., AE--STONE & WEBSTER
DATE OF EVENT - 010278. POWER LEVEL - 100%. CAUSE - INVERTER MALFUNCTION. THE OUTPUT FREQUENCY OF INVERTER 4 WAS DISCOVERED OSCILLATING. VITAL BUS 4 WAS TRANSFERRED TO THE AUXILIARY SUPPLY BUT THE TRANSFER WAS DONE OUT OF PHASE AND THE SUPPLY BREAKER TRIPPED. LOSS OF VITAL BUS 4 CAUSED HIGH STEAM FLOW TRIPS ON 3 STEAM LINES. WHEN BUS 4 WAS RETURNED TO THE INVERTER, A VOLTAGE SURGE CAUSED FALSE LOW LOW TAVG SIGNAL WHICH RESULTED IN A REACTOR TRIP AND SAFETY INJECTION. THIS TRANSIENT RESULTED IN CONDENSER CONDENSATE AND STEAM GENERATOR CATION CONDUCTIVITIES EXCEEDING THE LIMITS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

REACTOR, PWR + BEAVER VALLEY I (PWR) + SCRAM, REAL + SAFETY INJECTION + ELECTRIC POWER + ELECTRIC POWER, VITAL + COOLANT QUALITY + COOLING SYSTEM, SECONDARY + TRANSIENT + FAILURE, EQUIPMENT + FAILURE, INHERENT + ON SITE + ELECTRIC POWER, VITAL + FAILURE

ACCESSION NUMBER 137296 1111-01 1 03/07/83

LOW PRESSURIZER PRESSURE AND LEVEL CAUSES SAFETY INJECTION AT BEAVER VALLEY I
DUQUESNE LIGHT CO., PITTSBURGH, PA
3 PGS, LTR W/LER 78-011/99X TO NRC DIRECTOR OF REGULATION, REGION I, MARCH 28, 1978, DOCKET 50-334, TYPE--PWR, MFG--WEST., AE--STONE & WEBSTER
DATE OF EVENT - 012478. POWER LEVEL - 0%. CAUSE - OPERATOR ERROR. AT 0348 HOURS, DURING NORMAL OPERATION, A REACTOR TRIP OCCURRED DUE TO A TURBINE TRIP. THE AUXILIARY FEEDWATER PUMPS AUTO STARTED ON LOW-LOW STEAM GENERATOR LEVEL. AT 0351 HOURS, A SAFETY INJECTION OCCURRED AS A RESULT OF LOW PRESSURIZER PRESSURE AND LEVEL. THE PRIMARY CAUSE OF THE SI WAS FAILURE OF THE OPERATOR TO ISOLATE STEAM TO THE MSRS IMMEDIATELY AFTER THE REACTOR TRIP. SECONDARY CAUSES WERE INTRODUCTION OF COLD AUXILIARY FEEDWATER AND A LEAKING PRESSURIZER SPRAY VALVE. THE OPERATOR INVOLVED WAS REPRIMANDED. THE PRESSURIZER SPRAY VALVE WILL BE REPLACED. AUTOMATIC MSR ISOLATION WILL BE IMPLEMENTED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + BEAVER VALLEY 1 (PWR) + SAFETY INJECTION + FAILURE, OPERATOR ERROR + OPERATOR ACTION + TRANSIENT + LICENSED OPERATOR

ACCESSION NUMBER 137918 10-01 11/18/82

REACTOR COOLDOWN RATE EXCEEDS LIMIT AT THREE MILE ISLAND 2
METROPOLITAN EDISON CO., READING, PA
5 PGS, LTR W/LER 78-33/1T TO NRC OFFICE OF I & E, REGION I, MAY 8, 1978, DOCKET 50-320, TYPE--PWR, MFG--B&W, AE--BURNS & ROE
DATE OF EVENT - 042378. POWER LEVEL - 30%. CAUSE - RELIEF VALVES FAIL TO RESEAT CORRECTLY. THE REACTOR TRIPPED FROM 30% R.T.P. FOLLOWING THE TRIP, THE REACTOR COOLANT SYSTEM RAPIDLY COOLED DOWN AND DEPRESSURIZED. DURING THE DEPRESSURIZATION, SAFETY INJECTION WAS INITIATED. DURING THE TRANSIENT, RCS AND PRESSURIZER COOLDOWN RATES WERE EXCEEDED AND PRESSURIZER VOLUME WAS BELOW TECHNICAL SPECIFICATION LIMITS. THE CORE REMAINED COVERED AT ALL TIMES AND NO RELEASE OF RADIOACTIVE MATERIAL RESULTED. THE MAIN STEAM RELIEF VALVES DID NOT RESEAT AT THE CORRECT PRESSURE. THE FEEDWATER SYSTEM RESPONSE WAS SLOW SINCE THE INITIAL INTEGRATED CONTROL SYSTEM TESTING WAS STILL IN PROGRESS. NO CAUSE FOR EXCESSIVE BLOWDOWN WAS DISCOVERED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + THREE MILE ISLAND 2 (PWR) + REACTOR STARTUP EXPERIENCE + REACTOR TRANSIENT + THERMAL TRANSIENT + MAIN COOLING SYSTEM + SYSTEM CAPACITY + PRESSURIZER + BLOWDOWN + VALVES + PRESSURE RELIEF + FAILURE, EQUIPMENT + FAILURE, INSTALLATION ERROR + SAFETY INJECTION + RATE + COOLING + ACTUATION + PSCDA + PSCDR

ACCESSION NUMBER 137919 10-01 11/18/82

STEEL LINERS IN MAIN STEAM LINE RELIEF VALVE DISCHARGE FAIL AT THREE MILE ISLAND 2
METROPOLITAN EDISON CO., READING, PA
3 PGS, LTR W/LER 78-34/1T TO NRC OFFICE OF I & E, REGION I, MAY 8, 1978, DOCKET 50-320, TYPE--PWR, MFG--B&W, AE--BURNS & ROE
DATE OF EVENT - 042478. POWER LEVEL - 30%. CAUSE - DESIGN DEFICIENCY. A REACTOR TRIP OCCURRED CAUSING SEVERAL MAIN STEAM RELIEF VALVES TO LIFT. DUE TO INADEQUATE DESIGN, SEVERAL STEEL LINERS USED TO PROTECT THE BELLOWS IN THE DISCHARGE PIPING WERE EJECTED. THIS EVENT WAS ON THE SECONDARY SIDE OF THE PLANT. AN INADEQUATE DESIGN ALLOWED THE USE OF BELLOWS LINERS QUALIFIED FOR 180 F/SEC. STEAM VELOCITY IN A LOCATION WHERE VELOCITIES UP TO 950 F/SEC. COULD BE EXPERIENCED. THE UNDER-DESIGNED BELLOWS LINERS ARE BEING REPLACED BY LINERS OF AN APPROPRIATE DESIGN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555, (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + THREE MILE ISLAND 2 (PWR) + REACTOR STARTUP EXPERIENCE + STEEL LINER + VALVES + COOLING SYSTEM, SECONDARY + FAILURE, COMPONENT + FAILURE, DESIGN ERROR + DISCHARGE + PRESSURE RELIEF + ACTUATION

ACCESSION NUMBER 138830 10-01 11/18/82

REACTOR COOLDOWN RATE EXCEEDS LIMIT FOLLOWING TRIP AT RANCHO SECO
SACRAMENTO MUNICIPAL UTILITY DISTRICT, SACRAMENTO, CA
6 PGS, LTR W/RO 78-1 TO NRC DIRECTOR OF REGULATORY OPERATIONS, REGION V, MARCH 31, 1978, DOCKET 50-312, TYPE--PWR, MFG--B&W, AE--BECHTEL
DATE OF EVENT - 032078. POWER LEVEL - 70%. CAUSE - OPERATOR ERROR. AN OPERATOR CHANGING A LIGHTBULB IN A BACKLIGHTED PUSHBUTTON IN THE CONTROL ROOM, SHORTED OUT DC POWER TO THE PLANT NON-NUCLEAR INSTRUMENTATION. LOSS OF ABOUT 2/3 OF THE TEMPERATURE, PRESSURE, FLOW AND LEVEL SIGNALS RESULTED IN A REACTOR TRIP AND SUBSEQUENT RCS COOLDOWN AT 300 F/HR. THE LOSS OF NNI CAUSED INVALID SYSTEMS TO BE DISPLAYED IN THE CONTROL ROOM AND USED AS INPUTS TO THE ICS. FOLLOWING THE REACTOR TRIP, LACK OF RCS TEMPERATURE INDICATION RESULTED IN RAPID COOLDOWN BECAUSE OPERATORS ALLOWED AUXILIARY FEEDWATER TO RUN UNTIL NNI WAS RESTORED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + RANCHO SECO (PWR) + RATE + COOLING + MAIN COOLING SYSTEM + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + REACTOR SHUTDOWN + ELECTRIC POWER + INSTRUMENTS, MISC. + CONTROL SYSTEM + INDICATORS + TEMPERATURE + PSCDA + PSCDR

ACCESSION NUMBER 139563 10-01 11/18/82

REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE DROPS BELOW LIMIT AT ST. LUCIE 1
FLORIDA POWER & LIGHT CO., MIAMI, FL
2 PGS, LTR W/RO 335-78-19 TO NRC OFFICE OF I & E, REGION II, JUNE 28, 1978, DOCKET 50-335, TYPE--PWR, MFG--COMB., AE--EBASCO
DATE OF EVENT - 053078. POWER LEVEL - 3%. CAUSE - FEEDWATER CONTROL SYSTEM DEFICIENCY. DURING A SPECIAL LOW POWER (ABOUT 3%) FEEDWATER CONTROL SYSTEM TEST, A 15% FEEDWATER BYPASS CONTROL VALVE INADVERTENTLY OPENED FULLY CAUSING EXCESSIVE COOLING OF 1A STEAM GENERATOR. REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE DROPPED SLIGHTLY BELOW 515 F (ABOUT 1 F). THE REACTOR WAS TRIPPED MANUALLY WHEN THIS OCCURRED. A VENDOR PROPOSED MODIFICATION OF THE 15% FEEDWATER CONTROL SYSTEM WAS TEMPORARILY INSTALLED ON THE A LOOP FOR FEASIBILITY TESTING (TO IMPROVE STABILITY OF THE AUTOMATIC MODE). GAIN SETTINGS BASED ON ANALYTICAL RESULTS PROVED TO BE UNREALISTIC AND CAUSED THE BYPASS VALVE TO OPEN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ST. LUCIE 1 (PWR) + TEMPERATURE + MAIN COOLING SYSTEM + THERMAL TRANSIENT + FEEDWATER + COOLING SYSTEM, SECONDARY + CONTROL SYSTEM + MODIFICATION + FAILURE, INSTRUMENT + SET POINT + SCRAM, REAL + REACTOR SHUTDOWN + FAILURE, DESIGN ERROR + PSCDR

ACCESSION NUMBER 139931 10-01 11/18/82

SPECIAL REPORT ON SAFETY INJECTION ACTUATION AT THREE MILE ISLAND 2
METROPOLITAN EDISON CO., READING, PA
5 PGS, LTR W/RPT TO NRC OFFICE OF I & E, REGION I, JULY 24, 1978, DOCKET 50-320, TYPE--PWR, MFG--B&W, AE--BURNS & ROE
ON 4-23-78, UNIT 2 EXPERIENCED A REACTOR TRIP WHILE AT 30% POWER. FOUR OF THE SIX MAIN STEAM RELIEF VALVES LIFTED ON THE B STEAM GENERATOR. ONE MAIN STEAM RELIEF VALVE ON THE A STEAM GENERATOR LIFTED. THE 4 B SIDE MAIN STEAM SAFETY VALVES AND THE 1 A SIDE VALVE FAILED TO PROPERLY RESEAT. THE SAFETY VALVES FAILING TO RESEAT AT THE PROPER PRESSURE COUPLED WITH OVERFEEDING THE STEAM GENERATORS CAUSED A RAPID DEPRESSURIZATION AND COOLDOWN OF THE REACTOR COOLANT SYSTEM. THE REACTOR COOLANT TEMPERATURE VARIED FROM 583 DEG F TO 464 DEG F IN 3 MINUTES. SAFETY INJECTION OCCURRED. THIS INJECTED NAOH INTO THE REACTOR COOLANT SYSTEM THROUGH THE HIGH PRESSURE INJECTION LINES.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

THREE MILE ISLAND 2 (PWR) + REACTOR, PWR + REACTOR STARTUP EXPERIENCE + SAFETY INJECTION + PRESSURE TRANSIENT + THERMAL TRANSIENT + CONTROL + FEEDWATER + COOLING SYSTEM, SECONDARY + FAILURE, EQUIPMENT + FAILURE, MAINTENANCE ERROR + SODIUM + HYDROXIDE + FAILURE + PRESSURE RELIEF + VALVES + PSCDR

ACCESSION NUMBER 142387 10-01 11/18/82

INSURGE OF FEEDWATER CAUSES DECREASE IN PRESSURIZER PRESSURE AT SALEM 1
PUBLIC SERVICE ELECTRIC & GAS CO., NEWARK, NJ
4 PGS, LTR W/RO 78-06/03L TO NRC OFFICE OF I & E, REGION I, FEB. 24, 1978, DOCKET 50-272, TYPE--PWR, MFG--WEST., AE--PSE&G
DATE OF EVENT - 013078. POWER LEVEL - 22%. CAUSE - OPERATOR ERROR. TRANSFER OF NO. 11 SGFP TO AUTOMATIC CAUSED SUDDEN INCREASE IN S/G LEVELS. THE INSURGE OF FEEDWATER CAUSED TAVE REDUCTION AND PRESSURIZER PRESSURE REDUCTION. PRESSURIZER PRESSURE DECREASED TO 2165 PSIA WHICH IS LESS THAN THE 2220 PSIA ALLOWED BY TECH SPEC. PRESSURIZER HEATERS RECOVERED THE PRESSURE IN 4 MINUTES. THE CAUSE WAS PERSONNEL ERROR IN THAT THE OPERATOR FAILED TO MATCH THE MASTER DEMAND TO THE PUMP SPEED PRIOR TO PLACING THE MASTER CONTROLLER TO AUTOMATIC. THE OPERATOR WAS REINSTRUCTED BY THE SHIFT SUPERVISOR.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 1 (PWR) + PRESSURE, INTERNAL + PRESSURIZER + MAIN COOLING SYSTEM + FEEDWATER + FLOW + RATE + CONTROL SYSTEM + COOLING SYSTEM, SECONDARY + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + TRANSIENT

ACCESSION NUMBER 142480 10-01 11/18/82

INADVERTANT SAFETY INJECTION OCCURS AT ZION 1
COMMONWEALTH EDISON CO., ZION, IL
4 PGS, LTR W/RO 50-295/78-95 TO NRC DIRECTORATE OF REGULATORY OPERATIONS, REGION III, DEC. 11, 1978, DOCKET 50-295, TYPE--PWR, MFG--WEST., AE--SGT & LUNDY
DATE OF EVENT - 091478. POWER LEVEL - 0%. CAUSE - FALSE DELTA STEAM PRESSURE SIGNAL. 12 MINUTES AFTER A REACTOR TRIP, AN INADVERTANT SAFETY INJECTION OCCURRED FROM A FALSE DELTA STEAM PRESSURE SIGNAL. THE FALSE SIGNAL WAS GENERATED FROM THE SHOCK OF A FEED WATER HAMMER. THE WATER HAMMER OCCURRED PRIOR TO RECOVERING THE FEED RING WITH A MODERATE FLOW.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 1 (PWR) + SAFETY INJECTION + PRESSURE PULSE + HYDRAULIC EFFECT + FEEDWATER + COOLING SYSTEM, SECONDARY + TRANSIENT

ACCESSION NUMBER 142627 10-01 11/18/82

UNSTABLE FEEDWATER CONTROL OCCURS AT COOK 2
INDIANA & MICHIGAN POWER CO., BRIDGMAN, MI
2 PGS, LTR W/RO 78-095/03L-0 TO NRC OFFICE OF I & E, REGION III, DEC. 20, 1978, DOCKET 50-316, TYPE--PWR, MFG--WEST., AE--UTILITY
DATE OF EVENT - 120678. POWER LEVEL - 100%. CAUSE - FAULTY HEATER DRAIN VALVE. TROUBLE DEVELOPED WITH 5-A HIGH PRESSURE HEATER LEVEL CONTROL. THE FEEDWATER CONTROL TO THE STEAM GENERATORS BECAME UNSTABLE. AN EMERGENCY LOAD REDUCTION OF 10% WAS PERFORMED. WHILE DECREASING THE TURBINE LOAD, THE REACTOR COOLANT AVERAGE TEMPERATURE EXCEEDED 576.2 F FOR 20 MINUTES. THE CAUSE WAS A FAULTY DRAIN VALVE ON 5-A HEATER, CREATING UNSTABLE FEEDWATER CONTROL AND TURBINE LOAD. TURBINE LOAD WAS LOWERED RAPIDLY CAUSING THE TEMPERATURE TRANSIENT OF THE REACTOR COOLANT. THE TRANSIENT WAS NOTED AND DEALT WITH BY THE REACTOR OPERATOR, BRINGING IT BACK INTO NORMAL RANGE WITHIN THE TIME LIMIT OF THE ACTION STATEMENT. THE DRAIN VALVES HAVE BEEN REPAIRED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + COOK 2 (PWR) + FLOW + CONTROL + FEEDWATER + COOLING SYSTEM, SECONDARY + FAILURE, EQUIPMENT + VALVES + TEMPERATURE + MAIN COOLING SYSTEM + TRANSIENT

ACCESSION NUMBER 143496 10-01 11/18/82

PRESSURIZER PRESSURE DROPS BELOW LIMIT AT SALEM 1
PUBLIC SERVICE ELECTRIC & GAS CO., NEWARK, NJ
5 PGS, LTR W/RO 77-61/03L TO NRC OFFICE OF I & E, REGION I, AUG. 29, 1977, DOCKET 50-272, TYPE--PWR, MFG--WEST., AE--PSE&G
DATE OF EVENT - 080477. POWER LEVEL - 14%. CAUSE - INCREASED FEED WATER FLOW. DURING MODE 1 OPERATION, NO. 12 S/G FEED PUMP CONTROL WAS PLACED IN AUTOMATIC. PUMP SPEED INCREASED RAPIDLY RESULTING IN DECREASED TAVE. PRESSURIZER PRESSURE DECREASED TO 2184 PSIA THEN RETURNED TO 2250 PSIA. NO FURTHER CORRECTIVE ACTION WAS REQUIRED. THIS IS THE 4TH OCCURRENCE OF THIS TYPE. RAPID REDUCTION IN TAVE CAUSED BY INCREASED FEED WATER FLOW FROM NO. 12 S/G FEED PUMP CAUSED RCS PRESSURE TO DECREASE TO 2184 PSIA.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 1 (PWR) + PRESSURE, INTERNAL + COOLING SYSTEM, SECONDARY + FLOW + TRANSIENT + FEEDWATER + FAILURE, EQUIPMENT + FAILURE, INHERENT + PRESSURIZER

ACCESSION NUMBER 145364 10-01 11/18/82

SAFETY INJECTION ACTUATION OCCURS AT DAVIS-BESSE 1
TOLEDO EDISON CO., TOLEDO, OH
4 PGS, LTR W/RO 78-066 TO NRC OFFICE OF I & E, REGION III, JULY 28, 1978, DOCKET 50-346, TYPE--PWR, MFG--B&W, AE--BECHTEL
DATE OF EVENT - 042978. POWER LEVEL - 20%. CAUSE - FEEDWATER VALVES OPENED. DURING A TRANSIENT WHILE THE UNIT WAS SHUTTING DOWN AND WAS OPERATING AT 20% REACTOR POWER, REACTOR COOLANT SYSTEM PRESSURE DROPPED TO ABOUT 1620 PSIG AND SAFETY FEATURES ACTUATION SYSTEM CHANNELS 1 AND 3 TRIPPED ON RCS LOW PRESSURE. THE CAUSE OF THIS HIGH PRESSURE INJECTION ACTUATION WAS A LOW REACTOR COOLANT SYSTEM PRESSURE TRIP RESULTING FROM THE OPENING OF FEEDWATER VALVES. THE CAUSE OF THE FEEDWATER VALVES OPENING WAS DUE BOTH TO THE SENSITIVITY OF THE FEEDWATER SYSTEM WHILE REPETITIVELY TRANSFERRING FROM FLOW CONTROL TO LEVEL CONTROL OF STEAM GENERATOR LEVEL AND IMPROPER OPERATOR RESPONSE TO THE TRANSIENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + DAVIS-BESSE 1 (PWR) + SAFETY INJECTION + PRESSURE TRANSIENT + VALVES + FEEDWATER + OPERATOR ACTION + FAILURE, EQUIPMENT + RESPONSE TIME + LICENSED OPERATOR + COOLING SYSTEM, SECONDARY + TRANSIENT

ACCESSION NUMBER 146533 10-01 11/18/82

FEEDWATER TRANSIENT FOLLOWING SCRAM ACTUATES HPI AT OCONEE 1
DUKE POWER CO.
LTR W/LER 78-027 TO U.S. NRC, REGION 2, JAN 15, 1979, DOCKET 50-269,
TYPE--PWR,MFG--B+W, AE--DUKE CONTROL--023398
DATE OF EVENT - 121478. POWER LEVEL - 098%. CAUSE - IMPROPER VALVE OPERATION. ON DECEMBER 13, 1978, THE TAVE STATALARM BEGAN TO ACT ERRATICALLY AND AN INVESTIGATION WAS INITIATED. DURING INVESTIGATION, ON 12/14/78 POWER CORD SUPPLYING TAVE RECORDER SHORTED CAUSING AN APPARENT (NOT REAL) DROP IN TAVE OF 13 F. AS ICS ATTEMPTED TO CORRECT TAVE UNIT TRIPPED ON HIGH PRESSURE/TEMPERATURE. FEEDWATER TRANSIENTS DURING COOLDOWN ALLOWED THE B OTSG TO GO DRY. WHEN IT WAS REFILLED IT CAUSED RCS PRESSURE TO DROP BELOW 1500 PSI WHICH ACTUATED THE HPI SYSTEM. THE CAUSE OF THE TAVE CORD SHORT HAS NOT BEEN IDENTIFIED. THE FEEDWATER TRANSIENTS WERE PROBABLY CAUSED BY IMPROPER VALVE OPERATION. THE POWER SUPPLY CORD WAS REPLACED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + OCONEE 1 (PWR) + REACTOR SHUTDOWN + TRANSIENT + FEEDWATER + COOLING SYSTEM, SECONDARY + STEAM GENERATOR + DRY + HPSI + SCRAM, REAL + RECORDERS + FAULT + CONTROL SYSTEM + PRESSURE TRANSIENT + VALVES + FAILURE, EQUIPMENT + FAILURE, SEQUENTIAL + FAILURE, LATENT + FAILURE, DESIGN ERROR + REACTOR CONTROL + PSCDR

ACCESSION NUMBER 146587 10-01 11/18/82

RCS COOLDOWN RATE EXCEEDS LIMIT AT ARKANSAS NUCLEAR 2
ARKANSAS POWER AND LIGHT CO.
LTR W/LER 78-030 TO U.S. NRC, REGION 4, JAN 25, 1979, DOCKET 50-368,
TYPE--PWR, MFG--COMB,AE--BECH CONTROL--023488
DATE OF EVENT - 122778. POWER LEVEL - 008%. CAUSE - STEAM RELIEF VALVE FAILS TO RESEAT. DURING MAIN TURBINE ROLL, MAIN STEAM RELIEF VALVES 2PSV1053 LIFTED, DUE TO STEAM DUMP VALVE FAILURES, AND FAILED TO RESEAT, CAUSING EXCESSIVE RCS COOLDOWN (SEE ALSO LER 50-368/78-29). THE REACTOR WAS MANUALLY TRIPPED AT 1 DEG F. BELOW THE 525 DEG F. TAVE THE RELIEF VALVE RESEATED AFTER > 1 HOUR OF BLOWDOWN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ARKANSAS NUCLEAR 2 (PWR) + REACTOR SHUTDOWN + RATE + MAIN COOLING SYSTEM + VALVES + TEMPERATURE + REACTOR STARTUP EXPERIENCE + REACTOR STARTUP + THERMAL TRANSIENT + PSCDR

ACCESSION NUMBER 147267 10-01 11/18/82

REACTOR COOLDOWN RATE EXCEEDS LIMIT AT RANCHO SECO
SACRAMENTO MUNICIPAL UTIL. DISTRICT
LTR W/LER 79-001 TO U.S. NRC, REGION 5, JAN 19, 1979, DOCKET 50-312,
TYPE--PWR, MFG--B+W, AE--BECH CONTROL--025159
DATE OF EVENT - 010579. POWER LEVEL - 100%. CAUSE - MAINTENANCE ERROR. A TECHNICIAN PERFORMING A MODIFICATION TO THE ICS SHORTED THE DC POWER SUPPLIES TO THE ICS WHICH TRIPPED THE AC POWER SUPPLIES. THE RESULTANT LOSS OF ICS CAUSED A REACTOR TRIP, AND IN CONJUNCTION WITH LOSS OF SFAS INDICATING LIGHTS, A SUBSEQUENT RCS COOLDOWN AT 120 DEGREES PER HOUR. THIS RATE EXCEEDED LIMITS ESTABLISHED IN T.S. FIGURE 3.1.2-2. THE REACTOR VENDOR ANALYZED THE TRANSIENT AND SANCTIONED A RETURN TO POWER WITHOUT IMPOSING ADDITIONAL RESTRICTIONS. LOSS OF ICS RESULTED IN THE REACTOR TRIP. DISTRICT ENGINEERS ARE EVALUATING LONG TERM PROJECTS TO PREVENT RECURRENCE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + RANCHO SECO (PWR) + REACTOR SHUTDOWN + THERMAL TRANSIENT + RATE + MAIN COOLING SYSTEM + SCRAM, REAL + FAILURE, MAINTENANCE ERROR + MAINTENANCE PERSONNEL + ELECTRIC POWER + CONTROL SYSTEM + INSTRUMENTS, MISC. + PSCDR

ACCESSION NUMBER 148074 10-01 11/18/82

MAIN STEAM RELIEF VALVE FAILS TO RESEAT AT ARKANSAS NUCLEAR 2
ARKANSAS POWER AND LIGHT CO.
LTR W/LER 78-030 TO U.S. NRC, REGION 4, JAN 25, 1979, DOCKET 50-368,
TYPE--PWR, MFG--COMB, AE--BECH CONTROL--023534
DATE OF EVENT - 122778. POWER LEVEL - 008%. CAUSE - STEAM DUMP VALVE FAILURES. MODE 1 OPERATION, DURING MAIN TURBINE ROLL, MAIN STEAM RELIEF VALVES 2PSV1053 LIFTED DUE TO STEAM DUMP VALVE FAILURES AND FAILED TO RESEAT, CAUSING EXCESSIVE RCS COOLDOWN (SEE ALSO LER 50-368/78-29). THE REACTOR WAS MANUALLY TRIPPED AT 1 DEG. F. BELOW THE 525 DEG. F. TAVE AS REQUIRED BY T.S. 3.1.1.5. NO SIMILAR OCCURRENCES. REPORTABLE PER T.S. 6.9.1.9(B). THE RELIEF VALVE RESEATED AFTER >1 HOUR OF BLOWDOWN. THE REACTOR WAS MANUALLY TRIPPED AND BROUGHT TO MODE 3 CONDITIONS AS REQUIRED BY ACTION STATEMENT T.S. 3.1.1.5.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ARKANSAS NUCLEAR 2 (PWR) + REACTOR SHUTDOWN + VALVES + PRESSURE RELIEF + STEAM + THERMAL TRANSIENT + RATE + MAIN COOLING SYSTEM + BLOWDOWN + FAILURE, EQUIPMENT + FAILURE, INHERENT + COOLING SYSTEM, SECONDARY + ACTUATION + PSCDR

ACCESSION NUMBER 148080 10-01 11/18/82

REACTOR TRIPS DUE TO OVERFEEDING STEAM GENERATOR AT THREE MILE ISLAND 2
METROPOLITAN EDISON CO.
LTR W/LER 78-069 TO U.S. NRC, REGION 1, FEB 28, 1979, DOCKET 50-320,
TYPE--PWR, MFG--B+W, AE--B+R CONTROL--023543
DATE OF EVENT - 120278. POWER LEVEL - 022%. CAUSE - DEFECTIVE PROCEDURES.
WHILE IN MODE 1 ON DECEMBER 2, 1978, WHILE SWITCHING FROM THE STARTUP TO
THE MAIN FEEDWATER REGULATING VALVES, A REACTOR TRIP OCCURRED FOLLOWED
BY SAFETY INJECTION ACTUATION DUE TO OVERFEEDING THE STEAM GENERATORS.
SINCE SAFETY FEATURE SYSTEMS FUNCTIONED AS DESIGNED, THIS EVENT DID NOT
AFFECT THE HEALTH AND SAFETY OF THE PUBLIC. THIS EVENT OCCURRED DUE TO
THE MAIN FEEDWATER REGULATING VALVE BEING PINNED OPEN. PROCEDURES HAVE
BEEN REVISED TO PRECLUDE REOCCURRENCES.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C.
20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + THREE MILE ISLAND 2 (PWR) + REACTOR SHUTDOWN +
REACTOR STARTUP EXPERIENCE + SCRAM, REAL + STEAM GENERATOR + TRANSIENT +
FEEDWATER + CONTROL + VALVES + COOLING SYSTEM, SECONDARY + FAILURE,
EQUIPMENT + FAILURE, ADMINISTRATIVE CONTROL + PROCEDURES AND MANUALS +
SAFETY INJECTION

ACCESSION NUMBER 148112 10-01 11/18/82

REACTOR COOLANT PUMP SEALS FAIL DURING A TRANSIENT AT FT. CALHOUN 1
OMAHA PUBLIC POWER DISTRICT, NE
9 PGS, LTR W/OI 74A-49 TO NRC DIVISION OF REACTOR LICENSING, JUNE 26,
1975, DOCKET 50-285, TYPE--PWR, MFG--COMB., AE--GIBBS & HILL
DATE OF EVENT - 041774. POWER LEVEL - 0%. CAUSE - ISOLATION OF
COMPONENT COOLING WATER. WHILE OPERATING AT 90% POWER, FEEDWATER
CONTROL VALVE TO SG 2A FAILED OPEN. WHEN MANUAL CONTROL PANEL CONTROL
COULD NOT BE GAINED, LOCAL MANUAL CONTROL WAS INITIATED. VALVE CONTROL
WAS LOST DUE TO FAILURE OF AN AIR LINE TO THE VALVE OPERATOR. DURING
THE RESULTING TRANSIENT, THE REACTOR SCRAMMED AND ECCS WAS INITIATED BY
LOW PRESSURIZER LEVEL AND PRESSURE. THIS ISOLATED COMPONENT COOLING
FLOW TO RCS PUMP SEALS AND CRDM'S. THE PUMPS WERE RUNNING. SEALS ON 2
PUMPS FAILED. MAIN FEEDWATER CHECK VALVE FW-162 GASKET FAILED. ALL
PUMP SEALS AND THE VALVE GASKET WERE REPLACED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + FT. CALHOUN I (PWR) + SEAL + PUMPS + MAIN
COOLING SYSTEM + DAMAGE + SAFETY INJECTION + TRANSIENT + THERMAL
TRANSIENT + PRESSURE TRANSIENT + FAILURE, EQUIPMENT + VALVES + FEEDWATER
+ COOLING SYSTEM, SECONDARY + FAILURE, INHERENT + REACTOR SHUTDOWN +
PSCDR

ACCESSION NUMBER 148766 10-01 11/18/82

SHUTDOWN COOLING FLOW LOST AT MILLSTONE 2
NORTHEAST NUCLEAR ENERGY CO.
LTR W/LER 79-008 TO U.S. NRC, REGION 1, APR 12, 1979, DOCKET 50-336,
TYPE--PWR, MFG--COMB, AE--BECH CONTROL--025576
DATE OF EVENT - 031479. POWER LEVEL - 000%. CAUSE - DEFECTIVE PROCEDURES.
WHILE STEAM GENERATOR (SG) EDDY CURRENT TESTING WAS IN PROGRESS AND WITH RCS ON SHUTDOWN COOLING (SDC) WITH RCS LEVEL AT HOT LEG CENTER LINE, A LOSS OF SDC WAS EXPERIENCED DUE TO LPSI SUCTION AIR BINDING. RECOVERY WAS ACCOMPLISHED BY FLOODUP OF THE RCS FROM THE RWST. PRIOR TO RESTORING FULL SDC FLOW, RCS TEMPERATURE REACHED 208 DEGREES F, RESULTING IN ENTRY TO MODE 4. RCS TEMPERATURE WAS SUBSEQUENTLY LOWERED TO 120 DEGREES F. LPSI SUCTION LOOP SEAL PRIME WAS LOST DURING INITIAL RCS DRAINDOWN. PARTIAL FLOW RESTORED ON RCS FLOODUP FROM RWST AND FULL FLOW RESTORED BY VACUUM PRIMING SUCTION LOOP SEAL.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

SI but not frozen/HP be unanticipated

FAILURE + REACTOR, PWR + MILLSTONE 2 (PWR) + FLOW + SHUTDOWN COOLING SYSTEM + TEMPERATURE + MAIN COOLING SYSTEM + PUMPS + AIR + FAILURE, EQUIPMENT + FAILURE, ADMINISTRATIVE CONTROL + PROCEDURES AND MANUALS + FLOW BLOCKAGE + REFUELING + LPCI

ACCESSION NUMBER 148767 10-01 11/18/82

REACTOR COOLANT SYSTEM COOLDOWN RATE EXCEEDS LIMIT AT MILLSTONE 2
NORTHEAST NUCLEAR ENERGY CO.
LTR W/LER 79-009 TO U.S. NRC, REGION 1, APR 13, 1979, DOCKET 50-336,
TYPE--PWR, MFG--COMB, AE--BECH CONTROL--025577
DATE OF EVENT - 031479. POWER LEVEL - 000%. CAUSE - COLD RWST WATER ADDED. WHILE REESTABLISHING SHUTDOWN COOLINGS COOLDOWN LIMIT OF T.S. 3.4.9.1 WAS EXCEEDED. ANALYSIS PER ASME III APPENDIX G AND 10 CFR 50. APPENDIX G INDICATED NO DOWNGRADE OF FRACTURE TOUGHNESS DESIGN BASIS LIMITATION. THIS WAS CAUSED BY ADDITION OF COLD RWST WATER TO LPSI PUMP SUCTION. COOLDOWN TERMINATED WHEN SDC WAS REESTABLISHED AND SUCTION FROM RWST SECURED (REFERENCE LER 79-08/3L-0).
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + MILLSTONE 2 (PWR) + THERMAL TRANSIENT + MAIN COOLING SYSTEM + FLOW + SHUTDOWN COOLING SYSTEM + REFUELING + PRESSURE VESSELS

ACCESSION NUMBER 149716 10-01 11/18/82

INADVERTENT SAFETY INJECTION INITIATED AT ZION 1
COMMONWEALTH EDISON CO.
LTR W/LER 79-012 TO U.S. NRC, REGION 3, JUN 01, 1979, DOCKET 50-295,
TYPE--PWR, MFG--WEST, AE--S+L CONTROL--025967
DATE OF EVENT - 030279. POWER LEVEL - 100%. CAUSE - TRIPPING STEAM PRESSURE BISTABLES. DURING NORMAL POWER OPERATIONS, A SAFETY INJECTION (& A CONSEQUENT WATERHAMMER) WAS INITIATED DUE TO A HI STEAM FLOW SIGNAL COINCIDENT WITH THE LOW STEAM PRESSURE BISTABLES WHICH WERE TRIPPED FOR SURVEILLANCE TESTING. VISUAL INSPECTIONS INDICATED NO STRUCTURAL DAMAGE OCCURRED IN THE CONTAINMENT OR REACTOR COOLANT SYSTEM AS A RESULT OF THE SAFETY INJECTION. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED. THE SI OCCURRED BECAUSE THE STEAM PRESSURE BISTABLES WERE TRIPPED. HI STEAM FLOW SIGNAL OCCURRED WHEN THE SETPOINT IMMEDIATELY DROPPED TO 40% DURING AN INADVERTENT REACTOR TRIP AND ACTUAL STEAM FLOWS REMAINED ABOVE THIS DUE TO THE NORMAL DELAY IN CLOSING THE GOVERNOR VALVES.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 1 (PWR) + REACTOR SHUTDOWN + TRANSIENT + SAFETY INJECTION + HYDRAULIC EFFECT + PIPES AND PIPE FITTINGS + STEAM + SENSORS, PRESSURE + ELECTRONIC FUNCTION UNITS + TESTING + FAILURE, SEQUENTIAL + FAILURE, INHERENT

ACCESSION NUMBER 149746 10-01 11/18/82

REACTOR COOLANT TEMPERATURE DROP EXCEEDS LIMIT AT ARKANSAS NUCLEAR 2
ARKANSAS POWER & LIGHT CO., LITTLE ROCK, AR
2 PGS, LTR W/LER 78-29 TO NRC OFFICE OF I & E, REGION IV, JAN. 25, 1979, DOCKET 50-368, TYPE--PWR, MFG--COMB., AE--BECHTEL
DATE OF EVENT - 122778. POWER LEVEL - 015%. CAUSE - STEAM RELIEF VALVE FAILED TO RESEAT. MODE 1 OPERATION, DURING MAIN TURBINE ROLL, MAIN STEAM RELIEF VALVE 2PSV1053 LIFTED, DUE TO STEAM DUMP VALVE FAILURES (REFERENCE LER 50-368/78-30), AND FAILED TO RESEAT, CAUSING THE RCS TEMPERATURE TO DROP 107 F WITHIN 52 MINUTES, EXCEEDING 100 F/HR. LIMIT FOR AN APPROXIMATE 10 MINUTE PERIOD. THE RELIEF VALVE RESEATED AFTER > 1 HOUR OF BLOWDOWN. RCS TAVE WAS RESTORED TO 545 F WITHIN 30 MINUTES.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ARKANSAS NUCLEAR 2 (PWR) + REACTOR STARTUP EXPERIENCE + THERMAL TRANSIENT + MAIN COOLING SYSTEM + VALVES + PRESSURE RELIEF + COOLING SYSTEM, SECONDARY + BLOWDOWN + FAILURE, EQUIPMENT + FAILURE, INHERENT + ACTUATION + CMASP + PSCDR

ACCESSION NUMBER 150394 10-01 11/18/82

EMERGENCY FEEDWATER FLOW CONTROL VALVE FAILS TO CLOSE AT ARKANSAS NUCLEAR 2
ARKANSAS POWER AND LIGHT CO.
LTR W/LER 79-040 TO U.S. NRC, REGION 4, JUN 27, 1979, DOCKET 50-368, TYPE--PWR, MFG--COMB, AE--BECH CONTROL--026248
DATE OF EVENT - 060379. POWER LEVEL - 000%. CAUSE - TRIPPER BAR OUT OF ADJUSTMENT. DURING MODE 3 OPERATION, EMERGENCY FEEDWATER FLOW CONTROL VALVE FROM THE ELECTRIC DRIVEN EFW PUMP, 2P7B, TO THE "B" STEAM GENERATOR, 2CV-1036-1, WOULD NOT STROKE SHUT, OR RESPOND TO A MAIN STEAM ISOLATION SIGNAL. THE REDUNDANT VALVE, 2CV-1075-1, WAS STROKED SHUT AND VERIFIED OPERABLE IMMEDIATELY. THIS OCCURRENCE IS SIMILAR TO 50-368/79-036. THE COIL AND TRIPPER BAR WAS OUT OF ADJUSTMENT. THE VALVE WAS REPAIRED AND SUCCESSFULLY TESTED. AN EFW VALVE ACTUATOR CHECKOUT WAS COMPLETED. ON 6/5/79, EACH EFW VALVE WAS STROKED 20 TIMES.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ARKANSAS NUCLEAR 2 (PWR) + VALVES + FEEDWATER + AUXILIARY + COOLING SYSTEM, SECONDARY + CONTROL + FLOW + FAILURE, EQUIPMENT + FAILURE, INHERENT + VALVE OPERATORS + REACTOR STARTUP EXPERIENCE + PSCDR

ACCESSION NUMBER 150731 10-01 11/18/82

RCS POWER OPERATED RELIEF VALVE FAILS TO FULLY CLOSE AFTER LIFTING AT CALVERT CLIFFS 1
BALTIMORE GAS & ELECTRIC CO.
LTR W/LER 79-020 TO U.S. NRC, REGION 1, JUL 19, 1979, DOCKET 50-317, TYPE--PWR, MFG--COMB, AE--BECH CONTROL--026395
DATE OF EVENT - 070679. POWER LEVEL - 000%. CAUSE - PILOT VALVE OUT OF ADJUSTMENT. WHILE PERFORMING A POST MODIFICATION TEST ON THE REACTOR COOLANT SYSTEM POWER OPERATED RELIEF VALVES, RC-402-ERV FAILED TO FULLY CLOSE AFTER LIFTING. THE VALVE WAS ISOLATED AND PLACED OUT OF SERVICE. THE REDUNDANT POWER OPERATED RELIEF REMAINED OPERABLE DURING THE EVENT. RCS PRESSURE DECREASED FROM 260 TO 220 PSIA DURING THIS EVENT. BOTH PRESSURIZER POWER OPERATED RELIEF VALVES WERE DISASSEMBLED & MODIFIED DURING RECENT REFUELING OUTAGE. SINCE IT WAS NOT POSSIBLE TO STROKE TEST THESE VALVES WITHOUT REACTOR COOLANT SYSTEM PRESSURIZED, TEST WAS PERFORMED AT EARLIEST OPPORTUNITY DURING UNIT STARTUP. PILOT VALVE STROKE WAS ADJUSTED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CALVERT CLIFFS 1 (PWR) + VALVES + PRESSURE RELIEF + MAIN COOLING SYSTEM + PRESSURIZER + TEST, SYSTEM OPERABILITY + REACTOR STARTUP + PRESSURE TRANSIENT + BLOWDOWN + FAILURE, EQUIPMENT + FAILURE, INHERENT + ACTUATION + PSCDR

ACCESSION NUMBER 152751 10-01 11/18/82

UNPLANNED GASEOUS RADIATION RELEASE FOLLOWS SAFETY INJECTION ACTUATION AT NORTH ANNA 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 79-128 TO U.S. NRC, REGION 2, OCT 09, 1979, DOCKET 50-338, TYPE--PWR, MFG--WEST, AE--S+W CONTROL--027116
DATE OF EVENT - 092579. POWER LEVEL - 078%. CAUSE - STEAM DUMP VALVE STUCK OPEN. AT 0614 A SAFETY INJECTION OCCURRED DUE TO A LOW PRESSURIZER LEVEL COINCIDENT WITH LOW PRESSURIZER PRESSURE. FOLLOWING THE EVENT A RADIATION RELEASE OCCURRED WHICH WAS LESS THAN 0.05% OF THE INSTANTANEOUS RELEASE RATE. BOUNDARY RADIATION DETECTORS SHOWED ONLY BACKGROUND RADIOACTIVITY. THE CAUSE OF THE SI WAS STEAM DUMP VALVE TCV-1408G STUCK OPEN FOLLOWING A REACTOR TRIP, WHICH DEPRESSURIZED THE REACTOR COOLANT SYSTEM. CORRECTIVE ACTIONS WERE TO RETURN THE PLANT TO STABLE CONDITIONS BY THE CONTROL ROOM OPERATORS AND LATER INITIATE A PLANT COOLDOWN TO THE COLD SHUTDOWN MODE OF OPERATION.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA I (PWR) + REACTOR SHUTDOWN + AIRBORNE RELEASE + SAFETY INJECTION + ACTUATION + PRESSURE TRANSIENT + VALVES + STEAM + PRESSURE, INTERNAL + CONTROL + COOLING SYSTEM, SECONDARY + FAILURE, EQUIPMENT + FAILURE, INHERENT + EASP

ACCESSION NUMBER 152983 10-01 11/18/82

RCS PRESSURE EXCEEDS OPERATING MODE LIMIT AT OCONEE 3
DUKE POWER CO.
LTR W/LER 79-011 TO U.S. NRC, REGION 2, NOV 16, 1979, DOCKET 50-287, TYPE--PWR, MFG--B+W, AE--DUKE CONTROL--027368
DATE OF EVENT - 101979. POWER LEVEL - 000%. CAUSE - LICENSED OPERATOR ERROR. WHILE OCONEE 3 WAS SUSPENDED IN A HEATUP MODE IN ORDER TO TAKE LPI SYSTEM OUT OF SERVICE FOR MAINTENANCE, VALVE 3HP-410 INADVERTENTLY OPENED FULLY, ALLOWING RCS PRESSURE TO REACH A MAXIMUM OF 360 PSIG FOR LESS THAN 1 MIN. TECH SPECS LIMIT RCS PRESSURE TO 350 PSIG WITH LPI SYSTEM OUT OF SERVICE. 3HP-410 IS THE HPI CROSS-CONNECT ISOLATION VALVE TO RCS LOOP A. ATTEMPT WAS MADE TO TEST VALVE 3HP-410 WITHOUT FIRST ISOLATING IT. VALVE MOTOR WAS MISWIRED, SO WHEN ATTEMPT WAS MADE TO CLOSE IT, IT OPENED FULLY. WIRING ERROR WAS CORRECTED AND VALVE WAS TESTED SUCCESSFULLY. APPROPRIATE PERSONNEL HAVE BEEN COUNSELED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + OCONEE 3 (PWR) + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + VALVE OPERATORS + HPSI + PRESSURE TRANSIENT + TESTING + MAIN COOLING SYSTEM + VALVES + PRESSURE, INTERNAL

ACCESSION NUMBER 153164 10-01 11/18/82

CORE DAMAGE AND RADIOACTIVITY RELEASE OCCURS AT THREE MILE ISLAND 2
METROPOLITAN EDISON CO.
LTR W/LER 79-012 TO U.S. NRC, REGION 1, MAR 29, 1979, DOCKET 50-320,
TYPE--PWR, MFG--B+W, AE--B+R CONTROL--027456
DATE OF EVENT - 032879. POWER LEVEL - 099%. CAUSE - LICENSED OPERATOR ERROR. BEGINNING ABOUT 0400, AN ACCIDENT OCCURRED CAUSING CORE DAMAGE AND RADIOACTIVE RELEASE. LOST FEEDWATER; TURBINE TRIPPED; PRESSURIZER RELIEF VALVE OPENED; REACTOR TRIPPED. AUX FEEDWATER VALVES WERE CLOSED 8 MIN. PORV STAYED OPEN; RCS PRESSURE DECREASED; ECCS STARTED, WAS THEN THROTTLED ON RISING PRESSURIZER LEVEL. AT 1TO 2 HR RCS PUMPS TURNED OFF. OPEN PORV ALLOWED RC DRAIN TANK TO OPEN TO REACTOR BLDG. SUMP. SUMP PUMPED TO AUX BLDG.; RADIOACTIVITY RELEASED. SEVERAL MAIN FACTORS INCLUDING PERSONNEL ERROR, DESIGN DEFICIENCIES, COMPONENT FAILURE. VALVES VALVED OUT. PORV FAILED TO CLOSE. INCORRECT PRESSURIZER LEVEL INDICATION. RADIOACTIVITY LEAKS FROM LETDOWN SYSTEM TO AUX BLDG. TRIPPING RCS PUMPS LED TO FUEL DAMAGE. SEVERAL LICENSEE/NRC/VENDOR ACTIONS UNDERWAY. PRESIDENTIAL COMMISSION AND OTHER STUDY GROUPS APPOINTED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + THREE MILE ISLAND 2 (PWR) + REACTOR SHUTDOWN + INCIDENT + FUEL MELTDOWN + FUEL CLAD INTERACTION + ACCIDENT, LOSS OF COOLANT + OPERATOR ACTION + HPSI + AIRBORNE RELEASE + THERMAL TRANSIENT + FEEDWATER + AUXILIARY + VALVES + COOLING SYSTEM, SECONDARY + PRESSURE RELIEF + MAIN COOLING SYSTEM + PRESSURE TRANSIENT + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + FAILURE, MAINTENANCE ERROR + SENSORS, LEVEL + PRESSURIZER + METAL WATER REACTION + ACCIDENT + PSCDA + PSCDR

ACCESSION NUMBER 153683 10-01 11/18/82

STEAM GENERATOR BLOWDOWN ISOLATION VALVE FAILS AT INDIAN POINT 2
CONSOLIDATED EDISON CO.
LTR W/LER 79-023 TO U.S. NRC, REGION 1, DEC 07, 1979, DOCKET 50-247,
TYPE--PWR, MFG--WEST, AE--UEC CONTROL--027660
DATE OF EVENT - 110979. POWER LEVEL - 100%. CAUSE - DISLODGED CLOSURE SPRING BEARING PLATE. WHILE TERMINATING BLOWDOWN FROM NO. 24 STEAM GENERATOR, ONE OF TWO SERIES ISOLATION VALVES, PCV-1217, IN THE BLOWDOWN LINE WOULD NOT CLOSE. THE REDUNDANT VALVE FUNCTIONED AS REQUIRED. THE AFFECTED VALVE WAS MANUALLY CLOSED AND CLAMPED IN POSITION. THE VALVE OPERATOR CLOSURE SPRING BEARING PLATE, FOR A 2-INCH, C.S., MASONEILAN AIR OPERATED VALVE, NO. 38-20561 AB, BECAME DISLODGED. THE BEARING PLATE WAS RESTORED TO ITS PROPER POSITION, THE ASSOCIATED AIR SOLENOID VALVE WAS CLEANED AND ITS COIL REPLACED. THE VALVE WAS THEN RETESTED SATISFACTORILY.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + INDIAN POINT 2 (PWR) + FAILURE, EQUIPMENT + FAILURE, INHERENT + VALVES + STEAM GENERATOR + BLOWDOWN + CONTAINMENT ISOLATION + VALVE OPERATORS

ACCESSION NUMBER 153713 10-01 11/18/82

ATMOSPHERIC RELIEF VALVE FAILS OPEN AT FARLEY 1
ALABAMA POWER CO., BIRMINGHAM, AL
2 PGS, LTR W/LER 79-061 TO NRC OFFICE OF I & E, REGION II, JAN. 2, 1980, DOCKET 50-348, TYPE--PWR, MFG--WEST., AE--BECH & SSI
DATE OF EVENT - 120279. POWER LEVEL - 001%. CAUSE - BROKEN POSITIONING ARM. WHILE CONDUCTING A MODE CHANGE FROM MODE 1 TO 2 AND SECURING THE SECONDARY STEAM LOADS, THE C ATMOSPHERIC RELIEF VALVE FAILED OPEN CAUSING A REDUCTION IN RCS TO 531F. TECH. SPEC. REQUIRES RCS LOWEST OPERATING LOOP TO BE 541F OR GREATER. C ATMOSPHERIC RELIEF VALVE FAILED OPEN DUE TO A BROKEN POSITIONING ARM. IMMEDIATE CORRECTIVE ACTION WAS TO MANUALLY ISOLATE C ATMOSPHERIC RELIEF VALVE WHICH ALLOWED RCS T(AVG) TO BE RESTORED ABOVE 541F. THE POSITIONING ARM WAS REPLACED AND THE C ATMOSPHERIC RELIEF VALVE WAS RETURNED TO SERVICE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + FARLEY 1 (PWR) + VALVES + PRESSURE RELIEF + COOLING SYSTEM, SECONDARY + THERMAL TRANSIENT + FAILURE, EQUIPMENT + FAILURE, INHERENT + EASP

ACCESSION NUMBER 153772 10-01 11/18/82

COOLDOWN RATE LIMIT EXCEEDED FOLLOWING LOSS OF ICS POWER AT OCONEE 3
DUKE POWER CO.
LTR W/LER 79-013 TO U.S. NRC, REGION 2, NOV 30, 1979, DOCKET 50-287, TYPE--PWR, MFG--B+W, AE--DUKE CONTROL--027697
DATE OF EVENT - 111079. POWER LEVEL - 099%. CAUSE - TRANSFER SWITCH FAILURE. THE RCS COOLDOWN RATE LIMIT WAS EXCEEDED AFTER POWER TO THE ICS WAS LOST FOR ABOUT TWO AND ONE-HALF MINUTES. NO ES ACTUATION SETPOINTS WERE REACHED, AND ADEQUATE RCS INVENTORY WAS MAINTAINED. NO DAMAGE WAS INCURRED. LOSS OF ICS POWER RESULTED FROM BLOWN FUSES IN NORMAL INVERTER (KI) & FAILURE OF TRANSFER SWITCH TO TRANSFER AUTOMATICALLY TO REGULATED AC POWER. WHEN ICS POWER WAS RESTORED, EXCESSIVE FEEDWATER FLOW CAUSED A RAPID RCS COOLDOWN. A REDUNDANT TRANSFER SWITCH HAS BEEN INSTALLED, & PERSONNEL HAVE BEEN INSTRUCTED IN ORDER TO RESPOND PROPERLY TO LOSS OF ICS PWR.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + OCONEE 3 (PWR) + REACTOR SHUTDOWN + FAILURE, EQUIPMENT + FAILURE, INHERENT + ELECTRIC POWER, VITAL + REACTOR TRANSIENT + THERMAL TRANSIENT + MAIN COOLING SYSTEM + COOLING + RATE + ELECTRIC POWER + ON SITE + FAILURE, SEQUENTIAL + INSTRUMENT, PROTECTIVE + INSTRUMENT, SWITCH + PSCDR

ACCESSION NUMBER 154291 10-01 11/18/82

VITAL BUS INVERTER FAILURE RESULTS IN REACTOR TRIPS AND SAFETY INJECTION AT BEAVER VALLEY 1
DUQUESNE LIGHT CO.
LTR W/LER 79-039 TO U.S. NRC, REGION 1, DEC 19, 1979, DOCKET 50-334, TYPE--PWR, MFG--WEST, AE--S+W CONTROL--028012
DATE OF EVENT - 092079. POWER LEVEL - 099%. CAUSE - ELECTRICAL. THE NO. 4 VITAL BUS INVERTER FAILED. WHEN THE INVERTER FAILED, THE REACTOR TRIPPED AND A HIGH STEAM FLOW SAFETY INJECTION ACTUATION OCCURRED. ALL SI SYSTEMS FUNCTIONED NORMALLY. THE OPERATORS VERIFIED THE SAFETY INJECTION WAS INADVERTENT AND RESTORED THE PLANT TO NORMAL SHUTDOWN CONDITIONS. SAFETY INJECTIONS TO DATE: 12 OPERATIONAL, 2 PREOPERATIONAL. INCIDENT RESULTED FROM A FAILURE OF THE #4 VITAL BUS INVERTER & THE SUBSEQUENT TRANSFER (SECONDS LATER) OF THE VITAL BUS TO ITS AUXILIARY POWER SUPPLY. THIS ACTION RESULTED IN A HIGH STEAM FLOW DUE TO OPENING OF CONDENSER STEAM DUMP VALVES. THE INVERTER WAS REPAIRED AND THE TRIP OPEN SIGNAL TO THE STEAM DUMPS DEFEATED TO PREVENT RECURRENCE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON D.C. 20555, (08¢/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + BEAVER VALLEY 1 (PWR) + REACTOR SHUTDOWN + GENERATORS + ELECTRIC POWER + ELECTRIC POWER, VITAL + SCRAM, REAL + SAFETY INJECTION + ACTUATION + FAILURE, EQUIPMENT + FAILURE, INHERENT + TRANSIENT + EASP

ACCESSION NUMBER 154451 10-01 11/18/82

PRESSURIZER PRESSURE RELIEF VALVE OPENS FOR ABOUT TWO MINUTES AT CONNECTICUT YANKEE
CONNECTICUT YANKEE ATOMIC POWER CO., EAST HAMPTON, CT
2 PGS, LTR W/LER 80-04/3L TO NRC OFFICE OF I & E, REGION I, FEB. 12, 1980, DOCKET 50-213, TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 020480. POWER LEVEL - 100%. CAUSE - PROBABLY A SPURIOUS SIGNAL. WHILE OPERATING AT STEADY STATE FULL POWER, POWER OPERATED PRESSURIZER RELIEF VALVE 570 AND ITS ISOLATION VALVE MOV-569 OPENED. REACTOR COOLANT SYSTEM PRESSURE WAS REDUCED TO 1992 PSIG (FROM 2016 PSIG) BEFORE MANUAL ISOLATION TERMINATED THE TRANSIENT. PRESSURE RETURNED TO NORMAL IN ABOUT 2 MINUTES. THIS EVENT HAS OCCURRED BEFORE. THE CAUSE APPEARS TO BE A SPURIOUS SIGNAL FROM PRESSURIZE PRESSURE CONTROL CHANNEL ONE. THE SIGNAL DID NOT STAY IN AND THOROUGH INVESTIGATION AND TESTING HAVE FOUND NOTHING. THE CHANNEL AND VALVE CONTROLS HAVE BEEN PLACED IN NORMAL OPERATING MODE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CONNECTICUT YANKEE (PWR) + VALVES + PRESSURE RELIEF + MAIN COOLING SYSTEM + PRESSURE TRANSIENT + ACTUATION + FAILURE, EQUIPMENT

ACCESSION NUMBER 154690 10-01 11/18/82

EXCESSIVE FEEDWATER FLOW TO STEAM GENERATOR RESULTS IN REACTOR TRIP AT COOK 2
INDIANA & MICHIGAN ELECTRIC CO., BRIDGMAN, MI
2 PGS, LTR W/LER 80-001/03L-0 TO NRC OFFICE OF I & E, REGION III, FEB. 6, 1980,DOCKET 50-316, TYPE--PWR, MFG--WEST, AE--UTILITY
DATE OF EVENT - 011980. POWER LEVEL - 010%. CAUSE - LICENSED OPERATOR ERROR. WHILE PERFORMING A NORMAL UNIT START-UP AND WITH THE REACTOR POWER LEVEL ABOUT 10%, THE OPERATORS WERE CHANGING FEEDWATER FROM AUXILIARY FEED TO NORMAL FEED. THE STEAM GENERATORS WERE INADVERTENTLY FED TOO MUCH FEEDWATER FLOW. THIS RESULTED IN A DROP IN T(AVE) TEMPERATURE TO 539F AND A PRESSURIZER PRESSURE OF 2170 PSIG WHICH CULMINATED WITH A REACTOR AND TURBINE TRIP. SIMILAR OCCURRENCES INCLUDE 79-165 AND 79-147. DURING THIS PHASE IN THE UNIT START-UP, STEAM GENERATOR LEVELS ARE CONTROLLED MANUALLY. TO RECOVER A LOW STEAM GENERATOR LEVEL, THE STEAM GENERATOR WAS OVER-FED CAUSING AN INCREASE IN THE LEVEL TO THE HIGH LEVEL TRIP POINT. THE EXCESSIVE AMOUNT OF COLD WATER RESULTED IN THE DECREASE IN REACTOR COOLANT SYSTEM T(AVE) AND PRESSURE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D.C. 20545(8 CENTS/PAGE -- MINIMUM CHARGE $2)

FAILURE + REACTOR, PWR + COOK 2 (PWR) + RATE + FLOW + FEEDWATER + COOLING SYSTEM, SECONDARY + STEAM GENERATOR + THERMAL TRANSIENT + SCRAM, REAL + REACTOR SHUTDOWN + REACTOR STARTUP

ACCESSION NUMBER 154747 10-01 11/18/82

INADVERTENT ACTUATION OF SAFETY INJECTION OCCURS AT TROJAN
PORTLAND GENERAL ELECTRIC CO., RAINIER, OR
2 PGS, LTR W/SPECIAL RPT. TO U.S. NRC, REGION V, DEC. 3, 1979, DOCKET 50-344, TYPE--PWR, MFG--WEST, AE--BECH
DATE OF EVENT - 090679. POWER LEVEL - 002%. CAUSE - FAULTY PROCEDURE. AN INADVERTENT ACTUATION OF THE ECCS SAFETY INJECTION OCCURRED WHILE TROUBLESHOOTING A TURBINE GENERATOR CONTROL CIRCUIT. DURING THE RECOVERY FROM A PLANT TRIP WHILE TROUBLESHOOTING THE EHC SYSTEM ELECTRONICS, A REACTOR TRIP AND SAFETY INJECTION OCCURRED. THE BACKUP ACCELERATION AMPLIFIER CARD WAS PULLED OUT OF THE RACK CAUSING THE MAIN STEAM CONTROL VALVES TO GO WIDE OPEN WHICH RESULTED IN A HIGH STEAM FLOW/LOW STEAM PRESSURE SAFETY INJECTION. THE APPARENT CAUSE IS A FAULTY PROCEDURE. THE I&C TECHNICIAN WAS FOLLOWING THE GE TROUBLESHOOTING PROCEDURE STEP BY STEP WHEN THE EVENT OCCURRED. GE HAS CHANGED THE PROCEDURE TO PREVENT THIS FROM RECURRING.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + TROJAN (PWR) + SAFETY INJECTION + ACTUATION + INADVERTENT + TESTING + PROCEDURES AND MANUALS + FAILURE, ADMINISTRATIVE CONTROL + EASP + PSCDR

ACCESSION NUMBER 154890 10-01 11/18/82

PORV INADVERTENTLY OPENS AT CALVERT CLIFFS 1
BALTIMORE GAS & ELECTRIC CO.
LTR W/LER B0-007 TO U.S. NRC, REGION 1, FEB 12, 1980, DOCKET 50-317,
TYPE--PWR, MFG--COMB, AE--BECH
DATE OF EVENT - 012980. POWER LEVEL - 000%. CAUSE - SPURIOUS SIGNAL FROM PRESSURE TRANSMITTER. AT APPROXIMATELY 1145 WHILE CONDUCTING A CORE FLUSH WITH THE REACTOR COOLANT SYSTEM (RCS) AT 258 PSIA, RELIEF VALVE (PORV) RC-404-ERV OPENED. THE OPERATOR STARTED ALL CHARGING PUMPS, TURNED ON ALL PRESSURIZER HEATERS AND STOPPED THE TWO OPERATING REACTOR COOLANT PUMPS. THE OPERATOR SHUT THE PORV BY GOING TO OVERRIDE. RCS PRESSURE HAD DECREASED TO 169 PSIA AT 1148 WHEN THE PORV WAS SHUT. SHUTDOWN COOLING WITH A FLOW OF 6000 GPM REMAINED IN OPERATION DURING THE EVENT. THIS IS NOT A REPETITIVE OCCURRENCE. THE APPARENT CAUSE WAS A SPURIOUS HIGH PRESSURE SIGNAL FROM TRANSMITTER PT-103-1. AT THE REPORTED TIME OF THE EVENT, A TECHNICIAN WAS CONDUCTING A REVIEW OF INSTRUMENTS INSTALLED IN THE CONTAINMENT BUILDING. THE CHECK REQUIRED REMOVING THE COVER OF PT-103-1 TO VISUALLY VERIFY INTERNAL COMPONENTS. THE TRANSMITTER WAS JOLTED WHEN THE COVER WAS UNSCREWED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CALVERT CLIFFS 1 (PWR) + FAILURE, EQUIPMENT + VALVE OPERATORS + MAIN COOLING SYSTEM + PRESSURE RELIEF + SENSORS, PRESSURE + INSTRUMENT, SWITCH + VALVES + ACTUATION + PRESSURE TRANSIENT + PRESSURIZER + FAILURE + REACTOR, PWR + CALVERT CLIFFS 1 (PWR) + FAILURE, INHERENT

ACCESSION NUMBER 155359 10-01 11/18/82

EMERGENCY CORE COOLING SYSTEM ACTUATED AT NORTH ANNA 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 80-028 TO U.S. NRC, REGION 2, MAR 7, 1980, DOCKET 50-338,
TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 022380. POWER LEVEL - 000%. CAUSE - TESTING FOR LOOSE PARTS. AN ACUTATION OF THE EMERGENCY CORE COOLING SYSTEM OCCURRED DUE TO A HIGH STEAMLINE DIFFERENTIAL PRESSURE. THIS WAS AN INADVERTENT ACTUATION (EQUIPMENT PERFORMED THE INTENDED INJECTION FUNCTION) AND THE TRANSIENT WAS NOT SEVERE. THE CAUSE OF THE HIGH STEAMLINE DIFFERENTIAL PRESSURE AND SUBSEQUENT SI WAS VALVE NRV-MS101C BEING OPENED WHILE NRV-MS101A AND B REMAINED CLOSED AND STEAM FLOW EXISTED. THE REASON NRV-MS101C WAS OPENED WAS TO ALLOW NON-RETURN VALVE NOISE DATA TO BE OBTAINED IN CONNECTION WITH PREVIOUSLY DETECTED LOOSE PARTS INDICATIONS. CORRECTIVE ACTION WAS TO RETURN THE PLANT TO STABLE CONDITIONS BY THE CONTROL ROOM OPERATORS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA 1 (PWR) + SAFETY INJECTION + ACTUATION + PRESSURE DROP + COOLING SYSTEM, SECONDARY + TESTING + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + TRANSIENT + VALVES + EASP

ACCESSION NUMBER 155451 10-01 11/18/82

MAIN STEAM SAFETY VALVE LIFTS AND FAILS TO RESEAT AT TURKEY POINT 4
FLORIDA POWER & LIGHT CO.
LTR W/LER 80-003 TO U.S. NRC, REGION 2, MAR 17, 1980, DOCKET 50-251,
TYPE--PWR, MFG--WEST, AE--BECH
DATE OF EVENTS - 021580. POWER LEVEL - 000%. CAUSE - BINDING IN LIFTING GEAR. AT HOT SHUTDOWN WHILE TESTING THE 4C STEAM GENERATOR (S/G) FEEDWATER CONTROL VALVE, A RAPID CHANGE IN STEAM GENERATOR LEVEL RESULTED IN LIFTING RV-1410 (THE LOW SETPOINT VALVE OF THE FOUR MAIN STEAM SAFETY VALVES ON THE 4C S/G HEADER). THE VALVE DID NOT FULLY RESEAT, AND STEPS WERE INITIATED TO BRING THE UNIT TO COLD SHUTDOWN. DURING THIS PERIOD, MAINTENANCE PERSONNEL RESEATED THE VALVE. OPERABILITY OF THE TWELVE MAIN STEAM SAFETY VALVES IS REQUIRED BY TS. THE ROOT CAUSE OF THE PROBLEM WAS BINDING BETWEEN THE LIFTING GEAR AND THE RELEASE NUT. THE VALVE WAS TESTED AND FOUND TO RELIEVE AT THE PROPER SETPOINT, 1805 PSI.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + TURKEY POINT 4 (PWR) + BLOWDOWN + STEAM GENERATOR + PRESSURE RELIEF + VALVES + COOLING SYSTEM, SECONDARY + THERMAL TRANSIENT + ACTUATION + OBSTRUCTION

ACCESSION NUMBER 156951 10-01 11/18/82

INADVERTENT ECCS ACTUATION WITHOUT CONTAINMENT INTEGRITY AT SURRY 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 80-024 TO U.S. NRC, REGION 2, MAY 22, 1980, DOCKET 50-280,
TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 042680. POWER LEVEL - 000%. CAUSE - OPERATOR ERROR. DURING POWER ESCALATION THE INDICATED AXIAL FLUX DIFFERENCE DEVIATED FROM ITS TARGET BAND FOR GREATER THAN ONE HOUR IN A TWENTY FOUR HOUR PERIOD. IN ACCORDANCE WITH TS, POWER WAS REDUCED TO LESS THAN 50%. THE POWER REDUCTION TOOK MORE THAN THIRTY (30) MINUTES WHICH IS CONTRARY TO T.S. THE AXIAL FLUX DIFFERENCE WAS RETURNED TO THE BAND. THE CAUSE WAS AN UNSCHEDULED BORON ADDITION WHICH INITIATED A RAPID POWER REDUCTION CAUSING AFD TO GO OUT OF THE BAND. POWER WAS REDUCED TO LESS THAN 50% AS REQUIRED BY T.S. POWER REDUCTION TOOK MORE THAN THIRTY (30) MINUTES BECAUSE OF THE DESIRE TO PREVENT CHALLENGING SAFEGUARDS TO RAPID POWER REDUCTION WITH UNSTABLE PLANT CONDITIONS PRESENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

boron inj. was unanticipated but resultant effects were not zbs for 80-026

FAILURE + REACTOR, PWR + SURRY 1 (PWR) + FAILURE, OPERATOR ERROR + FAILURE, EQUIPMENT + NONLICENSED OPERATOR + ACTUATION + CONTAINMENT INTEGRITY + DILUTION + POISON, SOLUBLE + INSPECTION + SAFETY INJECTION + MAINTENANCE AND REPAIR + FAILURE, MAINTENANCE ERROR + MAINTENANCE PERSONNEL

6 ACCESSION NUMBER 156952 10-01 11/18/82

AXIAL FLUX DIFFERENCE EXCEEDS LIMIT AT SURRY 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 80-026 TO U.S. NRC, REGION 2, MAY 22, 1980, DOCKET 50-280,
TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 051280. POWER LEVEL - 080%. CAUSE - BORON ADDITION
DURING MAINTENANCE WITH THE UNIT AT COLD SHUTDOWN AND PARTIALLY DRAINED
EMERGENCY CORE COOLING SYSTEM (ECCS), WATER FLOWED TO THE REACTOR
WITHOUT CONTAINMENT INTEGRITY. THE PRIMARY SYSTEM BORON CONCENTRATION
WAS INITIALLY 3217 PPM AND WAS CHANGED TO 2567 PPM, THIS IS FAR ABOVE
THE 820 PPM REQUIRED FOR COLD SHUTDOWN AT THIS STAGE IN CORE LIFE. THE
FIRST SEGMENT OF THIS EVENT OCCURRED BECAUSE PROCEDURES DID NOT
PRECAUTION AGAINST GRAVITY FLOW FROM THE RWST TO THE CORE DURING VALVE
CYCLING. THE SUBSEQUENT SEGMENTS OF THIS EVENT STEMMED FROM AN
INADVERTENT ACTUATION OF SAFETY INJECTION BY ELECTRICIANS PERFORMING
MAINTENANCE IN THE RELAY RACKS RESULTING FROM IEB 79-25.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

abs. for 80-024

FAILURE + REACTOR, PWR + SURRY 1 (PWR) + FAILURE, OPERATOR ERROR + POWER
DISTRIBUTION + FLUX DISTRIBUTION + POISON, SOLUBLE + LICENSED OPERATOR +
CORE + HOT CHANNEL + DILUTION + MAIN COOLING SYSTEM

✓ ACCESSION NUMBER 157093 10-01 11/18/82

ECCS ACTUATED INADVERTENTLY AT NORTH ANNA 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 80-037 TO U.S. NRC, REGION 2, APR 16, 1980, DOCKET 50-338,
TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 040380. POWER LEVEL - 100%. CAUSE - MAIN STEAM TRIP
VALVE CLOSED. AT 2314, ACTUATION OF THE EMERGENCY CORE COOLING SYSTEM
WAS INITIATED ON HIGH STEAM LINE FLOW COINCIDENT WITH LOW STEAM LINE
PRESSURE DUE TO FULL CLOSURE OF MAIN STEAM TRIP VALVE TV-MS-101A DURING
PERIODIC PART-STROKE EXERCISING OF THE VALVE. THIS WAS AN INADVERTENT
ACTUATION (EQUIPMENT PERFORMED THE INTENDED INJECTION FUNCTION) AND THE
TRANSIENT WAS NOT SEVERE. IT IS BELIEVED THAT THE HIGH STEAM FLOW,
RESULTING FROM PERFORMING THE PARTIAL CLOSURE TEST AT FULL FLOW
CONDITIONS, ACTED TO PULL THE VALVE DISC DOWN AND THEN THE SMALL
REDUCTION IN AIR CYLINDER PRESSURE CAUSED THE TRIP VALVE TO CLOSE.
CORRECTIVE ACTION WAS FOR THE CONTROL ROOM OPERATORS TO PERFORM THE
APPLICABLE EMERGENCY PROCEDURE AND RETURN THE PLANT TO STABLE
CONDITIONS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA 1 (PWR) + FAILURE, EQUIPMENT +
FAILURE, INHERENT + TESTING + EMERGENCY COOLING SYSTEM + ACTUATION +
COOLING SYSTEM, SECONDARY + STEAM + FLOW + PRESSURE TRANSIENT + VALVES +
SAFETY INJECTION + INADVERTENT + SCRAM, REAL + REACTOR SHUTDOWN

ACCESSION NUMBER 157130 10-01 11/18/82

ECCS ACTUATES DUE TO PRESSURE EXCURSION AT ARKANSAS NUCLEAR 2
ARKANSAS POWER AND LIGHT CO.
LTR W/LER 80-015 TO U.S. NRC, REGION 4, MAY 13, 1980, DOCKET 50-368,
TYPE--PWR, MFG--COMB, AE--BECH
DATE OF EVENT - 012980. POWER LEVEL - 000%. CAUSE - VALVES NOT FULLY CLOSED. DURING THE RECOVERY FROM A PLANNED UNIT TRANSIENT PER POWER ESCALATION 100% TURBINE TRIP TEST, THE RCS DEPRESSURIZED TO A MINIMUM OF 1350 PSIA (SATURATION MARGIN = 57 F). ECCS SUBSYSTEM, SAFETY INJECTION ACTUATION SIGNAL (SIAS) ACTIVATED AT 1740 PSIA PER DESIGN. ENGINEERING EVALUATION DETERMINED THAT 1 CYCLE OF THE 160 ALLOWABLE SAFETY INJECTION NOZZLE THERMAL CYCLES WAS EXPERIENCED DURING THE TRANSIENT. THE RCS PRESSURE EXCURSION WAS LARGER THAN NORMAL AFTER A TURBINE TRIP DUE TO PRESSURIZER SPRAY VALVE 2CV-4651 AND STEAM DUMP & BYPASS VALVE 2CV-0301 NOT FULLY RECLOSING ON DEMAND. THE SPRAY VALVE OPERATOR AND THE DUMP VALVES ARE BEING MODIFIED FOR INCREASED RELIABILITY.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ARKANSAS NUCLEAR 2 (PWR) + REACTOR STARTUP EXPERIENCE + PRESSURIZER + VALVES + SAFETY INJECTION + VALVE OPERATORS + MODIFICATION + SPRAY + FAILURE, INHERENT + EASP + PRESSURE TRANSIENT + MAIN COOLING SYSTEM + ACTUATION

ACCESSION NUMBER 158645 10-01 11/18/82

EMERGENCY CORE COOLING SYSTEM ACTUATED AT NORTH ANNA 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 80-047 TO U.S. NRC, REGION 2, JUN 4, 1980, DOCKET 50-338,
TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 052380. POWER LEVEL - 000%. CAUSE - LOSS OF VITAL BUS. AT 0146, WITH THE UNIT AT HOT STANDBY, ACTUATION OF THE EMERGENCY CORE COOLING SYSTEM WAS INITIATED ON HIGH STEAM LINE FLOW SIGNAL COINCIDENT WITH LO-LO T-AVE. THIS WAS AN INADVERTENT ACTUATION (EQUIPMENT PERFORMED THE INTENDED INJECTION FUNCTION) AND THE REACTOR WAS ALREADY SHUTDOWN. THE HIGH STEAM FLOW SIGNAL RESULTED FROM A LOSS OF VITAL BUS 1-III. THE LO-LO T-AVE SIGNAL RESULTED FROM THE TRIPPING OF ALL THREE REACTOR COOLANT PUMPS WHICH WAS REQUIRED DUE TO THE LOSS OF VITAL BUS 1-III. CORRECTIVE ACTION WAS FOR THE CONTROL ROOM OPERATORS TO PERFORM THE APPLICABLE EMERGENCY PROCEDURE AND RETURN THE PLANT TO STABLE CONDITIONS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA 1 (PWR) + FAILURE, EQUIPMENT + FAILURE, INHERENT + ENGINEERED SAFETY FEATURE + SAFETY INJECTION + ACTUATION + ELECTRIC POWER + ELECTRIC POWER, VITAL + CONTROL + MAIN COOLING SYSTEM + PUMPS + GENERATORS + EASP

ACCESSION NUMBER 158860 10-01 11/18/82

DECAY HEAT FLOW LOST AT DAVIS-BESSE 1
TOLEDO EDISON CO.
LTR W/LER 80-029 TO U.S. NRC, REGION 3, APR 30, 1980, DOCKET 50-346,
TYPE--PWR, MFG--B&W, AE--BECH
DATE OF EVENT - 041980. POWER LEVEL - 000%. CAUSE - BREAKER OPENS DUE TO HIGH VIBRATION. BREAKER HBBF2 WAS OPENED BY A GROUND FAULT RELAY WHICH PROBABLY ACTUATED FROM MECHANICAL VIBRATION DUE TO CONSTRUCTION PERSONNEL IN THE ROOM. TWO ESSENTIAL BUSES WERE LOST DUE TO THE ELECTRICAL LINEUP FOR MAINTENANCE. SAFETY FEATURES ACTUATION RESULTED. THE DECAY HEAT PUMP HAD TO BE SHUTDOWN TO STOP THE INJECTION. AIR IN THE PIPING DELAYED THE PUMP RESTART. REACTOR COOLANT TEMPERATURE ROSE ABOVE THE 140F SPECIFICATION. THE DECAY HEAT PUMP WAS VENTED AND RESTARTED AT A MAXIMUM REACTOR COOLANT SYSTEM TEMPERATURE OF 170F.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + DAVIS-BESSE 1 (PWR) + REFUELING + SAFETY INJECTION + FAILURE, EQUIPMENT + FAILURE, INHERENT + ACTUATION + ELECTRIC POWER + BREAKER + VIBRATION + SHUTDOWN COOLING SYSTEM + FLOW + PUMPS + EASP + THERMAL TRANSIENT

ACCESSION NUMBER 159219 10-01 11/18/82

LOW BORON CONCENTRATION IN INJECTION TANK AT SEQUOYAH 1
TENNESSEE VALLEY AUTHORITY
LTR W/LER 80-112 TO U.S. NRC, REGION 2, JUL 31, 1980, DOCKET 50-327,
TYPE--PWR, MFG--WEST, AE--TVA
DATE OF EVENT - 062380. POWER LEVEL - 000%. CAUSE - INADVERTENT SAFETY INJECTION. INADVERTENT SI CAUSED THE BORON INJECTION TANK TO FALL BELOW THE REQUIRED BORON CONCENTRATION. WHEN CONTENTS OF BIT WERE PUMPED INTO THE REACTOR COOLANT SYSTEM, BORATED WATER FROM THE RWST WAS PUMPED IN BY CHARGING PUMPS. THE BORIC ACID TRANSFER PUMPS WERE REALIGNED TO RECIRCULATE THROUGH THE BIT FROM BORIC ACID STORAGE TANK TO RESTORE CONCENTRATION.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SEQUOYAH 1 (PWR) + REACTOR STARTUP EXPERIENCE + SAFETY INJECTION + ACTUATION + POISON, SOLUBLE + EMERGENCY COOLING SYSTEM + CONCENTRATION + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR

ACCESSION NUMBER 159240 10-01 11/18/82

SAFETY INJECTION INITIATION OCCURS AT SURRY 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 80-029 TO U.S. NRC, REGION 2, JUL 2, 1980, DOCKET 50-280,
TYPE--PWR, MFG--WEST, AE--S&W
DATE OF EVENT - 060380. POWER LEVEL - 100%. CAUSE - FIRE IN VITAL BUS.
VITAL BUS IV SOLA FAILED TO INITIATE AN AUTOMATIC TURBINE RUNBACK. THE
REACTOR TRIPPED ON STEAM GENERATOR LO LEVEL WITH FEED WATER MISMATCH. A
SAFETY INJECTION WAS INITIATED ON HI STEAM LINE FLOW WITH LO TAVE. A
SECOND SAFETY INJECTION WAS INITIATED DUE TO AN UNSUCCESSFUL CROSS
CONNECT ATTEMPT TO REENERGIZE VITAL BUS IV IN WHICH VITAL BUS II WAS
LOST. A FILTERING CAPACITOR IN THE SOLA-TRANSFORMER FAILED AND CAUGHT
FIRE DAMAGING THE SOLA AND CAUSING LOSS OF POWER TO VITAL BUS IV. THE
VITAL BUS WAS RE-ENERGIZED AND THE TRANSFORMER REPLACED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

? don't you always get a trip when you lose a vital bus?

FAILURE + REACTOR, PWR + SURRY 1 (PWR) + ACTUATION + SAFETY INJECTION +
REACTOR SHUTDOWN + EMERGENCY COOLING SYSTEM + FAILURE, EQUIPMENT +
FAILURE, INHERENT + TRANSFORMERS + ELECTRIC POWER + FIRE + CABLES AND
CONNECTORS + ELECTRICAL CONDUCTION

ACCESSION NUMBER 159553 10-01 11/18/82

SAFETY INJECTION ACTUATES AT SALEM 1
PUBLIC SERVICE ELECTRIC & GAS CO.
LTR W/LER 80-031 TO U.S. NRC, REGION 1, JUL 8, 1980, DOCKET 50-272,
TYPE--PWR, MFG--WEST, AE--PSE&G
DATE OF EVENT - 060880. POWER LEVEL - 100%. CAUSE - LIGHTNING HIT
CONTAINMENT PENETRATION. DURING A LIGHTNING STORM, THE SOUTH
PENETRATION AREA WAS HIT AT 0844 HOURS RESULTING IN A SEVERE TRANSIENT
ON 7 MAIN STEAM PRESSURE TRANSMITTERS CAUSING 2 TO FAIL. A SAFETY
INJECTION OCCURRED FOR FOUR MINUTES. PREVIOUS OCCURRENCES: 77-50/01T.
A TRANSIENT ON MAIN STEAM LINE PRESSURE TRANSMITTERS WAS CAUSED BY
LIGHTNING STRIKING THE SOUTH PENETRATION AREA. THE TRANSMITTERS WERE
REPLACED IN KIND AND ARE TO BE ANALYZED TO DETERMINE THE CAUSE OF THEIR
FAILURE. SUITABLE SURGE PROTECTION IS BEING INVESTIGATED AND THE
RESULTS REPORTED IN A FOLLOW-UP REPORT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 1 (PWR) + REACTOR SHUTDOWN + SAFETY
INJECTION + ACTUATION + FAILURE, INSTRUMENT + LIGHTNING + STEAM +
SENSORS, PRESSURE + REACTOR PROTECTION SYSTEM + INSTRUMENT, PROTECTIVE +
FAILURE, COMMON MODE + CONTAINMENT PENETRATION

ACCESSION NUMBER 159786 10-01 11/18/82

MAXIMUM PRESSURIZER HEATUP RATE EXCEEDED AT SEQUOYAH 1
TENNESSEE VALLEY AUTHORITY
LTR W/LER 80-082 TO U.S. NRC, REGION 2, JUN 23, 1980, DOCKET 50-327,
TYPE--PWR, MFG--WEST, AE--TVA
DATE OF EVENT - 052680. POWER LEVEL - 000%. CAUSE - LICENSED OPERATOR ERROR. THE PRESSURIZER HEATUP RATE EXCEEDED THE MAXIMUM HEATUP RATE OF 100 F IN ANY ONE HOUR PERIOD DURING THE RECOVERY PHASE OF A SAFETY INJECTION. TEMPERATURE CHANGED AT A RATE OF 151 F IN 30 MINUTES. THE SAFETY INJECTION CAUSED AN INSURGE INTO THE PRESSURIZER CAUSING THE TEMPERATURE TO DROP. DURING THE RECOVERY THE LEVEL WAS RESTORED TO NORMAL CAUSING THE COOLER WATER TO BE DISPLACED OUT AND THE TEMPERATURE TO INCREASE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SEQUOYAH 1 (PWR) + REACTOR STARTUP EXPERIENCE + PRESSURIZER + TEMPERATURE + SAFETY INJECTION + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + THERMAL TRANSIENT

ACCESSION NUMBER 160405 10-01 11/18/82

SAFETY INJECTION OCCURS AT SEQUOYAH 1
TENNESSEE VALLEY AUTHORITY
LTR W/LER 80-004S TO U.S. NRC, REGION 2, SEP 17, 1980, DOCKET 50-327,
TYPE--PWR, MFG--WEST, AE--TVA
DATE OF EVENT - 062380. POWER LEVEL - 000%. CAUSE - OPERATOR OPENED ALL FOUR MSIV'S. THE REACTOR COOLANT SYSTEM TEMPERATURE FELL BELOW 541F (LO-LO TAVE) TO 538F. THE OPERATOR OPENED ALL FOUR MSIV'S WHICH BROUGHT IN THE HI STEAM FLOW SIGNAL. APPROXIMATELY 540 GALLONS OF BORATED WATER WAS INJECTED INTO THE RCS AT 190F. THE SAFETY INJECTION WAS TERMINATED, TWO MINUTES AFTER THE SI SIGNAL WAS ACTIVATED, AND THE REACTOR COOLANT SYSTEM WAS RESTORED TO PRE-INITIATION CONDITIONS. OPERATIONS PERSONNEL HAVE BEEN CAUTIONED TO MORE CAREFULLY OBSERVE PLANT CONDITIONS THAT COULD LEAD TO A SAFETY INJECTION DURING ABNORMAL PLANT OPERATIONS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SEQUOYAH 1 (PWR) + SAFETY INJECTION + ACTUATION + EMERGENCY COOLING SYSTEM + FAILURE, OPERATOR ERROR + VALVES + LICENSED OPERATOR + STEAM + REACTOR STARTUP EXPERIENCE + THERMAL TRANSIENT + THERMAL CONSIDERATION

ACCESSION NUMBER 160846 10-01 11/18/82

UPDATE REPORT ON FEB. 26, 1980 TRANSIENT AT CRYSTAL RIVER 3
FLORIDA POWER CORPORATION
LTR W/LER 80-10-01T-1 TO U.S. NRC, REGION 2, MAR 25, 1980, DOCKET 50-302, TYPE--PWR, MFG--B&W, AE--GIL
DATE OF EVENT - 022680. POWER LEVEL - 100%. CAUSE - LOSS OF 24 VDC NON-NUCLEAR INSTRUMENT POWER SUPPLY. THE 24 VDC NON-NUCLEAR INSTRUMENTATION "X" POWER SUPPLY WAS LOST DUE TO A FAILURE OF A BUFFER CARD. THIS CAUSED THE EVENTS TO OCCUR: PORV OPENED AND LATCHED AT A RCS PRESSURE BELOW ITS SETPOINT, INTEGRATED CONTROL SYSTEM (ICS) RECEIVED ERRONEOUS SIGNALS AND DEMANDED 103% FULL POWER, FEEDWATER FLOW WAS RUN BACK AND STEAM FLOW INCREASED. THIS CAUSED REACTOR TO TRIP ON HIGH PRESSURE. THE PORV OPENING CAUSED RAPID DEPRESSURIZATION AND ESFAS WAS ACTUATED WITH 2 HPI PUMPS STARTING. STEAM GENERATOR BOILED DRY DUE TO LOW FEEDWATER FLOW. FOLLOWING INSTRUMENTATION WAS LOST: CONTAINMENT ATMOSPHERE RADIATION MONITORING SYSTEM, REMOTE SHUTDOWN INSTRUMENTATION SUCH AS MAKEUP TANK LEVEL, PRESSURIZER LEVEL, RCS LOOP A AND B T(SUB H), POST ACCIDENT MONITORING INSTRUMENTATION SUCH AS PRESSURIZER LEVEL LT-1, -2, -3, RCS FLOW AND OTSG "A" OUTLET PRESSURE, FW FLOW "A" & "B". CORE COVERAGE WAS MAINTAINED, NO FUEL DAMAGE OCCURRED AND RCS PRESSURE REMAINED WITHIN ALLOWABLE LIMITS, AND NO APPRECIABLE RADIATION WAS RELEASED TO ENVIRONMENT. PLANT WAS STABILIZED AND PLACED IN COLD SHUTDOWN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CRYSTAL RIVER 3 (PWR) + INCIDENT + SAFETY INJECTION + INSTRUMENT, NON-NUCLEAR + ACTUATION + STEAM GENERATOR + REACTOR CONTROL + UPDATE + REACTOR SHUTDOWN + CIRCULATION, NATURAL + PRESSURE RELIEF + VALVES + FEEDWATER + FLOW + PRESSURIZER + COOLING SYSTEM, SECONDARY + MAIN COOLING SYSTEM + SCRAM, REAL + TRANSIENT + ACCIDENT, LOSS OF POWER + ON SITE + SOLID STATE DEVICE + FAILURE, INSTRUMENT + ELECTRIC POWER + FAILURE, EQUIPMENT + FAILURE, SEQUENTIAL + FAILURE, COMMON MODE + REACTOR PROTECTION SYSTEM

ACCESSION NUMBER 160915 10-01 11/18/82

REACTOR COOLANT TEMPERATURE FALLS BELOW LIMIT AT SEQUOYAH 1
TENNESSEE VALLEY AUTHORITY
LTR W/LER 80-162 TO U.S. NRC, REGION 2, OCT 31, 1980, DOCKET 50-327, TYPE--PWR, MFG--WEST, AE--TVA
DATE OF EVENT - 100280. POWER LEVEL - 005%. CAUSE - OVER FEEDING WATER TO STEAM GENERATORS. THE REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE DECREASED BELOW T.S. LIMIT OF 541 DEGREES F. OVER FEEDING WATER TO THE STEAM GENERATORS CAUSED THE RCS TEMPERATURE TO DECREASE BELOW LIMIT. FEEDWATER FLOW WAS DECREASED ALLOWING THE RCS TEMPERATURE TO INCREASE ABOVE THE LOWER LIMIT WITHIN A FEW MINUTES ON BOTH OCCASIONS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SEQUOYAH I (PWR) + MAIN COOLING SYSTEM + REACTOR COOLANT + TEMPERATURE + HEAT EXCHANGERS + STEAM GENERATOR + WATER + FAILURE, EQUIPMENT + FAILURE, DESIGN ERROR + REACTOR STARTUP EXPERIENCE + THERMAL TRANSIENT + FEEDWATER + COOLING SYSTEM, SECONDARY

ACCESSION NUMBER 161826 10-01 11/18/82

INADVERTENT SAFETY INJECTION OCCURS AT ZION 2
COMMONWEALTH EDISON CO.
LTR W/LER 80-026 TO U.S. NRC, REGION 3, DEC 2, 1980, DOCKET 50-304,
TYPE--PWR, MFG--WEST, AE--S&L, DCS NO.--8012150093
DATE OF EVENT - 090380. POWER LEVEL - 000%. CAUSE - PRESSURE TRANSMITTER SPIKE DUE TO WATER HAMMER. FOLLOWING A REACTOR TRIP AN INADVERTANT SAFETY INJECTION OCCURRED WHEN SECONDARY SIDE LOOPS 2A AND 2C SENSED A FALSE HIGH PRESSURE DIFFERENTIAL SIGNAL. VISUAL INSPECTIONS INDICATED NO STRUCTURAL DAMAGE IN THE CONTAINMENT OR REACTOR COOLANT SYSTEM. THE FALSE SIGNAL WAS INITIATED BY A WATER HAMMER IN THE 2B FW LINE. THE RESULTING SHOCK CAUSED 2B AND 2D STEAM LINE PRESSURE TRANSMITTERS TO SPIKE HIGH. MODIFICATIONS ARE CURRENTLY UNDERWAY TO INSTALL "J" TUBES IN THE FEEDWATER RINGS OF THE STEAM GENERATORS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 2 (PWR) + FAILURE, INSTRUMENT + FAILURE, DESIGN ERROR + MODIFICATION + SENSORS, PRESSURE + COOLING SYSTEM, SECONDARY + STEAM + IMPACT SHOCK + WATER + SAFETY INJECTION + ACTUATION + HYDRAULIC EFFECT + PRESSURE PULSE + CONDENSATION + OSCILLATION

ACCESSION NUMBER 162394 see 166648 10-01 11/18/82

INADVERTENT SAFETY INJECTION ACTUATION AT CRYSTAL RIVER 3
FLORIDA POWER CORPORATION
LTR W/LER 80-003S TO U.S. NRC, REGION 2, JAN 12, 1981, DOCKET 50-302,
TYPE--PWR, MFG--B&W, AE--GIL, DCS NO.--8101210512
DATE OF EVENT - 101680. POWER LEVEL - 097%. CAUSE - RELAY FAILURE DURING CHANNEL TEST. DURING NORMAL OPERATION CHANNEL 1 PRESSURE MODULE WAS IN "TEST GIVING A 1 OF 3 CONDITION FOR A ROUTINE SURVEILLANCE TEST. A FUSE IN CHANNEL RB2 BLEW, GIVING THE 2 OF 3 CONDITION REQUIRED FOR ACTUATION OF ES CHANNEL "B". MAKEUP PUMP 1B OPERATED FOR LESS THAN A MINUTE WITH SUCTION VALVES CLOSED BEFORE IT WAS SHUT DOWN. THE TRANSIENT CAUSED APPROXIMATELY 250 GALLONS OF WATER FROM THE BORATED WATER STORAGE TANK TO BE INJECTED INTO THE REACTOR COOLANT SYSTEM WITH AN ATTENDANT PRESSURE INCREASE FROM 2145 TO 2185 PSIG. THE CAUSE IS A FAILED RELAY (63Z3D/2) IN ENGINEERING SAFEGUARDS CABINET 5B.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CRYSTAL RIVER 3 (PWR) + FAILURE, INSTRUMENT + FAILURE, INHERENT + TESTING + SAFETY INJECTION + ACTUATION + RELAYS

I-012
I-242

ACCESSION NUMBER 162919 10-01 11/18/82

NRC STAFF PROPOSES TO FINE CONSOLIDATED EDISON $210,000
U.S. NUCLEAR REGULATORY COMMISSION
NRC NEWS RELEASE 80-224 +. 1 PG, FOR WEEK ENDING DEC. 16, 1980
ON OCTOBER 20 CONSOLIDATED EDISON REPORTED TO NRC THAT LEAKS IN A FAN COOLER UNIT HAD RESULTED IN THE FLOODING OF THE REACTOR VESSEL PIT AND CONTAINMENT FLOOR WITH ABOUT 100,000 GALLONS OF RIVER WATER. THE LOWER PART OF THE HOT REACTOR VESSEL WAS WETTED WITH COLD RIVER WATER. THE FAN COOLERS ARE USED TO COOL THE AIR INSIDE THE CONTAINMENT BUILDING. THE WATER WAS DISCOVERED BY WORKERS WHO ENTERED THE CONTAINMENT BUILDING ON OCTOBER 17 FOLLOWING A SHUTDOWN OF THE PLANT.
AVAILABILITY - NRC, OFFICE OF PUBLIC AFFAIRS, WASHINGTON, D.C. 20555

AGENCY, NRC + INDIAN POINT 2 (PWR) + REACTOR, PWR + ECONOMICS + LEAK + CONTAINMENT + FLOOD + PRESSURE VESSELS + REFUELING

ACCESSION NUMBER 163421 1111-01 2 03/07/83

SAFETY INJECTION ACTUATION AT FARLEY 1
ALABAMA POWER CO.
LTR W/LER 80-000S TO U.S. NRC, REGION 2, JAN 21, 1980, DOCKET 50-348, TYPE--PWR, MFG--WEST, AE--BECH, DCS NO.--8001290457
DATE OF EVENT - 102479. POWER LEVEL - NA. CAUSE - FAULTY LOGIC CARD. WHILE THE PLANT WAS IN MODE 4, THE A TRAIN SI (SAFETY INJECTION) BLOCKED LIGHT WENT OFF AND A SI ACTUATED ON A TRAIN. THE SHIFT FOREMAN VERIFIED THAT PLANT PARAMETERS DID NOT WARRANT A SI AND TERMINATED THE SI. SINCE THE EXACT CAUSE OF THE SPURIOUS SI COULD NOT BE DETERMINED, THE MOST SUSPECT ELEMENT IN THE CHAIN WAS REPLACED, P/O UNIVERSAL CARD A416. THERE HAVE BEEN A TOTAL OF SEVEN (7) INADVERTENT SAFETY INJECTIONS (IN ALL MODES) TO DATE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + FARLEY 1 (PWR) + EMERGENCY COOLING SYSTEM + SAFETY INJECTION + ACTUATION + FAILURE, INSTRUMENT + FAILURE, INHERENT + ELECTRONIC FUNCTION UNITS

ACCESSION NUMBER 163551 10-01 11/18/82

REACTOR COOLANT TEMPERATURE BELOW LIMIT TWICE AT NORTH ANNA 2
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 81-001 TO U.S. NRC, REGION 2, JAN 15, 1981, DOCKET 50-339, TYPE--PWR, MFG--WEST, AE--S&W, DCS NO.--8101230593
DATE OF EVENT - 010181. POWER LEVEL - 002%. CAUSE - OVERBORATION OF VOLUME CONTROL TANK. THE AVERAGE REACTOR COOLANT SYSTEM TEMPERATURE WAS LESS THAN 541F ON TWO OCCASIONS. HOWEVER, ON EACH OCCASION, THE TEMPERATURE WAS RESTORED TO WITHIN ITS LIMITS WITHIN 15 MINUTES. THE FIRST EVENT OCCURRED AS A RESULT OF THE VOLUME CONTROL TANK (VCT) BEING INADVERTENTLY OVERBORATED WHILE EXCESS LETDOWN WAS IN SERVICE, AND PRIOR TO INITIATING NORMAL LETDOWN. THE SECOND EVENT OCCURRED AS A RESULT OF OVERCOOLING THE PRIMARY WITH THE CONDENSER STEAM DUMPS. TEMPERATURE WAS RESTORED BY SECURING STEAM GENERATOR BLOWDOWN, WITHDRAWING RODS AND DILUTING THE RCS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA 2 (PWR) + REACTOR COOLANT + TEMPERATURE + POISON, SOLUBLE + CONCENTRATION + STORAGE CONTAINER + ACCUMULATORS + MAIN COOLING SYSTEM + FAILURE, EQUIPMENT + FAILURE, ADMINISTRATIVE CONTROL + PROCEDURES AND MANUALS + HEAT TRANSFER + CONDENSER + BLOWDOWN

ACCESSION NUMBER 164149 10-01 11/18/82

L5003

LETDOWN PIPING LEAKS AT ROBINSON 2
CAROLINA POWER & LIGHT CO.
LTR W/LER 81-005 TO U.S. NRC, REGION 2, FEB 12, 1981, DOCKET 50-261, TYPE--PWR, MFG--WEST, AE--EBASCO, DCS NO.--8102180486
DATE OF EVENT - 012981. POWER LEVEL - 000%. CAUSE - PIPE CAP BLOWN OFF AND VALVE OPENED BY SI. A SPURIOUS SAFEGUARDS ACTUATION AND REACTOR TRIP WERE RECEIVED RESULTING FROM A HIGH STEAM FLOW SPIKE CAUSED BY TURBINE GOVERNOR VALVE/E-H OIL PROBLEMS. LETDOWN FLOW WAS RESTORED. RCS PRESSURE BEGAN DECREASING WITH CONTAINMENT PRESSURE AND DEW POINT INCREASING. UPON CONTAINMENT ENTRY FOLLOWING LETDOWN ISOLATION A LETDOWN LINE DRAIN VALVE (CVCS-200E) WAS FOUND PARTIALLY OPENED AND LEAKING THROUGH WITH ITS PIPE END-CAP MISSING. CAUSE IS THE VALVE VIBRATING OPEN AND THE PIPE CAP, WHICH WAS SERVING AS A PRESSURE BOUNDARY, FAILING TO HOLD AT SOMETIME FOLLOWING THE FIRST SI. APPROXIMATELY 4500-6000 GALLONS OF WATER LEAKED TO THE CONTAINMENT SUMP DURING THE EVENT. LETDOWN WAS SECURED. THE VALVE WAS SECURED IN THE CLOSED POSITION AND A NEW CAP WAS INSTALLED.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ROBINSON 2 (PWR) + SAFETY INJECTION + ACTUATION + LEAK + PIPES AND PIPE FITTINGS + VIBRATION + FAILURE, EQUIPMENT + FAILURE, INHERENT + EMERGENCY COOLING SYSTEM + VALVES + MAIN COOLING SYSTEM

ACCESSION NUMBER 164791 10-01 11/18/82

UPDATE ON SAFETY INJECTION ACTUATION AT BEAVER VALLEY 1
DUQUESNE LIGHT CO.
LTR W/LER 80-094R TO U.S. NRC, REGION 1, MAR 2, 1981, DOCKET 50-334, TYPE--PWR, MFG--WEST, AE--S&W, DCS NO.--8103170508
DATE OF EVENT - 111780. POWER LEVEL - 000%. CAUSE - LICENSED OPERATOR ERROR. THE STATION OPERATING SUPERVISOR (SOS) WAS NOTIFIED OF A STEAM LINE LOW PRESSURE BISTABLE (BS-MS-485) INTERMITTENTLY TRIPPING AND RESETTING. WHEN THE SOS ACCOMPANIED THE OPERATOR TO THE PROCESS RACKS TO TRIP THE BISTABLE, THEY OPERATED THE WRONG CHANNEL (BS-MS-484). SAFETY INJECTIONS TO DATE: 15 OPERATIONAL AND 2 PRE-OPERATIONAL. THE SOS DID NOT LOOK AT THE STATUS PANEL HIMSELF, BUT TOOK THE WORD OF ANOTHER PERSON INVOLVED IN PLANT OPERATIONS. THE REQUIRED READING INCIDENT REPORT DISTRIBUTION IS BEING USED TO INDICATE THE NEED TO RECHECK IMPORTANT EVOLUTIONS SUCH AS TRIPPING BISTABLES. A MALFUNCTIONING SIGNAL CONVERTER CARD WAS THE CAUSE FOR FLUTTERING BISTABLE (BS-MS-485).
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + BEAVER VALLEY 1 (PWR) + FAILURE, INSTRUMENT + UPDATE + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + SAFETY INJECTION + ACTUATION + REACTOR SHUTDOWN + SCRAM, REAL + IDENTIFICATION

ACCESSION NUMBER 164865 10-01 11/18/82

UPDATE REPORT ON VITAL INSTRUMENT BUS FAILURE AT SALEM 1
PUBLIC SERVICE ELECTRIC & GAS CO.
LTR W/LER 78-073R TO U.S. NRC, REGION 1, MAR 25, 1981, DOCKET 50-272, TYPE--PWR, MFG--WEST, AE--PSE&G, DCS NO.--8104020481
DATE OF EVENT - 112778. POWER LEVEL - 100%. CAUSE - SUPPLY INVERTER FAILURE. DURING NORMAL OPERATION, LOSS OF 1B VITAL INSTRUMENT BUS DE-ENERGIZED "RCP BREAKER OPEN" RELAY IN SSPS INITIATING A REACTOR TRIP. SEVEN MINUTES LATER, A SAFETY INJECTION OCCURRED AND 1B DIESEL, NO. 11 CHARGING PUMP AND NO. 12 RHR PUMP FAILED TO START. THIS IS THE THIRD OCCURRENCE OF THIS TYPE (78-51 AND 78-57). THE CAUSE OF THE BUS FAILURE WAS DUE TO FAILURE OF THE OUTPUT TRANSFORMER IN THE SUPPLY INVERTER. THE TRANSFORMER AND REGULATING RESISTORS WERE REPLACED. FAILURE OF EQUIPMENT TO START ON SEC INITIATION WAS DUE TO LOSS OF THE BUS AND IS CONSIDERED NORMAL FOR THIS OCCURRENCE. NO. 13 AUXILIARY FEED PUMP OVERSPEED TRIP RESET LATCH WAS ADJUSTED FOR MORE RELIABLE LATCHING.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 1 (PWR) + ENGINEERED SAFETY FEATURE + REACTOR SHUTDOWN + SCRAM, REAL + SAFETY INJECTION + ACTUATION + ON SITE + ELECTRIC POWER + GENERATORS + FAILURE, EQUIPMENT + FAILURE, INHERENT + UPDATE

ACCESSION NUMBER 165876 10-01 11/18/82

SAFETY INJECTION ACTUATED AT NORTH ANNA 1
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 81-018 TO U.S. NRC, REGION 2, APR 20, 1981, DOCKET 50-338, TYPE--PWR, MFG--WEST, AE--S&W, DCS NO.--8104240565
DATE OF EVENT - 032981. POWER LEVEL - 000%. CAUSE - LICENSED OPERATOR ERROR. BOTH POWER OPERATED RELIEF VALVES LIFTED IN ORDER TO MITIGATE A REACTOR COOLANT SYSTEM PRESSURE TRANSIENT CAUSED BY AN INADVERTENT SAFETY INJECTION WHILE OPERATING IN SOLID PLANT CONDITIONS. THE CAUSE WAS AN INADVERTENT UNBLOCKING OF THE LOW PRESSURIZER PRESSURE SAFETY INJECTION SIGNAL DURING TRAIN B SOLID STATE PROTECTION TESTING. THE BORON INJECTION TANK DISCHARGE WAS STOPPED, THE SAFETY INJECTION SIGNAL WAS RESET, AND RCS PRESSURE WAS REDUCED BELOW PORV SETPOINT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA 1 (PWR) + FAILURE, INSTRUMENT + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + TESTING + SAFETY INJECTION + ACTUATION + PRESSURIZER + SENSORS, PRESSURE + PRESSURE RELIEF + VALVES + PRESSURE TRANSIENT + EASP

ACCESSION NUMBER 166342 10-01 11/18/82

INADVERTENT SAFETY INJECTION DILUTES BIT AT SALEM 2
PUBLIC SERVICE ELECTRIC & GAS CO.
LTR W/LER 81-009 TO U.S. NRC, REGION 1, MAY 22, 1981, DOCKET 50-311, TYPE--PWR, MFG--WEST, AE--PSE&G, DCS NO.--81060204O
DATE OF EVENT - 042381. POWER LEVEL - 000%. CAUSE - INADVERTENT VALVE OPENING. DURING COOLDOWN OPERATION, VALVE 24MS10 OPENED, CAUSING ALARM "24 S/G STM. PRESS. 100 PSI LO" TO INITIATE SAFETY INJECTION. INVESTIGATION OF VALVE 24MS10 COULD FIND NO APPARENT REASON FOR ITS OPENING. THE INADVERTENT OPENING OF VALVE 24MS10 CAUSED THE BORON CONCENTRATION TO DROP BELOW THE MINIMUM OF 20,000 PPM IN THE BORON INJECTION TANK. THE BORON INJECTION TANK WAS BORATED. PREVIOUS SIMILAR OCCURRENCES: 77-01, 77-26, 77-29, 78-04, 78-74 (ON UNIT 1) AND 80-31 (ON UNIT 2).
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 2 (PWR) + EMERGENCY COOLING SYSTEM + SAFETY INJECTION + ACTUATION + VALVES + POISON, SOLUBLE + CONCENTRATION + STORAGE CONTAINER + FAILURE, EQUIPMENT + FAILURE, INHERENT + REACTOR STARTUP EXPERIENCE + DILUTION + COOLING SYSTEM, SECONDARY + STEAM

ACCESSION NUMBER 166453 10-01 11/18/82

TWO INADVERTENT SAFETY INJECTION ACTUATIONS AT SEQUOYAH 1
TENNESSEE VALLEY AUTHORITY
LTR W/LER 81-002S TO U.S. NRC, REGION 2, JUN 5, 1981, DOCKET 50-327,
TYPE--PWR, MFG--WEST, AE--TVA, DCS NO.--81060904 1
DATE OF EVENT - 042381. POWER LEVEL - NA. CAUSE - PROCEDURAL
DEFICIENCY. ON APRIL 23, 1981, DURING A MAIN TURBINE OVERSPEED TEST, AN
INADVERTENT SAFETY INJECTION OCCURRED. THE SIGNAL WHICH BROUGHT IN THE
SAFETY INJECTION WAS FROM A COMBINATION OF HIGH STEAMLINE FLOW
COINCIDENT WITH LOW-LOW TAVE. THIS SI CAUSED THE CENTRIFUGAL CHARGING
PUMPS TO PUMP WATER THROUGH THE BIT (170F) INTO THE RCS (539.8F) FOR
ABOUT 2-3 MINUTES. ON MAY 1, 1981 DURING GENERATOR TRIP TEST RECOVERY,
AN INADVERTENT SI OCCURRED. IT WAS BROUGHT IN BY THE RECEPTION OF TWO
HIGH STEAMLINE DIFFERENTIAL PRESSURE SIGNALS. THIS SI CAUSED THE
CENTRIFUGAL CHARGING PUMPS TO PUMP WATER THROUGH THE BIT (170F) INTO THE
RCS (500F) AND EVENTUALLY BROUGHT IN WATER FROM THE RWST.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SEQUOYAH 1 (PWR) + TESTING + EMERGENCY COOLING
SYSTEM + SAFETY INJECTION + ACTUATION + FAILURE, EQUIPMENT + FAILURE,
ADMINISTRATIVE CONTROL + PIPES AND PIPE FITTINGS + NOZZLE + FATIGUE +
REACTOR STARTUP EXPERIENCE + PROCEDURES AND MANUALS

ACCESSION NUMBER 166648 see 162394 10-01 11/18/82

UPDATE ON ECCS ACTUATION AT CRYSTAL RIVER 3
FLORIDA POWER CORPORATION Jan 12, 81
LTR W/LER 80-003SR TO U.S. NRC, REGION 2, JUN 11, 1981, DOCKET 50-302,
TYPE--PWR, MFG--B&W, AE--GIL, DCS NO.--8106170137
DATE OF EVENT - 101680. POWER LEVEL - 097%. CAUSE - RELAY FAILURE.
CHANNEL I PRESSURE MODULE WAS IN "TEST", GIVING A 1 OF 3 CONDITION FOR A
ROUTINE SURVEILLANCE TEST. A FUSE IN CHANNEL RB2 FAILED, GIVING THE 2
OUT OF 3 CONDITION REQUIRED FOR ACTUATION OF ES CHANNEL "B". MAKEUP 1B
OPERATED FOR LESS THAN ONE MINUTE WITH SUCTION VALVES CLOSED BEFORE IT
WAS SHUT DOWN. OTHER AUTOMATIC ACTIONS WERE MANUALLY OVERRIDDEN UNTIL
THE ES "B" ACTUATION CONDITION COULD BE CLEARED BY TAKING CHANNEL 1
PRESSURE MODULE OUT OF THE "TEST" MODE. THE TRANSIENT CAUSED
APPROXIMATELY 250 GALLONS OF WATER FROM THE BORATED WATER STORAGE TANK
TO BE INJECTED INTO THE REACTOR COOLANT SYSTEM WITH AN ATTENDANT
PRESSURE INCREASE FROM 2145 TO 2185 PSIG. THE FLOWPATH WAS THROUGH
MUV-25 AND MUV-26. THE CAUSE IS A FAILED RELAY (6373D/2) IN ENGINEERING
SAFEGUARDS CABINET 5B. THE RELAY WAS A CLARKE 5UK8-33-76 RELAY.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + CRYSTAL RIVER 3 (PWR) + EMERGENCY COOLING SYSTEM
+ ACTUATION + UPDATE + TESTING + RELAYS + FUSE + FAILURE, EQUIPMENT +
FAILURE, INHERENT + SAFETY INJECTION

ACCESSION NUMBER 166686 10-01 11/18/82

SAFETY INJECTION ACTUATED TWICE AT MCGUIRE 1
DUKE POWER CO.
LTR W/LER 81-037 TO U.S. NRC, REGION 2, APR 28, 1981, DOCKET 50-369,
TYPE--PWR, MFG--WEST, AE--DUKE, DCS NO.--8106290348
DATE OF EVENT - 033081. POWER LEVEL - 000%. CAUSE - MAINTENANCE PERSONNEL ERROR. DURING MAINTENANCE WORK ON THE SOLID STATE PROTECTION SYSTEM CABINETS, AN INADVERTENT SAFETY INJECTION WAS INITIATED. THE SAFETY INJECTION PUMP AND THE CENTRIFUGAL CHARGING PUMP BREAKERS HAD BEEN DISCONNECTED AND THE RHR SYSTEM WAS ALIGNED IN THE NORMAL SHUTDOWN COOLING MODE. NO SIGNIFICANT VOLUME OF WATER WAS ADDED TO THE RCS. TECHNICIANS WORKING ON THE SOLID STATE PROTECTION SYSTEM CABINET PLACED THE "INPUT ERROR INHIBIT" SWITCH IN THE "INHIBIT" POSITION AND THE MODE SELECTOR SWITCH IN THE "TEST" POSITION. BEFORE COMPLETING THE WORK THEY LEFT THE AREA AND THEIR SUPERVISOR, ASSUMING THE WORK WAS DONE, RETURNED THE SWITCHES TO THEIR NORMAL POSITIONS. THIS ACTUATED THE SI SYSTEM. PERSONNEL WERE RETRAINED AND PROCEDURES WERE REWRITTEN TO PRECLUDE SUCH OCCURRENCES FROM HAPPENING AGAIN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + MCGUIRE 1 (PWR) + REACTOR STARTUP EXPERIENCE + EMERGENCY COOLING SYSTEM + SAFETY INJECTION + ACTUATION + FAILURE, EQUIPMENT + FAILURE, MAINTENANCE ERROR + MAINTENANCE PERSONNEL + ENGINEERED SAFETY FEATURE + MAINTENANCE AND REPAIR

ACCESSION NUMBER 166746 10-01 11/18/82

INSTRUMENT INVERTER FAILURE CAUSES REACTOR TRIP AT ZION 1
COMMONWEALTH EDISON CO.
LTR W/LER 81-022 TO U.S. NRC, REGION 3, JUN 26, 1981, DOCKET 50-295,
TYPE--PWR, MFG--WEST, AE--S&L, DCS NO.--8107080234
DATE OF EVENT - 052881. POWER LEVEL - 098%. CAUSE - INVERTER SHORTED OUT. INSTRUMENT INVERTER 114 FAILED CONCURRENT WITH PERFORMANCE OF INSTRUMENT FUNCTIONALS, REQUIRING REACTOR PROTECTION BISTABLES TO BE TRIPPED, WHICH RESULTED IN A REACTOR TRIP. THE TRIP GENERATED A HIGH STEAM FLOW SIGNAL WHICH CONCURRENT WITH THE INVERTER FAILURE YIELDED A SAFETY INJECTION. THE PRIMARY WINDING OF THE MASTER SOLA OUTPUT TRANSFORMER WAS DETERMINED TO BE SHORTED. THE TRANSFORMER WAS SENT TO THE MANUFACTURER FOR FAILURE ANALYSIS AND A TASK FORCE FORMED TO FURTHER INVESTIGATE THE PROBLEM WITH INPUT FROM THE VENDOR.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 1 (PWR) + ELECTRIC POWER + REACTOR SHUTDOWN + TESTING + FAILURE, EQUIPMENT + FAILURE, INHERENT + SCRAM, SPURIOUS + SAFETY INJECTION + ACTUATION + EASP + GENERATOR, MOTOR

ACCESSION NUMBER 166854 10-01 11/18/82

BIT CONCENTRATION FALLS BELOW LIMIT AT SALEM 2
PUBLIC SERVICE ELECTRIC & GAS CO.
LTR W/LER 81-032 TO U.S. NRC, REGION 1, JUN 30, 1981, DOCKET 50-311, TYPE--PWR, MFG--WEST, AE--PSE&G, DCS NO.--8107130243
DATE OF EVENT - 060381. POWER LEVEL - 020%. CAUSE - DILUTION DUE TO SAFETY INJECTION. A HIGH LEVEL IN NO. 22 STEAM GENERATOR RESULTED IN A TURBINE TRIP, REACTOR TRIP AND SAFETY INJECTION, BORON INJECTION TANK CONCENTRATION DECREASED TO LESS THAN 20,000 PPM OF BORON. PREVIOUS SIMILAR OCCURRENCE: 81-09. THE SAFETY INJECTION CAUSED DILUTION OF THE BORON INJECTION TANK. THE TANK WAS BORATED UNTIL CONCENTRATION WAS WITHIN THE ALLOWABLE LIMITS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 2 (PWR) + REACTOR SHUTDOWN + SAFETY INJECTION + ACTUATION + EMERGENCY COOLING SYSTEM + STORAGE CONTAINER + POISON, SOLUBLE + CONCENTRATION + FAILURE, EQUIPMENT + FAILURE, INHERENT + DILUTION + REACTOR STARTUP EXPERIENCE

ACCESSION NUMBER 167124 10-01 11/18/82

UPDATE ON SAFETY INJECTION ACTUATION AT ZION 2
COMMONWEALTH EDISON CO.
LTR W/LER 81-008R TO U.S. NRC, REGION 3, JUN 29, 1981, DOCKET 50-304, TYPE--PWR, MFG--WEST, AE--S&L, DCS NO.--8107280217
DATE OF EVENT - 050781. POWER LEVEL - 092%. CAUSE - MASTER INVERTER SHORTED OUT. UNIT TWO TRIPPED AND SAFETY INJECTED DUE TO THE LOSS OF INVERTER 213. LOSS OF 213 CAUSED ONE FEEDWATER PUMP TO TRIP DUE TO SPEED CONTROL. ALL SAFETY EQUIPMENT PERFORMED ITS INTENDED FUNCTION. THE MASTER INVERTER OUTPUT SOLA TRANSFORMER SHORTED TO GROUND WHICH TRIPPED THE INVERTER AND CAUSED THE SLAVE TRANSFORMER TO BURN UP. THE STATION IS TRENDING THE PERFORMANCE OF THE SOLA TRANSFORMER AND HAS FORMED A TASK FORCE TO EVALUATE POSSIBLE GENERIC PROBLEMS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 2 (PWR) + UPDATE + REACTOR SHUTDOWN + SCRAM, REAL + SAFETY INJECTION + ACTUATION + INVERTER + GENERATOR, MOTOR + FAILURE, EQUIPMENT + FAILURE, INHERENT + ELECTRIC POWER, VITAL + TRANSFORMERS

ACCESSION NUMBER 167550 10-01 11/18/82

SPURIOUS SAFETY INJECTION ACTUATION AT NORTH ANNA 2
VIRGINIA ELECTRIC & POWER CO.
LTR W/LER 81-055 TO U.S. NRC, REGION 2, JUL 15, 1981, DOCKET 50-339,
TYPE--PWR, MFG--WEST, AE--S&W, DCS NO.--8107280592
DATE OF EVENT - 070381. POWER LEVEL - 018%. CAUSE - TRANSFORMER FIRE.
A REACTOR TRIP OCCURRED DUE TO A FAULT IN B PHASE MAIN TRANSFORMER. THE FAULT CAUSED A FIRE WHICH AFFECTED FEEDER CABLES FROM 1C RESERVE STATION SERVICE TRANSFORMER RESULTING IN A LOSS OF POWER TO 2J EMERGENCY BUS. A VOLTAGE TRANSIENT WHICH AFFECTED VITAL BUSSES 2-III AND 2-IV CAUSED A SPURIOUS ACTUATION OF ECCS. THE CAUSE OF THE OIL RELEASE AND FIRE FROM THE TRANSFORMER AND THE CAUSE OF THE VOLTAGE TRANSIENT THROUGH THE VITAL BUS INVERTERS 2-III AND 2-IV ARE UNDER INVESTIGATION. THE SAFETY INJECTION WAS CAUSED BY LOW RCS TEMPERATURE FOLLOWING THE REACTOR TRIP AND SPURIOUS HIGH STEAM LINE FLOW SIGNAL FROM THE BISTABLE TRIPS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + NORTH ANNA 2 (PWR) + TRANSFORMERS + FIRE + ELECTRIC POWER + TRANSIENT + REACTOR SHUTDOWN + SAFETY INJECTION + ACTUATION + FAILURE, EQUIPMENT + FAILURE, INHERENT + OFF SITE + EASP

ACCESSION NUMBER 167625 10-01 11/18/82

INADVERTENT SAFETY INJECTION ACTUATION OCCURS AT SALEM 2
PUBLIC SERVICE ELECTRIC & GAS CO.
LTR W/LER 81-045 TO U.S. NRC, REGION 1, JUL 21, 1981, DOCKET 50-311,
TYPE--PWR, MFG--WEST, AE--PSE&G, DCS NO.--8107290131
DATE OF EVENT - 042381. POWER LEVEL - 000%. CAUSE - OPERATOR SET STEAM GENERATOR VALVE SET POINT LOW. AN INADVERTENT SAFETY INJECTION OCCURRED DURING UNIT COOLDOWN TO MODE 5. THE AUTOMATIC SET POINT FOR VALVE 24MS10 WAS DECREASED TOO LOW AND THE CONTROLLER RESPONDED BY OPENING 24MS10, DECREASING 24 STEAM GENERATOR PRESSURE BELOW 100 PSI INITIATING A SAFETY INJECTION. THE CAUSE IS PERSONNEL ERROR. THE AUTOMATIC SET-POINT FOR 24MS10 WAS DECREASED TOO LOW. DURING THIS INITIATION ALL SAFEGUARD EQUIPMENT FUNCTIONED AS DESIGNED. THIS IS THE FIRST SAFETY INJECTION FOR UNIT 2. PERSONNEL WERE COUNSELED ON REQUIREMENTS TO FOLLOW PROCEDURES WHEN MS10'S ARE USED TO CONTROL COOLDOWN IN ORDER TO PREVENT INADVERTENT SI'S ON STEAM GENERATOR DIFFERENTIAL PRESSURE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 2 (PWR) + SAFETY INJECTION + EMERGENCY COOLING SYSTEM + VALVES + SET POINT + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + REACTOR STARTUP EXPERIENCE + STEAM GENERATOR + CALIBRATION + EASP + ACTUATION + PRESSURE RELIEF

ACCESSION NUMBER 167692 10-01 11/18/82

INADVERTENT SAFETY INJECTION ACTUATION AT MCGUIRE 1
DUKE POWER CO.
LTR W/LER 81-078 TO U.S. NRC, REGION 2, JUN 4, 1981, DOCKET 50-369, TYPE--PWR, MFG--WEST, AE--DUKE, DCS NO.--8108030249
DATE OF EVENT - 050781. POWER LEVEL - 000%. CAUSE - OPERATOR DEPRESSED INCORRECT SWITCH. AN INADVERTENT SAFETY INJECTION WAS INITIATED ON TRAIN B WHEN THE WRONG BUTTON ON THE CONTROL BOARD WAS PUSHED. DURING VALVE TIMING TESTS THE RESET PUSHBUTTON ON THE 'PRESSURIZER SAFETY INJECTION TRAIN B' WAS INADVERTENTLY DEPRESSED INSTEAD OF THE 'SAFETY INJECTION RESET TRAIN 1B' BUTTON. THE WRONG PUSHBUTTON WAS DEPRESSED UNBLOCKING THE PRESSURIZER LOW PRESSURE SIGNAL. A DIFFERENT METHOD WILL BE USED TO RESET THE VALVE TIMING COMPUTER PROGRAM WITHOUT DEPRESSING THE SAFETY INJECTION RESET SWITCH. A SLIDING COVER WILL BE PLACED OVER THESE PUSHBUTTONS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + MCGUIRE 1 (PWR) + EMERGENCY COOLING SYSTEM + SAFETY INJECTION + ACTUATION + TESTING + FAILURE, INSTRUMENT + INSTRUMENT, SWITCH + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + MODIFICATION + REACTOR STARTUP EXPERIENCE

ACCESSION NUMBER 167772 10-01 11/18/82

REACTOR SCRAMMED AND SAFETY INJECTION ACTUATED AT ZION 1
COMMONWEALTH EDISON CO.
LTR W/LER 81-029 TO U.S. NRC, REGION 3, JUL 31, 1981, DOCKET 50-295, TYPE--PWR, MFG--WEST, AE--S&L, DCS NO.--8108110488
DATE OF EVENT - 052881. POWER LEVEL - 099%. CAUSE - SOLA TRANSFORMER SHORTED. INVERTER 114 FAILED CONCURRENT WITH PERFORMANCE OF INSTRUMENT FUNCTIONALS, REQUIRING REACTOR PROTECTION BISTABLES TO BE TRIPPED, WHICH RESULTED IN A REACTOR TRIP. THE TRIP GENERATED A HIGH STEAM FLOW SIGNAL WHICH CONCURRENT WITH THE INVERTER FAILURE YIELDED A SAFETY INJECTION. PREVIOUS LER: 50-304/81-8. THE PRIMARY WINDING OF THE MASTER SOLA TRANSFORMER SHORTED TO GROUND WHICH TRIPPED THE INVERTER AND CAUSED THE SLAVE TRANSFORMER TO BURN UP. THE CORRECTIVE ACTION WILL BE TO MINIMIZE CIRCULATING CURRENT IN THE INVERTER. A PERIODIC SURVEILLANCE HAS BEEN WRITTEN TO CHECK THE CIRCULATING CURRENT AND ADJUST AS NECESSARY BY TUNING THE OUTPUT CAPACITANCE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 1 (PWR) + INVERTER + GENERATOR, MOTOR + REACTOR SHUTDOWN + TRANSFORMERS + ELECTRIC POWER + TESTING + FAILURE, EQUIPMENT + FAILURE, INHERENT + SAFETY INJECTION + ACTUATION + SCRAM, SPURIOUS

ACCESSION NUMBER 167784 10-01 11/18/82

INADVERTENT SAFETY INJECTION ACTUATION OCCURS AT MCGUIRE 1
DUKE POWER CO.
LTR W/LER 81-074 TO U.S. NRC, REGION 2, MAY 28, 1981, DOCKET 50-369,
TYPE--PWR, MFG--WEST, AE--DUKE, DCS NO.--8107310360
DATE OF EVENT - 042981. POWER LEVEL - 000%. CAUSE - OPERATOR ERROR
DURING MODIFICATIONS. AN INADVERTENT SAFETY INJECTION WAS INITIATED ON
TRAIN A WHEN THE TRAIN A SOLID STATE PROTECTION SYSTEM CABINETS WERE
RETURNED TO SERVICE WITHOUT BLOCKING THE SAFETY INJECTION SIGNALS.
DURING MODIFICATION WORK, CERTAIN SAFETY INJECTION PERMISSIVE BLOCKS
WERE CLEARED FROM THE SOLID STATE PROTECTION SYSTEM TRAIN A LOGIC. WHEN
THE SYSTEM WAS RETURNED TO SERVICE SAFETY INJECTION WAS INITIATED. THE
SAFETY INJECTION WAS TERMINATED AND NEW PROCEDURES WERE WRITTEN TO KEEP
THIS INCIDENT FROM OCCURRING AGAIN.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + MCGUIRE 1 (PWR) + REACTOR STARTUP EXPERIENCE +
EMERGENCY COOLING SYSTEM + SAFETY INJECTION + ACTUATION + FAILURE,
INSTRUMENT + INSTRUMENT, PROTECTIVE + FAILURE, OPERATOR ERROR +
NONLICENSED OPERATOR

ACCESSION NUMBER 168077 10-01 11/18/82

SAFETY INJECTION ACTUATES AT ZION 2
COMMONWEALTH EDISON CO.
LTR W/LER 81-015 TO U.S. NRC, REGION 3, JUL 31, 1981, DOCKET 50-304,
TYPE--PWR, MFG--WEST, AE--S&L, DCS NO.--8108100322
DATE OF EVENT - 050781. POWER LEVEL - 092%. CAUSE - INVERTER FAILURE.
UNIT 2 SAFETY INJECTED DUE TO LOSS OF INVERTER 213 THAT CAUSED PI-526A
AND 536A TO FAIL LOW, WHICH WITH HIGH STEAM FLOW MADE UP SAFEGUARDS
LOGIC. ALL SAFETY EQUIPMENT PERFORMED ITS FUNCTION. THE USAGE FACTOR
ON THE INJECTION NOZZLES IS .03125, ADDED TO PREVIOUS USAGE FACTOR GIVES
.15879. THE TOTAL LIFETIME OF THE INJECTION NOZZLES IS A 1.0 USAGE
FACTOR. THE MASTER INVERTER OUTPUT SOLA TRANSFORMER SHORTED TO GROUND
WHICH TRIPPED THE INVERTER AND CAUSED THE SLAVE TRANSFORMER TO BURN UP.
THE CORRECTIVE ACTION WILL BE TO MINIMIZE CIRCULATING CURRENT IN THE
INVERTER. A PERIODIC SURVEILLANCE HAS BEEN WRITTEN TO CHECK THE
CIRCULATING CURRENT AND ADJUST AS NECESSARY BY TUNING THE OUTPUT
CAPACITANCE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C.
20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + ZION 2 (PWR) + EMERGENCY COOLING SYSTEM + SAFETY
INJECTION + ACTUATION + GENERATOR, MOTOR + INVERTER + ELECTRIC POWER +
FAILURE, EQUIPMENT + FAILURE, INHERENT + TRANSFORMERS + REACTOR SHUTDOWN
+ SCRAM, REAL

ACCESSION NUMBER 168079 10-01 11/18/82

SAFETY INJECTION ACTUATES AT SALEM 2
PUBLIC SERVICE ELECTRIC & GAS CO.
LTR W/LER 81-051 TO U.S. NRC, REGION 1, JUL 27, 1981, DOCKET 50-311,
TYPE--PWR, MFG--WEST, AE--PSE&G, DCS NO.--8108060216
DATE OF EVENT - 062781. POWER LEVEL - 030%. CAUSE - STEAM GENERATOR FUEL PUMP TRIP. NO. 22 STEAM GENERATOR FEED PUMP TRIPPED, CAUSING A REACTOR TRIP AND SAFETY INJECTION. THE BORON INJECTION TANK CONCENTRATION DECREASED TO LESS THAN 20,000 PPM OF BORON. SIMILAR OCCURRENCES: 81-09 AND 81-32. THE SAFETY INJECTION CAUSED DILUTION OF THE BORON INJECTION TANK. THE TANK WAS BORATED UNTIL THE CONCENTRATION WAS WITHIN THE ALLOWABLE LIMITS.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + SALEM 2 (PWR) + REACTOR STARTUP EXPERIENCE + STEAM GENERATOR + COOLING SYSTEM, SECONDARY + FEEDWATER + PUMPS + REACTOR SHUTDOWN + EMERGENCY COOLING SYSTEM + SAFETY INJECTION + ACTUATION + CONCENTRATION + FAILURE, EQUIPMENT + FAILURE, INHERENT + POISON, SOLUBLE + STORAGE CONTAINER + DILUTION + SCRAM, REAL

ACCESSION NUMBER 168144 10-01 11/18/82

AVERAGE RCS TEMPERATURE FALLS BELOW LIMIT AT COOK 2
INDIANA & MICHIGAN ELEC. CO.
LTR W/LER 81-031 TO U.S. NRC, REGION 3, AUG 10, 1981, DOCKET 50-316,
TYPE--PWR, MFG--WEST, AE--UTILITY, DCS NO.--8108190265
DATE OF EVENT - 071181. POWER LEVEL - 003%. CAUSE - EXCESSIVE FEEDWATER FLOW TO STEAM GENERATOR. DURING A STARTUP WITH THE REACTOR CRITICAL, TAVG DECREASED BELOW 541 DEGREES. TAVG WAS INCREASED ABOVE 541 WITHIN APPROXIMATELY 1 MIN. PREVIOUS SIMILAR EVENTS INCLUDE 315/81-18, 316/78-10, 316/78-092, 316/78-053, AND 316/78-032. WHILE ATTEMPTING TO RAISE STEAM GENERATOR LEVELS TO PROGRAM LEVEL AN EXCESSIVE FEED RATE WAS USED. THIS IN EFFECT CREATED A LARGER HEAT SINK FOR THE RCS THEREBY LOWERING ITS TEMPERATURE. THE FEED RATE WAS REDUCED AND CONTROL RODS WITHDRAWN TO STOP THE TEMPERATURE DECREASE BUT THIS DID NOT PREVENT REACHING THE T.S. MINIMUM. ALL LICENSE OPERATORS WILL BE REQUIRED TO REVIEW THIS REPORT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + COOK 2 (PWR) + MAIN COOLING SYSTEM + TEMPERATURE + REACTOR COOLANT + STEAM GENERATOR + FEEDWATER + FLOW + RATE + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR + EFFECT

ACCESSION NUMBER 168503 10-01 11/18/82

PRESSURIZER HEATUP RATE EXCEEDS LIMIT AT COOK 1
INDIANA & MICHIGAN ELEC. CO.
LTR W/LER 81-029 TO U.S. NRC, REGION 3, AUG 24, 1981, DOCKET 50-315,
TYPE--PWR, MFG--WEST, AE--UTILITY, DCS NO.--8109030251
DATE OF EVENT - 072381. POWER LEVEL - 000%. CAUSE - OPERATOR ERROR.
WHILE FILLING AND VENTING THE RCS AN APPARENT PRESSURIZER HEATUP TRANSIENT OCCURRED IN EXCESS OF THE MAXIMUM 100 F/HR. ALLOWABLE TECH. SPEC. HEATUP RATES. DURING THE TRANSIENT THE RCS AVERAGE TEMPERATURE WAS 175 F AND THE PRESSURE WAS BEING INCREASED FROM 100 PSIG TO 325 PSIG FOR RUNNING OF THE REACTOR COOLANT PUMPS. THE TRANSIENT OCCURRED DURING A RECOVERY OF A LOW PRESSURIZER LEVEL. RELATIVELY COOL WATER (175 F) WAS SURGING INTO THE PRESSURIZER FROM THE REACTOR COOLANT LOOP WHERE IT CONTACTED THE HOT PRESSURIZER METAL (400 F). THIS CAUSED AN INCREASE IN THE PRESSURIZER LIQUID TEMPERATURE OF APPROXIMATELY 130 F OVER A 1-HOUR PERIOD. THIS INCREASE WAS NOT A TRUE INDICATION OF CHANGES IN THE PRESSURIZER METAL TEMPERATURE. HAD PRESSURIZER METAL TEMPERATURE INDICATION BEEN AVAILABLE, A DECREASE IN METAL TEMPERATURE WOULD HAVE BEEN INDICATED. THE APPARENT EXCESSIVE HEATUP RATE WAS IN REALITY A COOLDOWN OF THE PRESSURIZER AND THE HEATUP LIMIT WAS NOT VIOLATED. TO PREVENT RECURRENCE THE RATE AT WHICH THE PRESSURIZER PRESSURE IS INCREASED MUST BE KEPT TO WITHIN THE CAPABILITY OF THE CHARGING PUMP TO MAINTAIN THE PRESSURIZER PROGRAMMED LEVEL. ALL LICENSED OPERATORS WILL BE REQUIRED TO REVIEW THIS EVENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + COOK 1 (PWR) + MAIN COOLING SYSTEM + PRESSURIZER + THERMAL TRANSIENT + HEAT TRANSFER + RATE + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR

ACCESSION NUMBER 169297 10-01 11/18/82

SAFETY INJECTION ACTUATES DUE TO FAST COOLDOWN RATE AT MCGUIRE 1
DUKE POWER CO.
LTR W/LER 81-151 TO U.S. NRC, REGION 2, OCT 9, 1981, DOCKET 50-369,
TYPE--PWR, MFG--WEST, AE--DUKE, DCS NO.--8110200580
DATE OF EVENT - 091481. POWER LEVEL - 020%. CAUSE - DESIGN, PERSONNEL AND PROCEDURAL ERRORS. DURING THE PERFORMANCE OF THE LOSS OF CONTROL ROOM TEST, OPERATORS HAD TROUBLE CONTROLLING THE AUXILIARY FEEDWATER SYSTEM. AN EXCESSIVE AMOUNT OF FEEDWATER WAS PUMPED INTO THE STEAM GENERATORS CAUSING AN ABNORMALLY FAST COOLDOWN RATE AND A SAFETY INJECTION ACTUATION. THE TRANSIENT WAS THE RESULT OF A COMBINATION OF DESIGN DEFICIENCY AND PERSONNEL ERRORS DURING THE TEST. PROCEDURAL DEFICIENCIES ALSO CONTRIBUTED TO THE SEVERITY OF THE COOLDOWN TRANSIENT. VARIOUS PROCEDURES WERE REVISED AND DUKE POWER CO. IS DEVELOPING PLANS TO MODIFY THE AUXILIARY SHUTDOWN PANELS. A REVIEW BY WESTINGHOUSE CONCLUDED THAT NO SIGNIFICANT DAMAGE RESULTED FROM THE TRANSIENT.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + MCGUIRE 1 (PWR) + TESTING + CONTROL PANEL/ROOM + FEEDWATER + AUXILIARY + SAFETY INJECTION + ACTUATION + THERMAL TRANSIENT + FAILURE, EQUIPMENT + FAILURE, DESIGN ERROR + REACTOR SHUTDOWN + MODIFICATION + PROCEDURES AND MANUALS + FAILURE, OPERATOR ERROR + FAILURE, ADMINISTRATIVE CONTROL + STEAM GENERATOR

ACCESSION NUMBER 170158 10-01 11/18/82

UPDATE ON PRESSURIZER HEATUP TRANSIENT AT COOK 1
INDIANA & MICHIGAN ELEC. CO.
LTR W/LER 81-029R1 TO U.S. NRC, REGION 3, NOV 6, 1981, DOCKET 50-315,
TYPE--PWR, MFG--WEST, AE--UTILITY, DCS NO.--811190481
DATE OF EVENT - 072381. POWER LEVEL - 000%. CAUSE - LICENSED OPERATOR ERROR. WHILE IN THE PROCESS OF FILLING AND VENTING RCS AN APPARENT PRESSURIZER HEATUP TRANSIENT OCCURRED IN EXCESS OF THE MAXIMUM 100 F/HR ALLOWABLE HEATUP RATES. DURING THE TRANSIENT THE RCS AVERAGE TEMPERATURE WAS APPROXIMATELY 175 F AND THE PRESSURE WAS BEING INCREASED FROM 100 PSIG TO 325 PSIG FOR RUNNING OF THE REACTOR COOLANT PUMPS. THE TRANSIENT OCCURRED DURING A RECOVERY OF A LOW PRESSURIZER LEVEL. RELATIVELY COOL WATER (175) WAS SURGING INTO THE PRESSURIZER FROM THE REACTOR COOLANT LOOP WHERE IT CONTACTED THE HOT PRESSURIZER METAL WHICH WAS AT APPROXIMATELY 400 F. THIS CAUSED AN INCREASE IN THE PRESSURIZER LIQUID TEMPERATURE OF APPROXIMATELY 130 OVER A 1-HOUR PERIOD. TO PREVENT RECURRENCE THE RATE AT WHICH THE PRESSURIZER PRESSURE IS INCREASED MUST BE KEPT TO WITHIN THE CAPABILITY OF THE CHARGING PUMP TO MAINTAIN THE PRESSURIZER PROGRAMMED LEVEL. ALL LICENSED OPERATORS WILL BE REQUIRED TO REVIEW THIS EVENT TO AVOID IT BEING REPEATED IN THE FUTURE.
AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

FAILURE + REACTOR, PWR + COOK 1 (PWR) + UPDATE + MAIN COOLING SYSTEM + PRESSURIZER + THERMAL TRANSIENT + FAILURE, EQUIPMENT + FAILURE, OPERATOR ERROR + LICENSED OPERATOR

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