Severity, Causes, and Frequencies of Pressure Vessel Thermal Shock at U.S. PWR Plants (1963-1981)

D.L. Phung
Institute for Energy Analysis, Oak Ridge Associated Universities, P.O. Box 117, Oak Ridge, Tennessee 37830, U.S.A.

W.B. Cottrell
Nuclear Operations Analysis Center, Oak Ridge National Laboratories, Oak Ridge, Tennessee 37830, U.S.A.

Abstract

A systematic search for events and precursors to events of pressure vessel thermal shock (PVTS) was made. The data base included some 16,000 licensee event reports of 47 U.S. PWRs covering 329 reactor years (RY) of operation between 1963 and 1981. Ninety-nine events were found, of which 34 were considered significant.

The analysis of the finds were based on five categories of severities, six modes of reactor operation, eight types of initiating sequences, and eight types of causes. After correcting for factors of completeness, the frequency of events and precursors to PVTS was found to be about 1 per RY. Two-thirds of the causes originated from equipment and system inadequacies. About a quarter of the causes were due to human errors.

Repressurization of the reactor coolant system was found to take place in the majority of cases considered significant.

1. Introduction

This article is a summary of a report that was prepared for the U.S. Nuclear Regulatory Commission [Ref. 1].* The report surveys licensee event reports (LERs) of operating U.S. PWRs for events and precursors of events of thermal shock at U.S. PWRs. The data base comprises about 16,000 PWR LERs stored in the computerized data bank of the Nuclear Safety Information Center (NSIC) in Oak Ridge from its conception (1963) through 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 and 47 PWR units.

2. Framework

The LER search methodology includes (1) a logic approach that narrows the data base to specific areas of interest, (2) the Boolean combination of keywords that turns up appropriate events from the computer data base, and (3) 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 completeness of the findings includes comparison against 10 percent of all PWR LERs and against data of NRC and of the industry.

The seven generalized event sequences leading to PVTS include: largebreak loss-of-coolant accident (LBLOCA), small-break loss-of-coolant accident (SBLOCA), overfeeding of steam generators (OF) (we define OF as including both high flow of normally warm feedwater and normal flow of cold feedwater), 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 through 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 [Ref. 1]. 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 data on all are not necessarily available for a 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."

3. 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 pressure vessel thermal shock events or precursors.

A listing of the 34 significant events is provided in Table 1. A more complete listing and description of all 99 events can be found in Ref. 1.

Forty-seven PWRs were in the data base between 1963 and 1981, covering 329 reactor years (RY). These do not include Shippingport and Indian Point 1 which did not have much information in the data base presumably because they were demonstration reactors.

4. 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?

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

The second question was addressed by checking our finds against industry's data (Oconee plant's [Ref. 2] and Westinghouse owners group's [Ref. 3]). Although these data themselves are incomplete, we found that our finds represented only about 25% of their insignificant and 67% of their significant events. The overall ratio is about 50%, a value remarkably in agreement, perhaps fortuitously so, with a 1978 ACRS observation [Ref. 4].

The third question was raised by realizing that the LER data is very scarce for the 1960s. Indeed, the earliest date 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 reactor year value is 320 instead of 329.

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

	Corrected Number of Events	Corrected Frequency Events/RY
Significant	51	0.159
Insignificant	260	0.813
Total	311	0.972

5. Categories and Causes

Table 2 shows the characteristics of the PVTS events and precursors. These characteristics are categorized in three groups: modes of reactor operation just before the event, initiating sequences of the event, and causes of initiation. For each characteristic the frequencies (corrected for completeness) of significant and insignificant events are shown.

About half of all events occurred when the reactors were in power operation (Mode 1) and about 85 percent occurred when the reactor systems were in hot conditions (Modes 1, 2, 3, and 4).

No large-break LOCA event took place, and small-break LOCA accounted for about 10 percent of all events. About a quarter of all events was initiated by the power conversion circuits, either by overfeeding of steam generators or by high steam flow. About half of all events were initiated by HPSI, either by instrumentation and control (I&C) inadequacy or by spurious signals. In terms of severity, one third of all SBLOCA events and 40 percent of all overfeeding/high steam flow events were significant (Severities 3 and 4). Hardly one out of 25 HPSI-initiated PVTS event was significantly severe.

A quarter of all events were directly caused by operator/maintenance/test errors, and an additional 10% can be considered as secondarily caused by those errors. When fully corrected for factors of completeness, only 1 out of 15 such errors resulted in a significant event.

Valve failures accounted for about 16% of all events, and 3 out of 10 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.

A third of events were caused by power failure or design inadequacies in the instrumentation and control system. Another quarter was caused by hardware and system failures that have not been categorized as valves and I&C failure. In fact, when all four categories are considered together--valves, I&C, power, and systems and hardware--then they account for over two-thirds of all PVTS events and precursors, one out of five of which was significant.

6. Distribution of Events Among Vendors and Utilities

The data base shows 8 reactors by B&W with 52.7 RYs of operation, 8 reactors by CE with 54.4 RYs of operation, and 31 reactors by W with 222.2 RYs of operation. Thus, either by reactor units or by RYs, the contribution to the experience for B&W/CE/W is in the proportion 17%/17%/66%.

In terms of the total number of PVTS-like events, the distribution among reactor vendors is pretty much the same as the distribution of reactor years or number of reactors. CE reactors show only half the overall frequencies of W and B&W reactors (0.49 versus 1.05 and 0.95 event/RY). In terms of severity of PVTS-like events, however, the frequencies appear to be in the ratio 3:2:1 among B&W:CE:W reactors (0.34, 0.19, 0.10 event/RY). The only obvious thermal-hydraulic difference between W and the other reactors is that the W reactors have 3 to 4 steam generators, each attached to a hot and a cold RCS leg, while CE and B&W reactors have only two steam generators, 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 non-nuclear instrumentation and integrated control systems which could initiate or aggrevate PVTS-like events. These weaknesses appear to have been corrected since 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 PVTS-like events among them--the statistics of such a study being certainly poor-we present mappings of the events on each reactor operating history both in terms of RYs and in real time. These mappings can be found in Reference 1. There appears to be no time pattern to the occurrence of PVTS events and precursors. An exception is at pre-operational 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 Roberts' learning theory [Ref. 5].

There were 13 reactors with no PVTS-like event. These include 2 by B&W, 2 by CE, and 9 by W; again quite proportional to the number of reac-tors and RYs of each vendor in the experience. 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, different LER reporting procedures, the deficiencies of the NSIC data bank, or all of these.

7. Which Was the Most Severe PVTS Event?

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 (the Three Mile Island 2 accident not being considered). 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 non-nuclear instrumentation system, which then fed wrong signals to the integrated control system, the Rancho Seco event was basically an overfeeding event, whereas the Crystal River 3 event was basically a small-break LOCA event followed by sustained safety injection. The Rancho Seco event took place for 70 minutes with the RCS decreasiing to 285°F and repressurized by HPSI to 2000 psi. The Crystal River 3 event took place for 87 minutes, with the RCS repressurized by HPSI up to 2400 psi and parts of the vessel exposed to cold water as low as 250°F. While detailed analyses need to be done, it appears that areas of the pressure vessel of Crystal River 3 could have seen even lower temperatures. Exactly how low such temperatures were depend on the nature of the reverse flow and whether or not the vent valves opened.

8. How Bad Were the 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), appears 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. 6] to study some PVTS cases bounding the most severe of our finds. These cases assume the existence of a longitudinal flaw up to 15 percent deep in the vessel weld, the existence of 0.35 percent copper content in the weld material, 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 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 Temp. Drop	Assumed RCS Pressure	Critical Neutron Fluence (fast neutrons/cm^2)
200°F	1500 psi	2.77×10^{19}
	2500 psi	1.43×10^{19}
300°F	1500 psi	0.62×10^{19}
	2500 psi	0.43×10^{19}

The material copper content and operational fast neutron fluxes for the 47 U.S. PWRs are given in Ref. 1. As of December 1981, some half dozen vessels had accumulated a fast neutron fluence higher than $1 \times 10^{19} m/cm^2$. While the copper contents in the Rancho Seco and Crystal River vessel welds are as high as 0.31 percent, the accumulated fast fluences at the time of their respective PVTS occurrences were much smaller than $0.4 \times 10^{19} m/cm^2$; 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 7 proposes a screening criterion whereby reactors must go through a thorough evaluation of the vessel integrity three years before the critical fast neutron fluence is attained.

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

Our observations, categorized in Ref. 1, suggest that in the majority of significant PVTS-like events the operators were aware of system upsets fairly soon following initiation of such events. Whether or not his corrective actions help or hurt depended on specific conditions and the competence of the operators. In half of the cases the RCS recirculation pumps were not shut down. In the majority of cases, repressurization of the RCS took place. This latter observation suggests that warm prestress probably should not be credited in analyzing the flaw propagation in the pressure vessel wall.

10. References

[1] Doan L. Phung, "Pressure Vessel Thermal Shock at U.S. Pressurized Water Reactors: Events and Precursors, 1963-1981," ORNL/NSIC-112, Oak Ridge National Laboratory, Oak Ridge, Tennessee (in print as of November 1982).

[2] Duke Power Company, "Oconee Nuclear Station Reactor Vessel Pressurized Thermal Shock Evaluation," DPC-RS-1001 (January 1982).

[3] Westinghouse Electric Corporation, Nuclear Technology Division, "Summary of Evaluation Related to Reactor Vessel Integrity Performed for the Westinghouse Owners Group," Report 20 72Q:1, June 1982. Also NRC Internal Memorandum by F. B. Litton, "Summary of WOG/NRC Meeting on Reactor Vessel Integrity on June 22, 1982 Concerning PTS Issue," June 30, 1982.

[4] USNRC Advisory Committee on Reactor Safeguards, Review of Licensee Event Reports (1976-1977), NUREG-0572 (1979), Washington, D.C.

[5] P. C. Roberts and C. C. Burwell, "The Learning Function in Nuclear Reactor Operation and Its Implications for Siting Policy," ORAU/IEA-81-4(M), Institute for Energy Analysis, Oak Ridge Associated Universities, Oak Ridge, Tennessee, May 1981.

[6] S. K. Iskander et al, OCA-1, "A Code for Calculating the Behavior of Flaws on the Inner Surface of a Pressure Vessel Subjected to Temperature and Pressure Transients," NUREG/CR-2113 (ORNL/NUREG-84), Oak Ridge National Laboratory, Oak Ridge, Tennessee, August 1981. Updated version, OCA-2, September 1982.

[7] U.S. Nuclear Regulatory Commission, NRC Staff Evaluation of Pressurized Thermal Shock, Draft, September 13, 1982, Washington, D.C.

Table 1. Short List of 34 Significant PVTS Events (1969 through 1981)

Date	Reactor	Nature of Event*	ΔT/Δt or Comments
		Severity 3	
10/12/70	Conn. 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, operator inexperience	≈40°F/10 min.
12/08/76	Calvert Cliffs 2	FW control, xenon effect	≈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/hr.
11/05/72	Robinson 2	PORV stuck open	SI for 87 min.
01/04/74	Oconee 2	Excessive auxiliary steam use	>100°F/hr.
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/hr.
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 caused RT but not trip of steam-driven FW pump	70°F/30 min.
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 hr.
03/20/78	Rancho Seco	Loss of NNI caused transient	310°F/1 hr.
04/23/78	Three Mile Island 2	Steam relief valves stuck open	149°F/6 min.
12/27/78	Arakansas 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. Operator turned off SI.	core damage; PVTS not considered
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 feet of pressure vessel under river water for extended period
09/14/81	McGuire 1	Overcooling during loss-of-control room test	117°F/20 min.

Note that "nature of the event" does not necessarily pinpoint the cause of the events.

Table 2. Nature and Frequencies of PVTS Events and Precursors (1963-1981 data, corrected for completeness)

Type of Frequency	Events/RY	% Insignificant Event	% Significant Event	% of Total
Modes of Reactor Operation				
1-Power	0.467	36	12	48
2-Startup	0.144	13	2	15
3-Hot standby	0.156	14	2	16
4-Hot shutdown	0.042	4	-	>4
5-Cold shutdown	0.100	10	-	10
6-Refueling	0.063	6	-	>6
Total	0.972	83	16	99
Initiating Sequences				
LBLOCA	0	-	-	0
SBLOCA	0.091	6	3	9
OF/Cold FW	0.161	11	6	17
HSF	0.097	5	5	10
HPSI	0.481	47	2	49
CF/LPSI	0.050	5	-	5
EX	0.005	<1	-	<1
Others	0.087	9	-	9
Total	0.972	84	16	100
Causes of Initiation				
Operator error	0.142	>14	<1	15
Maintenance/test error	0.080	>7	<1	8
Valve failure	0.153	9	7	16
Power failure	0.114	10	1	12
I&C design error/ spurious signal	0.156	14	2	16
System/equipment failure	0.233	18	5	23
Waterhammer	0.075	8	-	8
Fire/lightning	0.025	3	-	3
Total	0.971	84	16	100