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

October 29, 1980

MEMORANDUM FOR: S. Fabic, Chief
Analysis Development Branch
Division of Reactor Safety Research
Office of Nuclear Regulatory Research

THRU: Gordon E. Edison, Section Leader
Systems Development Section
Division of Systems and Reliability Research
Office of Nuclear Regulatory Research

FROM: Merrill A. Taylor
Systems Development Section
Division of Systems and Reliability Research
Office of Nuclear Regulatory Research

SUBJECT: INSIGHTS ON OVERCOOLING TRANSIENTS IN PLANTS WITH THE B&W NSSS

At our September 12, 1980 meeting SAB/SRR agreed to survey the B&W LER file for insights on actual overcooling events experienced by the B&W NSSS. It was felt that information on actual plant transients would be helpful in validating BNL calculations and in exploring potentially more severe overcooling transients. Subsequent to the 9/12 meeting, a workscope with specific tasks for the RSR program on Analyses of Overcooling Transients was mutually agreed to. Tasks I, II and III (Phase I) were to be accomplished by SAB/SRR. This memo is intended to fulfill these specific tasks. It is recognized however, that further dialogue and DSRR inputs will likely be needed throughout the Phase I work.

Table 1 summarizes the results of the LER survey. Figure 1 illustrates the severity of the transients experienced. As discussed in the following sections (I, II and III), we have recommended several of the more severe events for BNL benchmark use. Section IV mentions other events that may result in greater overcooling and some LWR failure experiences that might be viewed as accident precursors. We have also attempted to give you a rough perspective on the frequency of various overcooling transients. We caution, however, that such estimates are made on a statistically limited base of experience with the B&W plants and they are not of high precision. These should, therefore, be used with recognition that considerable uncertainty may exist around such estimates.

8105110190

I. LER SURVEY RESULTS

A survey of the B&W LERs was undertaken to identify the more significant overcooling events experienced. The LERs used covered a time period through early 1979 when about 22 reactor years of operating[1] experience had been accumulated by the B&W plant designs. Table 1 summarizes the results of the survey. Other experiences have occurred after mid-1979 (e.g., Oconee 3, 11/10/79) and these are also included in Table 1. A total of roughly 28 reactor years is the present base of B&W operating experience. Approximately 1/2 of this experience has been accumulated by the Oconee units.

Fifteen events were identified that have exceeded the cooldown rate limits ($\sim 100^{\circ}$F/hr) set forth in the technical specifications. This would suggest a frequency of roughly 0.5 events per reactor year exceeding the technical specification limits. Review of these events indicate that about 60% (9 events) occurred prior to the plant having attained commercial operations. This may reflect the plant burn-in/tuning-up experiences usually seen.

In regard to the LERs, these set forth little detail of the type that should be of interest to the BNL analyses. In a few cases, we have dug out additional details from the docket files. Examples are also being sent (separately) for your review. (We could perhaps ask the licensees for recorded plant data on the transients if you think more information detail will be needed by BNL. Let's discuss these additional needs at your convenience.)

Based on the LER survey, five (5) overcooling events were identified that should reflect a spectrum of cooldown rates from about 120°F/hr through 300°F/hr. In terms of decreasing severity, the events are:

1. Rancho Seco - 03/20/78
2. Oconee #1 - 05/05/73 (occurred prior to commercial operation)
3. Crystal River #3 - 03/22/77 (occurred prior to commercial operation)
4. Oconee #1 - 12/14/78
5. Oconee #3 - 11/10/79

These events are illustrated in Figure 1 and briefly discussed below.

II. RECOMMENDED BENCHMARKS

1. The Rancho Seco event of March 20, 1978 is believed to represent the most severe (and prolonged) overcooling transient experienced

---

[1] Based on NRC Gray Book data on accumulated number of critical hours. Actual time from start of commercial operations is larger by less than a factor of two.

to date ($\sim 300^{\circ}F/hr$) and it is recommended as an important benchmark for the BNL analyses. Not only did the Rancho Seco event greatly exceed the cooldown rate limitations of $\sim 100^{\circ}F/hr$ specified in Technical Specifications, it also appears to have exceeded the pressure, temperature limits specified therein for the RCS. These RCS limits are currently based on RPV irradiation of only 5 effective full power years.

2. The Oconee 1 event of May 5, 1973 involved a high initial rate (10^6 lb/hr @ $100^{\circ}F$) of delivery by the main feedwater system while the system was under manual control. This event occurred prior to commercial operation and is believed to be the reason for Duke installing a safety grade high level trip for the main feedwater system - this high level trip being independent of the Integrated Control System (ICS). This event may also be of interest to the BNL analyses because of the initial high cooldown run rate experienced and the fact that the RCS pressure diminished to ~ 1330 psig and shrinkage caused a loss of pressurizer coolant.

3. The Crystal River 3 event also involved a high initial cooldown ($\sim 164^{\circ}F/15$ min) rate but this stemmed largely from excessive steam relief occurring when the atmospheric dump (and possibly turbine bypass) valves remained partly open. This event may also be of interest to the BNL analyses because of the system response to high steam loads. This event also occurred just prior to commercial operations.

4. The Oconee 1, December 14, 1978 event involved overcooling due to OTSG fill levels being specified higher than was found to be needed. Emergency feedwater was used to fill the OTSG to about a 95% level - a specification that subsequently was revised downward. A cooldown to ~ 1430 psig occurred resulting in an actuation of ECCS.

5. The Oconee 3, November 10, 1979 overcooling event resulted from some delays in stopping the main feedwater flow (slow valve operation or setpoint errors) combined with inadequate secondary pressure control due to apparent turbine bypass valve malfunctions and the presence of high auxiliary steam loads. The minimum RCS temperature, pressure conditions reached were $420^{\circ}F$ and 1660 psig, respectively.

These latter two events should be of lesser interest to the BNL analyses.

III. DISCUSSION ON NATURE AND CAUSES OF OVERCOOLING EVENTS

The B&W plant experience indicates that the more severe overcooling events have happened after the plant has experienced an undercooling event causing near depletion or dry-out of the OTSGs. These have stemmed largely from operator actions taken to reestablish feedwater delivery into the depleted OTSGs. The cooldown has also been made somewhat more severe by the presence of high auxiliary steam loads and/or by high steam relief via open turbine bypass or atmospheric relief valves (e.g., 50% open).

With exception of the Oconee 1, May 5, 1973 event (prior to commercial operations), the above overcooling transients have involved various power faults that influenced the response of the plant non-nuclear instrumentation and controls (i.e., NNI/ICS). Such faults have contributed to the loss of (or somewhat erratic response from) the steam and main feedwater portions of the plant. Undoubtedly, these faults have also contributed to some degree of confusion by the plant operators in their actions taken subsequently to restore feedwater and achieve stable plant conditions.

The Rancho Seco event included a number of factors that contributed to the severity of the cooldown and to the fact that the extent of the RCS cooldown was not fully recognized for a prolonged period of time (i.e., ∾1 hour). Some of these factors include:

- Power fault (short) that affected the response of nearly 2/3 of the NNI/ICS equipment,
- Essentially "dried-out" the OTSGs via loss of feedwater,
- Gagged PZR-PORV existed such that relief through PZR safety valve occurred when feedwater was lost. (This factor has relevance to potential repressurization conditions.),
- Confusion existed about the status of feedwater delivery into OTSGs largely because of NNI/ICS faulting,
- High auxiliary steam loads were present,
- Emergency feedwater design has an SIS actuation signal for starting unlike other B&W designs (overcooling can be and was in fact made more severe by this feature),
- Initial overcooling occurred from human actions taken to reestablish main feedwater flow. This challenged the SIS actuation signal and caused initiation of (1) 100% emergency

feedwater delivery of cold condensate to both OTSGs in the presence of main feedwater being delivered into at least one of the OTSGs and (2) 100% high pressure injection (i.e., 2 HPI pumps) delivering cold BWST coolant to the RCS while an additional high pressure pump was also delivering to the RCS in a normal mode from the makeup tank,

- All main RCS pumps were allowed to remain running to low RCS pressure-temperature conditions in violation of usual procedures and precautions.

- Operators of the plant were apparently preoccupied with restoring NNI/ICS equipment and did not promptly rectify the overcooling transient, although part of HPI and emergency feedwater delivery was secured during the overcooling transient. As a result, the operators did not realize until ~1 hour into the transient that the RCS temperature had decreased to ~285°F.

- Although not yet clear, it [illegible]ssible that the NNI/ICS faulting also caused additional stea[illegible]nds. This is so because the turbine bypass and/or atmosp[illegible]ic dump valves are designed open to about 50% position as an expected null position on loss of power to the ICS. (Not all NNI/ICS power was lost in this event however, and the particular response of the turbine bypass and atmospheric steam dump valves remains unknown to us at this time).

As mentioned above, most of the moderate to severe overcooling events have involved power faults of various kinds in the NNI/ICS equipment but were the result of human actions taken later into the transient to restore feedwater to the OTSGs.

NUREG-0667 reveals[1] that there have been 29 failures of the NNI/ICS in the B&W plants through about the spring of 1980. Approximately 20 of these failures have resulted in a reactor trip while the plant was above about 30% power. A feedwater transient was experienced in nearly all of these reactor trips. About 6 of these events resulted in excess cooldown rates of the severity noted above and illustrated in Figure 1.

According to NUREG-0667 information, four (4) automatic actuations of the HPI were experienced during these NNI/ICS failures. These actuations could have resulted from RCS depressurization caused either by a stuck open PZR-PORV or from the overcooling by the steam-feedwater conditions in the secondary side of the plant

[1] See Table 4.2 of NUREG-0667

(these do not necessarily occur together). This experience indicates a frequency of moderate to severe overcooling events (or events sufficient to actuate HPI) at roughly 0.1 to 0.2 per reactor year. If, on the other hand, Rancho Seco were to be taken as the singular most severe overcooling event thus far experienced in commercial operations, a frequency of roughly $3x10^{-2}$ per reactor year might be estimated. In light of the post-TMI-2 and post-Crystal River improvements required of the B&W plants (particularly in relationship to the NNI/ICS failures and to the human training in response to such feedwater disturbances), the frequency of overcooling events as severe as Rancho Seco may have been considerably reduced.

The Rancho Seco event could conceivably have been made even more severe through prolonged inattention to the OTSG heat removal, by mainfeed delivering to more than one of the OTSGs, or by failure of the human to partly secure the emergency feedwater and HPI delivery after the such was actuated by the SIS signal. The probability of such additional errors is speculative on our part, but for purposes of establishing RSR analysis priorities for B&W reactors, we suggest an overall frequency of $\sim 3x10^{-3}$/RY for the Rancho Seco overcooling to have been made more severe than actually experienced.

IV. OVERCOOLING EVENTS MORE SEVERE THAN REVEALED BY B&W EXPERIENCE

As mentioned above, the Rancho Seco event might have resulted in somewhat greater overcooling largely through human inactions. There also exists the possibility of other severe cooldown events beyond those revealed by B&W experience. Examples would include a rupture of large main steam (or feedwater) piping - perhaps with additional coincidental failures taking place. At this time, we do not have very good estimates on the frequency to be associated with these more severe overcooling events. More work would be needed to derive such frequency estimates and these could vary somewhat from one plant design to another. We know, however, that the world experience with various commercial LWR designs is roughly 1000 reactor years[1] with about 1/2 of this being U.S. experience. To our knowledge, there has been no large rupture of main steam piping during this commercial experience which suggests a frequency of large pipe ruptures of the order of 10^{-3} per reactor year or less. Failures coincidental with rupture of large piping would either be caused by common interactions or result independently from the rupture. Some examples of coincidental failures that might result in greater overcooling would be: failure of the ICS, failure of main steam stop valves, failure of the steam line rupture matrix, uncontrolled emergency feedwater flow, rupture of steam line interacting and causing failure of another, human error, etc. In the overall,

[1] This estimate reflects actual time from start of commercial operations. The accumulated number of critical hours should be about 50-60% smaller.

events involving coincidental failures would be expected to be at lower frequency perhaps by as much as an order of magnitude. Bush[1] has made a survey of incidents pertinent to the reliability of piping in LWRs and has considered these incidents as possible precursors to piping failure. A basic finding of his was that "failure statistics confirm 10^{-4} to 10^{-6} per reactor year in large pipes, with higher rates as size decreases." Earlier, WASH-1400 had estimated 10^{-4} per reactor year as a median value for failure of large piping ($\gtrsim 6"$ dia.) giving an error spread of 10 up and down about the median (i.e., 10^{-3} to 10^{-5}). For purposes of the RSR analysis tasks, we believe it reasonable to assume a frequency of 10^{-4} per reactor year as being applicable to rupture of large steam piping. Given a rupture of large size steam piping, the plant operators would be faced with events not yet experienced. As the Rancho Seco and TMI-2 incidents would suggest, the possibility of human errors being committed is large when the operators are faced with a new experience or one for which training and procedures are lacking. Hindsights from the TMI-2 accident would suggest roughly a 30% chance of error existed given the unanticipated sequence of events. We would therefore suggest use of $\sim 3\times10^{-5}$/RY as an estimated frequency of overcooling caused by a large steam piping rupture coincidental with other failures.

V. SUMMARY

The B&W experience reveals a spectrum of overcooling transients including several that have been rather severe. This experience suggests that as the plant equipment and operators become "tuned-up" prior to commercial operations the frequency of overcooling transients diminishes. All of the more severe overcooling events that occurred during commercial operations appear to have some commonality in that they involved various faults in the instrumentation and controls (NNI/ICS) causing disturbance and/or loss of feedwater to the OTSGs. The severe overcooling events resulted largely from human actions to restore feedwater to depleted or "dry" OTSGs. In these cases, it is likely that the human faced some degree of confusion about equipment status because of the faulted NNI/ICS. Therefore, he cannot, in our view, be held totally accountable as the root cause for the overcooling events.

We have attempted to give a perspective on the frequency of overcooling events experienced and our judgments on the frequency of events that may result in greater overcooling. These frequency estimates

---

[1] See paper IAEA-SM-218/11, Reliability of Piping in Light Water Reactors, S. H. Bush, USNRC, ACRS, Washington, DC, USA (International Symposium on Applications of Reliability Technology to Nuclear Power Plants, Vienna 10-13, October 1977).

are intended to assist RSR calculational priorities and should not be taken to be of high certainty and precision. For RSR convenience, these estimates are summarized in Figure 2.

Merrill A. Taylor

Merrill A. Taylor,
Systems Development Section
Division of Systems & Reliability Research
Office of Nuclear Regulatory Research

Enclosures:
1. Table 1
2. Figure 1
3. Figure 2

cc: G. Edison
F. Rowsome
R. Bernero
L. Shao
J. Strosnider
M. Vagins
C. Serpan
P. K. Niyogi
J. Jenkins
W. Vesely
M. Cullingford
D. Basdekas

ENCLOSURE 1

Summary of Overcooling Events in

Plants with the B&W NSSS

TABLE 1. Overcooling Events - B&W NSSS Designs

	Plant and Docket No.	Event Date	Letter Date	Event Summary
1.	Oconee 1 (50-269)	05/05/73	-	Booster and MFW pumps tripped while at ~18% power. Reactor was manually tripped. Emergency feedwater did not start automatically and one (A) OTSG dried out due to improper pressure control in that loop. Turbine bypass valve control was in manual and the pressure setpoint did not increase in reactor trip as designed. About 4 minutes after initial feedwater loss, the MFW flow was reestablished to both SGs at high flow rates ($\sim 10^6$ lb/hr @ 100^0F). The rapid cooldown resulted in a minimum pressure of ~1330 psi and a loss of pressurizer level.
2.	Oconee #2 (50-270)	04/04/74	01/17/74	Spurious signal cause isolation of the 230 Kv switchyard plus trip of MFW and reactor while at ~70% power. About 7 minutes after reactor trip, emergency feedwater pump was started and filled SGs to ~95% level (original setpoint). This fill level and high auziliary steam loads produced an overcooling of $\sim 140^0$F/hr in 1 loop and $\sim 135.5^0$ in the other.
3.	Crystal River #3 (50-302)	03/02/77	03/10/77	Loss of AC vital bus led to loss of inverter power to ICS while reactor was at ~40% power. The reactor and turbine tripped and atmospheric dump valves opened to ~50%. A cooldown rate of about 164^0F/15 minutes was experienced.

TABLE 1. Overcooling Events - B&W NSSS Designs (Cont.)

	Plant and Docket No.	Event Date	Letter Date	Event Summary
4.	Crystal River #3 (50-302)	04/15/77	04/21/77	During test of remote shutdown equipment, an overfilling transient was caused by pressure transmitter for MFW valves FWV 39 and 40 being out of calibration and valves did not fully close. Power level of plant was negligible and a small cooldown of ~101^{0}F in about 20 minutes was experienced.
5.	Rancho Seco (50-312)	01/05/79	01/19/79	While plant was at ~100% power, a technician caused a DC short to ICS. This led to a MFW transient and subsequent overcooling on refill of about 120^{0}F/hr.
6.	Rancho Seco (50-312)	03/20/78	03/31/78	Human was replacing a lightbulb for NNI/ICS panel and caused a short to occur while plant was at ~70% power. About 2/3 of the pressure, temperature and level indication was lost. The reactor tripped, feedwater was lost and OTSGs dried out. Subsequent refilling by the MFW caused an overcooling and actuation of HPI and emergency feedwater. Actuation of HPI and EFW caused severe overcooling rates (~300^{0}F/hr) until pumps were partly secured by plant operators. Operators did not, however, recognize until 1 hr that primary system temperature had been reduced to about 285^{0}F (because of preoccupation with restoration of NNI/ICS equipment).
7.	Rancho Seco (50-312)	-	10/08/79	Condenser vacuum was lost while at ~15% power causing main steam relief valves to lift. HPI was manually actuated and contributed to overcooling at ~110^{0}F over ~1 hour period.

TABLE 1. Overcooling Events - B&W NSSS Designs (Cont.)

	Plant and Docket No.	Event Date	Letter Date	Event Summary
8.	Rancho Seco (50-312)	-	10/22/74	Decreasing condenser vacuum resulted in human closure of TBP valves and reactor trip. Atmospheric relief valves were isolated and pressure was controlled by steam safety relief valves. Due to low power (little decay heat) plus auxiliary steam load demands an overcooling occurred.
9.	Oconee 1 (50-269)	12/14/78	01/15/79	A power cord fault is thought to have caused spurious ICS response and MFW transient while at ~98% power. The "B" OTSG went dry; then when it was refilled, an overcooling occurred and caused the RCS to reach ~1430 psi actuating the HPI system.
10.	Oconee 3 (50-287)	-	06/27/75 08/08/75	While reactor power was being reduced from 100% to ~15% for a maintenance shutdown, the PZR-PORV malfunctioned and did not reclose. The reactor tripped and the subsequent RCS depressurization by the PORV remaining open led to overcooling of ~101^{0}F/hr during the first hour.
11.	Oconee 3 (50-287)	11/10/79	-	A maintenance act while at ~99% power caused trip of "A" condensate pump and a reactor trip after ICS runback to ~75% power. ICS power was then lost for several minutes along with NNI. Restart of condensate and hotwell pumps caused increase in "B" OTSG and subsequent overcooling. The overcooling was made somewhat more severe because (1) turbine bypass valves were partly open (~50%), (2) auxiliary steam loads were high, and (3) four RC pumps were left running from ~500^{0}F to 450^{0}F violating limits and precautions procedures. A cooldown of ~112^{0}F over about 20 minutes was experienced.

TABLE 1. Overcooling Events - B&W NSSS Designs (Cont.)

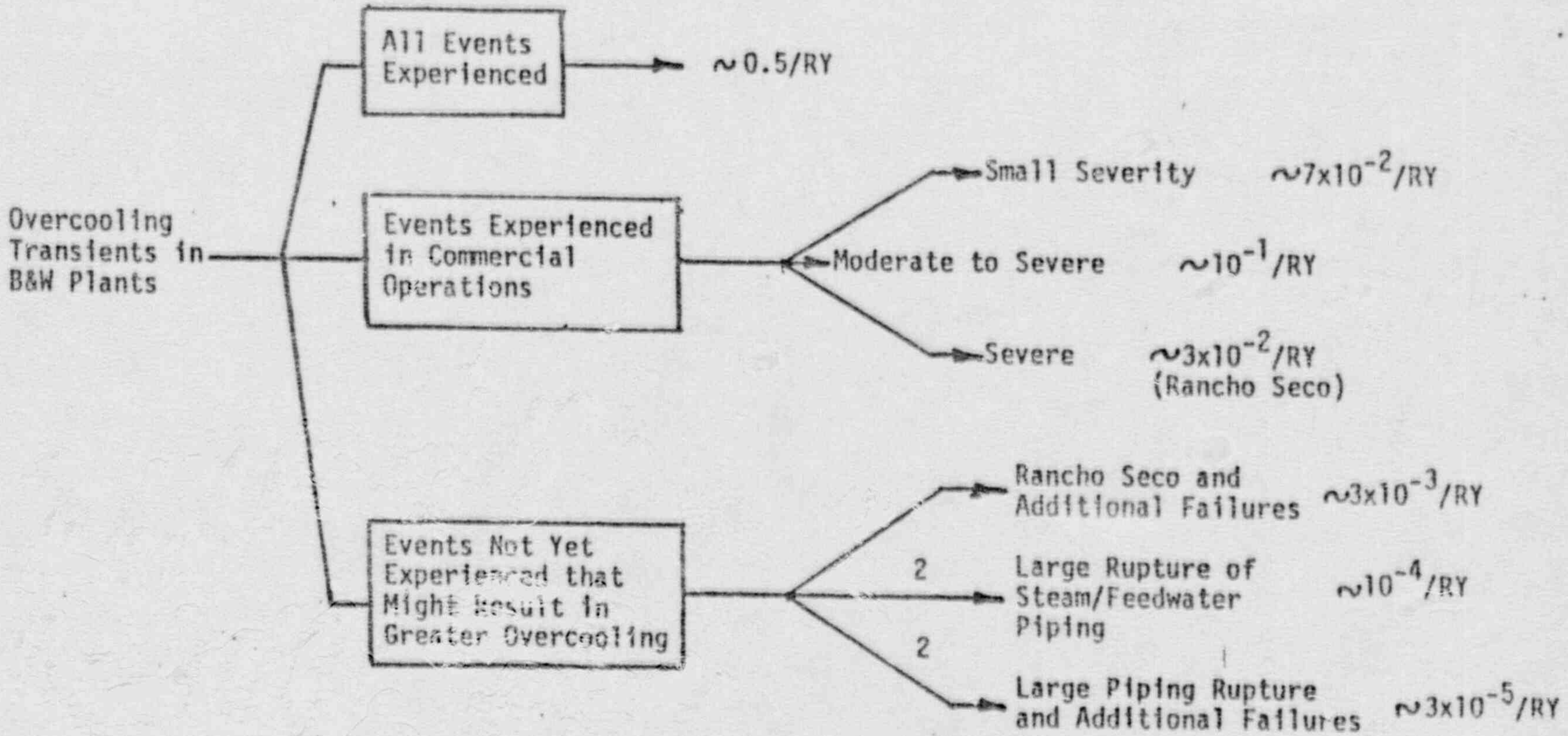
	Plant and Docket No.	Event Date	Letter Date	Event Summary
12.	TMI-2 (50-320)	12/02/78	02/28/79	While at ~22% power and ascending, defective procedures were used in switching from startup to MFW regulating valves. This resulted in reactor trip and an overfeeding of the OTSGs. Actuation of SIS occurred.
13.	Davis Besse	04/29/78	07/28/78	While at ~20% power and shutting down, SIS actuation occurred (≈1620 psi) due to low RCS pressure caused by opening MFW valves on transfer from flow and level control by plant operator. Overcooling was also made more severe by incorrect operator response to the transient.
14.	Davis Besse	03/25/78	04/20/78	After shutting down the plant and raising the level of SG1-2, the level continued to increase >86% maximum specified level. The fault was attributed to failure of FW valve to fully close. Operator took actions and opened SG drain valves for control of level.
15.	TMI-2	04/23/78	05/08/78	The reactor tripped while at ~30% power. The main steam relief valves did not reseat at correct pressure after the trip. During RCS depressurization caused by the blowdown, SIS was actuated. The MFW system response was slow since initial ICS tuning was still in progress.

Figure 1. ILLUSTRATION OF APPROXIMATE SEVERITY OF OVERCOOLING EVENTS

Events Exceeding Tech Spec. Cooldown Rates of $\sim 100^\circ$F/hr.	Small Overcooling Rate $\lesssim 110^\circ$F/hr.	Moderate Overcooling Rate $\gtrsim 120^\circ$F/hr.	Severe Overcooling Rates to $\sim 300^\circ$F/hr.: High Initial Rate	Severe Overcooling Rates to $\sim 300^\circ$F/hr.: Prolonged Overcooling
I. PRIOR TO COMMERCIAL OPERATION				
1. Rancho Seco (~July 1974)	(?)			
2. Rancho Seco (~October 1974)*	($\lesssim 110^\circ$F/hr.)			
3. Davis Besse (03/25/78)	(?)			
4. Oconee #2 (01/04/74)		# ($\sim 140^\circ$F/hr.)		
5. TMI-#2 (12/02/78)		(?)		
6. TMI-#2 (04/23/78)		(?)		
7. Davis Besse (04/29/78)		(?)		
8. Crystal River #3 (03/02/77)			# ($\sim 164^\circ$F/15 m	
9. Oconee #1 (05/05/73)			(?, To ~1330 ps	
II. AFTER COMMERCIAL OPERATION				
1. Oconee #3 (~June 1975)	($\sim 101^\circ$F/hr.)			
2. Crystal River #3 (04/16/77)	($\sim 101^\circ$F/20 min)			
3. Rancho Seco (01/05/79		# ($\sim 120^\circ$F/hr.)		
4. Oconee #3 (11/10/79)		# ($\sim 112^\circ$F/20 min)		
5. Oconee #1 (12/14/78)			# (?, to 1430 p	
6. Rancho Seco (03/20/78)				# ($\sim 300^\circ$F/hr to $\sim 285^\circ$F)

*Possibly includes 2 occurrences when the plant was at low power (~15%)
(?) Actual rate of cooldown not set forth in LER writeup.
These events involved various power faults affecting NNI/ICS response and main feedwater delivery. Overcooling resulted largely during human restoration of feedwater to the OTSGs.
[dark bar] Recommended Benchmark Events.

Figure 2. Estimated Frequency[1] for Overcooling Transients



[1] These estimates are intended as an aid in RSR analysis priorities and should not be taken to be of high precision and certainty.

[2] This frequency estimate assumes actual time of experience as the time from start of commercial operations and it may be larger by less than a factor of two if the relevant time of experience is measured by the accumulated number of critical hours.