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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

Nov. 3, 2000

MEMORANDUM TO: Kress, Bonaca, Uhrig, Wallis, Powers, Apostolakis Larkins, Lyons

MEMORANDUM #: AWC-114.2000

FROM: A. W. Cronenberg

SUBJECT: Summary of Boron Dilution Issues and Regulatory Actions (Prepared for ACRS Presentation to the German-RSK Board)

SUMMARY: Boron has a large thermal-neutron cross-section and as such is used for reactivity control in both PWR and BWR systems. Any dilution of boron in the reactor coolant system (RCS) thus adds reactivity to the system and is therefore of regulatory concern. In PWRs boron is also used for optimization of fuel burnup, compensate for excess reactivity at the beginning of the fuel cycle, and to provide shutdown margin for maintenance and refueling operations.

Boron concentration adjustments are an integral part of the operation of any plant. Examples include changes in boron concentration in the RCS during: (1) Reactor startup -- boron concentration must be decreased from shutdown concentration before taking the reactor to criticality, (2) Load change -- boron concentration must be either increased or decreased to compensate for changes in xenon inventory following a fuel load change, (3) Fuel burnup -- boron concentration must be decreased to compensate for fuel burnup and the buildup of fission products in the fuel, and (4) Cold Shutdown -- boron concentration must be increased to the cold shutdown concentration. The operator determines by use of instrumentation and/or computer analysis, the desired boron concentration for the desired operating mode of the plant. He then either adds boric acid solution or primary water to the RCS as needed, which is governed by administrative controls to limit the rate and amount of boron addition. Changes in boron are made through a process of feed and bleed via the chemical and volume control system. Boron concentration in the RCS is generally decreased (diluted) by controlled addition of un-borated makeup water, with a corresponding removal of previously borated reactor coolant. Another option is the use of ion exchangers to de-borate reactor coolant. Controlled dilution is therefore a necessary operation and is performed at a preselected rate. *Uncontrolled boron dilution* on the other hand is a the decrease in the RCS boron concentration caused by the *inadvertent* addition of un-borated water.

Inadvertent boron dilution events usually occur during shutdown or refueling outages, generally due to human error or equipment failures, including failures of non-process hardware such as leaking seals. During the late 60's and 70's such events were common enough to be of heightened regulatory concern, a time during which approximately 25 reported instances (LER) of inadvertent boron dilution occurred. None of these events however lead to criticality. The issue was one of the earliest to be addressed under the agency's Generic Safety Issues (GSI) program; namely GSI-22 (*Inadvertent Boron Dilution Events).*

More recently the issue of boron dilution has again come to the forefront for the specific case of re-entry of de-borated water into PWR cores with high-burnup fuel, specifically for recovery actions following a small-break LOCA. This issue is under current analysis as GSI-185 (*Control of Recriticality Following Small-Break LOCAs in PWRs).* These two GSIs and related regulatory actions are summarized here.

GSI-22--Inadvertent Boron Dilution Events: The issue of boron dilution was addressed early in the GSI process with publication of NUREG/CR-2798 (*Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants*, 1982) where a survey of dilution related LERs (License Event Report) was made. Results of that investigation indicated that about 80-% of the LERs were for unplanned boron dilutions significantly different from those postulated FEARS (Final Safety Analysis Report). In the FEARS boron dilution events were viewed as principally concerned with the malfunctioning of equipment for control of borated water. However, a survey of the LERs indicated that human errors were the cause for about most dilution incidents. The human errors were associated with incomplete or faulty procedures, selection of wrong controls or equipment, miscalculations, improper sampling, incorrect use of tools, and misapplication of equipment. Actual equipment failures were random and much less of a problem then human errors.

Resolution of GSI-22 thus centered on efforts at to improve operator attentiveness to the problem, with agency action in two areas. The first was to increase the dissemination of information about operating experiences and the feedback of such information to licensees, vendors, and architectural engineers. A number of NRC Information Notices on the subject were thus issued. The agency also recommended more in-depth training for plant operators on boron dilution, with increased administrative controls during maintenance and refueling operations. These actions resulted in a decrease in the number of dilution related LERs during the 1990s, although like LOOP (loss of off-site power), have not been completely eliminated. GSI-22 is now classified as RESOLVED with no new requirements.

GSI-185--Control of Recriticality Following Small-Break LOCAs in PWRs: This issue was recently identified (1999) based on analysis by the B&W Owners' Group (B&WOG) of high burn-up core response to small-break LOCAs. Specifically analysis indicated that cores with high burn-up fuel may be more susceptible to reactivity increases than previously expected for recovery actions for small-break (SB) LOCA events. The problem stems from fact that for SB-LOCAs, a substantial amount of de-borated water may accumulate in the reactor coolant pump (RCP) suction piping. The Analysis showed that RCP restart (for accident recovery) could pump the de-borated water into the core and might cause a re-criticality, particularly for cores incorporating high burnup fuel with beginning of life excess reactivity. The scope of the issue has been expanded to include additional locations for holdup of de-borated water, such as in the steam generators, cold legs, reactor vessel down-comer, and reactor vessel lower plenum. The problem is perceived to be greater in B&W NSSS (Nuclear Steam Supply System) than in the Westinghouse and CE designs because the B&W lowered-loop geometry may favor the accumulation of more de-borated water. Nevertheless, Framatome Technologies, Inc.

developed guidance to restrict RCP restart to prevent potential fuel damage under such conditions for French reactors, which more closely resemble Westinghouse NSSS designs.

Recently B&WOG prepared a progress report which reiterated that, with conservative assumptions, displacement of de-borated water had the potential to cause a prompt-critical condition due to insertion of several dollars of excess reactivity. B&WOG however concluded that this was an operational issue, not a safety concern, and that potential plant consequences under 10CFR50.46 assumptions need not be determined. B&WOG has advised licensees with B&W-designed NSSS to restrict RCP restart following SB-LOCAs, until the de-borated water has been adequately mixed with borated water. This industry voluntary action is under consideration for possible regulatory guidance for issue resolution.

At this time GSI-185 is under current investigation by the agency. RES is supporting a test program at the University of Maryland to simulate B&W-NSSS conditions. Although test data have been obtained for restart of RCS and natural circulation, the applicability to the issue of de-borated water has not been established. The issue is ranked HIGH in the GSI process, where emphasis is being placed at resolution of the thermal-hydraulic modeling uncertainties and consequent re-criticality, particularly for cores incorporating high burnup fuel with beginning of life excess reactivity.

<u>Attachments (Larkins only)</u>

GSI-22, *Inadvertent Boron Dilution Events*, in NUREG-0933 (*Prioritization of Generic Safety Issues)*

Memo from Farouk Eltawila to Ashok Thadani: GSI-185, *Control of Recriticality Following Small-Break LOCAs in PWRs,* (July 7, 2000).

NUREG/CR-2798, *Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants, (1982).*

ISSUE 22: INADVERTENT BORON DILUTION EVENTS

DESCRIPTION

Historical Background

Many PWRs have no positive means of detecting boron dilution during cold shutdown.[25] Some operations carried out during outages (e.g., steam generator decontamination) reduce the RCS volume, thus speeding up dilution. Boron dilution has taken place during such operations although, thus far, criticality has not occurred.[26] An analysis of this issue was provided in a DST memorandum.[108]

Safety Significance

There have been 25 reported instances of inadvertent boron dilution during maintenance and refueling.[109] Although none has yet occurred, the safety concern is the possibility of an inadvertent criticality. If the boron is sufficiently diluted and the reactor core is near beginning of cycle, it is possible to bring the reactor to criticality with all of the control rods inserted into the core. The only way to shut the core down again in such a circumstance would be to reborate the moderator, which could take considerable time. The events have occurred with sufficient frequency to raise the question whether, considering their possible consequences, the degree of protection is appropriate.

Possible Solution

All 43 operating PWRs are affected by this condition. The fix is to install instrumentation to detect the event and stop the dilution either automatically or, if the detection is sufficiently early, by alerting the operator.

PRIORITY DETERMINATION

Frequency Estimate

Boron dilution events during a shutdown or refueling have usually been caused either by human error or by failures of special, nonprocess equipment such as inflatable seals. Therefore, event frequencies cannot be easily calculated by fault tree analysis. Moreover, because no event has yet resulted in criticality, it is not possible to simply add up the number of events in operating history.

The fact that no inadvertent criticalities have happened in 337 PWR-years allows us to estimate an upper bound to the frequency. By assuming a Poisson distribution and using a 95% confidence level, the frequency of inadvertent criticalities is, at most, 9×10^{-3} event per PWR-year.

However, an upper limit is not sufficient to gauge the significance of boron dilution events; a "best estimate" (in some sense) is needed. The only information available is contained in the frequency of boron dilution events which have happened but which did not result in criticality. Most of these events can be considered "precursor" events to an actual inadvertent criticality.

The severity of a precursor event is defined here in terms of the shutdown margin remaining at the end of the event. That is, an event which was halted with 2% shutdown remaining is considered more severe than an event which was halted with 10% remaining shutdown margin.

Using the information in the NUREG/CR-2798,[110] a histogram shows that the number of events goes down as the severity increases. To estimate an expectation value for the number of critical events, a two-parameter exponential distribution was fitted to the data. Extrapolation of this distribution to the point of zero shutdown margin gives a value of 0.67 event in a time interval of (currently) 337 PWR-years. Thus, we expect the frequency of inadvertent criticalities to be on the order of 2×10^{-3} event per PWR-year.

This calculation, although rough, gives an answer that is reasonable. With 43 PWRs presently operating, we would expect an inadvertent criticality roughly every 11 years, if nothing were done.

However, this number does not take into account the effect of the neutron monitoring instrumentation. As a reactor core approaches criticality, neutron flux does not rise linearly. Instead, the reciprocal of the flux drops linearly as shutdown margin decreases. The net effect is that neutron flux rises slowly as the reactor core goes from 10% to 9% shutdown, but rises very dramatically as shutdown margin drops below 0.5%. None of the events tabulated in NUREG/CR-2798[110] came close enough to criticality for the neutron monitoring channels to trigger alarms. Thus, to realistically estimate the frequency of an event that continues in dilution to criticality, we must give some credit for the neutron flux channel alarms, which are usually set one-half to one decade above background.

Since the control rods are already fully inserted into the core in this event, the only actions which will prevent criticality are stopping the dilution or reborating the moderator. Both are done by the operator. Thus, the credit to be given for neutron flux alarms is governed by the reliance which can be placed on the operator. We will assume (based purely on judgment) that the operator will be able to correctly diagnose the problem and successfully prevent criticality 90% of the time. This drops the frequency of a criticality by one order of magnitude, to 2×10^{-4} event/RY. Of these, roughly one sixth will take place with the reactor head removed. Thus, the frequency of radioactivity-releasing events is 3×10^{-5}/RY.

Consequence Estimate

In the PWR case under consideration here, all rods are either already in the core or are disconnected from their drives. Either way, there is no scram reactivity available. Shutdown by emergency boration will take much more time than shutdown via scram. The important parameter is the peak level achieved by the core.

Once the core becomes critical, it will heat up with a positive period governed by the rate of dilution and by moderator temperature and Doppler feedback. Eventually the coolant may boil and the peak power level will be limited by void generation in the moderator. Preliminary calculations indicate that, assuming BOC parameters (worst case), a power level of about 3% of rated would be reached.[111] (These calculations are limited in their ability to model the multidimensional aspects of void feedback.)

A core power of 3% of rated is not likely to fail fuel that must withstand decay heat rates of this same order. The only likely consequence is the release of gap activity from any leak already present. If we make the standard assumption of users of the GALE codes that 0.16% of the fuel leaks, the total activity released to the coolant would be roughly 69,000 Ci. This is not enough activity to be significant unless the vessel head is removed. If the vessel head were not in place, about 10% of this activity, or 6,900 Ci, would escape from containment, based on analyses of dropped fuel assembly events. Consequences for this event are expressed in man-rem. The total whole-body man-rem dose is obtained by using the CRAC Code[64] for the particular release category. The calculations assume a uniform population density of 340 people per square mile (which is average for U.S. domestic sites) and a typical (midwest plain) meteorology. Therefore, the dose for such as event would be 700 man-rem.

For 43 PWR operating plants with an average remaining life of 30 years, the total risk reduction is $(43)(3 \times 10^{-5})(7 \times 10^{2})(30)$ man-rem or 27.1 man-rem.

Cost Estimate

Industry Cost: Since these events are caused by a wide spectrum of causes, it is not practical to reduce the frequency of boron dilution events other than by bringing the matter to the attention of plant operations personnel and having them upgrade their procedures (if and where appropriate). It has been proposed to install a microprocessor-based monitor on the source range neutron flux instrumentation. Such a monitor, if connected to a display panel such as the safety parameter display system (SPDS), could give earlier warning of loss of shutdown margin than is possible with the present instrumentation, and thus would reduce the probability of a boron dilution event leading to criticality.

We have evaluated[112] the cost of such a system. The results are:

Control grade instrument, alarm only	-- \$ 50,000
Safety grade instrument, alarm plus automatic initiation of emergency boration	-- \$ 300,000

To be conservative, we assumed that the cheapest hardware fix at a cost of \$50,000/plant would be used. Therefore, the total industry cost is \$(0.05)(43)M = \$2.2M.

NRC Cost: The cost to the NRC is estimated to be 2 staff-months plus 1 staff-week for each of the 43 operating PWRs. This corresponds to an NRC cost of \$84,000 which is small in comparison to the cost of industry.

Value/Impact Assessment

Based on a total public risk reduction of 27.1 man-rem, the value/impact score is given by:

$$S = \frac{27.1 \text{ man-rem}}{\$2.2M}$$

$$= 12 \text{ man-rem/\$M}$$

Uncertainties

The upper limit (95% confidence) on inadvertent criticality frequency without credit for neutron flux alarms was a factor of 5 over the "best" estimate. If we assume a symmetrical distribution and also assume a factor of 5 error in the credit for the neutron flux alarms and a factor of 3 error in the chance of the head being off the vessel, the estimated error in the frequency of radioactive release is plus or minus a factor of 8.

The release is expected to be on the order of 6,900 Ci, primarily noble gases. We will use an estimated error of a factor of 5, again based on judgment.

The uncertainty in the costs, which are dominated by the $50,000/plant, is at most a factor of 2.

CONCLUSION

Based on the low value/impact score and low risk reduction associated with an inadvertent criticality, DST concluded that boron dilution events do not constitute a significant risk to the public and recommended that the issue be dropped from further consideration. However, DSI disagreed with this evaluation and obtained permission from the NRR Director to pursue the issue further.

As a result of DSI's work, it was determined that the consequences of an unmitigated boron dilution event, although undesirable, are not severe enough to warrant backfit of additional protective features at operating plants. DSI recommended that DL issue a generic letter to OLs informing them of this result and pointing out that the event represents a breakdown in a licensee's ability to control its plant. DSI concluded that the criteria in SRP[11] Section 15.4.6 are adequate for plants currently undergoing license review.[693] Furthermore, because offsite consequences following the event are likely to be insignificant, DSI also recommended that SRP[11] Section 15.4.6 be considered for deregulation.[694] This recommendation is covered in Issue 104. Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

25. Memorandum for F. Schroeder from T. Novak, "Application of SRP 15.4.6 Acceptance Criteria to Operating Reactors," December 12, 1980.

26. IE Information Notice 80-34, "Boron Dilution of Reactor Coolant During Steam Generator Decontamination," U.S. Nuclear Regulatory Commission, September 26, 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

108. Memorandum for R. Mattson from S. Hanauer, "Inadvertent Boron Dilution," March 10, 1982.

109. Memorandum for T. Murley from R. Mattson, "Inadvertent Boron Dilution," September 15, 1981.

110. NUREG/CR-2798, "Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1982.

111. Letter to R. T. Curtis (NRC) from N. S. DeMuth (LANL), "Analyses of Unmitigated Boron Dilution Events," November 18, 1981.

112. Final Report, "Determination of the Cost of Modifications Needed to Mitigate Boron Dilution Events in PWRs during Cold Shutdown," NRC Contract FIN A6452 (DOE Work Order 20-81-297), December 14, 1981.

693. Memorandum for H. Denton from R. Bernero, "Resolution of Generic Issue No. 22, Inadvertent Boron Dilution Events (BDES)," September 17, 1984.

694. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issue No. 22, 'Inadvertent Boron Dilution Events (BDE),'" October 15, 1984.

July 7, 2000

MEMORANDUM TO: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

FROM: Farouk Eltawila, Acting Director /RA/
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: GENERIC ISSUE NO. 185, "CONTROL OF RECRITICALITY FOLLOWING SMALL-BREAK LOCAs IN PWRs"

The technical screening of Generic Safety Issue (GSI) No. 185, "Control of Recriticality Following Small-Break LOCAs in PWRs," shows that procedural resolution of the issue can be accomplished without backfit. The staff found that some small-break LOCAs in PWRs involve steam generation in the core and condensation in the steam generators which cause deborated water to accumulate in part of the reactor coolant system (RCS). Under these circumstances, restart of the RCS circulation may cause a deboration event by moving this deborated water into the core. The problem is perceived to be greater in most B&W-designed plants than in those designed by Westinghouse and CE because the lowered-loop geometry of B&W plants may favor the accumulation of more deborated water. The staff's technical screening evaluation is attached. This evaluation was reviewed by NRR and comments have been incorporated.

We recommend that work on the GSI continue and that your approval be granted for the performance of a technical assessment. The technical contact for this issue is Harold Vandermolen (301-415-6236).

Approved: M.V. Federline for:
Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

Date: 07/06/2000

Attachment: Technical Screening Evaluation - Issue 185
Control of Reactivity Following
Small-Break LOCAs in PWRs

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ISSUE 185: CONTROL OF RECRITICALITY FOLLOWING SMALL-BREAK LOCAs IN PWRs

DESCRIPTION

Historical Background

This issue was identified[1730] following an NRR request for reconsideration of the safety priority ranking (DROP) of GSI-22, "Inadvertent Boron Dilution Events," based on new information on high burn-up fuel and new calculations provided by the B&W Owners' Group (B&WOG). Reactivity insertion event tests indicated that high burn-up fuel may be more susceptible to reactivity events than previously expected, and fuel failure may occur at fuel enthalpy values that were previously judged acceptable. In addition, B&WOG calculations predicted prompt criticality with significant heat generation under conditions that may result from small-break (SB) LOCAs. NRR believed that there is no regulatory guidance applicable to this issue.

NRR had previously reviewed studies of deborated water formation during SBLOCAs in PWRs and concluded that: (1) recovery of natural circulation was unlikely to lead to core damage from reactivity transients; and (2) starting or "bumping" of RCPs could lead to a large reactivity transient. However, recent B&WOG calculations predict prompt criticality from natural circulation restart with an accompanying significant heat generation, which raised serious questions about potential reactivity events.

NRR was informed in June 1995 that, if a B&W-designed NSSS spends some time in a boiling/condensing mode following an SBLOCA, a substantial amount of deborated water may accumulate in the RCP suction piping.[1728] Analysis showed that RCP restart would pump the deborated water into the core and might cause a criticality. In July 1995, the scope of the issue was expanded to include: (1) deborated water in the steam generators, cold legs, reactor vessel downcomer, and reactor vessel lower plenum; (2) restart of natural circulation as a mechanism for causing deborated water to flow into the core, and possibly result in criticality; and (3) the potential for prompt criticality.[1728] In late 1996, Framatome Technologies, Inc. (FTI) developed guidance to restrict RCP restart to prevent potential fuel damage.[1728]

In June 1998, the B&WOG prepared a progress report which reiterated that, with conservative assumptions, displacement of deborated water had the potential to cause a prompt-critical condition due to insertion of several dollars of excess reactivity.[1729] In this report the B&WOG concluded that this was an operational issue, not a safety concern, and that potential plant consequences under 10 CFR 50.46 assumptions need not be determined. The June 1998 report was not sufficient to assess the work that had been completed and NRR did not concur with the B&WOG conclusions.

On September 11, 1998, the B&WOG reported new calculation results, provided PRA values to clarify the significance of the safety concern, committed to provide an in-depth investigation to substantiate the September 11, 1998, results, and stated that three utilities had responded to the FTI recommendations regarding RCP restart and two others were in the process of responding.[1728]

Safety Significance

Although the original request from NRR requested a re-opening of Generic Issue 22, "Inadvertent Boron Dilution Events," the scope of GI-22 covered inadvertent boron dilution events when the reactor was in shutdown or refueling modes, a completely different scenario with different conditions, causes, and potential fixes. Thus, this new issue was initiated to address this new scenario.

Some SBLOCAs in PWRs involve steam generation in the core and condensation in the steam generators, causing deborated water to accumulate in part of the RCS. Restart of RCS circulation may cause a deboration event by moving this deborated water into the core. The problem is perceived to be greater in most NSSS designed by B&W than in the Westinghouse and CE designs because the B&W lowered-loop geometry may favor the accumulation of more deborated water.

Although the B&WOG calculated that the restart of natural circulation following some SBLOCAs may result in prompt criticality with deposition of significant energy in the fuel, similar information has not been provided for operating Westinghouse- and CE-designed NSSS, although Westinghouse representatives have written that RCP restart with a large quantity of deborated water must be prevented.

Potential core damage associated with RCP restart is not addressed in the B&WOG PRA and ideally would be included, since operator error may lead to inappropriate RCP restart and there are uncertainties associated with the analysis underlying restart guidance. Consequently, NRR did not concur with the B&WOG conclusion that there is no regulatory concern associated with potential recriticality due to restart of natural circulation. Although this analysis focused on B&W reactors, the generic issue is applicable to all PWRs.

Possible Solution

Because of the potential consequences of an inappropriate RCP start, the B&WOG advised licensees with B&W-designed NSSS to restrict RCP restart following SBLOCAs until the deborated water has been adequately mixed with borated water. This industry voluntary action could be included in regulatory guidance to be issued to all plants.

RES is supporting a test program at the University of Maryland thermal-hydraulic test facility that represents the B&W NSSS configuration. Test data have been obtained for restart of RCPs and of natural circulation, but applicability to the issue of deborated water has not been established. (When confronted with a similar problem with the CE System 80^{+}, the planned boron concentration in the refueling water storage tank was increased to ensure non-criticality.)

PRIORITY DETERMINATION

In its request for prioritization of this issue,[1730] NRR stated, "The fuel damage probability indicates that a significant safety problem is unlikely. Further, we judge that a backfit would not be cost-beneficial and would not be justified under 10 CFR 50.109. Nonetheless, modeling uncertainties are high and the potential consequences associated with prompt criticality are of sufficient concern that further assessment may be necessary."

The essence of the issue, as defined by NRR, is thus the thermal-hydraulic modeling uncertainty and the uncertainty in the potential consequences associated with prompt criticality. This analysis will therefore assess the importance of the thermal-hydraulic phenomena and the consequences of prompt criticality, i.e., the "worst" will be assumed for these two effects, namely that the boron dilution phenomenon will occur and that a prompt criticality will result in significant fuel damage, and the risk importance of the two effects, assuming the worst, will be estimated. (These assumptions are appropriate for the prioritization phase. The actual evaluation of the thermal-hydraulic phenomena and the consequences of prompt criticality is reserved for the resolution phase of this generic issue.

Description of sequence (B&W NSSS Design):

The event sequence for a B&W design will be explored first, since, as will be seen later, the thermal-hydraulic phenomena are somewhat simpler. (Other PWR designs will be examined in a later section.) The plant chosen for analysis is Crystal River Unit 3, a fairly typical 177-fuel assembly lowered-loop design. This plant was chosen primarily because of the ready availability of a RELAP model and considerable design information.

The event of interest begins with an "S2" small LOCA. As reactor coolant escapes, ECCS and auxiliary feedwater start on low pressurizer pressure. (The emergency procedures instruct the operator to trip the reactor coolant pumps once successful operation of high pressure injection is verified.) The high pressure injection pumps attempt to replace the lost coolant. However, the break size is too large, and the primary system pressure too high, for the HPI pumps to maintain inventory, and the coolant level in the pressurizer drops. Eventually, the pressurizer empties, and steam spaces form in the tops of the hot leg pipes, just above the steam generators, because these locations are the highest points in the system (see Figure 1, taken from NUREG/CR-5640[1759]). When the level drops to the point where there is no longer a liquid pathway to the top of the steam generators, natural circulation ceases, and the coolant in the reactor core region heats up and begins to boil, keeping system pressure high. The coolant level continues to drop and the upper portion of the steam generator tubes fill with steam.

The AFW systems in B&W plants spray feedwater into the upper portion of the steam generators. As the primary level drops further, more and more cool steam generator tube surface is exposed to the steam in the primary system, condensing it back into liquid. Eventually, as more and more steam generator tube surface is exposed to the vapor phase, the heat removal from condensation matches the heat generation in the core.

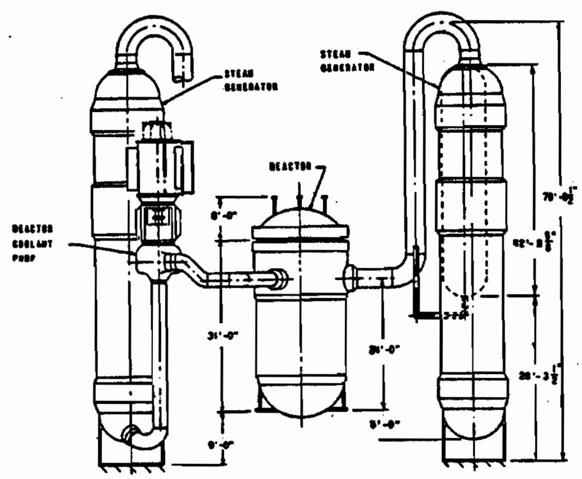

Figure 1: B&W NSSS

An equilibrium condition would be achieved, with the coolant boiling in the core and condensing in the steam generators, if it were not for the continued loss of coolant through the "S2" break. As level drops further, and still more cool steam generator tube surface is exposed to the vapor phase, primary pressure drops. (The heat generation rate in the core is also slowly decreasing due to radioactive decay, which contributes to the pressure drop.) As the pressure decreases, the flow rate from the high pressure coolant injection trains increases, and eventually the injection rate will equal the loss through the break.

This scenario is actually a successful operation of the ECCS, which would avoid severe core damage. However, this method of core cooling, which boils coolant in the core, condenses coolant in the steam generator, and returns coolant to the core through the cold leg, also removes the soluble boron from the coolant via distillation. The condensed coolant in the steam generator lower plena and cold leg piping will have a nearly zero boron concentration, while the boron concentration in the reactor vessel core volume will increase. (There will be some injection of borated coolant at the reactor coolant pump seals, but the coolant return flow will carry this boron into the reactor vessel.)

The deborated coolant region will not be troublesome as long as the system remains in the "reflux boiling" state, since deborated coolant entering the reactor will mix with the more concentrated boron solution in the core region. However, if the system is refilled to the point where liquid natural circulation restarts, or if the reactor coolant pumps are started, the deborated, relatively cool coolant which has accumulated in the cold legs and steam generators will be swept into the reactor core. In a typical 177-fuel assembly B&W NSSS (including Crystal River), the tube side free water volume of each steam generator is 2030 cubic feet,[1759] while the water volume of the reactor vessel is 3910 cubic feet (from the Crystal River RELAP model). Thus, the two steam generators would contain a water volume slightly larger than that of the reactor vessel. Thus, it appears plausible that, should natural circulation be re-established, the deborated coolant could momentarily flush the borated coolant out of the core with relatively little mixing. As was stated above, it will be assumed that this happens, consistent with the "worst-case" assumption. It should be noted that there is considerable uncertainty as to the reality of this phenomenon.

After shutdown, decay heat will drop rapidly to about 2% of rated thermal power, and continue to decrease. At this power level, a simple hand calculation shows that, if natural circulation is lost, the core will boil enough coolant to fill the steam generators with condensed coolant in about 25 minutes. Thus, the scenario is credible. Since there is return flow of condensed coolant from the steam generators to the reactor through the cold legs, it is unlikely that any dissolved boric acid will diffuse back into the steam generator volumes. However, it is possible that deborated coolant will gradually fill the reactor vessel downcomer and lower plenum, with soluble boron concentrating (and possibly precipitating) in the core region. How much mixing will occur in the lower plenum and downcomer is a source of uncertainty that will ultimately need to be resolved, but for prioritization purposes, it will be assumed that the deborated volume in the steam generators will be sufficient to (at least momentarily) flood the core region.

If the accident should occur early in the fuel cycle, there may be sufficient excess reactivity in the core for the deborated coolant to bring the core to criticality even though all the control rods have been inserted. The possible power excursion may be sufficient to cause severe damage to the core, even though the ECCS has successfully kept the core covered with coolant. It is this power excursion that is the basis for this generic issue.

Event Tree: An event tree was constructed to quantify this scenario (see Figure 2).

Small Break LOCA: The initiating event for this scenario is a LOCA of the proper size - large enough for the high pressure injection to not keep up with coolant loss at full primary system pressure, but small enough to not depressurize the system. This is an "S2" break in the language of NUREG-1150,[1081] a break of ½ to 2 inches equivalent diameter, corresponding to a fluid loss rate of approximately 100 to 1500 gallons per minute (gpm). The frequency of such breaks in NUREG-1150 is 10^{-3} per reactor-year.

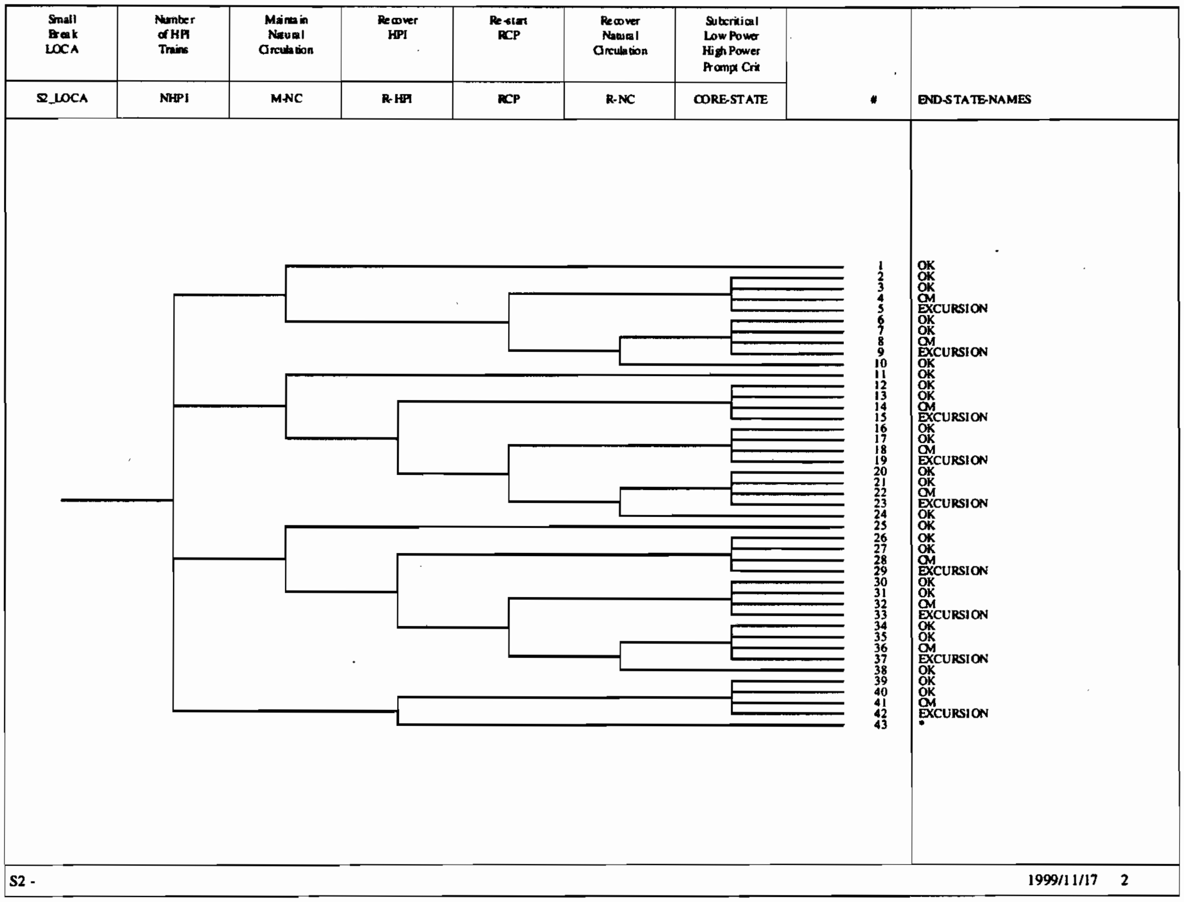

Figure 2: Event Tree

Number of HPI trains: Once the break occurs, high pressure injection will initiate. This particular plant has three HPI trains, two of which will start automatically, and one of which is kept "in reserve," and may be manually initiated by the operator. For this analysis, which is intended to be more generic, it will be assumed that all three trains will be started shortly after the onset of coolant loss. Thus, there can be four possible outcomes, corresponding to zero, one, two, or three trains operating. A full calculation of the probabilities of these four system states is beyond the scope of this prioritization analysis. Instead, it will be assumed that the likelihood of a single train failure is dominated by the unavailability of the pump (3.8×10^{-3} in the Crystal River SPAR-2QA model[1761]). For the failure of the entire system, the SPAR-2QA figure for the entire system will be used (1.019×10^{-4}, again from the SPAR-2QA model[1761]). If the unavailability of one pump is "p," the four probabilities, using the rare event approximation, are as follows:

$P(0) = 1.019 \times 10^{-4}$ (the SPAR-2QA number for the entire system[1761])

$P(1) = 3(1-p)p^2 = 4.32 \times 10^{-5}$

$P(2) = 3(1-p)^2 p = 1.113 \times 10^{-2}$

$P(3) = 1 - [P(0) + P(1) + P(2)] = 0.9887$

Two caveats should be noted: first, the number of significant figures being used is for the convenience of forming differences between numbers and for the reader who wishes to reproduce the calculation, not because the unavailabilities are known to such high accuracy, and appropriate rounding will be performed at the end of the calculation. Second, the approximation being used assumes that all common cause failures will fail all three trains, and also that failure other than pump failures will fail all three trains. For this reason, P(0), the probability of no trains operating, is higher than P(1).

It will be assumed that the operator will shut down the reactor coolant pumps with a probability of unity. This is a standard "no miracles" assumption in all PRA calculations - a failure to follow procedures is never credited as a positive outcome.

Maintain Natural Circulation: If the flow out the break is less than or equal to the injection flow from the HPI trains, the coolant level will not drop out of the pressurizer, and natural circulation will be maintained. If the HPI trains cannot keep up with the break flow, the level will drop and natural circulation will be lost. (Eventually, pressure will drop to the saturation pressure for the existing coolant temperature, and HPI flow will increase as pressure drops.)

The likelihood of a particular break size decreases as the equivalent diameter increases, which is why large break "A" LOCAs are less likely than small break "S1" LOCAs, which in turn are less likely than very small break "S2" LOCAs. However, for purposes of prioritization, it will be assumed that the likelihood of a particular break size is constant over the S2 size interval, which will be assumed to be equivalent to the "G3" coolant loss rate assessed by the former AEOD.[1760] Comparing these coolant loss rates with the capability of the HPI pumps:

Number of pumps	Flow at 1600 psi[1759] (gpm)	Flow at 2255 psi[1759] (gpm)	Fraction of 100-1500 gpm "G3" spectrum covered	Probability of loss of natural circulation
1	400	270	21.4%	79%
2	800	540	50%	50%
3	1200	810	78.6%	21%

Thus, the likelihood of loss of natural circulation depends on the number of HPI trains running. If all three trains of HPI fail, the probability of loss of natural circulation is unity.

Recover HPI: There is some likelihood that the operator will be able to recover a train of HPI. To estimate this probability, the operator's probability of recovery for the "SLOCA" sequences in the Crystal River SPAR-2QA model[1761] will be used. This parameter, designated "SLOCA-XHE-NOREC" is 43% of non-recovery, implying a recovery probability of 57%.

Re-start reactor coolant pumps (RCPs): For the usual small break LOCA sequences, procedures call for the operator to trip the reactor coolant pumps once it is verified that a train of HPI is operating. (The RCPs add a significant amount of energy to the primary system.) However, if the operator discovers that natural circulation has been lost and coolant is boiling in the core, the operator may elect to re-start a reactor coolant pump to ensure that the upper portion of the core does not rise above the liquid/vapor interface but instead is cooled by two-phase flow. There is essentially no precedent for this situation, and thus, based purely on judgment, a probability of 10% will be used for this parameter.

Recover Natural Circulation: The operator may be able to recover natural circulation, possibly by using the charging pumps (for which we have heretofore given no credit - the Crystal River plant does not have separate charging pumps, but other plants may be so equipped), by isolating the break (which might be a stuck-open valve for a LOCA in this size range), by manually starting a reserve train of HPI (in plants so equipped, such as Crystal River), or by blowing down the secondary side of a steam generator, thereby reducing the temperature and pressure in the primary, reducing flow out the break in the system, and permitting more injection flow from the HPI trains. Eventually, as decay heat slowly drops, the coolant level will rise. Again, there is no available estimate for this situation. Based on judgment, 50% will be used for this parameter.

Core State: PWR cores must be designed with sufficient excess reactivity to be able to remain at power throughout the fuel cycle. At the end of the cycle, there is no soluble boron in the coolant. Conversely, a high boron concentration is present at the beginning of the cycle, to compensate for the excess reactivity designed into the core. The longer the cycle, the more excess reactivity must be designed into the core, and the higher the beginning-of-cycle boron concentration. However, there is a limit to how high a boron concentration can be used, since the presence of soluble boron causes the moderator temperature coefficient (MTC) to be less negative. At the beginning of the cycle, the MTC is usually close to zero. The core designer may (and usually does) use burnable poison to further extend the cycle. The burnable poison holds reactivity "down" at the beginning of the cycle without causing the MTC to become excessively positive.

Boron concentration thus drops during the course of the cycle, very rapidly at first as xenon and samarium build up to equilibrium levels. Boron concentration as a function of burnup (commonly called "boron letdown curves") for the reactor under study is shown in Figure 3 (from the Crystal River updated FSAR). (It should be noted that the full equilibrium cycle for this plant is 310 effective full power days, even though the curve reaches zero boron concentration slightly before 300 days. It is at this point that the transient rod bank is moved out of the core, which extends core life by approximately 30 days.)

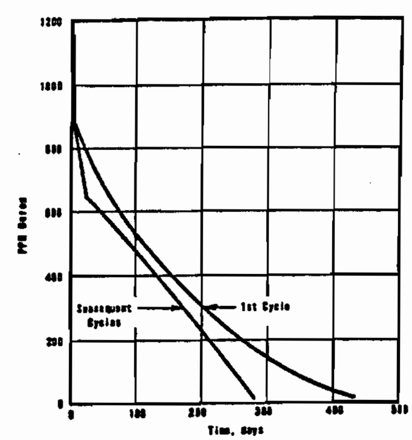

Figure 3: Boron Letdown

The significance of this for current purposes is that, at the beginning of the cycle, the reactivity worth of the soluble boron is greater than the worth of the control rods, Thus, if the soluble boron is swept out of the core and replaced with deborated coolant, the control rods do not have sufficient worth to keep the core in a subcritical state.

The boron letdown and reactivity characteristics can vary considerably from plant to plant or even from cycle to cycle, since the core designer may be aiming for a longer cycle, a flatter power distribution, maximum burnup on older fuel assemblies, or any number of other factors. Thus, although this calculation must of necessity be based on one set of core parameters, these numbers must not be taken as being universally applicable to all plants and all cycles.

This particular cycle (the equilibrium cycle described in the Crystal River updated FSAR) has a soluble boron worth of 0.01 %Δk/k per ppm of boron, a total rod worth of 7.0 % (not including a stuck rod allowance of 1.6 %), and moderator and Doppler deficits of 0.2% and 1.7%, respectively. The excess reactivity was estimated and is shown in Figure 4.

As can be seen from the graph in Figure 4, there is an interval of approximately 24 days at the beginning of the cycle during which the control rod worth is insufficient to render the core subcritical. The probability of occurrence of such a criticality is just the number of days where this is possible (24) divided by the total number of days in the cycle (310), giving a probability of approximately 7.7 percent.

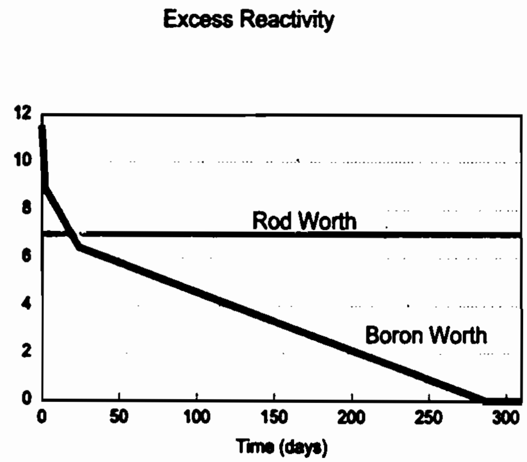

Figure 4: Excess Reactivity vs. Time

However, criticality does not automatically equate to severe core damage. In this scenario, AFW is operating, and both steam generators are capable of removing heat from the primary system. This plant is equipped with two AFW pumps, each capable of supplying 740 gpm of feedwater,[1761] which would accommodate approximately 7% of the reactor's rated thermal power. With both AFW pumps operating, and subtracting off 2% for the decay heat being produced in the reactor core, the steam generators should be able to accommodate fission heat up to approximately 12% of rated power. However, the fission heat will not be continuous, but will "chug" as the deborated coolant sweeps in and out of the core. Therefore, it will be assumed that the steam generators can accommodate power pulses of up to double the continuous power, or approximately 25% of rated thermal power. Any power pulse above 25% will be assumed to result in core damage.

If the net reactivity is greater than approximately 0.5% Δk/k, the core will be in a state of prompt criticality, and will experience a power excursion. This will also be assumed to result in severe core damage, consistent with the "worst-case" assumption discussed previously.

If the deborated coolant fills the core area relatively slowly, as would be expected in the case of a refill of the system and a restart of natural circulation, there will be time for the moderator temperature coefficient to limit core power. The situation is different if the reactor coolant pumps are restarted. The design forced coolant flow rate (131.3 x 10^6 lb/hr) corresponds to a core transit time of approximately 0.6 seconds. Of course, all four coolant pumps will not be switched on simultaneously, so the deborated coolant may take two or three seconds to flood the core. This is still significantly less than the thermal time constant of the fuel rods (roughly 6 seconds for most designs), and there will be little negative feedback provided by the moderator temperature coefficient. Moreover, there is a fairly strong tendency for the incremental axial reactivity worth to concentrate near the top in any core with significant burnup, which will accelerate the incremental reactivity insertion rate. Therefore, only Doppler feedback will be assumed for event sequences involving restart of the reactor coolant pumps. (The moderator temperature coefficient is only slightly negative at the beginning of the cycle, and thus the two situations are not vastly different.)

There is also a timing window effect due to the xenon transient, as is shown in Figure 5 (from the NRC training manual for PWR plants). If the core is operating at full power and has achieved an equilibrium xenon concentration, the xenon concentration will increase and insert still more negative reactivity after the reactor shuts down. For a shutdown from full power, the negative reactivity peaks about eight hours after shutdown, returns to the equilibrium value after approximately one day, and then continues to decrease, which implies that still more shutdown reactivity is needed to keep the core in a subcritical condition. It will be assumed that the operators will have the plant stabilized by the time a full day has gone by, and thus the effects of the xenon "tail" will not be considered here.

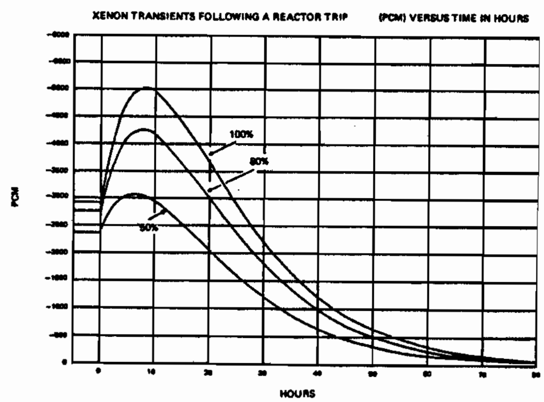

Figure 5: Typical Xenon Transients

It should be noted that, for the first few hours after reactor trip, if natural circulation or pump restart occurs later in time, the likelihood of a re-criticality is less, because of the xenon transient. The excess reactivity at the very beginning of the cycle is sufficient to overcome the xenon overshoot even at its peak, but the xenon effect might prevent a criticality if the boron dilution event occurred after an hour or so and if the event occurred a little later in the fuel cycle.

The boron curve was digitized and the excess reactivity compared with the various deficits. Of the 310 days in the fuel cycle, criticality is possible with all rods in for approximately the first 20 days. The probabilities of the various branches were as follows:

	Probability of prompt criticality	Probability of overpower	Probability of criticality, low power	Probability of no criticality
Slow reactivity insertion	2/310 (0.6%)	13/310 (4.2%)	5/310 (1.6%)	290/310 (93.6%)
Fast reactivity insertion	4/310 (1.3%)	11/310 (3.5%)	5/310 (1.6%)	290/310 (93.6%)

In summary, after the first four days of the fuel cycle, a reactivity excursion is no longer possible, and after 15 days, significant core damage is no longer possible. These figures can vary somewhat from plant to plant and cycle to cycle, however.

Results:

The results of the event tree calculation for this B&W design were a frequency of core damage of 5.7×10^{-6} events per reactor-year, of which 9×10^{-7} events per reactor-year involved a reactivity excursion.

The highest frequency scenario corresponded to sequences 8 and 9 on the event tree. The scenario is initiated by a small break LOCA, all three HPI trains operate, but flow is not sufficient to maintain natural circulation. The RCPs are not re-started, but natural circulation re-starts after the steam generators fill with deborated coolant. The frequency of a reactivity excursion is 2 x 10^{-7}, and the frequency of severe core damage is an additional 4 x 10^{-6}, per reactor-year.

The second highest frequency scenario, corresponding to sequences 4 and 5, is similar, but instead of recovering natural circulation, the reactor coolant pumps are re-started. The total frequency is 1 x 10^{-6} per reactor-year, which includes a frequency of excursion of 3 x 10^{-7}.

The third highest frequency scenario, 14 and 15, starts with a small break LOCA, but one train of HPI fails. Natural circulation is lost, the steam generators fill with deborated coolant, and then the inoperable HPI train is recovered. The frequency of this scenario is 1 x 10^{-7}, which includes a frequency of excursion of 2 x 10^{-8}.

Description of Sequence (Westinghouse design)

The Westinghouse design differs significantly from the B&W design, and the thermal-hydraulic effects can be affected. The design is shown in Figures 6 and 7 (from NUREG/CR-5640[1759]).

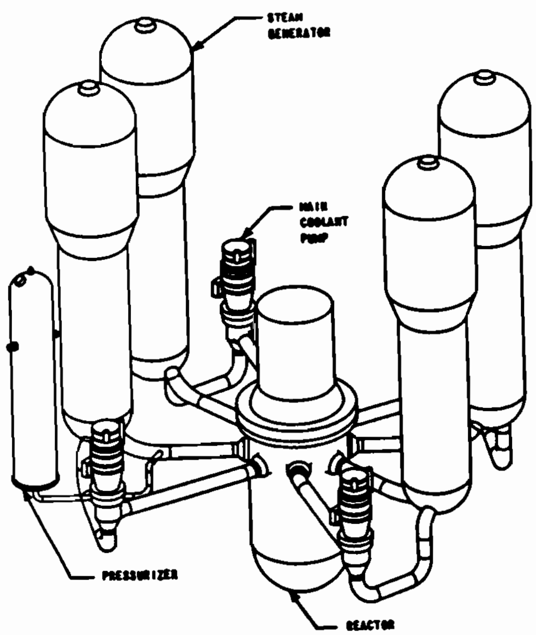

Figure 6: Westinghouse NSSS

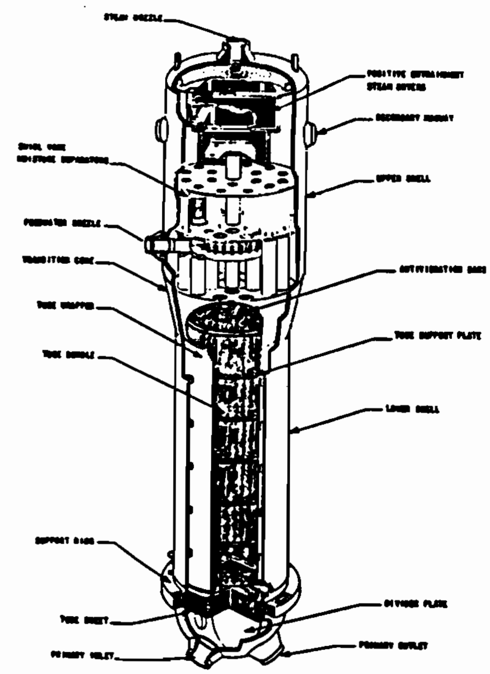
Figure 7: U-Tube Steam Generator

First, the steam generators are of the U-tube design, and these tubes are completely submerged in liquid water on the secondary side. After a small LOCA, as coolant is lost out of the break, the pressurizer will empty, pressure will drop, and voids will form in the core area.

Unlike the situation in the B&W design, where the voids will naturally collect and form a vapor space at the top of the hot leg, voids will be carried into the ascending half of the U-tubes and condense back into the liquid phase. As pressure and coolant inventory continue to drop, a greater fraction of the volume above the core and in the hot legs will be in the vapor phase. It is likely that re-condensed (and deborated) coolant will first flow back down the ascending half of the U-tubes and run down on the lower surfaces of the pipes back down to the upper plenum of the reactor, where it will mix rapidly with the more concentrated, turbulently boiling coolant just above the core. As more inventory is lost, eventually a state will be reached where the primary system is at saturation pressure, coolant in the vapor phase condenses in the steam generators, and at least some of the condensed, deborated coolant collects in the descending half of the U-tubes, and the outlet plena, cold legs, pump volume, and, eventually, the lower plenum of the reactor vessel.

Second, unlike the B&W "lowered loop" design, the steam generators are located at a higher elevation than the top of the reactor core. In this design, as the coolant level in the primary system drops, it will be more difficult for deborated coolant to remain in the steam generators. In contrast to this, in the B&W lowered loop design, the coolant level can drop to the top of the active core, and there will still be some deborated coolant in the steam generators.

Third, the available volume in the steam generators is somewhat less. The total volume of coolant in the reactor vessel is 4333 cubic feet (from the RELAP model for this plant), while the primary side of a "Model F" steam generator is 962 cubic feet.[1759] The total primary volume of the four steam generators is thus about 90% of the reactor volume. However, because of the U-tube design of the steam generators, it is not clear that the entire primary volume of the steam generators will fill with deborated coolant. If only the descending portion of the tubes are filled, the total liquid inventory in the steam generators will be only 45% of the reactor volume, and it is not clear that, should natural circulation be restored, the core area will be flooded temporarily with deborated coolant. Conversely, the reactor downcomer and lower plenum volumes may slowly fill with unmixed, deborated coolant, as was discussed earlier, and this would be a sufficient volume to sweep the dissolved boron out of the core region. Thus, for this design, there is even more uncertainty regarding the credibility of this scenario than in the B&W example discussed previously. However, some experimental work at a test facility at the University of Maryland strongly suggests that the deborated coolant will sweep through the primary system as a "slug," with relatively little mixing. Again, consistent with the "worst case" guidelines of the prioritization process, it will be assumed that the accumulation of deborated coolant will occur.

<u>Event Tree:</u> The event tree structure is essentially unchanged, but the values of certain split fractions must be changed because of the differences in the various systems. The Seabrook plant was chosen for analysis, again because of the ready availability of design information and the existence of a RELAP model.

Small Break LOCA: As before, the NUREG-1150[1081] S2 frequency of 10^{-3} per reactor-year will be used.

Probability of maintaining natural circulation: Seabrook is equipped with three charging pumps, two of which are centrifugal, and one of which is a positive displacement pump.[1759] In addition, the plant is equipped with a two-train high-pressure safety injection (HPSI) system. The two HPSI pumps are centrifugal pumps, but have a shutoff head close to the saturation pressure of the primary system; they cannot inject at operating pressure. Pump capacities are as follows:

Pump type	Flow at 1750 psi[1759]	Flow at PORV setpoint[1759]
Charging, Centrifugal (2)	(unknown)	150 gpm (each)
Charging, Positive Displacement	98 gpm	98 gpm
HPSI, Centrifugal (2)	425 gpm (each)	zero

For current purposes, we will largely neglect the positive displacement pump, because of its low capacity. The flow near saturation pressure for the two centrifugal charging pumps is not given in NUREG/CR-5640[1759]. However, the SPAR-2QA model[1761] event tree for small break LOCA has, as success criteria, either of the two HPSI pumps, or both of the two centrifugal charging pumps. Thus, the two charging pumps will be treated together as if they were a third HPSI train, with a combined flow of 425 gpm.

Split fractions can now be calculated, using the same assumptions as before:

Number of pumps	Flow at 1750 psi	Fraction of 100-1500 gpm "G3" spectrum covered	Probability of loss of natural circulation
1	425 gpm	23.2%	76%
2	850 gpm	53.6%	46%
3	1275 gpm	83.9%	16%

Number of HPSI "trains:" The SPAR-2QA model's HPSI fault tree for this plant[1761] is much more tractable than that of the B&W plant. From the SPAR-2QA model for this plant, calculations of the three total system and the individual trains gave the following results:

Probability of failure of:	Parameter in SPAR-2QA model[1761]	Value
Entire HPSI system, including charging pumps	HPI	1.096E-5
Two centrifugal charging pump trains	CHV-SYS-F	8.77E-3
Both HPSI trains (including common cause failures)	HPI-TRAINS-F	1.624E-5
One HPSI train	HPI-TRAINA-F or HPI-TRAINB-F	4.030E-3

Again, the numbers above do not have four significant figure accuracy. The extra digits are given for the convenience of the reader who wishes to repeat the calculation. The probability of a certain number of trains operating, P(n), is then:

Probability of n trains operating	Parameters in SPAR-2QA model[1761]		Value
P(0)	HPI		1.096E-5
P(1)	(HPI-TRAINS-F)(1-CHV-SYS-F) + [(HPI-TRAINA-F)(CHV-SYS-F)](1-HPI-TRAINB-F) + [(HPI-TRAINB-F)(CHV-SYS-F)](1-HPI-TRAINA-F)	1.61E-5 + 3.52E-5 + 3.52E-5	8.65E-5
P(2)	HPI-TRAINA-F + HPI-TRAINB-F + CHV-SYS-F	4.03E-3 + 4.03E-3 + 8.77E-3	1.683E-2
P(3)	1 - P(0) - P(1) - P(2)		0.983

Recover HPSI: The Seabrook SPAR-2QA model will be used.[1761] This parameter, designated "SLOCA-XHE-NOREC" indicates a 43% probability of non-recovery, implying a recovery probability of 57%.

Re-start reactor coolant pumps (RCPs): As in the B&W case, a probability of 10% will be used, based purely on judgment.

Recover Natural Circulation: As in the B&W case, the operator may be able to recover natural circulation by isolating the break, using the positive displacement charging pump, or blowing down a steam generator. Based on judgment, 50% will again be used for this parameter.

Core State: The boron letdown curve for the Seabrook core (fairly typical of a Westinghouse "low leakage" design, and plotted versus burnup in megawatt-days per metric ton of uranium instead of days in the cycle) is shown in Figure 8 (from the Seabrook updated FSAR). As can be seen by comparing this curve with the B&W curve shown earlier, there are some marked differences. First, it should be noted that the licensee did not include the xenon and samarium build-in at the very beginning of the cycle, and thus the curve does not begin at zero burnup. Second, the full power boron concentration actually increases slightly at the beginning of the cycle, then decreases slowly, eventually becoming linear for the latter portion of the cycle until it becomes zero at the end of the cycle (17 GWD/MTU). This is due to the burnable poison loading, which is typically higher in Westinghouse cores.

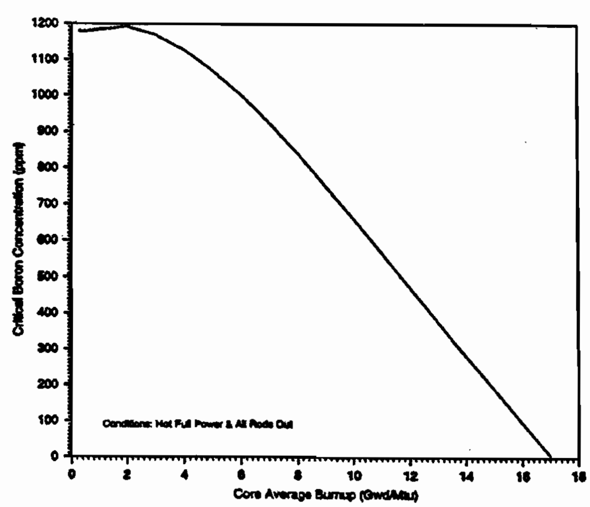

Figure 8: Westinghouse Boron Letdown

This curve was digitized and combined with other information in the Seabrook FSAR to produce a plot of boron worth and control rod worth over the cycle (with the xenon buildup added at the beginning of the cycle. For this core design, it is possible to achieve criticality for about 36 percent of the cycle, almost five times the 7.7% figure for the B&W core.

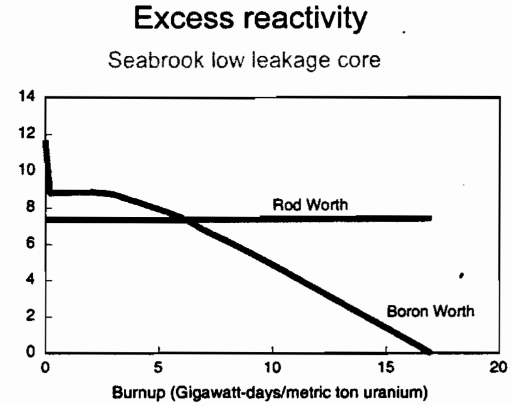

Figure 9: Excess Reactivity vs. Burnup

As before, criticality does not automatically equate to severe core damage. The Seabrook plant is equipped with two AFW trains, one motor-driven and one turbine-driven, and each capable of supplying 710 gpm at a secondary side pressure of 1322 psi.[1759] This is somewhat less than the capacity of the Crystal River plant's AFW, and the rated thermal power of the Seabrook reactor core is actually greater than that of Crystal River. A rough calculation similar to the one done for the B&W design indicates that the auxiliary feedwater supply is capable of removing about 4.8% of rated thermal power per AFW train. If both trains are operating, allowing 2% of rated power for decay heat removal, and assuming the fission heat pulses with a 50% duty cycle, the AFW system can accommodate fission power of about 15 percent of rated - significantly less than that of the B&W design. However, unlike the B&W design, the Westinghouse steam generators are likely to contain a significant inventory of secondary coolant, completely submerging the tubes on the secondary side, and are far less likely to dry out before the power pulses in the primary side die out due to boron mixing in the primary. There is no easy way to estimate this effect quantitatively. However, the probability of damage is not a very strong function of the power level assumed to be the threshold of severe fuel damage. Using the digitized curves, the following estimates were made:

Fuel damage assumption	Percentage of fuel cycle
Fuel melts at criticality	36%
Fuel melts at AFW limit (15% power)	33%
Fuel melts at 50% power	25%
Fuel melts at 100% power	15%

It is difficult to believe that a 100% power pulse would not result in damage. It is even more difficult to believe that a subcritical core would sustain any damage. The extreme range in damage threshold only leads to a range of 15% to 36% in the probability of severe core damage, given a boron dilution event. It will be assumed, based purely on judgment, that severe core damage will result at 50% of rated power.

Regarding prompt criticality, a hand calculation indicates this to be possible only during the time of xenon buildup - about 1 percent of the fuel cycle. Once equilibrium is achieved, the burnable poison loading is such that the excess reactivity curve is relatively flat, and does not rise sufficiently above the shutdown rod worth to permit a prompt criticality event.

The digitized boron curve was used to calculate the probabilities of the various branches:

	Probability of prompt criticality	Probability of overpower	Probability of criticality, low power	Probability of no criticality
Slow reactivity insertion	1%	24%	11%	64%
Fast reactivity insertion	1%	24%	11%	64%

Results: The results of the event tree calculation for this Westinghouse design were a frequency of core damage of 2.2 x 10^{-5} events per reactor-year, of which 1 x 10^{-6} events per reactor-year involved a reactivity excursion.

As in the B&W case, the highest frequency scenario corresponded to sequences 8 and 9 on the event tree. This scenario is initiated by a small break LOCA, all HPSI trains operate, but flow is not sufficient to maintain natural circulation. The RCPs are not re-started, but natural circulation re-starts after the steam generators fill with deborated coolant. The frequency of a reactivity excursion is 7 x 10^{-7} per reactor-year, and the frequency of severe core damage is an additional 2 x 10^{-5}, per reactor-year.

The second highest frequency scenario, corresponding to sequences 4 and 5, is similar, but instead of recovering natural circulation, the reactor coolant pumps are re-started. The total frequency is 4 x 10^{-6} per reactor-year, which includes a frequency of excursion of 2 x 10^{-7} per reactor-year.

The third highest frequency scenario, corresponding to sequences 14 and 15, starts with a small break LOCA, but one train of HPSI fails. Natural circulation is lost, the steam generators fill with deborated coolant, and then the inoperable HPSI train is recovered. The frequency of this scenario is 1 x 10^{-6} per reactor-year, which includes a frequency of excursion of 4 x 10^{-8} per reactor-year.

Discussion

The core damage frequency results are quite similar for both designs. This is not too surprising, in that the same event tree was used for both, and many of the split fractions were the same. Results for 2-loop or 3-loop Westinghouse designs, or a Combustion Engineering design, are not likely to be greatly different. The Westinghouse core damage frequencies are about a factor of four higher than that estimated for the B&W design. This appears to be primarily due to the higher burnable poison loading in the Westinghouse core, which causes the core to have a potential for criticality for almost five times as long a fraction of the fuel cycle. There is, however, somewhat less uncertainty in the thermal-hydraulic effects in the B&W design.

The nature of the highest frequency scenarios suggest that a procedural fix may be appropriate for this issue. All three scenarios involve natural circulation re-starting due to actions taken by the operators, re-starting the reactor coolant pumps, or recovering a train of high pressure injection.

Consequences

To estimate consequences and risk, the standard analysis described in the introduction to NUREG-0933 was used, i.e, the WASH-1400 release categories[16] and a generic site. For the portion of the core damage frequency associated with overpower damage to the fuel, the spectrum of consequences across the seven PWR release categories for the S2 LOCA in WASH-1400 was re-normalized to this issues core damage frequency. For the reactivity excursions, the entire event frequency was put into the PWR-1 release category, consistent with the worst case assumption discussed earlier. The results were as follows:

Release category	1	2	3	4	5	6	7	Total
WASH-1400 spectrum of release categories[16]								
WASH-1400 S2 frequencies	1.0e-07	3.0e-07	3.0e-06	3.0e-07	3.0e-07	2.0e-06	2.0e-05	2.6e-05
WASH-1400 normalized frequencies	0.38%	1.15%	11.54%	1.15%	1.15%	7.69%	76.92%	100.00%
Westinghouse design:								
Frequencies, overpower sequences	8.1e-08	2.4e-07	2.4e-06	2.4e-07	2.4e-07	1.6e-06	1.6e-05	2.1e-05
Excursion event frequency	1.0e-06							1.0e-06
Sum	1.1e-06	2.4e-07	2.4e-06	2.4e-07	2.4e-07	1.6e-06	1.6e-05	2.2e-05
Release category consequences (person-rem)	5.4e+06	4.8e+06	5.4e+06	2.7e+06	1.0e+06	1.5e+05	2.3e+03	
Risk (person-rem per reactor-year)	5.8e+00	1.2e+00	1.3e+01	6.5e-01	2.4e-01	2.4e-01	3.7e-02	2.1e+01
B&W design:								
Frequencies, overpower sequences	1.8e-08	5.5e-08	5.5e-07	5.5e-08	5.5e-08	3.7e-07	3.7e-06	4.8e-06
Excursion event frequency	9.0e-07							9.0e-07
Sum	9.2e-07	5.5e-08	5.5e-07	5.5e-08	5.5e-08	3.7e-07	3.7e-06	5.7e-06
Release category consequences (person-rem)	5.4e+06	4.8e+06	5.4e+06	2.7e+06	1.0e+06	1.5e+05	2.3e+03	
Risk (person-rem per reactor-year)	5.0e+00	2.7e-01	3.0e+00	1.5e-01	5.5e-02	5.5e-02	8.5e-03	8.5e+00

The net risk associated with this issue is thus estimated to be 8.5 person-rem per reactor-year for the B&W design, and 21 person-rem per reactor-year for the Westinghouse and CE designs.

As of the beginning of the year 2000, the net benefit of this issue is estimated as follows:

	Number of plants	Remaining aggregate life (reactor-years)	person-rem per reactor-year	Risk benefit (person-rem)
B&W	10	190	8.5	1615
Westinghouse	54	1100	21	23100
CE	15	300	21	6300
			Total:	31015

Thus, the total risk benefit is estimated to be 31000 person-rem. This does not include the effect of license renewal, which would increase the number significantly.

Cost Estimate

Industry Cost: The cost to a licensee would be the cost of writing and putting in place a complex change in emergency procedures. According to Table 4.1 of NUREG/CR-4627,[961] such a change would cost $3420 to $4350, with a point estimate of $3900. This complex procedure may well be an above-average cost, and therefore the upper limit of $4350 will be used. For approximately 80 PWRs, this works out to a total licensee cost of $348000.

NRC Cost: The cost to the NRC would be significant, since considerable work would need to be done to resolve the thermal-hydraulic uncertainties, plus all of the administrative effort involved in any type of regulatory action. Based purely on judgment, a cost of two million dollars will be assumed.

Total Cost: Total cost is then estimated to be on the order of $2,400,000, which is dominated by the cost of confirmatory thermal-hydraulic research.

Impact/Value Assessment

The cost-benefit ratio for this issue is estimated to be $2,400,000/31000 person-rem, or roughly 80 dollars per person-rem - well into the cost-beneficial range.

Other Considerations

(1) Because the contemplated fix is procedural in nature, there are no implications for increased occupational radiation exposure to plant workers.

(2) Because the issue is well into the cost-beneficial range, avoided offsite costs of a potential accident have not been estimated. Inclusion of these costs would not change the conclusion.

(3) License Renewal: Assuming a license renewal period for 79 plants, the public risk reduction would be approximately doubled, to 60,000 person-rem.

Uncertainties:

The calculations presented above are point estimates only. The Rev. 2QA SPAR models from which many of the parameters were taken do not include uncertainty distributions. Moreover, some of the parameters were based only on judgment. Thus, a standard PRA uncertainly analysis is not presently feasible.

Nevertheless, it is appropriate to point out several limitations of this analysis:

- The estimates of the fraction of the fuel cycle during which the core can be brought to a critical state with all control rods inserted are based on hand calculations performed on FSAR data. These calculations are very primitive, core nuclear design parameters may differ for each fuel cycle, and the two estimates of this fraction, 7.7% for the B&W core and 36% for the Westinghouse core, can vary. However, it is doubtful that these fractions will vary by orders of magnitude, which would be necessary to change the conclusion.

- The xenon reactivity transient was included only as a window effect. In reality, the xenon transient will become steadily more important as core burnup increases, and the "window" of time after shutdown during which it is possible to achieve criticality will steadily decrease.

- Conversely, the fact that the xenon will eventually decay away has not been included. The assumption was made that, by the time the xenon transient turned around, the operators would have taken appropriate corrective action. This "delayed criticality" effect is, in reality, still another accident scenario which should be incorporated into the resolution of this issue.

- The options available to the operator to refill the primary system (and thereby recover natural circulation) are plant-specific. In the particular case of Crystal River, this analysis assumes that all three HPI trains will be started to mitigate the loss of coolant. However, only two trains start automatically on an SI signal. If the operator manually starts the third train at the beginning of the accident sequence, this will be a good approximation. However, if the operator delays starting the manual train, and then starts the third train after observing that the automatically-initiated trains have either failed or are not sufficient to maintain primary coolant inventory, this late start will actually increase the likelihood of a return to criticality.

- The core power level associated with the onset of severe fuel damage is, at best, an educated guess. If there is any high burnup fuel in the core, severe damage might occur as a result of even a relatively mild reactivity excursion. Conversely, the steam generators are sized to accommodate full power operation, and should be able to remove the integrated energy of a significant power pulse, limited primarily by the capacity of the AFW system and the capacity of the secondary side safety valves and ADVs.

- The actions of the operators are worthy of much more study, given the time windows involved in these scenarios and the lack of information on core reactivity. The plant operators would be faced with some confusing decisions about whether to restore failed trains, initiate forced circulation, etc.

- The thermal-hydraulic phenomena need further investigation. Although the estimate for this was 2 million dollars (roughly 10 staff-years), the investigation would be cost-effective even if this expense were much higher.

It should also be noted that, in its evaluation of the B&WOG PRA, NRR believed that the deborated water accumulation modeling, transport modeling, and reactivity analyses are highly approximate, incompletely understood, and subject to large uncertainties. Although the staff recognized these shortcomings, it expanded the B&WOG PRA to include approximations of additional variables and concluded that the fuel damage probability for natural circulation restart is probably between approximately 10^{-7} and 10^{-5}/reactor-year.[1730] This was completely independent of the analysis presented here, but nevertheless yielded similar results.

CONCLUSION

The core damage frequency change associated with the issue was estimated above to be 2.2×10^{-5} events per reactor-year, and the cost/benefit ratio is approximately 80 dollars per person-rem, for Westinghouse and CE plants. This class of PWRs dominates primarily because of a higher burnable poison loading, and consequently a longer fraction of the fuel cycle in which re-criticality is possible. The cost/benefit ratio is particularly favorable because the cost is low, and is likely to be dominated by NRC research costs.

Based on the current cost-benefit criteria (shown in Figure 1 of the introduction to NUREG-0933), this issue should be assigned a HIGH priority ranking.

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NUREG/CR-2798
ORNL/NSIC-208

Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants

Manuscript Completed: March 1982
Date Published: July 1982

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NRC FIN B0755

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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is sponsored by the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data. Support for the technical progress review *Nuclear Safety* (see last page of this report) is provided by both the Breeder Reactor and Light-Water Reactor Safety Programs of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of operational safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. The Center prepares reports and bibliographies as listed on the inside covers of this document. NSIC has developed a system of keywords to index the information it catalogs. The title, author, installation, abstract, and keywords for each document reviewed are recorded at the central computing facility in Oak Ridge.

Computer programs have been developed that enable NSIC to (1) prepare monthly reports with indexed summaries of Licensee Event Reports, (2) make retrospective searches of the stored references, and (3) produce topical indexed bibliographies. In addition, the Center Staff is available for consultation, and the document literature at NSIC offices is available for examination. NSIC reports (i.e., those with ORNL/NSIC and ORNL/NUREG/NSIC numbers) may be purchased from the National Technical Information Service (see inside front cover). All of the above services are available free of charge to U.S. Government organizations as well as their direct contractors. Persons interested in any of the services offered by NSIC should address inquiries to:

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PREFACE

The Nuclear Regulatory Commission (NRC) Division of Safety Technology in the Office of Nuclear Reactor Regulation assigned the project entitled *Special Studies of Reactor Operating Experience* to the Nuclear Safety Information Center (NSIC) in the early part of FY-1981. The object of this project was to identify safety significant implications of current nuclear power plant operating experience by special studies of the following specific subsystems: compressed air and backup nitrogen, service water, decay heat removal, and boron dilution.

About two to three man-months of engineering assessment was devoted to each of the studies. The information used was basically that found in NSIC's files. The documents containing this information are available to the public in the NRC Public Document Room, 1717 H Street, Washington, DC 20555. The scope of the project did not include visits to the plants or meetings with inspectors of the NRC Office of Inspection and Enforcement.

Project personnel for the studies were

NRC Cognizant Manager	M. L. Ernst
NRC Technical Manager	R. J. Colmar
NRC Cognizant Branch Chief	W. Minners
ORNL Program Director	A. L. Lotts
ORNL Program Manager	W. B. Cottrell
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ORNL Principal Investigator	W. R. Casto
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EVALUATION OF EVENTS INVOLVING UNPLANNED BORON DILUTIONS IN NUCLEAR POWER PLANTS

E. W. Hagen

ABSTRACT

This report reviews and evaluates events concerned with the inadvertent dilution of boron concentrations to the reactor coolant system for pressurized-water-cooled thermal reactors in commercial service. The safety concern is the unplanned addition of reactivity. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and regulatory documents. The results are collated and analyzed for significance and impact on power plant safety performance.

Several operating experience events were selected for analysis because they meet the criteria for safety significance. However, no boron dilution incidents resulted in a reactivity excursion or transient that scrammed a unit, nor was a reactor protection system challenged by any of the events. The most common cause for unplanned boron dilutions was human error, of which one was a common-mode/common-cause failure. For each recorded event, the operator had sufficient time to diagnose and correct the cause of the inadvertent dilution before the shutdown safety margin was lost or seriously challenged.

1. INTRODUCTION

The Safety Program Evaluation Branch (SPEB) in the Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) has the responsibility for evaluating reactor operating experience to detect events or trends that may be of safety significance to nuclear reactors so that these results may be factored into the licensing process. The SPEB commissioned this study of operating experiences involving the inadvertent dilution of boron concentration, particularly those experiences, occurring in the reactor coolant system (RCS) for pressurized-water-cooled thermal reactors (PWRs), that were outside the plant

Technical Specifications but including as well those within the Technical Specifications. The safety concern is the unplanned addition of reactivity. The purpose of this review was to identify and place in perspective any possible significant implications for reactor safety.

Computerized reference files of the Nuclear Safety Information Center (NSIC) [containing more than 24,500 Licensee Event Report (LER) descriptions plus abstracts of thousands of other operational and licensing documents] were systematically searched for those events associated with boron dilutions and impairment of boric acid delivery to the RCS. The computer selected and retrieved 555 references for 79 units in 49 plants from mid-1969 or unit initial operation (whichever was first) through early 1981. LERs by the electric power generating utilities were the major source of information for this study. However, other sources were also reviewed for information. The generic or vendor Safety Analysis Reports (SARs) were used to obtain background information, design, and operating philosophies. NRC guides, notices, and letters were reviewed for regulatory direction. The operating experiences were systematically compiled, categorized, and evaluated. This study analyzes the events involving inadvertent dilution of the RCS and of the boric acid supplies available to the RCS. The LERs used in this study are summarized in the Appendix.

Because boric acid is used as a reactivity control in PWR's, this study begins with a brief discourse on boric acid application; then the safety-related operating experiences are reviewed (i.e., those instances where less than the desired amount of boron was present in the RCS for that unit's operational mode). These instances constituted unwanted and unplanned boron dilution and thus the insertion of unanticipated reactivity. The safety relevance is noted, and a discussion of observaions and comments follows. Conclusions and recommendations conclude this study.

Boron, an absorber of thermal neutrons, is used to control excess reactivity. Controlled boron concentration in the RCS is used (1) to obtain optimum positioning of control element assemblies; (2) to compensate for reactivity changes associated with major changes in reactor coolant temperature between cold shutdown and hot full-power operation, fuel burnup, and burnable poisons buildup of fission products in the fuel; (3) to compensate for xenon variations; and (4) to provide a shutdown margin for maintenance and refueling operation. Boron concentration adjustment is a manual operation under strict administrative controls with procedures setting limits for the rate and duration of changes. Changes are made in the reactor coolant boron concentration for the following conditions:

1. Reactor startup — boron concentration must be decreased from shutdown concentration before taking the reactor to criticality.

2. Load change — boron concentration must be either increased or decreased to compensate for the xenon transient following a change in load.

3. Fuel burnup — boron concentration must be decreased to compensate for fuel burnup and the buildup of fission products in the fuel.

4. Cold shutdown — boron concentration must be increased to the cold shutdown concentration. Thus, boron in the form of the BO_3 ion in the reactor coolant controls reactivity.

The operator determines, by the use of nomographs or the in-plant computer, the desired boron concentration for the desired operating mode of the plant. He then either adds boric acid solution or primary water to the RCS as needed. These changes are made to the RCS through a process of feed and bleed using the chemical and volume control system (CVCS). Boron concentration in the RCS can be decreased (diluted) either by controlled addition of unborated makeup water with a corresponding removal of reactor coolant or by using the deborating ion exchangers. Controlled dilution has a purpose, that is, a preselected quantity of primary water is added at a preselected rate. Uncontrolled boron dilution is defined as the decrease in the RCS boron concentration caused by the inadvertent addition of unborated water. Uncontrolled boron dilution could result from equipment failure or human error.

Typically, the Final Safety Analysis Reports (FSARs) carry a statement, applicable only to the RCS makeup systems, that inadvertent or unplanned dilution of the reactor coolant due to the addition of unborated water can be terminated by (1) isolating the makeup water system, (2) stopping either the makeup water pumps or the charging pumps, or (3) closing the charging system isolation valves. A charging pump must be running in addition to a makeup water pump for boron dilution to take place. However, a review of the operating experiences associated with inadvertent or unplanned boron dilution in this report found that 81% of these events were not consistent with this FSAR statement. The SARs also state that the maximum possible rate of boron dilution is limited by design such that the operator has sufficient time to identify and terminate a boron dilution incident prior to serious loss of shutdown margin. Experience reports show this to be usually or conditionally true.

2. DESCRIPTION OF SYSTEM CONTROL

A boric acid blend system is provided to permit the operator to match the boron concentration of the reactor coolant makeup water to that in the RCS during normal charging of the RCS. The makeup and purification system normally has one pump in operation, which supplies makeup to the RCS and the required seal flow to the reactor coolant pumps. For makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump. Boric acid solutions can be supplied to the charging pump from (1) the boric acid (makeup storage) tank (BAT), (2) the refueling water storage tank (RWST), and (3) the volume control tank (VCT) and to the RCS from the boron injection tank (BIT) (Fig. 1).

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are activated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction. The signals initiating these alarms will also cause the closure of control valves, thus terminating the addition of boric acid to the RCS. Thus, the CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value that, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The controlled rate of addition of unborated makeup water to the RCS when it is not at pressure is limited by the capacity of the primary water makeup pumps. With the RCS at pressure, the maximum delivery rate is limited by the control valve. The highest rate of dilution can be handled by the automatic control system, which inserts rods to maintain the power level and the RCS system temperature. Control rod insertion to the predetermined limit will cause the feed block valve to close, terminating the addition of deborated water. If the reactor is under

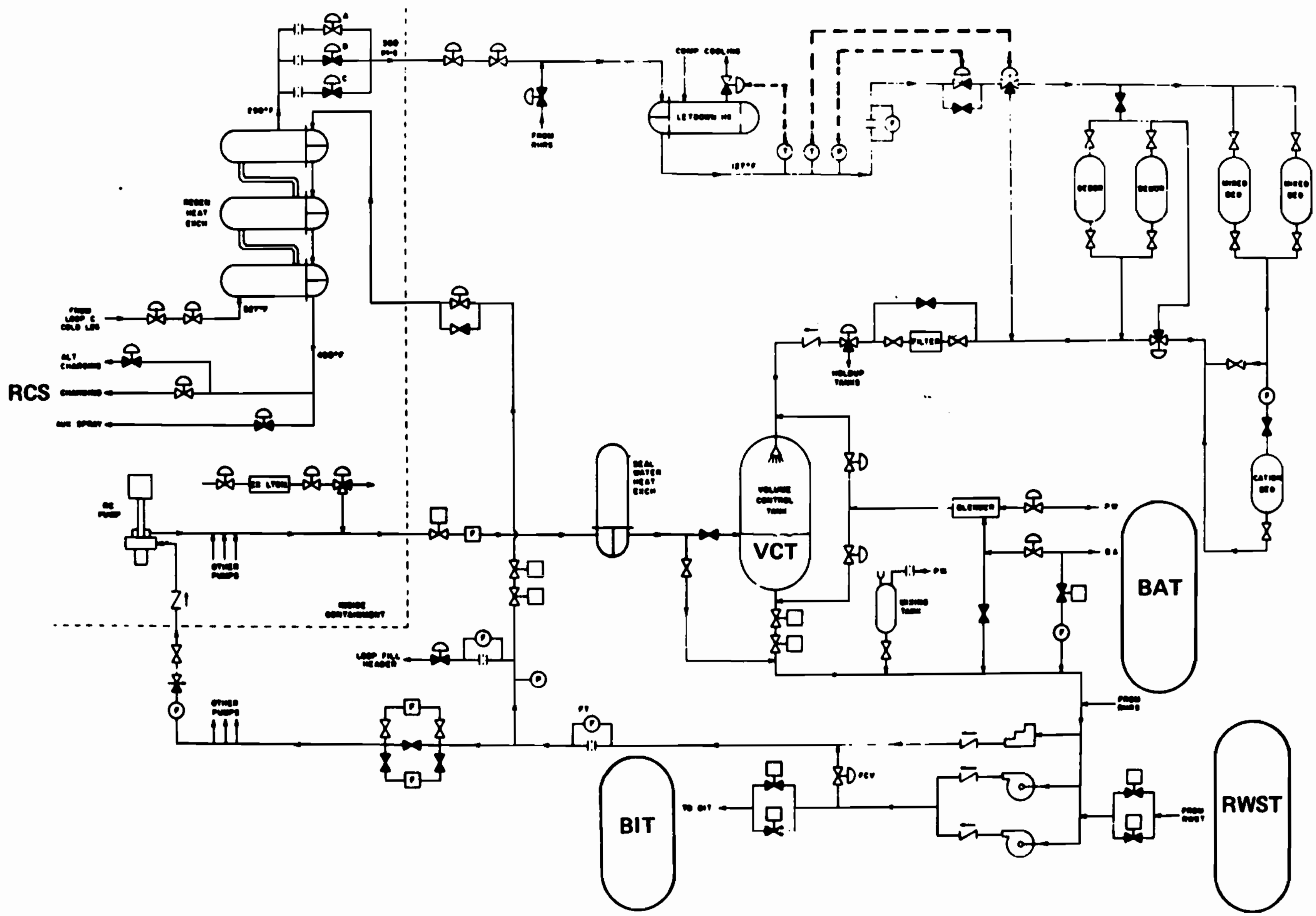

Fig. 1. Typical chemical and volume control system.

manual control with no control rod insertion, reactivity additions will raise power and cause a high-temperature or high-pressure trip.

3. TECHNICAL SPECIFICATIONS

Technical Specifications define plant variables, operating conditions, surveillance requirements, and administrative controls that are considered necessary to ensure the health and safety of the public. The scope of these specifications is set forth in the *Code of Federal Regulations*, Title 10, Section 50.36 (Ref. 1). Typical Technical Specifications concerned with the boric acid system are similar to the following:

1. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.
2. The reactor shall not be critical unless the following chemical and volume control system conditions are met.
   a. Two boric acid transfer pumps shall be operable.
   b. The boric acid tanks together shall contain a total minimum of (*) gallons of (*) weight percent boric acid solution at a temperature of at least (*)°F.
   c. System piping instrumentation, controls, and valves shall be operable to the extent of establishing one flow path from the boric acid tanks and one flow path from the refueling water storage tank to the RCS.
   d. Two channels of heat tracing shall be operable for the boric acid tank flow paths.
3. During power operation, the requirements of No. 2 may be modified to allow any one of the following conditions to exist at any one time. If the system is not restored to meet the requirements of No. 2 within the time period specified, the reactor shall be placed in the hot shutdown condition in (*) hours utilizing normal operating procedures. If the requirements of No. 2 are not satisfied within an additional (*) hours, a cold shutdown shall be initiated utilizing normal operating procedures.

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*Plant specific condition or parameter.

a. One out of two boric acid transfer pumps may be out of service provided that the pump is restored to operable status within 24 h.

b. One boric acid tank may be out of service provided a minimum of (*) gallons of (*) weight percent boric acid solution at a temperature of at least (*)°F is contained in the operable tank.

c. One channel of heat tracing may be out of service for (*) hours.

4. The quantity of boric acid in storage from either the boric acid tanks or the RWST shall be sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

Each time a Technical Specification is violated, that utility is required to submit to the NRC an LER describing the event, its cause, and the corrective action taken.[2] During the period under review, unplanned dilution of the RCS was reported 26 times and dilution of tanks containing solutions of boric acid occurred 58 times. From 1978 through 1980, 55 individual heat tracing failures occurred. Unit power reductions were recorded 18 times to comply with Technical Specifications: 2 for heat tracing failures, 2 for unplanned RCS dilution during power operation, and 14 for holding or storage tank dilutions.

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*Plant specific condition or parameter.

4. CATEGORIZATION OF EVENTS

A review of the reported operating experiences from June 1969 through January 1981 (Table 1) that related to unwanted dilutions of boric acid solutions determined that these events were caused either by equipment failures or by human errors. Twenty-six unwanted dilutions of the RCS occurred in 16 units at 12 PWR plants. This was an average of three unplanned dilution events of the RCS per year from 1973 through 1980 (Table 2) or $\sim$0.09 events per reactor year (i.e., one such event per PWR per every 11 years): of these 26 unwanted dilutions, 5 were caused by equipment failures and 21 by human errors.

Concurrently, 52 events of unwanted dilutions in makeup and storage systems of PWR plants occurred, for an average of about twice that of the inadvertent RCS dilutions. In each of three cases of inadvertent dilution, the initial corrective procedures implemented were apparently not effective. Repetitive events stemmed from two common causes, occurring a total of 14 times; these repetitions indicated a situation that could have been, and eventually was, improved. Even though unit and public safety were not jeopardized, plant availability and operational economics were degraded.

During this same review interval, only five unplanned boron dilutions occurred in boiling-water-reactor (BWR) standby liquid control systems at five plants. This yielded an average of 0.6 events per year or $\sim$0.03 events per reactor year (i.e., one event per BWR per every 33 years).

4.1 Dilutions in PWR Reactor Coolant System

Equipment failures. Five equipment failures resulted in unexpected RCS dilutions (see Table 1). Three of the failures were random and of the type expected during normal service life: (a) an internal set point error in a boric acid controller, (b) a failed capacitor in the primary-grade water valve flow controller for the blender, and (c) a cut seat and bent stem on an acid pump discharge relief valve. (A review of equipment performance for normal and emergency boration systems is under

Table 1. Distribution of RCS dilution events

Unit	LER No. or date	Equipment failures		Human errors		
		Steam generator	Miscellaneous	Operator	Maintenance	Procedure
Calvert Cliffs 1	78-9					X
Crystal River 3	77-8			X		
Crystal River 3	77-12[a]			X		
Ginna	75-6			X		
Ginna	74-1				X	
Indian Point 1	6/5/69				X	
Indian Point 2	2/8/74					X
Millstone 2	78-5					X
Point Beach 1	79-19				X	
Point Beach 1	6/8/74		X			
Oconee 1[b]	74-6[a]					X
Oconee 1	7/17/73					X
Oconee 2	80-3[a]			X		
San Onofre 1	80-29	X				
San Onofre 1	80-34	X				
Surry 1	75-1					X
Surry 1	5/12/80			X		
Surry 2	11/26/77					X
Surry 2	78-12		X			
Surry 2	76-3				X	
St. Lucie 1	79-15				X	
St. Lucie 1[b]	80-71		X			
Trojan	80-10				X	
Zion 1	76-62					X
Zion 2	77-9[a]					X
Zion 2	5/16/74			X		

[a] Repeated events.

[b] Forced reduction in power.

Table 2. Number of unplanned RCS dilutions by year

Year	Number of dilutions: RCS	SLCS[a]	Storage tanks
1969	1	0	0
1970	0	0	0
1971	0	0	0
1972	0	0	1
1973	1	2	3
1974	5	3	10
1975	2	0	13
1976	2	0	9
1977	4	0	4
1978	3	0	1
1979	2	0	0
1980	6	0	5
1981	0	0	6

[a]Standby liquid control system.

way for the NRC's NRR Division of Risk Analysis as a part of the Light-Water-Reactor Systems Survey.) The other two equipment failures were caused by leaking tubes or faulty maintenance seals in steam generators.

Other less frequent disturbances to the boric acid system concerned the interruption of electric power to pumps and some valves and failures of the heat tracing on some lines. These were all immediately recognized and fixed. Of the 55 reported heat tracing interruptions from 1978 through 1980, only in three reported instances was redundancy of heat tracing lost; in one of these a shutdown was required, and in another a power reduction was necessary. During the period November 1974 to February 1977, a number of cracking incidents were experienced in safety-related stainless steel piping systems, one of which involved cracks in piping containing stagnant or essentially stagnant borated water. This potential failure was recognized by the NRC in their bulletin No. 79-17 (Ref. 3). In 1980, another NRC letter informed of RCS dilution problems during steam generator decontamination.[4] A recent study by NRC Office of Analysis and Evaluation of Operational Data reaffirmed that problems caused by cold-weather freezing of instrumentation lines (boron for one) are still being reported despite the Inspection and Enforcement Bulletin of September 27, 1979 (Ref. 5).

Human error. Twenty-one inadvertent boron dilution events of the RCS were attributed to human errors (i.e., 80% of the boron dilution events). Six of these involved steam generator maintenance such as (a) omission of procedural steps (not plugging sectioned tubes), (b) miscalculations (the amount of excess boron needed for tube header washing and the decision as to when to open a manway for maintenance action), and (c) execution of prescribed actions but choosing either an inadequate seal plug during a maintenance action or the incorrect use of tools (accidentally slicing or nicking other tubes during the process of removing or sectioning specific tubes). Nine other events were concerned with operating procedures and were caused by misinterpretations, omissions, or inadequate surveillance. Six other events were associated with the general problem of commission (e.g., when executing a prescribed course of action, the operator selected the wrong controls or valves).

System operation, as opposed to equipment/system automation, is largely dependent on the human touch. Therefore, in operation, performance, and maintenance, the high rate of inadvertent boron dilutions of the RCS attributed to human errors should come as no surprise.

4.2 Dilutions in PWR Makeup and Storage Systems

Boric acid is also used in PWRs to provide shutdown margin. As such, sufficient quantities of boric acid in proper concentrations must always be available. Maintaining such a supply and monitoring the concentration of boron is a necessary routine for fueled units. On 52 occasions this routine was broken by an inadvertent or unplanned dilution. The disruptions generally caused a Technical Specification violation; of those violations, 54% were due to equipment failures and 46% to human errors (i.e., operational, procedural, or administrative) (Table 3). The three reports that follow mentioned that the events were repeats of previous occurrences.

1. Turkey Point 4 (Ref. 6). On three occasions, the boron concentration in the boron injection tank dropped below the 20,000-ppm Technical Specification limit. A unit shutdown was initiated each time, and recirculation of the boric acid storage tanks was initiated. The low concentration resulted from minor inleakage and insufficient recirculation. Through recirculation, the boron concentration was returned to within allowable limits prior to realizing cold shutdown, and the reactor was returned to normal power operation.

2. Salem 1 (Ref. 7). While the plant was in mode 3, the boron injection tank was declared inoperable on seven occasions during a 1-month period because of inadvertent safety injection actuations that reduced the boron concentration below the minimum allowed in the Technical Specifications. Appropriate procedures were implemented and plant conditions were returned to normal. At no time was there any hazard to the general public or site personnel.

3. Sequoyah 1 (Ref. 8). On four occasions over a 10-month interval, the boron concentrations in the refueling water storage tank were discovered to be below the 20,000-ppm Technical Specification limit.

Failure to maintain proper boron concentration was caused by the addition of insufficient concentrated boron during tank refilling. Unreliable analytical results were due to failure to follow proper sampling procedures. Each time (per licensing requirements), the unit was placed into a status of limited operating conditions. However, there was no effect on public health or safety.

4.3 Dilutions in BWR Liquid Control Systems

LERs that occurred during the period under review were examined, and inadvertent dilution of the standby liquid control system, a secondary BWR shutdown system, occurred only at five BWRs, all during 1973/74. The unintentional dilutions were caused by operator error. On one of these occasions, the dilution decreased the boron concentration from 14.7 to 7.4%, and the unit was shut down until the proper concentration could be restored. Apparently the remedial actions were effective because there have been no such reported events in the last 7 years. These five events are also listed in Table 3.

Table 3. Low concentration of boric acid in tanks (reduced below Technical Specifications)

Unit	Docket No./LER No.	Date	Reactor power level (%)	Refueling water storage tank	Boron injection tank and core flood tank	Standby liquid control system storage tank	Boric acid storage tank	Boric acid mix tank	Cause	Remarks
Ft. Calhoun 1	285/76-23	8/4/76	96[a]	X					Equipment	Valve leakage increased level
Trojan	344/76-33	5/13/76	42[a]		X				Operator	Makeup due to routine sampling
Pilgrim	293/	10/10/74				X			Operator	Opened water valve instead of air sparging valve
Turkey Point 4	251/	9/23/74					X		Maintenance	Flush and isolation valves left partially open
Sequoyah 1	327/81-037[b]	4/10/81	0	X					Chemical operator	Failure to follow sampling procedures
North Anna 2	339/81-021	3/2/81	100		X				Equipment	Leaky check valve
Beaver Valley 1	334/80-091R	1/29/81	0		X				Operator	Insufficient mixing before sampling
Surry 2	281/80-013	9/12/80	0				X		Operator	Sampling procedures not followed
Oconee 2	270/80-005	7/2/80	0[a]		X					Improper mixing prior to sampling during filling
Oconee 1	269/80-017	7/11/80	73				X		Maintenance	Pump removed from service (68/IVS)
Salem 1	272/76-31/3L[b]	1/28/77	0		X					Inadvertent safety injection reduced concentration
Salem 1	272/76-26/3L	1/14/77	17		X					Inadvertent safety injection reduced concentration
Robinson 2	261/76-3	2/9/76			X				Equipment	Check valve leak
Cook 1	315/75-59	9/22/75	80		X					Boron crystallization due to heater failure
Cook 1	315/75-61	9/29/75	81		X					Flow blockage due to heat loss
Palisades	255/75-22	9/28/75	40		X					Poor mixing when tank was topped off
Surry 1	280/75-14	8/12/75	0[a]		X					Inleakage through check valves
Surry 2	281/75-14	8/8/75	100[a]		X				Equipment	Inlet valve leakage
Surry 1	280/75-81	8/12/75	0[a]		X					Check valve leakage
Zion 2	304/74-59	6/25/75					X			Primary water added to tank
Rancho Seco	312/75-9	5/13/75	c				X		Operator	Tank not refilled after use and before critical operation
Indian Point 2	247/	5/5/75	a	X	X					Low level from filling BIT
Zion 2	304/75-17	4/11/75	84				X		Operator	Makeup not of proper concentration
Zion 2	304/75-6	2/5/75					X			Valve leakage
Ginna	244/74-16	9/30/74			X					Valve leakage
Oconee 1	269/74-12	8/22/74			X					Unknown
Zion 1	295/74-40	9/4/74	42				X			Unknown, possible dilution through recirculation
Zion 2	304/74-35	9/3/74	4				X			Valve leakage
Zion 1	295/74-32	8/8/74	a				X			Improper valve lineup, procedures
Oconee 2	269/	7/29/74	80[a]				X			Demineralizer valve leakage
Surry 2	281/	7/12/74	50		X					Leaking valves
Turkey Point 4	251/[b]	7/5/74	a		X					Minor inleakage and insufficient recirculation
Turkey Point 3	250/	4/19/74			X					Unknown, possibly due to injection testing
Peach Bottom 2	277/74-4	2/27/74				X				Testing procedure
Cooper	298/74-2	2/7/74	0			X				Faulty chemical analysis
Point Beach 1	266/	11/3/73					X		Operator	Failure to sample transfer
Maine Yankee	309/73-6	11/6/73		X						Results from testing
Turkey Point 4	251/73-11	9/73			X					Primary water leak into suction of boric acid pump
Monticello	263/	6/4/73				X				Procedures for water addition
Dresden 3	249/	1/9/73	a			X				Demineralizer water valve and air sparging valves open
Maine Yankee	309/	12/15/72								Check valve leakage
Maine Yankee	309/80-017	9/19/80	65	X						Improper sampling procedure
Trojan	344/80-016	8/18/80	3				X		Operator	Procedures not followed
North Anna 1	338/78-13	4/25/78	0	X						Poor sampling
Zion 2	304/77-14	4/29/77	50		X					Back leakage through valves
Zion 2	304/77-15	4/29/77	48		X					Back leakage through valves
Salem 1	272/76-01	9/3/76	0				X			Charging pump discharge valve leaking
Ft. Calhoun 1	285/76-23	8/4/76	96[a]		X					Valve leaking
Turkey Point 4	250/76-4	6/25/76	100		X					Inleakage through isolation valve
Turkey Point 3	251/76-4	6/25/76	100		X					Inleakage through isolation valve
Turkey Point 4	251/76-5	6/25/76	100		X					Inleakage through isolation valve
Cook 1	315/76-26	6/15/76	75		X		X			Dilution from flushing lines
Indian Point 2	247/75-2	12/12/75	80[a]				X		Operator	Sampling inaccuracies and excessive flushing water
Turkey Point 4	251/	1/30/75	100[a]				X			Dilution during flushing
Beaver Valley 1	334/81-034	5/7/81	0				X			Procedural deficiency
Cook 1	315/81-001	2/13/81	100		X		X			Unknown
Surry 2	281/81-006	2/13/81	100		X					Leakage through check valve

[a]Forced reduction in power.

[b]Repeated events.

[c]Not given.

5. EVENTS OF POTENTIAL SIGNIFICANCE

The significance of all the reported events relating to unplanned boron dilution was evaluated, and the events were listed as to their safety relevance. The top six events are briefly described. These events resulted from either a common-mode failure, a control system failure that degraded a safety system operation, or a recurring single failure.

Human factor deficiencies in tagging, labeling, and both operational and administrative procedures appear to be the causative factors for the recurrent events. There were seven recurring, single failure events, four of which constituted 27% of all unplanned RCS boron dilution incidents.

Human errors were the causative factors in 81% of the RCS dilution incidents and in 46 of the unplanned dilutions of the boron concentrations in the holding and storage tanks (see Chap. 4).

1. At Oconee 1, a common-mode failure in boron analysis occurred during physics testing.[9] A sodium hydroxide solution used for titration in the determination of boron concentration in the RCS had been improperly evaluated as to normality. Therefore, for 3 d, during zero-power physics testing, the boron concentration determinations for the RCS were in error, and criticality was attained at other than the extrapolated value. The normality of the sodium hydroxide changed over a period of several months due to an absorption of carbon dioxide. This incident was an example of a common-mode failure. "Although the absolute boron concentrations were not accurately known, plant performance was accurately predicted and controlled using relative concentrations."[9]

The following four-step procedure[9] was inaugurated to prevent recurrence: (1) Fresh batches of standard sodium hydroxide solutions are to be prepared weekly for use in titrating boron samples. Quantities are to be kept small so that new solutions must be prepared. (2) A boron standard is checked in duplicate twice each day. (3) Each sample is run in duplicate twice each day. (4) Each sample is run in duplicate on a boron titrator. (5) A meter for measuring boron concentrations is used to provide an independent check of lab results.

2. Four-day system malfunction caused dilution of the RCS at Surry 1 (Ref. 10). Prior to establishing containment integrity in preparation for startup, the RCS boron concentration was found to be low. Intermittent unknown dilutions, which had been occurring over a period of 4 d, were discovered by routine chemistry analysis of the RCS. The primary-grade water flow controller was found to be in the manual mode of operation with a demand position corresponding to full open position for the water valve. Therefore, whenever the level controller on the VCT demanded makeup, full instead of proportional water flow was fed to the acid blender, then to the VCT, and on to the RCS.

This flow controller, which is normally kept in the automatic operating mode, will change to the manual mode on a loss of electric power. A temporary loss of power to the vital bus had been experienced 4 d earlier. Lights that indicate the status of the controller were burned out and therefore could not alert the operator to the fact that the water flow controller was in the manual mode of control.

The RCS dilution was not discovered by its effect on the source range detectors because of the large magnitude of the prevailing shutdown margin. Had the dilution continued indefinitely, the RCS boron concentration would have leveled off and the required shutdown margin would be maintained. All operations personnel were instructed to maintain the boron concentration during shutdown periods to ensure that integrity is set before dilutions are begun. The operators were also reinstructed to scrutinize all indicating lights periodically to detect system malfunctions.

3. At St. Lucie 1, failure of the acid pump relief valve caused dilution of the RCS (Ref. 11). Boric acid was diverted back to the acid makeup tank instead of being pumped into the VCT when the acid pump relief valve failed during unit power operation. This resulted in dilution of the VCT and therefore dilution of the RCS.

The plant is designed to safely accommodate this transient, but the event is significant because of the misleading indications given. Because the relief valve is downstream from the boric acid flow element, the operators had an indication of normal blended flow to the VCT.

Because of the small amount of acid involved (a few gallons) and the large size boric acid tank (8000 gal), tank level did not respond noticeably. Increasing reactor power was the first indication of the problem; the cause was not readily apparent. Because dilution via the VCT is a slow process, the fault had been in progress for several minutes before any indication was noticed. A review by plant operators of this LER and of the proper actions taken should increase awareness of this possibility and should help the operator to deal with similar events if they should occur in the future.

4. Three plants (Crystal River,[12] Oconee,[13] and Zion[14]) experienced recurrent unplanned RSC dilutions. While Crystal River was recovering from a trip, a makeup and purification demineralizer was inadvertently placed into service. The corrective action consisted of tagging to alert the operators which piece of equipment was to be out of service. One week later, with the unit in cold shutdown, the mixed-bed demineralizer was put into service instead of the cation bed causing another unplanned dilution of the RCS. The corrective actions taken this time were that all drums of resin were to be clearly marked, and the operators were cautioned to carefully check the resins before placing the system in service.

5. Following procedures was a problem repeated three times at Oconee. Boration was performed when the power level was above 90% to obtain a better control rod alignment. No dilution problem ensued, but Technical Specifications modifications were initiated. The chemistry records were not kept up to date, and an improper demineralizer was placed into service causing a dilution incident. New administrative controls were developed. Another event occurred when an incorrectly prepared titrant placed boron concentration determinations in error for 4 d and produced unplanned dilutions of the RCS. Steps taken to prevent recurrence included limiting the time for titrating stock, checking daily the boron standard, taking duplicate samples, and using a meter for measuring boron concentrations as an independent check.

6. At Zion, an uncontrolled water addition to the RCS occurred through a normally closed isolation valve in the makeup water line. To

prevent recurrence, (1) the operating personnel were informed of the event and its cause, (2) additional leak rates or tests were to be performed when the unit was in cold shutdown, and (3) the shutdown margin due to boron concentrations was to be recorded in graphic form so that trends would be obvious in the future. However, 5 months later at Unit 2, makeup water isolation valves were inadvertently left open and an unplanned RCS dilution incident occurred. The unit was in cold shutdown. Prevention of recurrence involved verifying valve position prior to testing and depressurizing the isolation valve seal water supply tank whenever the unit is in cold shutdown.

6. CONCLUSIONS AND RECOMMENDATIONS

Accident analyses in the FSARs are quite different from the problems discovered in reviewing the actual operating experiences. The analytical exercise provides (1) for the verification of the initial design, (2) for its control concepts, and (3) for design-basis accidents; but feedback from the field via review of operating experiences does not appear to find its way back into the designs for the later plants. The same design philosophies seem to prevail in the recently licensed plants as did when the vintage plants were designed. Because the systems and their control are simple and conventional, the procedure of replication maintains the status quo. However, 81% of the operating experiences for unplanned dilutions of the RCS were other than those postulated in the design analyses in the PWR FSARs that were reviewed. Reviewers of SARs are principally concerned with the functioning of the systems as per the design criteria and safety philosophies. In the opinion of the author, operations, maintenance, and plant availability apparently are given only superficial consideration, if they receive any at all. (This opinion is based on the author's reviews of questions and answers made to plant SARs.) For example, there were five inadvertent dilution events of the standby liquid control system for BWRs, but all were within the period January 1973 to October 1974. Apparently, some lessons were learned at BWRs.

Human errors were the causative factors in 80% of the unplanned RCS dilution incidents and in 46% of the inadvertent dilutions of the boron concentrations in holding and storage tanks (BAT, BIT, RWST, and boric acid refueling tank). The errors were caused by incomplete or faulty procedures, selection of wrong controls or equipment, miscalculations, improper sampling, incorrect use of tools, and misapplication of equipment. Equipment failures were random and of the kind expected during normal service life. All of this is not unique to the nuclear industry, its modus operandi, or its regulations; but the situation can be improved.

On the average over the past 8 years, there have been 3 unplanned dilutions of the RCS per year, for a rate of about 0.05 such events per reactor-year. (For inadvertent tank dilutions, the rate was about 0.1 per reactor-year.) No boron dilution incident resulted in a reactivity excursion or transient, scram of a unit, or in a challenge to the reactor protection systems. For each recorded event, the operator had sufficient time to diagnose and correct the cause of the inadvertent dilution before the shutdown safety margin was seriously challenged.

This evaluation of the operating experience for boron dilution events in nuclear power plants indicates that system improvements can be achieved in two areas. The first is in the dissemination of information about operating experiences and the feedback of such information to the vendors and the architectural engineers. The second is concerned with human factors experiences.

Neither these conclusions nor recommendations are novel. Actually, they reflect a general awareness that has been recognized for some time. In each inadvertent boron dilution event, the operations personnel have had more than sufficient time to take corrective or mitigative action before the safety margin was seriously challenged. Therefore, to date the safety significance of these events has been small. Because the findings of this study show very little or no safety significance, justification of whether corrective action should be taken yields to the economics of the event situation. However, because these types of events continue to occur and the number of events caused by human error are significant, more thorough training and/or review of procedures appears warranted.

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boron concentration would have leveled off at 1312 ppm. This concentration would have maintained the required shutdown margin."

7. Ginna Nuclear Power Plant (244/75-06, March 25, 1975)

A deborating demineralizer instead of a cation demineralizer was valved onstream. This error in the procedures caused the RCS boron concentration to be unintentionally reduced from 2068 to 1975 ppm.

8. Surry Power Station, Unit 2 (281/76-03, July 30, 1976)

Leakage from three tubes that had been cut during the removal of a section of the seventh tube support plate on the secondary side of the SG diluted the boron concentration in the RSC from 2356 to 1836 ppm. This resulted in a shutdown margin of 11.6% as compared with a required minimum of 1%.

9. Zion Nuclear Plant, Unit 1 (295/76-62, October 15, 1976)

Isolation water entered the RCS through a normally closed makeup water valve and in 17 h reduced the boron concentration from 1081 to 964 ppm. "This posed no safety problem because the reactor was at all times shut down by the required margin."

10. Crystal River Nuclear Plant, Unit 3 (302/77-12, February 16, 1977)

Placing the mixed-bed demineralizer into service during RCS cleanup instead of the cation bed caused a 230 ppm reduction in the boron concentration. "Shutdown margin was maintained at least 8.51% $\Delta K/K$, assuming the highest worth control rod withdrawn, 11% $\Delta K/K$ with highest worth rod inserted."

11. Zion Nuclear Plant, Unit 2 (304/77-9, March 24, 1977)

Incorrectly positioned (open) seal water system isolation valves caused the RCS level to increase 3 ft in 3 h and the boron concentration to decrease from 2182 to 2090 ppm. "The boron concentration was at all times above the refueling concentration of 1888 ppm, and so there were no safety implications."

12. Calvert Cliffs Nuclear Power Plant, Unit 1 (317/78-9, February 2, 1978)

The RCS had been drained down to about 12 in. above the bottom of the hot leg to allow for an inspection of the SG. Due to the slow mixing of the stagnant coolant in the hot and cold legs with coolant in the vessel, the RCS boron concentration slowly decreased from 1720 to 1660 ppm.

13. Surry Power Station, Unit 2 (281/78-12, April 6, 1978)

A failed flow controller caused the primary-grade water valve to overfeed during blend operation. The boron concentration in the RCS decreased from 1372 to 1259 ppm in 13 h. The failure also affected the primary-grade water flow deviation. "Had the event proceeded undetected for an extended period of time, the reactor would never have been less than about 2.3% shut down."

14. Millstone Nuclear Power Station (336/78-5, May 4, 1978)

A partially open bypass valve for the primary makeup water flow control valve allowed the RCS boron concentration to be diluted from 2068 to 1634 ppm. This happened three times on consecutive days, but on two of them the unplanned dilution was masked by planned dilutions. While the bypass valve was thought to be locked closed, it actually was locked three-quarters of a turn open. This resulted from a visual check of valve position rather than a physical verification. "The required shutdown margin of greater than 1% $\Delta K/K$ was maintained during the dilution."

15. Surry Power Station, Unit 1 (280/---, May 12, 1980)

Inadvertent boration of the RCS occurred when a mixed-bed demineralizer was placed in service without verifying that boron concentration of the effluent was equalized with that of the RCS.

16. Oconee Nuclear Plant, Unit 2 (270/80-03, June 5, 1980)

The chemistry records for the deborating demineralizer had not been kept up to date. When the demineralizer, which was thought to be boron saturated, was placed into service, the RCS boron concentration changed from 1895 to 1539 ppm in about 2 h. "A margin of 296 ppm existed above that necessary to maintain the 1% $\Delta K/K$ required shutdown margin."

17. San Onofre Nuclear Generating Station, Unit 1 (260/80-29, July 21, 1980)

Leaky inflatable plugs used to isolate an SG during channel head decontamination resulted in the RCS boron concentration changing from 3357 to 2957 ppm. The inadvertent dilution occurred during a time when containment integrity was not established. "However, at no time did boron concentration decrease below 2400 ppm, which represents a K_{eff} of 0.90."

18. San Onofre Nuclear Generating Station, Unit 1 (206/80-34, September 17, 1980)

While an SG tube removal was being performed, unexpected water in the secondary side flowed by a nonwatertight plug in the nozzle and entered the RCS. The water had leaked past a block valve downstream of the feedwater regulator valve. A maximum dilution of 35 ppm occurred. "At no time since refueling has the RCS boron concentration dropped below 2400 ppm, which represents a 10% shutdown margin."

Hot Shutdown/Standby

19. Oconee Nuclear Plant, Unit 1 (269/---, July 17, 1973)

During zero-power testing, the predicted boron concentration for criticality was 1334 ppm. However, indications were that criticality would be reached at about 1000 ppm. The normality of the sodium hydroxide solution used for the boric acid titration was incorrect, and this condition had existed for 4 d. The errors in the boron analysis were systematic (common-mode failure). "Although the absolute boron concentrations were not accurately known, plant performance was accurately predicted and controlled using relative concentrations."

20. Surry Power Station, Unit 2 (281/---, November 26, 1977)

The RCS was diluted to 873 from 950 ppm due to a misinterpretation of the dilution nomograph. "The reactor was at all times more than 2% shut down."

21. Crystal River Nuclear Plant, Unit 3 (302/77-8, February 9, 1977)

A makeup and purification demineralizer was inadvertently placed into service while the unit was recovering from a trip. RCS boron concentration was diluted about 50 ppm. "Shutdown margin remained greater than 4% $\Delta K/K$."

22. Zion Nuclear Plant, Unit 2 (304/---, May 16, 1974)

The reactor was made critical with one control rod bank below the low-low insertion limit due to an incorrect RCS boron sampling technique. The boron concentration used for the estimated criticality calculation was reported as 1108 ppm when actually it was 1053 ppm. "This incident did not degrade, hamper, or jeopardize the safety of the plant."

23. Point Beach Nuclear Plant, Unit 1 (266/---, June 8, 1974)

The controls for the boric acid blender permitted the output concentration to be 20% less than the desired value, which resulted in a boron concentration of 1435 ppm instead of the expected 1470 ppm in the RCS. "Throughout this occurrence a shutdown margin in excess of 4% $\Delta K/K$ existed. At all times the reactor was sufficiently subcritical to protect the core from all potential reactivity accidents."

Critical Operation

24. Indian Point Station, Unit 2 (247/A04-2-6, February 8, 1974)

Criticality was achieved with the control rods about 27 steps below the insertion level for criticality. The boron concentration was 1125 ppm. "With the assumption that one rod is stuck in its fully withdrawn position, the boron concentration in the RCS was still more than sufficient to shut the reactor down."

25. Oconee Nuclear Plant, Unit 1 (269/74-6, March 28, 1974)

A series of power level changes left the unit operating at 90% of full power. However, continued control rod insertion to compensate for xenon burnup would have resulted in the violation of the control-rod-withdrawal Technical Specification limitation. Therefore, boration was begun to keep the control rods within the insertion limits. Shortly afterwards, a review of the situation revealed that xenon had not been at equilibrium, and the addition of boron was contrary to Technical Specifications. The power level was immediately reduced to below 80% full power. "The principal cause of the occurrence was misevaluation of transient xenon reactivity changes following a series of changes in power level. A contributing case to the occurrence was misunderstanding of the intent of Technical Specification 3.5.2.5-d, which prohibits changes in boron concentration above 80 percent full power to compensate for transient xenon but allows boration to compensate for reactivity effects other than transient xenon at any time."

26. St. Lucie Plant, Unit 1 (335/80-71, January 1, 1981)

While blending boric acid to slightly dilute the RCS, a relief in the line to the blending valve failed. This returned the boric acid to the boric acid makeup tank instead of diluting it and sending it into the VCT. This resulted in the dilution of the RCS because pure water reached the RCS. The plant is designed to safely accommodate this transient, but the event is potentially significant because of the misleading indication given. Because the relief valve is downstream from the boric acid flow element, the operators had indication of a normal blend to the VCT. The small amount of acid involved (a few gallons) and the large boric acid tank size (8000 gal) did not cause a noticeable change in tank level. Increasing power was the first indication of the transient, and the cause was not readily apparent. Because dilution via the VCT is a slow process, the transient had been in progress for several minutes before there was any indication of dilution.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-2798
ORNL/NSIC-208

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
E.W. Hagen

5. DATE REPORT COMPLETED
MONTH March | YEAR 1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Oak Ridge National Laboratory
Oak Ridge, TN 37830

DATE REPORT ISSUED
MONTH July | YEAR 1982

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.
B0755

13. TYPE OF REPORT
TECHNICAL (FORMAL)

PERIOD COVERED *(Inclusive dates)*

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

This report reviews and evaluates events concerned with the inadvertent dilution of boron concentrations to the reactor coolant system for pressurized-water-cooled thermal reactors in commercial service. The safety concern is the unplanned addition of reactivity. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and regulatory documents. The results are collated and analyzed for significance and impact on power plant safety performance.

Several operating experience events were selected for analysis because they meet the criteria for safety significance. However, no boron dilution incidents resulted in a reactivity excursion or transient that scrammed a unit, nor was a reactor protection system challenged by any of the events. The most common cause for unplanned boron dilutions was human error, of which one was a common-mode/common-cause failure. For each recorded event, the operator had sufficient time to diagnose and correct the cause of the inadvertent dilution before the shutdown safety margin was lost or seriously challenged.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

- Boron Dilution
- Nuclear Power Plant Operating Experience

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
UNLIMITED

19. SECURITY CLASS *(This report)*
UNCLASSIFIED

21. NO. OF PAGES

20. SECURITY CLASS *(This page)*
UNCLASSIFIED

22. PRICE
$

NRC FORM 335 (7-77)