Commonwealth Edison Company
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Downers Grove, IL 60515-5701

ComEd

June 19, 2000
RS-00-28

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Request for Technical Specification Change
Revise the Applicability of Technical Specification 3.3.9, "Boron Dilution Protection System (BDPS)"

References: (1) Letter from L. R. Wharton (U. S. NRC) to Licensees (Commonwealth Edison, Texas Utilities Electric, Union Electric, Wolf Creek Nuclear Operating Corporation, and Westinghouse), "Utility Subgroup Technical Approach to Modify or Delete the Boron Dilution Mitigation System," dated February 8, 1993.

(2) Letter from T. A. Bergman (U. S. NRC) to W. J. Cahill, Jr. (Texas Utilities Electric), "Comanche Peak Steam Electric Station, Units 1 and 2 - Amendment Nos. 20 and 6 to Facility Operating License Nos. NPF-87 and NPF-89," dated November 3, 1993.

(3) Letter from J. C. Stone (U. S. NRC) to N. S. Carns (Wolf Creek Nuclear Operating Corporation), "Wolf Creek Generating Station - Amendment No. 96 to Facility Operating License No. NPF-42," dated March 1, 1996.

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In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we are proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes revise the Applicability of TS 3.3.9, "Boron Dilution Protection System (BDPS)." During the Braidwood Station and Byron Station refueling outages, required modifications will be installed and the existing automatic valve actuation features of the BDPS (i.e., to reposition the valves to isolate dilution sources and to re-start boration of the Reactor Coolant System (RCS) on detection of the doubling of the neutron flux (i.e., neutron flux doubling) by the source range nuclear instrumentation) will be removed. The revised Applicability of TS 3.3.9 will make the TS applicable to the unit only until the BDPS is eliminated during that respective unit's refueling outage. Upon startup of the unit following its refueling outage in which BDPS was eliminated, TS 3.3.9 will no longer be applicable.

In conjunction with the implementation of the proposed changes, a number of enhancements to plant hardware, procedures, and controls will be implemented under the provisions of 10 CFR 50.59, "Changes, tests and experiments." The installation of two new redundant Volume Control Tank high level alarms, and revised procedures and controls, will provide the necessary functions required for boron dilution protection. The revised procedures will allow plant operators to take manual actions to implement the requirements for mitigating an unanticipated boron dilution event, without the need to take credit for the current automatic actuation of the dilution and boration valves on detection of neutron flux doubling by the source range nuclear instrumentation. The proposed changes are consistent with the methodology determined to be feasible by the NRC as documented in Reference 1, and as previously approved by the NRC for the Comanche Peak and Wolf Creek units as documented in Reference 2 and Reference 3, respectively.

We request approval of the proposed changes prior to February 1, 2001. This would support installation of the required modifications during the Braidwood Station and Byron Station refueling outages, beginning with the Byron Station Unit 2, spring 2001 refueling outage.

This request is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachments B-1 and B-2 include the marked-up TS page for the proposed changes for Braidwood Station and Byron Station, respectively. Attachments B-3 and B-4 include the associated TS page with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1), "Notice for public comment," which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c), "Issuance of amendment.

4. Attachment D provides information supporting an environmental assessment and a finding that the proposed changes satisfy the criteria for a categorical exclusion.

5. Attachment E provides the revised analysis of the proposed changes.

The proposed changes have been reviewed by the Braidwood Station and Byron Station Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

Commonwealth Edison Company is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Ms. Kelly M. Root at (630) 663-7292.

Respectfully,

R. M. Krich
Vice President - Regulatory Services

Enclosure: Westinghouse NSAC-183 Final Report, "Risk of PWR Inadvertent Criticality During Shutdown and Refueling"

Attachments:
- Affidavit
- Attachment A: Description and Safety Analysis for Proposed Changes
- Attachment B-1: Marked-Up Page for Proposed Changes for Braidwood Station
- Attachment B-2: Marked-Up Page for Proposed Changes for Byron Station
- Attachment B-3: Incorporated Proposed Changes for Braidwood Station
- Attachment B-4: Incorporated Proposed Changes for Byron Station
- Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
- Attachment D: Information Supporting an Environmental Assessment
- Attachment E: Revised Analysis of Proposed Changes

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

| | | |
|---|---|---|
| **STATE OF ILLINOIS** | **)** | |
| **COUNTY OF DUPAGE** | **)** | |
| **IN THE MATTER OF** | **)** | |
| **COMMONWEALTH EDISON (COMED) COMPANY** | **)** | **Docket Nos.** |
| **BRAIDWOOD STATION - UNITS 1 and 2** | **)** | **STN 50-456 and STN 50-457** |
| **BYRON STATION - UNITS 1 and 2** | **)** | **STN 50-454 and STN 50-455** |

**SUBJECT:** **Request for Technical Specification Change**
**Revise the Applicability of Technical Specification 3.3.9, "Boron Dilution Protection System (BDPS)"**

## AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Vice President - Regulatory Services

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 19 day of

June, 2000.

OFFICIAL SEAL
JEFFERY A BATARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:02/24/02

Notary Public

( OFFICIAL SEAL )

# ATTACHMENT A

## DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

### A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we are proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes revise the Applicability of TS 3.3.9, "Boron Dilution Protection System (BDPS)." During the Braidwood Station and Byron Station refueling outages, required modifications will be installed and the existing automatic valve actuation features of the BDPS (i.e., to reposition the valves to isolate dilution sources and to re-start boration of the Reactor Coolant System (RCS) on detection of the doubling of the neutron flux (i.e., neutron flux doubling) by the source range nuclear instrumentation) will be removed. The revised Applicability of TS 3.3.9 will make the TS applicable to the unit only until the BDPS is eliminated during that respective unit's refueling outage. Upon startup of the unit following it's refueling outage in which BDPS was eliminated, TS 3.3.9 will no longer be applicable.

In conjunction with implementation of the proposed changes, a number of enhancements to plant hardware, procedures, and controls will be implemented under the provisions of 10 CFR 50.59, "Changes, tests and experiments." The installation of two new redundant Volume Control Tank (VCT) high level alarms set at 70 percent, and revised procedures and controls, will provide the necessary functions required for boron dilution protection. The revised procedures will allow plant operators to take manual actions to implement the requirements for mitigating an unanticipated boron dilution event, without the need to take credit for the current automatic actuation of the dilution and boration valves on detection of neutron flux doubling by the source range nuclear instrumentation. The proposed changes are consistent with the methodology determined to be feasible by the NRC, as documented in Reference 1, and as previously approved by the NRC for the Comanche Peak and Wolf Creek units, as documented in Reference 2 and Reference 3, respectively.

The proposed changes are described in detail in Section E of this Attachment. The marked-up TS page is shown in Attachments B-1 and B-2 for Braidwood Station and Byron Station, respectively. Attachments B-3 and B-4 include the associated TS page with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

We request approval of the proposed changes prior to February 1, 2001. This would support installation of the required modifications during the Braidwood Station and Byron Station refueling outages, beginning with the Byron Station Unit 2, spring 2001 refueling outage.

### B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS 3.3.9 requires both trains of BDPS to be Operable in Modes 3, 4, and 5 (i.e., Hot Standby, Hot Shutdown, and Cold Shutdown, respectively) to ensure that the BDPS is capable of providing the necessary boron dilution protection. The BDPS must be Operable in Modes 3, 4 and 5 because the Updated Final Safety Analysis Report (UFSAR) Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant," accident analysis identifies the BDPS as the primary means to mitigate an inadvertent boron dilution of the RCS in the event of a Chemical and Volume Control System (CVCS) malfunction. However, the BDPS is not required to be Operable in Modes 1 and 2 (i.e., Power Operation and Startup, respectively) because an inadvertent boron dilution would be terminated by plant operator actions after being alerted to the dilution event by a reactor trip on source range neutron flux high, power range neutron flux high, or overtemperature delta temperature (OTΔT), or after being alerted by the low and low-low control rod insertion limit alarms. The BDPS is also not required to be Operable in Mode 6 (i.e., Refueling) because a dilution event is precluded by administrative controls which require valves to be secured closed to isolate the RCS from the potential source of unborated water.

The primary purpose of the BDPS is to mitigate the consequences of the inadvertent addition of unborated primary grade water into the RCS when the reactor is in Mode 3, 4 or 5 before a complete loss of shutdown margin occurs. The BDPS utilizes inputs from both channels of source range nuclear instrumentation. Upon detection of a neutron flux doubling by either source range nuclear instrumentation channel over a prescribed time period, an alarm is initiated to alert the plant operators, and valve movements are automatically initiated to terminate the dilution from the assumed dilution source and to re-start boration of the RCS. Valves that isolate the Refueling Water Storage Tank (RWST) are opened to supply borated water to the suction of the CVCS Centrifugal Charging Pumps, and valves which isolate the VCT are closed to terminate the assumed dilution source.

## C. BASES FOR THE CURRENT REQUIREMENTS

The current BDPS TS Limiting Condition for Operation (LCO) 3.3.9 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii), "Technical specifications," which requires that a TS LCO must be established for a "structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The BDPS senses abnormal increases in source range neutron flux level and actuates VCT and RWST valves to mitigate the consequences of an inadvertent boron dilution event as described in Chapter 15 of the UFSAR. The accident analysis relies on the automatic BDPS actuation to mitigate the consequences of an inadvertent boron dilution event.

The inadvertent boron dilution event is analyzed to demonstrate compliance with the requirements of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 10, "Reactor Design," GDC 15, "Reactor Coolant System Design," and GDC 26, "Reactivity Control System Redundancy and Capability." The Standard Review Plan (SRP), NUREG-0800, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)," dated July 1981, provides the following specific acceptance criteria necessary to meet these GDC requirements.

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the 95/95 DNBR limit.

3. An incident of moderate frequency (i.e., boron dilution) should not generate a more serious plant condition without other faults occurring independently.

4. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.

5. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time when an alarm announces an unplanned moderator dilution and the time of loss of shutdown margin:

   a. During Refueling: 30 minutes.

   b. During Startup, Cold Shutdown, Hot Standby, and Power Operation: 15 minutes.

## D. NEED FOR REVISION OF THE REQUIREMENT

NRC Information Notice 93-32, "Nonconservative Inputs for Boron Dilution Event Analysis" (Ref. 4), documents various issues that have been raised regarding the non-conservative assumptions and boundary conditions used in the Chapter 15 accident analysis of the boron dilution event. The inverse neutron count rate ratio data is used to predict the time at which a source range neutron flux doubling signal would occur as criticality is approached during a boron dilution event. However, the flux doubling signal and alarm detected by the BDPS may not meet the acceptance criteria for the boron dilution event because an appreciable amount of dilution may already have occurred before the signal and alarm are generated. Thus, an actual neutron flux signal could exceed the neutron flux doubling setpoint used in the analysis, resulting in a loss of shutdown margin before the BDPS terminates the event. Because of the large uncertainties associated with the indication of a true neutron flux doubling by the source range nuclear instrumentation, the mitigation effectiveness of the BDPS may be unreliable. It may not be feasible to demonstrate the adequacy of the BDPS under certain reactor core configurations when accounting for the possible effects of these non-conservatisms.

On December 15, 1992, Commonwealth Edison (ComEd) Company, along with Westinghouse Electric Corp. and three other utilities with similar BDPS designs, met with the NRC to discuss an approach for mitigating an inadvertent boron dilution event without the use of the BDPS, but which was consistent with the guidance provided in the SRP, Section 15.4.6. The methodology that was proposed provides a number of enhancements to plant hardware, procedures, and controls, thereby eliminating the need for the current automatic valve actuation features of the BDPS. In Reference 1, the NRC documented the results of this meeting and indicated that the proposed approach was feasible. Alternate potential solutions, such as relocating detectors, refining methods used to determine the inverse neutron count rate ratio, and attempts to reduce instrument error, have not produced a satisfactory resolution. Based on the approach presented at the December 1992 meeting, the proposed changes will eliminate the concerns expressed during the meeting with the inverse neutron count rate ratio versus boron concentration curves, instrument uncertainties with the neutron flux doubling setpoint, dilution rates, etc.

The proposed changes will resolve the significant operator "workaround" resulting from the current BDPS inoperability issue as documented in the Braidwood Station Licensee Event Report (LER) Number 1998-007-00 and Byron Station LER Number 1998-020-00, "Non-conservative error detected in vendor's analysis code resulted in the Boron Dilution Protection System being determined inoperable," dated December 16, 1998, and December 17, 1998, respectively.

## E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes revise the Applicability of TS 3.3.9, "Boron Dilution Protection System (BDPS)." During the Braidwood Station and Byron Station refueling outages, required modifications will be installed and the existing automatic valve actuation features of the BDPS (i.e., to reposition the valves to isolate dilution sources and to re-start boration of the RCS on detection of neutron flux doubling by the source range nuclear instrumentation) will be removed. The revised Applicability of TS 3.3.9 will make the TS applicable to the unit only until the BDPS is eliminated during that respective unit's refueling outage. Upon startup of the unit following it's refueling outage in which BDPS was eliminated, TS 3.3.9 will no longer be applicable.

In conjunction with the proposed changes, a number of enhancements to plant hardware, procedures, and controls will be implemented under the provisions of 10 CFR 50.59. These changes will provide the necessary functions required for boron dilution protection without the need to take credit for the current automatic actuation of the dilution and boration valves on detection of neutron flux doubling by the source range nuclear instrumentation, and include the following provisions.

- Installation of two new redundant VCT high level alarms as an indication of RCS / CVCS mass imbalance and primary predictor of a potential boron dilution transient. The two new redundant VCT high level alarms will each be set at 70 percent increasing, which is lower than the existing VCT high-high level alarm set at 95 percent, and provide improved timeliness in identifying a potential boron dilution event. The two new alarm annunciator windows in the Main Control Room (MCR) will be titled "Boron Dilution Alert Channel A" and "Boron Dilution Alert Channel B." The Channel A alarm will be initiated by inputs from its respective new VCT high level channel A and the new RCS letdown divert valve CV112A position indication. An alarm on the RCS letdown divert valve CV112A will be installed to annunciate when the valve is not aligned to the VCT (i.e., not in the "VCT" position) to heighten plant operator awareness of the potential for a dilution event during and following planned plant evolutions. The Channel B alarm will be initiated by inputs from its respective new VCT high level channel B and the existing neutron flux doubling channels.

- Revisions to operating procedures to heighten plant operator awareness during evolutions that potentially impact boron dilution and to include the new alarms and indications for timely event recognition as well as the necessary actions required to terminate the event.

- Revisions to the Technical Requirements Manual (TRM) to ensure the assumptions used in the revised ComEd analysis remain valid, i.e., requirements for the two new VCT level channels, and at least one Reactor Coolant Pump (RCP) is in operation and all RCS loop stop valves are open when in Modes 3, 4 and 5. Meeting these assumptions assures proper mixing of the reactor coolant throughout the full RCS volume, as credited in the revised analysis. Any closed RCS loop stop valve creates the potential for that isolated loop to represent an unanalyzed dilution source. If these conditions are not satisfied, the TRM will require the flow paths of potential boron dilution sources to be isolated to prevent the boron dilution event, unless unisolated intermittently under administrative controls as necessary for planned evolutions.

These hardware changes are similar to those implemented at Comanche Peak (Ref. 5) and Wolf Creek (Ref. 6). A key difference is our decision not to revise the normal operating mode of the RCS letdown divert valve CV112A from the "AUTO" position to the "VCT" position. We chose to keep the normal operating mode of valve CV112A in its current "AUTO" position. We made this decision because the two new redundant VCT high level alarms are being set at 70 percent VCT level, which will alert plant operators to any potential boron dilution event in advance of valve CV112A beginning to automatically (i.e., auto) divert RCS letdown flow at 73 percent VCT level. With the two new redundant alarms set to annunciate before receipt of the auto divert signal, the potential for masking an inadvertent boron dilution during routine plant operation when diverting RCS letdown flow is eliminated.

A revised analysis supporting the proposed changes was performed and is provided in Attachment E. As discussed in Attachment E, the revised ComEd analysis to manually isolate potential boron dilution sources and to re-start boration of the RCS is based on the new VCT high level annunciation at 70 percent, and administrative controls to prevent boron dilution if at least one RCP is not in operation and all RCS loop stop valves are not open in Modes 3, 4 and 5. The revised analysis determines the limiting ratio of the initial boron concentration to the critical boron concentration in Modes 3, 4 and 5 that must be exceeded for all reactor core designs to ensure that plant operators would have enough time to prevent criticality as specified in the SRP. The maximum initial to critical boron concentration ratio was calculated, so that, as long as this ratio of the actual reactor core design exceeds the limit, plant operators would have sufficient time to manually prevent criticality in accordance with the applicable SRP acceptance criteria. In this way, the automatic function of the existing BDPS would no longer need to be credited in mitigating an inadvertent boron dilution event. As such, the revised analysis meets the acceptance criteria that the plant operators will perform all actions required to prevent criticality in Modes 3, 4 and 5, in less than 15 minutes after the annunciation of the two new redundant VCT high level alarms, including valve stroke times and system purge times. In those instances where the two new redundant VCT high level alarms will not be sufficient to alert the plant operators in time to prevent the criticality, other alarms are available, i.e., high flux at shutdown, both indicated and audible source range neutron flux count rate, boric acid flow deviation, and primary water flow deviation. In this case, the revised analysis meets the acceptance criteria that the plant operators will diagnose the event and perform all actions required to prevent criticality in Modes 3, 4 and 5, in less than 30 minutes after the dilution event occurs, including valve stroke times and system purge times.

Because the revised analysis does not apply during Modes 1, 2 and 6, bypass switches are being added that will enable the new "Boron Dilution Alert Channel A" and "Boron Dilution Alert Channel B" alarms to be bypassed in these modes.

## F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

As documented in Generic Letter 85-05, "Inadvertent Boron Dilution Events" (Ref. 7), the NRC performed analyses of unmitigated boron dilution events for a typical plant for each Pressurized Water Reactor (PWR) vendor. The NRC determined that while power excursions during boron dilution events are possible, the excursion should be self-limiting and should not exceed SRP acceptance criteria. The NRC concluded that the possible consequences of boron dilution events are not severe enough to jeopardize the health and safety of the public.

The Westinghouse NSAC-183 Final Report, "Risk of PWR Inadvertent Criticality During Shutdown and Refueling" (Ref. 8), included an assessment of the probability and consequences of PWR reactivity events that could occur during shutdown operation. Although PWRs have the potential for reactivity addition due to boron dilution events, no cases were found in operating experience of inadvertent criticality in a PWR during shutdown operation. For gradual boron dilution, due to operator action or equipment malfunction, the initiating event frequency in recent years was found to be no greater than 2E-2 per reactor-year. With a low initiating event frequency, and industry experience showing that criticality has not occurred from any of the boron dilution events to date, the estimated frequency of inadvertent criticality due to gradual boron dilution is less than 1E-4 per reactor-year. Because criticality would cause low power generation, too low to cause reactor core damage, the report concluded that gradual boron dilution events are not considered to be significant contributors to reactor core damage. Similarly, the report found rapid boron dilution to be a low frequency event, estimated to range from 1E-4 to 1E-7 per reactor-year.

The proposed changes are consistent with the methodology presented to the NRC on December 15, 1992, by ComEd, Texas Utilities Electric, Union Electric, Wolf Creek Nuclear Operating Corporation, and Westinghouse Electric Corp. As documented in Reference 1, the NRC determined that the proposed methodology was feasible. The NRC has since specifically evaluated the deletion of automatic actuation functions of the BDPS and use of manual operator actions to implement the requirements for mitigating an unanticipated boron dilution event, similar to the proposed changes. For example, the operation of the Comanche Peak units without automatic BDPS actuation was found to be acceptable based on Texas Utilities Electric's evaluation (Ref. 5), as documented in the NRC Safety Evaluation (Ref. 2). Additionally, the operation of the Wolf Creek unit without automatic BDPS actuation was found to be acceptable based on Wolf Creek Nuclear Operating Corporation's evaluation (Ref. 6), as documented in the NRC Safety Evaluation (Ref. 3).

Attachment E provides the revised ComEd analysis of the CVCS malfunction mitigated by operator action in Modes 3, 4 and 5 using the revised analytical methodology discussed with the NRC as documented in Reference 1. With the revised method, it is recognized that the CVCS and the RCS form a closed system, and mass imbalances, which may affect the RCS, may be detected in the CVCS. The revised analysis demonstrates that positive indication of the occurrence of an inadvertent boron dilution event in Modes 3, 4 and 5 is provided to the plant operators with the two new redundant VCT high level alarms, and that with the alarm setpoint at 70 percent, sufficient time is available for the plant operators to diagnose this event and perform all requisite activities necessary to terminate the event prior to the loss of all shutdown margin. Each cycle reactor core re-load design ensures that sufficient shutdown margin is maintained. This is verified by including cycle-specific limits in the re-load design key parameters checklist which is completed for every reactor core re-load. Revisions are being made to the TRM in order to maintain the assumptions used in the revised analysis, including requirements for the two new VCT level channels. In addition, administrative controls will require potential dilution sources to be isolated to prevent the boron dilution event in Mode 3, 4 or 5, if at least one RCP is not operating and all of the RCS loop stop valves are not open. Meeting these conditions assures proper mixing of the reactor coolant throughout the full RCS volume, as credited in the revised analysis. Any closed RCS loop stop valve creates the potential for that isolated loop to represent an unanalyzed dilution source.

## G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding their impact on any previous submittals and have determined that there is no impact on any previous submittals.

## H. SCHEDULE REQUIREMENTS

We request approval of the proposed changes prior to February 1, 2001, to support installation of the required modifications during the Braidwood Station and Byron Station refueling outages, beginning with the Byron Station Unit 2, spring 2001 refueling outage. Because the modifications are outage related, implementation of the proposed changes will occur upon completion of each unit's respective refueling outage as follows: ($1^{st}$) Byron Station, Unit 2, spring 2001 refueling outage (i.e., B2RO9), ($2^{nd}$) Braidwood Station, Unit 1, fall 2001 refueling outage (i.e., A1RO9), ($3^{rd}$) Byron Station, Unit 1, spring 2002 refueling outage (i.e., B1R11), and ($4^{th}$) Braidwood Station, Unit 2, spring 2002 refueling outage (i.e., A2RO9).

## I. REFERENCES


1. Letter from L. R. Wharton (U. S. NRC) to Licensees (Commonwealth Edison, Texas Utilities Electric, Union Electric, Wolf Creek Nuclear Operating Corporation, and Westinghouse), "Utility Subgroup Technical Approach to Modify or Delete the Boron Dilution Mitigation System," dated February 8, 1993.

2. Letter from T. A. Bergman (U. S. NRC) to W. J. Cahill, Jr. (Texas Utilities Electric), "Comanche Peak Steam Electric Station, Units 1 and 2 - Amendment Nos. 20 and 6 to Facility Operating License Nos. NPF-87 and NPF-89," dated November 3, 1993.

3. Letter from J. C. Stone (U. S. NRC) to N. S. Carns (Wolf Creek Nuclear OperatingCorporation), "Wolf Creek Generating Station - Amendment No. 96 to Facility Operating License No. NPF-42," dated March 1, 1996.

4. NRC Information Notice 93-32, "Nonconservative Inputs for Boron Dilution Event Analysis," dated April 21, 1993.

5. Letter from W. J. Cahill, Jr. (Texas Utilities Electric) to U. S. NRC, submittal of Comanche Peak, Units 1 and 2, License Amendment Request 93-01 Reanalysis of Inadvertent Boron Dilution Event, dated April 30, 1993.

6. Letter from R. C. Hagan (Wolf Creek Nuclear Operating Corporation) to U. S. NRC, submittal of Wolf Creek License Amendment Request, dated November 22, 1995.

7. NRC Generic Letter 85-05, "Inadvertent Boron Dilution Events," dated January 31, 1985.

8. Westinghouse NSAC-183 Final Report, "Risk of PWR Inadvertent Criticality During Shutdown and Refueling," dated December 1992.

## ATTACHMENT B-1

## THE PROPOSED CHANGES FOR BRAIDWOOD STATION

MARKED-UP TS PAGE

3.3.9-1

## 3.3 INSTRUMENTATION

### 3.3.9 Boron Dilution Protection System (BDPS)

LCO 3.3.9 Two trains of the BDPS shall be OPERABLE.

---------------------------NOTE---------------------------
The boron dilution flux doubling signal may be blocked in MODE 3 during reactor startup.
----------------------------------------------------------

APPLICABILITY: MODES 3, 4, and 5 **for Braidwood Unit 1 through cycle 9 and for Braidwood Unit 2 through cycle 9**.

ACTIONS

------------------------------------NOTE------------------------------------
Unborated water source isolation valves may be unisolated intermittently under administrative controls.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One train inoperable. | A.1 Restore train to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Close unborated water source isolation valves. | 1 hour |
| | <u>AND</u> | |
| | B.2 Verify unborated water source isolation valves closed. | Once per 31 days |

(continued)

**ATTACHMENT B-2**

**THE PROPOSED CHANGES FOR BYRON STATION**

MARKED-UP TS PAGE

3.3.9-1

## 3.3 INSTRUMENTATION

### 3.3.9 Boron Dilution Protection System (BDPS)

LCO 3.3.9 Two trains of the BDPS shall be OPERABLE.

---------------------------NOTE---------------------------
The boron dilution flux doubling signal may be blocked in MODE 3 during reactor startup.
----------------------------------------------------------

APPLICABILITY: MODES 3, 4, and 5 **for Byron Unit 1 through cycle 11 and for Byron Unit 2 through cycle 9**.

ACTIONS

------------------------------------NOTE------------------------------------
Unborated water source isolation valves may be unisolated intermittently under administrative controls.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One train inoperable. | A.1 Restore train to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Close unborated water source isolation valves. | 1 hour |
| | <u>AND</u> | |
| | B.2 Verify unborated water source isolation valves closed. | Once per 31 days |

(continued)

**ATTACHMENT B-3**

**PROPOSED CHANGES INCORPORATED FOR BRAIDWOOD STATION**

TS PAGE

3.3.9-1

## 3.3 INSTRUMENTATION

### 3.3.9 Boron Dilution Protection System (BDPS)

LCO 3.3.9 Two trains of the BDPS shall be OPERABLE.

---------------------------NOTE---------------------------
The boron dilution flux doubling signal may be blocked in MODE 3 during reactor startup.
----------------------------------------------------------

APPLICABILITY: MODES 3, 4, and 5 for Braidwood Unit 1 through cycle 9 and for Braidwood Unit 2 through cycle 9.

ACTIONS

------------------------------------NOTE------------------------------------
Unborated water source isolation valves may be unisolated intermittently under administrative controls.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One train inoperable. | A.1 Restore train to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Close unborated water source isolation valves. | 1 hour |
| | <u>AND</u> | |
| | B.2 Verify unborated water source isolation valves closed. | Once per 31 days |

(continued)

**ATTACHMENT B-4**

**PROPOSED CHANGES INCORPORATED FOR BYRON STATION**

TS PAGE

3.3.9-1

## 3.3 INSTRUMENTATION

### 3.3.9 Boron Dilution Protection System (BDPS)

LCO 3.3.9 Two trains of the BDPS shall be OPERABLE.

---------------------------NOTE---------------------------
The boron dilution flux doubling signal may be blocked in MODE 3 during reactor startup.
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APPLICABILITY: MODES 3, 4, and 5 for Byron Unit 1 through cycle 11 and for Byron Unit 2 through cycle 9.

ACTIONS

------------------------------------NOTE------------------------------------
Unborated water source isolation valves may be unisolated intermittently under administrative controls.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One train inoperable. | A.1 Restore train to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Close unborated water source isolation valves. | 1 hour |
| | <u>AND</u> | |
| | B.2 Verify unborated water source isolation valves closed. | Once per 31 days |

(continued)

# ATTACHMENT C

## INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), "Issuance of amendment," a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Commonwealth Edison (ComEd) Company is proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes revise the Applicability of TS 3.3.9, "Boron Dilution Protection System (BDPS)." During the Braidwood Station and Byron Station refueling outages, required modifications will be installed and the existing automatic valve actuation features of the BDPS will be removed. The revised Applicability of TS 3.3.9 will make the TS applicable to the unit only until the BDPS is eliminated during that respective unit's refueling outage. Upon startup of the unit following it's refueling outage in which BDPS was eliminated, TS 3.3.9 will no longer be applicable. In conjunction with the proposed changes, a number of enhancements to plant hardware, procedures, and controls will be implemented under the provisions of 10 CFR 50.59, "Changes, tests and experiments." The installation of two new redundant Volume Control Tank (VCT) high level alarms set at 70 percent, and revised procedures and controls, will provide the necessary functions required for boron dilution protection. The revised procedures will allow plant operators to take manual actions to mitigate a boron dilution event, without the need to take credit for the current automatic actuation of the dilution and boration valves on detection of neutron flux doubling by the source range nuclear instrumentation.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The only accident potentially impacted by the proposed changes is the inadvertent boron dilution event.

   The Boron Dilution Protection System (BDPS) is not considered an initiator of any analyzed event. The BDPS performs detection and mitigative functions for the inadvertent boron dilution event. Therefore, the proposed changes have no impact on the probability of an event previously analyzed. Therefore, the proposed change does not involve a significant increase in the probability of occurrence of an accident previously evaluated.

The proposed changes impact the consequences of an inadvertent dilution event due to the new requirement to manually reposition the Chemical and Volume Control System (CVCS) valves that isolate the boron dilution sources and that re-start boration of the Reactor Coolant System (RCS) in Modes 3, 4, and 5 (i.e., Hot Standby, Hot Shutdown, and Cold Shutdown, respectively). The revised detection and mitigation methodology being proposed achieves the same basic function as the existing BDPS, i.e., to prevent a return to critical during an inadvertent boron dilution event. The proposed changes will provide an improved response to the inadvertent boron dilution event compared to the BDPS, and thereby will prevent a return to critical. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed changes to manually isolate potential dilution sources and to re-start boration of the RCS do not create the potential for a new or different kind of accident because the change results in plant configurations that have always been allowed. In conjunction with these proposed changes, enhancements to plant hardware, revisions to procedures, and administrative controls will be implemented. The proposed enhancements to plant hardware include the addition of two new redundant Volume Control Tank (VCT) high level alarms, which are passive in nature (i.e., do not provide any control function), and therefore do not create the possibility of a new or different kind of accident. Administrative controls and revisions to procedures will increase the operator's awareness of a potential boron dilution event and will provide the steps necessary to respond to a boron dilution event. As a result, the administrative controls and revisions to procedures do not create the possibility of a new or different kind of accident.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

The design criterion and margin of safety for the existing BDPS is that the inadvertent boron dilution event is terminated within a specified period prior to the complete loss of shutdown margin. This criterion will continue to be satisfied following implementation of the proposed changes. The proposed changes were evaluated to ensure that the plant operators prevent criticality in Modes 3, 4 and 5 following an inadvertent boron dilution event, based on the revised analytical methodology previously discussed with the NRC and found to be feasible as documented in a letter from L. R. Wharton (U. S. NRC) to Licensees (Commonwealth Edison, Texas Utilities Electric, Union Electric, Wolf Creek Nuclear Operating Corporation, and Westinghouse), "Utility Subgroup Technical Approach to Modify or Delete the Boron Dilution Mitigation System," dated February 8, 1993. The proposed method of detecting and mitigating this event has been shown by the analysis supporting this Technical Specifications change request to prevent a return to critical following an inadvertent boron dilution event, and meets the same NRC acceptance criteria as specified in the Standard Review Plan (SRP), NUREG-0800, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)," dated July 1981, as applicable to the existing BDPS. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

## ATTACHMENT D

### INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." ComEd has determined that the proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9), "Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b), "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

**(i) The amendment involves no significant hazards consideration.**

As demonstrated in Attachment C, the proposed changes do not involve any significant hazards consideration.

**(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.**

The proposed changes are limited to revising the Applicability of Technical Specification 3.3.9, "Boron Dilution Protection System (BDPS)." The proposed changes do not allow for an increase in the unit power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or by-products. Therefore, the proposed changes do not affect actual unit effluents. As documented in Attachment C, there will be no change in the types or increase in the amounts of any effluents released offsite.

**(iii) There is no significant increase in individual or cumulative occupational radiation exposure.**

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.

# ATTACHMENT E

## REVISED ANALYSIS OF PROPOSED CHANGES

**Chemical and Volume Control System (CVCS) Malfunction**
**Mitigated by Operator Action in Modes 3, 4 and 5**
(From Commonwealth Edison (ComEd) Company Calculation No. PSA-B-95-12)

## CALCULATION TABLE OF CONTENTS

## Purpose and Objective

This calculation determines the limiting ratio of the initial boron concentration to the critical boron concentration in Modes 3, 4 and 5, which must be exceeded for all reactor core designs. This limit incorporates at least 15 minutes (Reference 1) for the plant operators to diagnose the event and prevent criticality following an inadvertent boron dilution in the reactor core. This dilution may occur due to a Chemical and Volume Control System (CVCS) malfunction. This method of analysis is implemented in Modes 3, 4 and 5 for Braidwood Station and Byron Station.

The existing Braidwood Station and Byron Station licensing basis credits the Boron Dilution Protection System (BDPS) as a means to mitigate criticality in the event of a CVCS malfunction in Modes 3, 4 and 5. Due to a computer modeling error, the BDPS has been declared inoperable. The purpose of this calculation is to eliminate the BDPS from the Braidwood Station and Byron Station design basis.

This calculation determines the ratio of the initial boron concentration to the critical boron concentration that must be exceeded for the actual reactor core design to ensure that plant operators would have enough time to prevent criticality according to the Standard Review Plan (SRP) acceptance criteria. This would allow the BDPS to not be credited in mitigating an inadvertent boron dilution. The SRP states that plant operators have 15 minutes to act on the alarm that announces the dilution event (Reference 1). In this calculation, the two new redundant Volume Control Tank (VCT) high level alarms set at 70 percent are assumed to announce the dilution event.

## Methodology and Acceptance Criteria

Under normal operating conditions, the Reactor Coolant System (RCS) charging and letdown flow is set to maintain a constant reactor vessel level (Modes 5 and 6) or pressurizer level (Modes 1, 2, 3 and 4). For an inadvertent boron dilution to occur, some event must lead to a step increase in RCS charging flow from a diluted source. The postulated source of dilution is from the primary water makeup system through the primary water makeup control valve, CV111A (Reference 7), which injects between the VCT and the centrifugal charging pump (CV pump) (Figure 1), and a failure of the boric acid blend system. Since control of RCS charging and letdown flow is not affected by the increase in flow at the CV pump suction, with the exception of a negligible increase in suction pressure, the level will begin to increase in the VCT. The plant operators are alerted to a CVCS malfunction when the VCT level reaches its new 70 percent high level setpoint, and the two new redundant alarms are sounded. The plant operators are then allowed 15 minutes to recognize the dilution and perform the necessary actions. These manual actions entail switching the CV pump suction from the VCT to the Refueling Water Storage Tank (RWST), a source of highly borated water. The dilution continues until all the diluted water is purged from the CVCS piping. At that point, boric acid enters the cold leg and the transient is terminated. The termination of the transient must occur before the reactor reaches criticality.

The simplified schematic of the RCS and CVCS interaction is illustrated in Figure 1. The dilution is shown to be between the VCT and the CV pump. The plant operators must open valves CV112D and/or CV112E and close valves CV112B and/or CV112C in order to mitigate the event.

Currently, the VCT has several alarm setpoints and automatic actuations associated with VCT level. The high level alarm, alerting the plant operators to the dilution event, must annunciate before the auto divert actuation, or a dilution could occur without a high level alarm annunciating. Therefore, the two new redundant high level alarms will be installed at the 70% VCT level. Figure 2 shows the existing setpoints and alarms, and the new high VCT level alarm. Reference 16 describes the current VCT level alarms and setpoints.

It is possible that, if a boron dilution were to occur during certain plant evolutions, the high VCT level alarm may not annunciate. A discussion of a possible boron dilution coincident with these evolutions is provided below. For all of these situations, there is significant operator interaction with the plant. Therefore, it is reasonable to assume that the plant operators would be able to diagnose and mitigate the event within 30 minutes of its initiation.

For very small dilution flow rates, the time required to fill the VCT to the high alarm setpoint may be greater than the time required to dilute the RCS to the critical boron concentration. However, it can be shown that the time from the initiation of the event to the time that the critical boron concentration is reached is significantly greater than 30 minutes. Alarms available to alert the plant operators of the dilution include the boric acid flow deviation and primary water flow deviation alarms. In addition, the alarms from the BDPS will still be available, without any associated automatic actions. This event is discussed in detail in Appendix A - Slow Dilution.

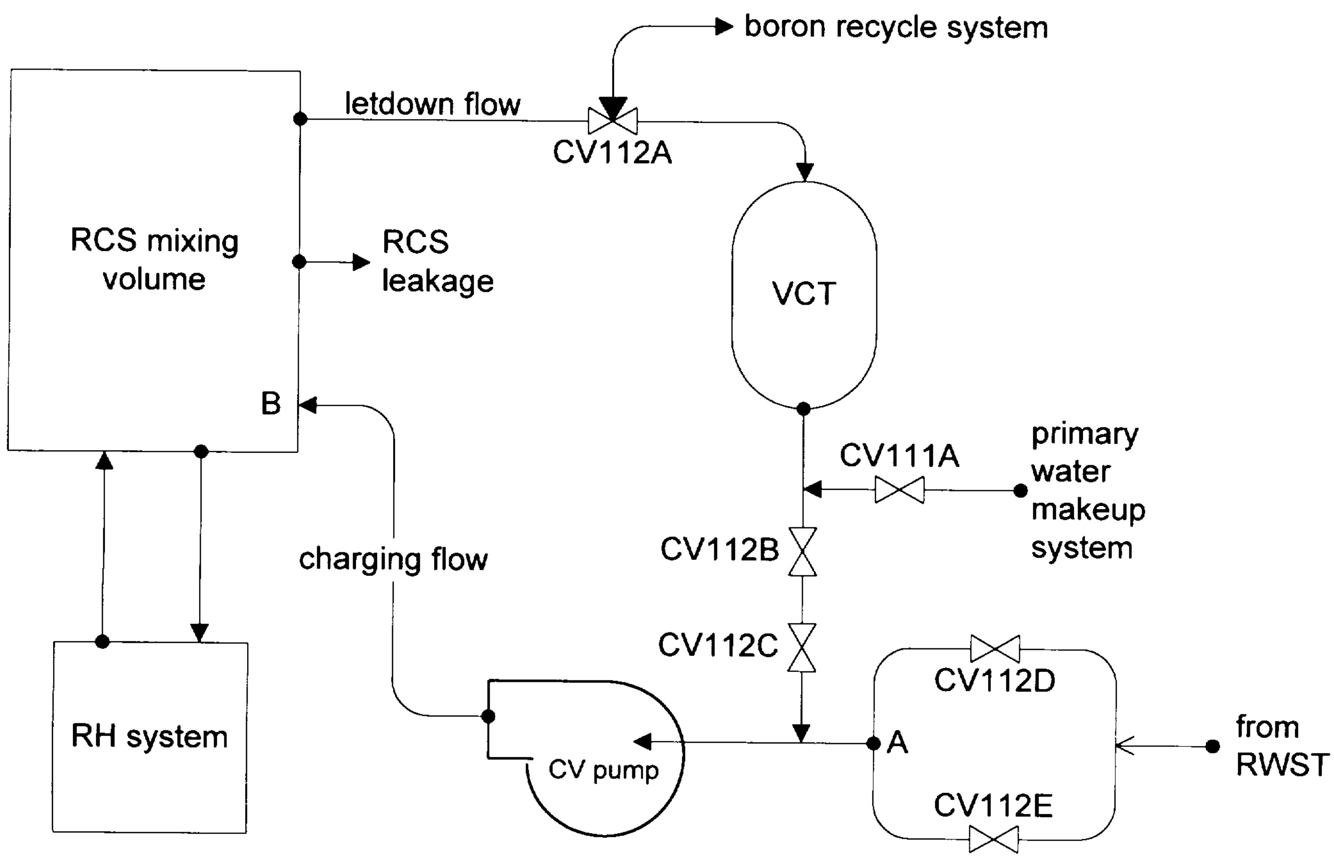


**Figure 1: Simplified Schematic Diagram of RCS and CVCS Flowpaths**

Another potential dilution event is a boron dilution during plant heatup. The effect of an RCS heatup is not considered in this analysis, because during an RCS heatup, water is expelled from the RCS to the VCT (or to the recycle holdup tank) due to the thermal expansion of the reactor coolant. Typically, dilution sources are isolated during heatup to minimize liquid radwaste processing. Furthermore, primary water flow deviation and boric acid flow deviation alarms are available during a plant heatup to alert plant operators to an inadvertent boron dilution, and the existing BDPS alarm without any associated automatic actions would also be available. During a plant heatup, it is therefore possible to administratively control diluted makeup water sources to preclude an inadvertent dilution without adversely affecting operations, since the VCT level will not need to be increased. Due to these administrative controls, inadvertent boron dilution can be prevented and need not be analyzed. In addition, the heatup evolution is performed in accordance with plant procedures, which require significant plant operator interaction. During this time, plant operators are expecting specific plant responses, and any deviations will receive prompt attention.

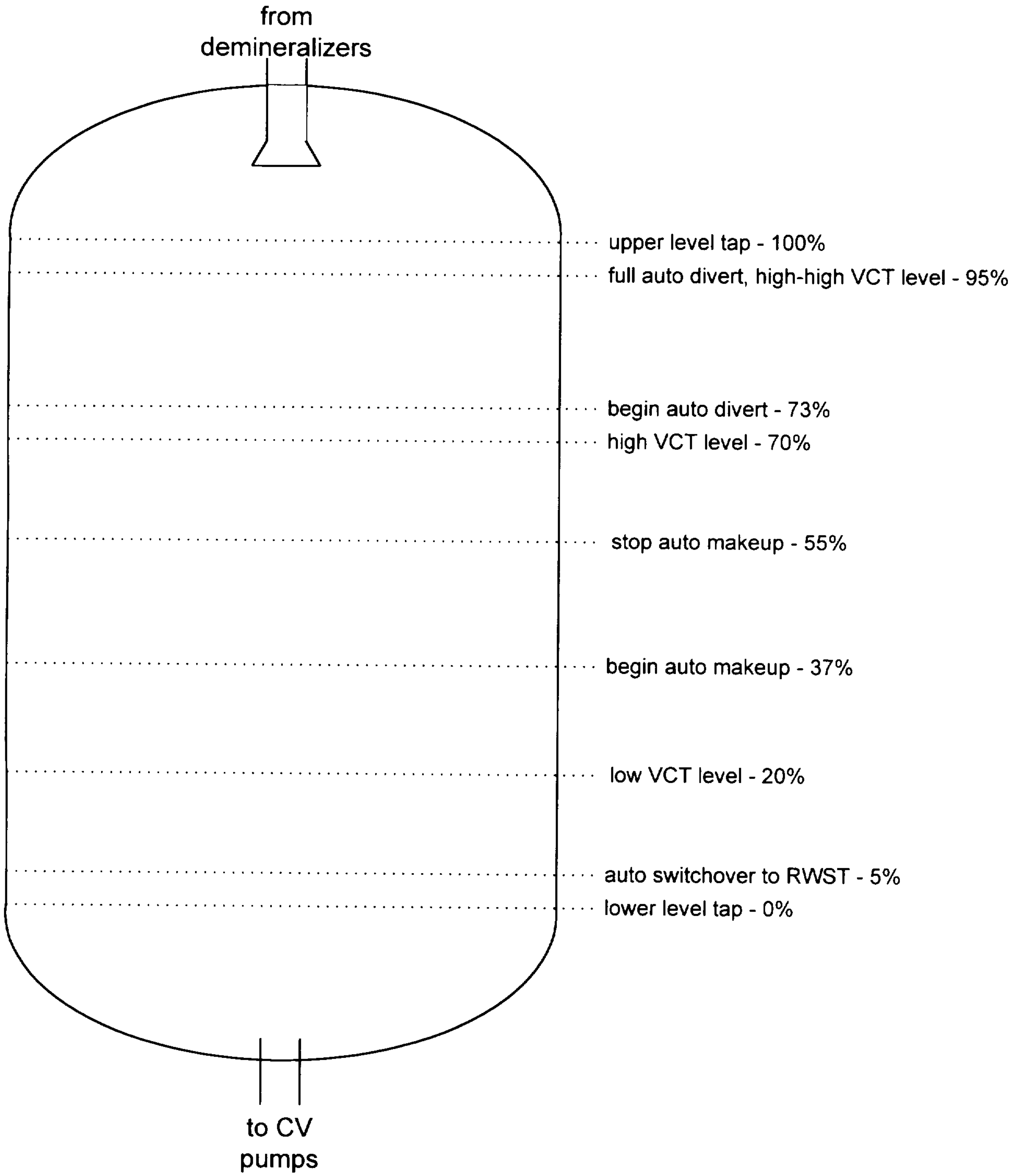


**Figure 2: VCT Level Setpoints (Reference 16)**

If the RCS is cooled rapidly, its density is increased and therefore, the RCS volume decreases. When this happens, normal RCS charging flow is increased to make up the lost pressurizer level, and as a consequence, the VCT level will decrease. While the method above crediting high VCT level alarm to alert plant operators to the dilution takes into account very small fluctuations in the RCS inventory, large changes associated with an RCS cooldown are not included. Moreover, if a dilution event were to occur at the same time as an RCS cooldown, the VCT may not see a significant rise in level, and the high level alarm may not annunciate. In this case, since the cooldown evolution requires significant plant operator interaction, it is reasonable to assume that the plant operators have 30 minutes to diagnose the dilution and mitigate criticality from the time of the dilution until all actions have been performed.

For this case, the plant operators will be alerted to the dilution by a number of possible alarms. These include the primary water flow deviation and boric acid flow deviation alarms. In addition, the nuclear instrumentation flux doubling alarms from BDPS will continue to be active to provide indication to the plant operators, without any automatic actions. Appendix B - Coincident RCS Cooldown shows that this scenario is less limiting than the fast dilution during stable plant conditions described above.

## ***Acceptance Criteria***

This is a calculation of the acceptance criteria for the ratio of the initial to critical boron concentration required for all Braidwood Station and Byron Station reactor core designs in Modes 3, 4 and 5. This calculation determines the ratio of the initial to critical boron concentration, required to meet one of the following acceptance criteria:

Criterion 1 The plant operators will perform all actions required to prevent criticality in Modes 3, 4 and 5 in less than 15 minutes after the annunciation of the VCT high level alarm, including valve stroke times and system purge times (Reference 1).

Criterion 2 The plant operators will diagnose the event and perform all actions required to prevent criticality in Modes 3, 4 and 5 in less than 30 minutes after the dilution event occurs, including valve stroke times and system purge times.

Criterion 2 is needed because there are some instances when the high VCT alarm will not be sufficient to alert the plant operators in time to prevent criticality. Other alarms are available in these situations. For a detailed discussion of these cases, see Appendix A - Slow Dilution, and Appendix B - Coincident RCS Cooldown.

## ***Methodology***

Table 1, "Variable Definitions for Boron Dilution Analysis," gives the definitions of all variables used for the calculation of the ratio of the initial to critical boron concentration.

The time from annunciation of the high VCT level alarm to the time that the plant operators can no longer mitigate criticality must be at least 15 minutes according to Criterion 1 (Reference 1). This time is the total time it would take for the reactor to become critical, less the time required to fill the VCT to the high level alarm, the valve stroke time, and the time required to purge the

CVCS (i.e., purge volume). This is written:

$$t_{op15} \le t_{crit} - t_{fill} - t_{swap} - t_{purge} \tag{1}$$

**Table 1: Variable Definitions for Boron Dilution Analysis**

| Type | Variable | Units | Definition | Conservative Direction |
|---|---|---|---|---|
| input | $V_{RCS}$ | $ft^3$ | RCS mixing volume | minimum |
| input | $T_{RCS}$ | °F | Initial temperature of RCS mixing volume | maximum |
| property | $\nu_{RCS}$ | $ft^3/lb$ | Specific volume of water in RCS | maximum |
| input | $V_{purge}$ | $ft^3$ | CVCS purge volume | maximum |
| input | $T_{purge}$ | °F | CVCS purge water temperature | minimum |
| property | $\nu_{purge}$ | $ft^3/lb$ | Specific volume of water in purge volume | minimum |
| input | $Q_c$ | gpm | Charging flow rate | maximum |
| input | $Q_m$ | gpm | Mismatch flow rate (net letdown - net charging) | most negative |
| input | $V_{VCT}$ | $ft^3$ | Volume of VCT between low and high level | maximum |
| input | $Q_{dil}$ | gpm | Dilution flow rate | maximum |
| N/A | $C_{bi}$ | ppm | Initial boron concentration | N/A |
| N/A | $C_{bc}$ | ppm | Critical boron concentration | N/A |
| input | $t_{swap}$ | min | Time of valve swap over | maximum |
| calc | $t_{purge}$ | min | Time to purge charging line | maximum |
| calc | $t_{fill}$ | min | Time to fill the VCT | maximum |
| input | $t_{op15}$ | min | Net operator response time | N/A |
| calc | $t_{crit}$ | min | Time to dilute RCS from initial to critical boron concentration | maximum |
| calc | $C_{bi}/C_{bc}$ | N/A | Ratio of initial boron concentration to critical boron concentration | maximum |

The time required to increase the combined volume of the RCS and CVCS by the volume of the VCT between the empty (5%) and high (70%) levels can be calculated by:

$$t_{fill} = \left(\frac{V_{VCT}}{Q_m + Q_{dil}}\right)\left(\frac{7.481\,\text{gal}}{1\,\text{ft}^3}\right) \tag{2}$$

The time required to purge the charging line after valve swapover to the time borated water from the RWST enters the RCS is calculated by:

$$t_{purge} = \left(\frac{V_{purge}}{Q_c}\right)\left(\frac{7.481\,\text{gal}}{1\,\text{ft}^3}\right) \tag{3}$$

The time to dilute the RCS from the initial boron concentration to the critical boron concentration, assuming the charging flow rate is at least as large as the dilution flowrate is:

$$t_{crit} = \frac{V_{RCS}\left(\frac{v_{purge}}{v_{RCS}}\right)}{Q_{dil}}\left(\frac{7.481\,\text{gal}}{1\,\text{ft}^3}\right)\ln\left(\frac{C_{bi}}{C_{bc}}\right) \tag{4}$$

Equation (4) does not take into account any additional benefits from additional mixing in the VCT or letdown path.

The limiting ratio of initial to critical boron concentration, where the plant operators have exactly 15 minutes from the time of the high VCT level alarm until shutdown margin is lost (i.e., the limiting ratio) in Modes 3, 4 and 5 to meet Criterion 1, is calculated by combining equation (1) and equation (4), to get:

$$\frac{C_{bi}}{C_{bc}} = \exp\left(\frac{t_{op15} + t_{fill} + t_{swap} + t_{purge}}{\left(\frac{V_{RCS}\cdot v_{purge}}{Q_{dil}\cdot v_{RCS}}\right)\left(\frac{7.481\,\text{gal}}{1\,\text{ft}^3}\right)}\right) \tag{5}$$

Maximizing this ratio will be the most conservative, so the design inputs will be chosen to maximize this ratio.

## Assumptions and Engineering Judgments

The following assumptions and engineering judgements are used in this calculation.

1. The RCS volume and temperature remain constant. This assumption is based on the plant operators setting the RCS charging and letdown flow to maintain a constant system volume during heatup, cooldown, and steady state operations if automatic systems are not available. The variable $Q_m$ is introduced to account for small fluctuations in the RCS volume and temperature.

2. The RCS and CVCS together are a closed system. Water may enter the system to cause a dilution, but no water will leave the boundary. All changes to the volume of this system are to the VCT water level, and not to any other part of the system. Automatic systems which divert water from the VCT to the Boron Recycle System at high VCT levels through valve CV112A will not actuate until after the high VCT level alarm is annunciated (Figure 2). As the RCS heats up, water expelled from the RCS fills the VCT. As the RCS cools down, water from the VCT enters the RCS. Note that small changes in the RCS volume are included in accordance with design input number 8.

3. All dilutions occur at a constant rate.

4. For Criterion 1 (Reference 1), the allowed plant operator action time begins when the VCT high level alarm annunciates, and ends when plant operator action would no longer mitigate criticality. This time includes the time necessary to manipulate valves and purge the piping system, before reboration can occur. For Criterion 2, the allowed plant operator action time begins when the dilution is initiated.

5. The subsystem temperatures are chosen to minimize the available mixing mass and maximize the dilution flow rate. The RCS mass is evaluated at the temperature boundaries, 557°F, 350°F, 200°F, and 40°F, and the charging and letdown system at 40°F. The maximum Mode 3 temperature of 557°F is selected as zero percent power temperature. The Mode 4 and 5 maximum temperatures of 350°F and 200°F are from Technical Specifications Table 1.1-1, "Modes." These values maximize the density of the charging system and minimize the density of the RCS, increasing the dilution rate.

6. The most reactive control rod is stuck out of the reactor core. This minimizes the shutdown margin.

7. The reactor is shutdown and meets the minimum shutdown margin described by Technical Specifications and the Core Operating Limits Report (COLR).

8. Since the reactor criticality is not modeled in this evaluation, no explicit assumptions are made regarding the moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, or radial power distribution.

9. Changes in fluid density and specific volume due to boric acid addition are neglected. The highest boron concentration of approximately 2000 ppm corresponds to about 0.2 weight percent, which would have an insignificant effect on the density of the fluid.

Note that additional assumptions, specific to the coincident RCS cooldown case, are listed in Appendix B - Coincident RCS Cooldown.

## Design Inputs

1. **RCS mixing volume** ($V_{RCS}$) - This is the volume of water that is diluted during the transient. This volume includes the reactor vessel, hot and cold legs, Reactor Coolant Pumps (RCPs), and steam generator tubes. It does not include the reactor vessel head, CVCS, Residual Heat Removal (RH) System, the pressurizer, the pressurizer surge line or the pressurizer spray line. Thermal expansion is not considered, as a minimum volume is conservative. The entire RCS mixing volume can be included because it is assumed at least one RCP is operating and providing forced flow to all four loops. Steam generator (SG) tube plugging of 5% for Unit 1 and 10% for Unit 2 is assumed, consistent with the Braidwood Station and Byron Station Power Uprate Project (Reference 14). From Reference 4 and Reference 5 for Unit 1 SG volumes, and Reference 6 for Unit 2 SG volumes and reactor vessel head volume, the total RCS volume is given in Table 2, "Design Inputs for RCS Mixing Volume." Therefore, an RCS mixing volume of 10,653 $ft^3$ is used for Unit 1 and 9331 $ft^3$ is used for Unit 2.

**Table 2: Design Inputs for RCS Mixing Volume**

| IMP data ($ft^3$) | Unit 1 (BWI SG) | Unit 2 (D5 SG) |
|---|---|---|
| Reactor vessel | 4767.926 | 4767.926 |
| Hot leg | 427.64 | 427.64 |
| Steam generator plenums | 1206.8 | 1260.0 |
| Steam generator tubes | 3611.52 | 2235.6 |
| Crossover leg | 558.36 | 558.36 |
| Cold leg | 439.16 | 439.16 |
| Reactor coolant pump | 324 | 324 |
| Reactor vessel head | -681.452 | -681.452 |
| Total RCS volume | 10,653.954 | 9331.234 |

For Modes 3, 4 and 5, at least one RCP shall be operating and all loop stop valves shall be open, or the potential dilution sources must be isolated. One RCP with all loop stop valves open ensures that adequate mixing occurs throughout the RCS as shown in Reference 15. Note that with an RCP operating, the reactor vessel will not be drained for refueling operations, therefore, a drained RCS is not considered.

2. **Initial temperature of RCS mixing volume** ($T_{RCS}$) - This is the temperature of the RCS and determines the density and mass of the RCS. The maximum temperature is assumed for Modes 3, 4 and 5 in order to maximize the specific volume. This analysis assumes values of 557°F, 350°F, 200°F, and 40°F, as the RCS temperature depending on the Mode. The 40°F case is included to provide a lower bound for all expected temperatures in Modes 3, 4 and 5.

3. **Specific volume of water in RCS** ($v_{RCS}$) - This is the saturated water specific volume found from the steam tables. This value is dependent on RCS temperature. The values used for this calculation are given in Table 3, "Specific Volume of Saturated Water."

**Table 3: Specific Volume of Saturated Water**

| Temperature (°F) | Specific Volume ($ft^3/lb_m$) |
|---|---|
| 557 | 0.021977 |
| 350 | 0.017989 |
| 200 | 0.016637 |
| 40 | 0.016019 |

4. **CVCS purge volume** ($V_{purge}$) - This is the amount of diluted water that must be purged from the CV System before borated water from the RWST reaches the cold leg. This represents the volume of piping and components from the "T" junction where the RWST supply meets the VCT supply line, to the cold leg charging nozzle. This volume includes a CV pump and a regenerative heat exchanger and associated piping. It is shown on Figure 1 as the volume from point "A" to point "B." As specified in Technical Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," only one CV pump may be capable of injecting into the RCS below 330°F. A value of 66.952 $ft^3$ is assumed for the purge volume in Modes 3 and 4, with two CV pumps in operation and 59.248 $ft^3$ is assumed for Mode 5, with only one CV pump in operation (Reference 9).

5. **CVCS purge water temperature** ($T_{purge}$) - This is the temperature of the water initially in the CVCS purge volume and determines its density and mass. This is also the temperature of the VCT. This temperature is set at a conservatively low 40°F to maximize the density of the CVCS water, which maximizes the dilution rate.

6. **Specific volume of water in purge volume** ($\nu_{purge}$) - This is the specific volume of saturated water at 40°F. This value is given in the steam tables as 0.016019 $ft^3/lb$.

7. **Charging flow rate** ($Q_c$) - This value is the sum of the seal injection and normal charging flow. The entire charging flow is assumed to enter the RCS mixing volume. A value of 150 gpm is specified in Reference 10, which corresponds to the high flow alarm. An uncertainty of 16.7 gpm is given in Reference 23 for Braidwood Station and 18 gpm uncertainty is given in Reference 24 for Byron Station. A bounding uncertainty of 18 gpm is applied to get a value of 168 gpm, which will be used in this analysis.

8. **Mismatch flow rate** ($Q_m$) - This term is used to account for small changes in RCS mass and volume, which are assumed to be constant. These changes could be caused by control systems fluctuations or temperature changes, and RCS leakage (Figure 1). Since these affect the charging/letdown balance, the mismatch flow can be defined as:

$$Q_m = \text{letdown (gpm)} - \text{charging (gpm)} \tag{6}$$

   The RCP seal water return is 12 gpm during normal plant operation (Reference 17). This represents a volume decrease since the seal water is returned downstream of the VCT. From Reference 3, the leakage out of the RCS and CVCS systems is limited to 12 gpm. This value bounds changing the VCT level by about 15% in 30 minutes, which would be easily detected by the plant operators. Therefore, system leakage and control system fluctuations can be accounted for by this 12 gpm. Leakage out of the system is represented by a negative value and leakage into the system is represented by a positive value of this flow mismatch term. Since leakage out of the system will delay the VCT high level alarm, a value of -24 gpm will be used.

9. **Volume of VCT between empty and high level** ($V_{VCT}$) - This is the amount of water which must enter the VCT in order to actuate the high level alarm assuming the volume initially is just above the low level setpoint. At 5% VCT level, the CV pump suction valves will automatically switchover from the VCT as a suction source to the RWST as a suction source, i.e., valves CV112D/E open and valves CV112B/C close (Reference 10). At 70% VCT level, the two new redundant VCT high level alarms will annunciate. The alarms are being installed as part of the modification to eliminate the automatic functions of the BDPS. The VCT volumes for these levels are given in References 11 and 12 (Figure 2). The uncertainties for these levels are given in Reference 25. The values used for this analysis are given in Table 4, "VCT Level Setpoints and Volumes." The conservative maximum

value used in this analysis for this volume is 202.00 $ft^3$.

**Table 4: VCT Level Setpoints and Volumes**

| Alarm | Percent Level | Nominal Tank Capacity ($ft^3$) Byron | Braidwood |
|---|---|---|---|
| VCT high level | 70% | 222.03 | 249.51 |
| VCT auto switchover | 5% | 54.34 | 83.89 |
| Difference | 65% | 167.69 | 165.62 |
| Uncertainty | 13.3% | 34.31 | 33.89 |
| VCT Volume | 78% | 202.00 | 199.51 |

10. **Dilution flow rate** ($Q_{dil}$) - This is the rate at which water flows from the dilution source to the CV pump suction. For the diluted water to reach the RCS, it must flow through the CV pump and the normal charging path. Therefore, the charging flow rate and dilution flow rate, as seen by the RCS, is the same. If the dilution flow rate is greater than the charging flow rate, then the VCT level rises faster without resulting in more rapid RCS dilution. This means that the $Q_{dil}$ term in equation (2) will increase, increasing the rate at which the VCT will reach the alarm setpoint. During this time, $Q_{dil}$ in equation (4) will stay the same as $Q_c$. For this reason, $Q_{dil}$ is limited by the charging flow rate, and is always less than or equal to $Q_c$. The most conservative dilution flow rate, used in this analysis, is equal to the charging flow rate of 168 gpm.

11. **Initial boron concentration** ($C_{bi}$) - This is the initial boron concentration in the reactor core when dilution occurs. This is a cycle specific value and is based on shutdown margin requirements. The ratio of the initial to critical boron concentration is used to limit the reactor core design. No specific value of the initial boron concentration is used for the calculations here, but will be used when the acceptance criteria generated in this calculation are applied to an actual reactor core design. Examples of this are given in Appendix C - Cycle Specific Analysis.

12. **Critical boron concentration** ($C_{bc}$) - This is the boron concentration at which the reactor core will be critical with the most reactive control rod withdrawn from the reactor core. It is also a cycle specific value. The ratio of the initial to critical boron concentration is used to limit the reactor core design. No specific value of the critical boron concentration is used for the calculations here, but will be used when the acceptance criteria generated in this calculation are applied to an actual reactor core design. Examples of this are given in Appendix C - Cycle Specific Analysis.

13. **Time of valve swap over** ($t_{swap}$) - This is the sum of the opening time of the RWST isolation valves (CV112D and CV112E) and the closing of the VCT isolation valves (CV112B and CV112C). This value is solely based on valve stroke times. Based on Braidwood Station and Byron Station procedures (Reference 13), the valve swapover times are verified to be less than or equal to 25 seconds. For this analysis, a valve swap over time of 25 seconds (i.e., 0.417 minutes) is used, which bounds Reference 2.

Note that additional design inputs, specific to the coincident RCS cooldown case, are listed in Appendix B - Coincident RCS Cooldown.

## References


1. U.S. Nuclear Regulatory Commission Standard Review Plan, NUREG-0800, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)," Revision 1, July 1981.

2. Westinghouse Calculation CN-TA-93-230, "Byron/Braidwood Boron Dilution for 24% SGTP," November 1993.

3. Byron/Braidwood Technical Specification 3.4.13, "Operational Leakage," Amendments 111 and 104.

4. Westinghouse Calculation SAE/FSE-C-CAE/CBE/CCE/CDE-0128, "Reactor Coolant Masses," Revision 1, December 1997.

5. BWI Calculation 222-7720-A19, "Byron/Braidwood RSG Design Data," Revision 4, December 1997.

6. Westinghouse Document WTD-TH-79-021 (Revision 2) and WNEP 8382 (Revision 1), "Model D5 Steam Generator Thermal Design Data Report," March 1988.

7. Westinghouse Safety Analysis Standard 4, "Uncontrolled Boron Dilution," Revision 2, December 1995.

8. Not used.

9. Westinghouse Letter NATD-FSD&A-91-1644, "Byron/Braidwood Boron Dilution Accident Analysis Input," September 11, 1991.

10. Byron/Braidwood Annunciator Response, "CHG LINE FLOW HIGH LOW."

| Annunciator Response | Revision |
|---|---|
| BAR 1-9-D3 | 3 |
| BAR 2-9-D3 | 3 |
| BwAR 1-9-D3 | 6 |
| BwAR 2-9-D3 | 6 |

11. Byron Procedure BTC 1.12, "Volume Control Tanks 1/2CV01T," Revision 0.

12. Braidwood Procedure 1/2BwOSR 3.4.13.1, Revision 0E1.

13. Byron/Braidwood Procedures

| Procedure | Revision |
|---|---|
| 1BOSR 0.5-2.CV.2 | 2 |
| 1BOSR 0.5-2.CV.2-2 | 3 |
| 2BOSR 0.5-2.CV.2 | 4 |
| 2BOSR 0.5-2.CV.2-2 | 4 |
| 1BwOSR 5.5.8.CV-5A | 0 |
| 1BwOSR 5.5.8.CV-5B | 0E1 |
| 1BwOSR 5.5.8.CV-15 | 0 |
| 2BwOSR 5.5.8.CV-5A | 0 |
| 2BwOSR 5.5.8.CV-5B | 0E1 |
| 2BwOSR 5.5.8.CV-15 | 0 |

14. NDIT NFM9900184, "Input to Performance Capability Parameters for the B/B Power Uprate Project," Sequence 0, September 27, 1999.

15. Westinghouse Calculation CN-TA-92-170, "CAE/CDE Boron Dilution in Modes 3, 4 and 5," Revision 0, July 1992.

16. Byron/Braidwood Annunciator Response, "VCT LEVEL HIGH LOW."

| Annunciator Response | Revision |
|---|---|
| BAR 1-9-A2 | 2 |
| BAR 2-9-A2 | 2 |
| BwAR 1-9-A2 | 7 |
| BwAR 2-9-A2 | 7 |

17. Reactor Coolant Pump Vendor Manual, Spec No. 2702-125, VETIP No. L-0041.

18. NFM Calculation SP-09, "BR1C8 SPIL - Boron Dilution at Power, Startup, Hot Standby, and Hot Shutdown," Revision 0, April 1998.

19. NFM Calculation SP-09, "BR2C8 SPIL - Boron Dilution in Modes 1, 2, 3, and 4," Revision 0, January 1999.

20. NFM Calculation SP-18, "BY1C9 SPIL - Boron Dilution in Modes 1, 2, 3, and 4," Revision 0, December 1997.

21. NFM Calculation SP-18, "BY2C8 SPIL - Boron Dilution in Modes 1, 2, 3, and 4," Revision 0, January 1998.

22. NFM Letter BR2C8/005, "Independent Review of the Braidwood 2 Cycle 8 Design SPIL Neutronics Calculations," February 23, 1999.

23. NDIT BRW-DIT-99-0119, "Charging Flow Rate Uncertainty (FI-0121A)," September 16, 1999.

24. NDIT BYR-99-245, "Charging Flow Rate Uncertainty (FI-0121A)," December 7, 1999.

25. NDIT BRW-DIT-99-0156, "Chemical Volume Control Tank Level Instrument Error," December 17, 1999.

26. Byron/Braidwood Procedure, "Plant Shutdown and Cooldown."

| Annunciator Response | Revision |
|---|---|
| 1BGP 100-5 | 31 |
| 2BGP 100-5 | 23 |
| 1BwGP 100-5 | 22 |
| 2BwGP 100-5 | 17 |

## Calculations

The calculation of the ratio of the initial to critical boron concentration, to meet Criterion 1 (Reference 1), are shown in Table 5, "Calculation of Initial to Critical Boron Concentration Required to Meet Criterion 1, Unit 1," for the limiting temperatures in Modes 3, 4 and 5 for Unit 1.

**Table 5: Calculation of Initial to Critical Boron Concentration Required to Meet Criterion 1, Unit 1**

| Mode | Units | 3 | 4 | 5 | 5c | Reference |
|---|---|---|---|---|---|---|
| $V_{RCS}$ | $ft^3$ | 10653 | 10653 | 10653 | 10653 | design input 1 |
| $T_{RCS}$ | °F | 557 | 350 | 200 | 40 | design input 2 |
| $v_{RCS}$ | $ft^3/lb_m$ | 0.021977 | 0.017989 | 0.016637 | 0.016019 | design input 3 |
| $V_{purge}$ | $ft^3$ | 66.952 | 66.952 | 59.248 | 59.248 | design input 4 |
| $T_{purge}$ | °F | 40 | 40 | 40 | 40 | design input 5 |
| $v_{purge}$ | $ft^3/lb_m$ | 0.016019 | 0.016019 | 0.016019 | 0.016019 | design input 6 |
| $Q_c$ | gpm | 168 | 168 | 168 | 168 | design input 7 |
| $Q_m$ | gpm | -24 | -24 | -24 | -24 | design input 8 |
| $V_{VCT}$ | $ft^3$ | 202 | 202 | 202 | 202 | design input 9 |
| $Q_{dil}$ | gpm | 168 | 168 | 168 | 168 | design input 10 |
| $t_{swap}$ | min | 0.417 | 0.417 | 0.417 | 0.417 | design input 13 |
| $t_{purge}$ | min | 2.98 | 2.98 | 2.64 | 2.64 | equation (3) |
| $t_{fill}$ | min | 10.49 | 10.49 | 10.49 | 10.49 | equation (2) |

**Criterion 1**

| | | | | | | |
|---|---|---|---|---|---|---|
| $t_{op15}$ | min | 15 | 15 | 15 | 15 | limiting value |
| $t_{crit}$ | min | 28.89 | 28.89 | 28.55 | 28.55 | equation (1) |
| **$C_b/C_{bc}$** | **-** | **1.087** | **1.071** | **1.064** | **1.062** | **equation (5)** |

The calculation of the ratio of the initial to critical boron concentration, to meet Criterion 1 (Reference 1), are shown in Table 6, "Calculation of Initial to Critical Boron Concentration Required to Meet Criterion 1, Unit 2," for the limiting temperatures in Modes 3, 4 and 5 for Unit 2.

**Table 6: Calculation of Initial to Critical Boron Concentration Required to Meet Criterion 1, Unit 2**

| Mode | Units | 3 | 4 | 5 | 5c | Reference |
|---|---|---|---|---|---|---|
| $V_{RCS}$ | $ft^3$ | 9331 | 9331 | 9331 | 9331 | design input 1 |
| $T_{RCS}$ | °F | 557 | 350 | 200 | 40 | design input 2 |
| $v_{RCS}$ | $ft^3/lb_m$ | 0.021977 | 0.017989 | 0.016637 | 0.016019 | design input 3 |
| $V_{purge}$ | $ft^3$ | 66.952 | 66.952 | 59.248 | 59.248 | design input 4 |
| $T_{purge}$ | °F | 40 | 40 | 40 | 40 | design input 5 |
| $v_{purge}$ | $ft^3/lb_m$ | 0.016019 | 0.016019 | 0.016019 | 0.016019 | design input 6 |
| $Q_c$ | gpm | 168 | 168 | 168 | 168 | design input 7 |
| $Q_m$ | gpm | -24 | -24 | -24 | -24 | design input 8 |
| $V_{VCT}$ | $ft^3$ | 202 | 202 | 202 | 202 | design input 9 |
| $Q_{dil}$ | gpm | 168 | 168 | 168 | 168 | design input 10 |
| $t_{swap}$ | min | 0.417 | 0.417 | 0.417 | 0.417 | design input 13 |
| $t_{purge}$ | min | 2.98 | 2.98 | 2.64 | 2.64 | equation (3) |
| $t_{fill}$ | min | 10.49 | 10.49 | 10.49 | 10.49 | equation (2) |
| **Criterion 1** | | | | | | |
| $t_{op15}$ | min | 15 | 15 | 15 | 15 | limiting value |
| $t_{crit}$ | min | 28.89 | 28.89 | 28.55 | 28.55 | equation (1) |
| $C_b/C_{bc}$ | - | 1.100 | 1.081 | 1.074 | 1.071 | equation (5) |

## Summary and Conclusions

The initial to critical boron concentration ratio, for all reactor core designs must exceed that calculated here for either Criterion 1 (Reference 1) or Criterion 2 for each of Modes 3, 4 and 5. Table 5 gives the limiting ratios for Modes 3, 4 and 5 to be, for Unit 1:

Mode 3 = 1.087
Mode 4 = 1.071
Mode 5 = 1.064

Table 6 gives the limiting ratios for Unit 2:

Mode 3 = 1.100
Mode 4 = 1.081
Mode 5 = 1.074

In addition, the calculations performed in Appendix B - Coincident RCS Cooldown require that the benefit received from removing the stuck control rod penalty must be at least 40 ppm boron concentration. This will ensure that the RCS cooldown case is less limiting than the case presented in the main calculation.

This analysis requires that a new VCT high level alarm be installed to annunciate when the VCT level reaches 70%. The two new redundant alarms will ensure that the plant operators will be able to diagnose the dilution event in Modes 3, 4 and 5. In addition, this analysis assumes that at least one RCP is in operation, and all loop stop valves are open in Modes 3, 4 and 5. If at least one RCP is not in operation, or any loop stop valves are closed, the plant must administratively isolate the potential source of the dilution.

## Appendix A - Slow Dilution

It is possible that, in the case of a very slow dilution, the VCT level may not increase fast enough to reach the high VCT level setpoint, and warn the plant operators that a dilution is occurring. In this case, the boric acid flow deviation and primary water flow deviation alarms are available to alert the plant operators of the dilution. In addition, the alarms from the existing BDPS will still be available, without any associated automatic actions. Therefore, it is reasonable to apply Criterion 2 for this case.

To show that Criterion 2 is easily met for a slow dilution, this additional calculation is performed. This calculation follows the same general methodology as described for the Criterion 1 calculation in the main calculation. Now, the plant operator response time is 30 minutes ($t_{op30}$), and the time to fill the VCT is not included in the calculation of the initial to critical boron concentration. To meet Criterion 2 we need to show that:

$$t_{op30} \leq t_{crit30} - t_{swap} - t_{purge} \qquad \text{(A1)}$$

Where $t_{crit30}$ is the same as $t_{crit}$ in the main calculation, but without the time required to fill the VCT. In all other respects, the calculation of this ratio is the same as in equation (5), so using equation (A1), we can find the limiting ratio of the initial to critical boron concentration:

$$\frac{C_{bi}}{C_{bc}} = \exp\left( \frac{t_{op30} + t_{swap} + t_{purge}}{\left( \frac{V_{RCS} \cdot v_{purge}}{Q_{dil} \cdot v_{RCS}} \right)\left( \frac{7.481\,\text{gal}}{1\,\text{ft}^3} \right)} \right) \qquad \text{(A2)}$$

All input parameters for this case are the same as in the main calculation, except that the dilution flow is greatly reduced. In addition, the charging flow is reduced to the same value as the dilution flow. A charging flow higher than the dilution flow will result in a smaller purge time, which will allow the plant operators more time from the time of the event initiation to the time of loss of shutdown margin. The charging flow cannot be less than the dilution flow rate, since this would result in filling of the VCT without any additional dilution of the RCS.

The RCS volume used here is the Unit 2 RCS volume. This is the limiting volume, the greater volume of Unit 1 would make the ratio required to prevent criticality lower, and is bounded by the Unit 2 volume.

A dilution flow rate of 25 gpm is chosen for this calculation, since it results in the highest required ratio of initial to critical boron concentration to meet Criterion 1. This value is found by iteration. The results of this calculation are given in Table 7, "Calculation of Initial to Critical Boron Concentration Required to Meet Criterion 1 and Criterion 2 for Slow Dilution Case." It can be seen that the ratio of initial to critical boron concentration required to meet Criterion 2 is much lower than that required to meet Criterion 1 in Table 5. This relationship is true for all values of dilution flow rate low enough to prevent the initiation of the high VCT level alarm before the plant operators can act to prevent criticality within the bounds of Criterion 1. Therefore, there are no new acceptance criteria needed to prevent criticality in the event of a slow dilution, in Modes 3, 4 and 5.

**Table 7: Calculation of Initial to Critical Boron Concentration Required to Meet Criterion 1 and Criterion 2 for Slow Dilution Case**

| Mode | Unit | 3 | 4 | 5 | 5c | Reference |
|---|---|---|---|---|---|---|
| $V_{RCS}$ | $ft^3$ | 9331 | 9331 | 9331 | 9331 | design input 1 |
| $T_{RCS}$ | F | 557 | 350 | 200 | 40 | design input 2 |
| $v_{RCS}$ | $ft^3/lb_m$ | 0.021977 | 0.017989 | 0.016637 | 0.016019 | design input 3 |
| $V_{purge}$ | $ft^3$ | 66.952 | 66.952 | 59.248 | 59.248 | design input 4 |
| $T_{purge}$ | F | 40 | 40 | 40 | 40 | design input 5 |
| $v_{purge}$ | $ft^3/lb_m$ | 0.016019 | 0.016019 | 0.016019 | 0.016019 | design input 6 |
| $Q_c$ | gpm | 25 | 25 | 25 | 25 | equal to dilution flow |
| $Q_m$ | gpm | -24 | -24 | -24 | -24 | design input 8 |
| $V_{VCT}$ | $ft^3$ | 202 | 202 | 202 | 202 | design input 9 |
| $Q_{dil}$ | gpm | 25 | 25 | 25 | 25 | most limiting value |
| $t_{swap}$ | min | 0.417 | 0.417 | 0.417 | 0.417 | design input 13 |
| $t_{purge}$ | min | 20.03 | 20.03 | 17.73 | 17.73 | equation (3) |
| $t_{fill}$ | min | 1511.16 | 1511.16 | 1511.16 | 1511.16 | equation (2) |
| | | | | **Criterion 1** | | |
| $t_{op15}$ | min | 15.00 | 15.00 | 15.00 | 15.00 | limiting value |
| $t_{crit}$ | min | 1546.61 | 1546.61 | 1544.31 | 1544.31 | equation (1) |
| **$C_b/C_{bc}$** | **-** | **2.138** | **1.863** | **1.776** | **1.739** | **equation (5)** |
| | | | | **Criterion 2** | | |
| $t_{op30}$ | min | 30.00 | 30.00 | 30.00 | 30.00 | limiting value |
| $t_{crit30}$ | min | 50.45 | 50.45 | 48.15 | 48.15 | equation (A1) |
| **$C_b/C_{bc}$** | **-** | **1.025** | **1.021** | **1.018** | **1.017** | **equation (A2)** |

## Appendix B - Coincident RCS Cooldown

As discussed previously, if a dilution event occurs simultaneously with an RCS cooldown, the high VCT level (70%) alarm may not be annunciated since the RCS shrinkage would cause an increase in the charging flow rate. In this case, compliance with Criterion 2 must be shown to ensure that criticality will not be reached before the dilution can be terminated.

The following calculation shows that, for the range of potential RCS critical boron concentrations, Criterion 2 will always be met for the coincident RCS cooldown case, as long as the critical boron concentration at the initial steady state conditions is sufficient to satisfy Criterion 1.

### *Methodology*

For the case of a boron dilution on Modes 3, 4 and 5, with a concurrent RCS cooldown, additional variables, listed in Table 8, "Additional Variable Definitions for Cooldown Analysis," are used.

**Table 8: Additional Variable Definitions for Cooldown Analysis**

| Type | Variable | Units | Definition | Conservativ Direction |
|---|---|---|---|---|
| input | $dT_{RCS}/dt$ | °F/hr | Cooldown rate of RCS | most negative |
| calc | $T_{targ30}$ | °F | Target RCS temperature | minimum |
| calc | $t_{op30}$ | min | Gross operator response time | N/A |
| calc | $t_{crit30}$ | min | Time to dilute RCS from initial to critical boron concentration with no withdrawn rods | maximum |
| calc | $C_b/(C_{bc}\text{-}40)$ | N/A | Ratio of initial boron concentration to critical boron concentration | maximum |

To meet Criterion 2, the plant operators have a total of 30 minutes to recognize, and terminate the dilution event, from the time that the dilution is initiated until the time of a loss of shutdown margin. This is similar to the net plant operator response time for Criterion 1, except that this value does not consider the amount of time it takes to reach the high level VCT alarm. This value is:

$$t_{op30} \le t_{crit30} - t_{swap} - t_{purge} \tag{B1}$$

Criterion 2 is used for the case with a concurrent RCS cooldown. For this case, the calculation of the time to dilute from the initial to critical boron concentration is basically the same as equation (4), except that the critical boron concentration is decreased by 40 ppm (see assumption B2 below). The time to reach the critical boron concentration can then be written:

$$t_{crit30} = \frac{V_{RCS}\left(\frac{v_{purge}}{v_{RCS}}\right)}{Q_{dil}}\left(\frac{7.48\text{ gal}}{1\text{ ft}^3}\right)\ln\left(\frac{C_{bi}}{C_{bc} - 40\text{ ppm}}\right) \tag{B2}$$

Combining equations (B1) and (B2), the following relationship can be used for calculation of the limiting value of Criterion 2:

$$\frac{C_{bi}}{C_{bc} - 40} = \exp\left(\frac{t_{op30} + t_{swap} + t_{purge}}{\left(\frac{V_{RCS} \cdot v_{purge}}{Q_{dil} \cdot v_{RCS}}\right)\left(\frac{7.481\text{ gal}}{1\text{ ft}^3}\right)}\right) \tag{B3}$$

Maximizing these ratios will be the most conservative, so the design inputs will be chosen to maximize these ratios.

The next step is to calculate the ratio in equation (B3) for the possible range of critical boron concentrations of the steady state conditions. Since the critical and initial boron concentrations can change as the RCS temperature changes, it is necessary to calculate a target temperature to estimate the critical boron concentration when the transient is terminated. This temperature is calculated based on the assumption that the event is terminated within 30 minutes of initiation (Criterion 2). The target temperature is:

$$T_{t\arg 30} = T_{RCS} + \frac{dT_{rcs}}{dt}\left(\frac{t_{swap} + t_{purge} + t_{op30}}{60\text{ min/hr}}\right) \tag{B4}$$

Based on the limits established for the dilution cases in the main calculation (Table 5), and assuming a critical boron concentration (see assumption B3 below), the initial boron concentration can be calculated from:

$$C_{bi} = \left(\frac{C_{bi}}{C_{bc}}\right)_{\lim} C_{bc} \tag{B5}$$

The initial and critical boron concentrations at the end of the event are calculated by linearly interpolating using the target temperature, based on the critical and initial boron concentrations calculated from equation (B5). The following values are used for this interpolation for:

**Unit 1**

| | | | | | |
|---|---|---|---|---|---|
| $T_{RCS}$ | F | 557 | 350 | 200 | 40 |
| $C_{bc}$ | ppm | 2000.0 | 2000.0 | 2000.0 | 2000.0 |
| $C_{bi}$ | ppm | 2174.3 | 2141.6 | 2129.0 | 2124.1 |

**Unit 2**

| | | | | | |
|---|---|---|---|---|---|
| $T_{RCS}$ | F | 557 | 350 | 200 | 40 |
| $C_{bc}$ | ppm | 2000.0 | 2000.0 | 2000.0 | 2000.0 |
| $C_{bi}$ | ppm | 2200.2 | 2162.4 | 2147.9 | 2142.3 |

These tables assume that the critical boron concentration is constant for Modes 3, 4 and 5 temperatures. While this is not necessarily the case, for this calculation, this assumption results in the lowest ratio of initial to critical boron concentrations.

Using these values for the initial and critical boron concentrations, the temperature specific value of the ratio, including the benefit of removing the stuck rod penalty is:

$$\begin{pmatrix} \text{temperature specifc ratio} \\ \text{of boron concentration} \end{pmatrix} = \frac{C_{bi}}{(C_{bc} - 40)} \quad \text{(B6)}$$

This value can then be compared to the ratio required to meet Criterion 2.

## ***Assumptions***

For the calculation with a concurrent RCS cooldown only, two assumptions (in addition to those given in the main calculation) are used:

B1. The RCS temperature decreases at a constant rate between 0 and 100°F/hr.

B2. The cooldown will always occur with the reactor trip breakers open and all control rods inserted. Since the critical boron concentration used in the main calculation includes a penalty for one control rod not inserted into the reactor core, and since this penalty is not required with the trip breakers open, an additional benefit can be incorporated into the analysis. This benefit is a 40 ppm reduction in the critical boron concentration. For shutdown margin calculations, a stuck control rod is worth at least 700 pcm. The differential boron worth is not more negative than -16 pcm/ppm for temperature over 300°F, typical values of which can be seen in curvebook calculations. Assuming these values, a conservative value of 40 ppm for the stuck control rod penalty equivalent can be assumed.

B3. The maximum initial critical boron concentration is 2000 ppm. This value is assumed in this analysis based on previous design results, which shows that for Modes 3, 4 and 5, the critical boron concentration is normally between 1100 and 1400 ppm. A value of 2000 ppm allows for a large margin over these values.

***Design Inputs***

Also, for the case with concurrent RCS cooldown, several other design inputs are required:

B4. **Initial temperature of RCS mixing volume** ($T_{RCS}$) - This is the temperature of the RCS and determines the density and mass of the RCS as in the constant RCS temperature case. For the concurrent RCS cooldown case, a range of initial temperatures are used from 300°F to 557°F. Temperature below 300°F will not have a significantly fast cooldown, since it is below the RH System temperature, at which point, the RH System is used for cooldown and the SGs are no longer used.

B5. **Specific volume of water in RCS** ($v_{RCS}$) - This is the saturated water specific volume found from the steam tables. This value is dependent on RCS temperature. The values used for this calculation are given below:

**Table 9: Specific Volume of Saturated Water**

| Temperature (°F) | Specific Volume ($ft^3/lb_m$) |
|---|---|
| 557 | 0.021977 |
| 500 | 0.020434 |
| 450 | 0.019432 |
| 400 | 0.018640 |
| 350 | 0.017989 |
| 300 | 0.017453 |

B6. **Cooldown rate of RCS** ($dT_{RCS}/dt$) - This is the rate of change of the RCS temperature during the cooldown. This cooldown results in a decrease in RCS volume, which may mask the mass addition of the dilution. Therefore it is conservative to assume the most negative value (negative signifies cooldown as opposed to RCS heatup). This value is set by Technical Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," (via PTLR) to -100°F/hr (Reference 26).

## Results

The results of this calculation are summarized in Table 10, "Calculation of Initial to Critical Boron Concentration for RCS Cooldown Case, Unit 1," for Unit 1, and Table 11, "Calculation of Initial to Critical Boron Concentration for RCS Cooldown Case, Unit 2," for Unit 2. These results show that, with the most limiting combination of critical boron concentrations, up to 2000 ppm, the ratio of the initial to critical boron concentration, with the penalty for a stuck control rod removed, is always greater than the ratio required to meet Criterion 2. Therefore, the case with RCS cooldown is always less limiting than the case included in the main calculation, and no additional limitations are required to prevent criticality due to a boron dilution with concurrent RCS cooldown. Note that the 40 ppm stuck control rod boron worth must be verified to ensure that this case is less limiting than that in the main calculation.

**Table 10: Calculation of Initial to Critical Boron Concentration for RCS Cooldown Case, Unit 1**

| Mode | Unit | 3 | 3 | 3 | 3 | 4 | 4 | Reference |
|---|---|---|---|---|---|---|---|---|
| $V_{RCS}$ | $ft^3$ | 10653 | 10653 | 10653 | 10653 | 10653 | 10653 | design input 1 |
| $T_{RCS}$ | F | 557 | 500 | 450 | 400 | 350 | 300 | B4 |
| $v_{RCS}$ | $ft^3/lb_m$ | 0.021977 | 0.020434 | 0.019432 | 0.018640 | 0.017989 | 0.017453 | B5 |
| $V_{purge}$ | $ft^3$ | 66.952 | 66.952 | 66.952 | 66.952 | 66.952 | 66.952 | design input 4 |
| $T_{purge}$ | F | 40 | 40 | 40 | 40 | 40 | 40 | design input 5 |
| $v_{purge}$ | $ft^3/lb_m$ | 0.016019 | 0.016019 | 0.016019 | 0.016019 | 0.016019 | 0.016019 | design input 6 |
| $Q_c$ | gpm | 168 | 168 | 168 | 168 | 168 | 168 | design input 7 |
| $t_{purge}$ | min | 2.98 | 2.98 | 2.98 | 2.98 | 2.98 | 2.98 | equation (3) |
| $dT_{RCS}/dt$ | F/min | -100 | -100 | -100 | -100 | -100 | -100 | B6 |
| $Q_{dil}$ | gpm | 168 | 168 | 168 | 168 | 168 | 168 | design input 10 |
| $t_{swap}$ | min | 0.417 | 0.417 | 0.417 | 0.417 | 0.417 | 0.417 | design input 13 |

**Criterion 2 Limiting Ratio**

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| $t_{op30}$ | min | 30 | 30 | 30 | 30 | 30 | 30 | limiting value |
| $t_{crit30}$ | min | 33.40 | 33.40 | 33.40 | 33.40 | 33.40 | 33.40 | equation (B1) |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.101 | 1.094 | 1.089 | 1.085 | 1.082 | 1.080 | equation (B3) |

**Temperature Specific Criterion 2 Ratio**

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| $T_{targ30}$ | F | 501 | 444 | 394 | 344 | 294 | 244 | equation (B4) |
| $C_{bc}$ | ppm | 2000.0 | 2000.0 | 2000.0 | 2000.0 | 2000.0 | 2000.0 | B3 |
| $C_{bi}$ | ppm | 2165.5 | 2156.5 | 2148.6 | 2141.1 | 2136.9 | 2132.7 | interpolation |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.105 | 1.100 | 1.096 | 1.092 | 1.090 | 1.088 | equation (B6) |

**Table 11: Calculation of Initial to Critical Boron Concentration for RCS Cooldown Case, Unit 2**

| Mode | Unit | 3 | 3 | 3 | 3 | 4 | 4 | Reference |
|---|---|---|---|---|---|---|---|---|
| $V_{RCS}$ | $ft^3$ | 9331 | 9331 | 9331 | 9331 | 9331 | 9331 | design input 1 |
| $T_{RCS}$ | F | 557 | 500 | 450 | 400 | 350 | 300 | B4 |
| $v_{RCS}$ | $ft^3/lb_m$ | 0.021977 | 0.020434 | 0.019432 | 0.018640 | 0.017989 | 0.017453 | B5 |
| $V_{purge}$ | $ft^3$ | 66.952 | 66.952 | 66.952 | 66.952 | 66.952 | 66.952 | design input 4 |
| $T_{purge}$ | F | 40 | 40 | 40 | 40 | 40 | 40 | design input 5 |
| $v_{purge}$ | $ft^3/lb_m$ | 0.016019 | 0.016019 | 0.016019 | 0.016019 | 0.016019 | 0.016019 | design input 6 |
| $Q_c$ | gpm | 168 | 168 | 168 | 168 | 168 | 168 | design input 7 |
| $t_{purge}$ | min | 2.98 | 2.98 | 2.98 | 2.98 | 2.98 | 2.98 | equation (3) |
| $dT_{RCS}/dt$ | F/min | -100 | -100 | -100 | -100 | -100 | -100 | B6 |
| $Q_{dil}$ | gpm | 168 | 168 | 168 | 168 | 168 | 168 | design input 10 |
| $t_{swap}$ | min | 0.417 | 0.417 | 0.417 | 0.417 | 0.417 | 0.417 | design input 13 |

**Criterion 2 Limiting Ratio**

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| $t_{op30}$ | min | 30 | 30 | 30 | 30 | 30 | 30 | limiting value |
| $t_{crit30}$ | min | 33.40 | 33.40 | 33.40 | 33.40 | 33.40 | 33.40 | equation (B1) |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.117 | 1.108 | 1.102 | 1.098 | 1.094 | 1.092 | equation (B3) |

**Temperature Specific Criterion 2 Ratio**

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| $T_{targ30}$ | F | 501 | 444 | 394 | 344 | 294 | 244 | equation (B4) |
| $C_{bc}$ | ppm | 2000.0 | 2000.0 | 2000.0 | 2000.0 | 2000.0 | 2000.0 | B3 |
| $C_{bi}$ | ppm | 2190.0 | 2179.6 | 2170.5 | 2161.9 | 2157.1 | 2152.2 | interpolation |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.117 | 1.112 | 1.107 | 1.103 | 1.101 | 1.098 | equation (B6) |

**Appendix C - Cycle Specific Analysis**

This appendix shows that, for the currently operating and previous cycles, the limiting ratio of the initial to critical boron concentration is met. For these cycles, either the BDPS limits have been met, or the BDPS is inoperable and boron dilution has been precluded by administrative controls in these Modes. However, this calculation is intended to show that for typical cycles the limiting ratio, calculated here, is feasible.

The cycle specific Mode 3 boron concentration data is taken from References 18 through 21. The Mode 4 and 5 data is taken from Reference 22.

**Mode 3**

| limiting $C_b/C_{bc}$ = | | 1.087 | (Unit 1) |
|---|---|---|---|
| | | 1.100 | (Unit 2) |
| Cycle | $C_{bc}$ (ppm) | $dC$ (ppm) | $C_b/C_{bc}$ |
| A1C08 | 1139 | 119 | 1.104 |
| A2C07 | N/A | N/A | N/A |
| A2C08 | 1131 | 139 | 1.123 |
| B1C09 | 954 | 121 | 1.127 |
| B1C10 | N/A | N/A | N/A |
| B2C08 | 1086 | 134 | 1.123 |

**Mode 4**

| limiting $C_b/C_{bc}$ = | | 1.071 | (Unit 1) |
|---|---|---|---|
| | | 1.081 | (Unit 2) |
| Cycle | $C_{bc}$ (ppm) | $dC$ (ppm) | $C_b/C_{bc}$ |
| A1C08 | 1167 | 119 | 1.102 |
| A2C07 | 1243 | 121 | 1.097 |
| A2C08 | 1191 | 123 | 1.103 |
| B1C09 | 1014 | 110 | 1.108 |
| B1C10 | 1107 | 117 | 1.106 |
| B2C08 | 1107 | 121 | 1.109 |

**Mode 5**

| limiting $C_b/C_{bc}$ = | | 1.064 | (Unit 1) |
|---|---|---|---|
| | | 1.074 | (Unit 2) |
| Cycle | $C_{bc}$ (ppm) | $dC$ (ppm) | $C_b/C_{bc}$ |
| A1C08 | 1262 | 115 | 1.091 |
| A2C07 | 1301 | 117 | 1.090 |
| A2C08 | 1324 | 121 | 1.091 |
| B1C09 | 1093 | 107 | 1.098 |
| B1C10 | 1201 | 116 | 1.097 |
| B2C08 | 1216 | 118 | 1.097 |

Based on this data, the most recent Braidwood Station and Byron Station cycle designs meet the acceptance criteria specified here. The following table repeats the calculation done for the concurrent RCS cooldown case with the cycle specific numbers above. The ratio of initial to critical boron concentration, with the stuck control rod penalty assumed to be 40 ppm removed, easily meets that ratio required, as calculated in Appendix B - Coincident RCS Cooldown.

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Limiting Ratio** | | | | | | | |
| $C_{bi}/(C_{bc} - 40)$ | Unit 1 | 1.101 | 1.094 | 1.089 | 1.085 | 1.082 | 1.080 |
| | Unit 2 | 1.117 | 1.108 | 1.102 | 1.098 | 1.094 | 1.092 |
| | | | | | | | |
| A1C08 | | | | | | | |
| $T_{targ30}$ | F | 501 | 444 | 394 | 344 | 294 | 244 |
| $C_{bc}$ | ppm | 1146.5 | 1154.2 | 1161.0 | 1170.6 | 1202.3 | 1233.9 |
| $C_{bi}$ | ppm | 1265.5 | 1273.2 | 1280.0 | 1289.4 | 1319.8 | 1350.1 |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.144 | 1.143 | 1.142 | 1.141 | 1.136 | 1.131 |
| | | | | | | | |
| A2C08 | | | | | | | |
| $T_{targ30}$ | F | 501 | 444 | 394 | 344 | 294 | 244 |
| $C_{bc}$ | ppm | 1147.1 | 1163.7 | 1178.1 | 1196.0 | 1240.4 | 1284.7 |
| $C_{bi}$ | ppm | 1273.1 | 1288.5 | 1302.0 | 1318.9 | 1362.6 | 1406.3 |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.150 | 1.147 | 1.144 | 1.141 | 1.135 | 1.130 |
| | | | | | | | |
| B1C09 | | | | | | | |
| $T_{targ30}$ | F | 501 | 444 | 394 | 344 | 294 | 244 |
| $C_{bc}$ | ppm | 970.1 | 986.7 | 1001.1 | 1017.0 | 1043.3 | 1069.6 |
| $C_{bi}$ | ppm | 1101.3 | 1109.9 | 1117.4 | 1126.9 | 1152.2 | 1177.5 |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.184 | 1.172 | 1.163 | 1.153 | 1.148 | 1.144 |
| | | | | | | | |
| B2C08 | | | | | | | |
| $T_{targ30}$ | F | 501 | 444 | 394 | 344 | 294 | 244 |
| $C_{bc}$ | ppm | 1091.6 | 1097.4 | 1102.5 | 1111.1 | 1147.4 | 1183.8 |
| $C_{bi}$ | ppm | 1222.2 | 1224.4 | 1226.3 | 1232.0 | 1267.3 | 1302.7 |
| $C_{bi}/(C_{bc} - 40)$ | - | 1.162 | 1.158 | 1.154 | 1.150 | 1.144 | 1.139 |

# ENCLOSURE

## WESTINGHOUSE NSAC-183
## RISK OF PWR INADVERTENT CRITICALITY DURING SHUTDOWN AND REFUELING

# NSAC/183

December 1992

# Risk of PWR Inadvertent Criticality During Shutdown and Refueling

## EPRI Outage Risk Assessment and Management (ORAM) Program

Prepared by
Westinghouse Electric Corporation

The Nuclear Safety Analysis Center is operated for
the electric utility industry by the Electric Power Research Institute

# Risk of PWR Inadvertent Criticality During Shutdown and Refueling

## EPRI Outage Risk Assessment and Management (ORAM) Program


NSAC-183

Final Report, December 1992



Prepared by

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## NSAC PERSPECTIVE

This report describes an assessment of the probability and consequences of PWR reactivity events that could occur during shutdown conditions. The intent of the assessment was to identify generic insights that can be used to support the development of PWR shutdown risk management and contingency planning guidelines. This assessment also provided valuable information to recent PWR probabilistic shutdown safety assessments (NSAC-176L) that have been performed by EPRI. All of these assessments are being performed as part of EPRI's Outage Risk and Management (ORAM) program.

The scope of this assessment included the probability and consequences of reactivity events during reactor shutdown, a review of existing design and administrative controls to preclude such events, and identification of other actions that may be helpful in further reducing their possibility. The assessment benefitted from the use of industry operating experience including utility surveys, and the use of ongoing research and experimentation.

The following types of reactivity events were included in this assessment:

- Gradual boron dilution events, in which the boron concentration in the reactor coolant system is gradually and somewhat uniformly reduced.
- Rapid boron dilution events, in which a relatively nonborated pocket of water in the reactor coolant system is swept into the core.
- Core loading errors during refueling.

In this assessment no events were found that involved inadvertent PWR criticality during shutdown conditions. This is based on the review of operating experience. Also, with the possible exception of rapid boron dilution, reactivity events do not appear to be significant contributors to PWR risk during shutdown operations.

The following are highlights of the results and insights for each of the types of events included in this assessment:

### Gradual Boron Dilution

- The frequency of gradual boron dilution events during shutdown conditions in U.S. PWRs has decreased by at least a factor of three from pre-1985 experience.
- Inadvertent criticality resulting from this type of event is estimated as less than 1E-4/reactor-year. Criticality is limited by void generation, with an expected power level too low to cause fuel damage.

### Rapid Boron Dilution

- There have been no reported events in which a pocket of unborated water has been swept into the core. However, there are some mechanisms in which pockets of unborated water can be introduced into the RCS, as indicated by a few precursor events that occurred in the U.S. before 1983.

- One way that rapid boron dilution could occur ("the Swedish scenario") appears to warrant precautions to ensure it has only a very low probability. Many of these precautions are already common U.S. practice.

**Fuel Loading Errors**

- PWR fuel loading errors are uncommon, but have occurred. Criticality from loading errors would require a cluster of 3 to 4 unrodded fresh assemblies. This can easily be prevented by experienced personnel who are observing the core loading pattern.

- The probability of inadvertent criticality from fuel loading errors is very low, and the results of this assessment indicate that the probability of core damage is insignificant. If core damage did occur, having the containment closed ensures public safety and a very low probability of offsite releases. Worker safety however would not be assured.

As a result of this assessment, several recommendations were developed to ensure that there is a low probability of these types of events. These recommendations are included in the report, and will also be included in the shutdown risk management and contingency planning guidelines currently under development by EPRI.

S. Pal Kalra
NSAC, Project Manager

## ABSTRACT

This report addresses the likelihood and consequences of reactivity accidents using PWR shutdown and refueling operations, and makes recommendations to reduce the possibility of their occurrence. Both criticality and core damage are addressed for several postulated mechanisms, including gradual boron dilution (in which the boron concentration of the entire core decreases relatively slowly), rapid boron dilution (in which boron concentration changes rapidly in both time and space), and refueling errors (involving loading fuel assemblies into improper locations). Available studies, scoping analyses, and engineering judgement are used to estimate the frequency and probable effects of these reactivity events. In addition, a refueling practices survey of PWR utilities is reported.

## ACKNOWLEDGEMENTS

During generation of this report, we had the opportunity to consult with numerous experts in various fields. Their input was a major factor in shaping the conclusions and recommendations in this report. Particular thanks are due:

To Sven Jacobson of Swedish State Power Board (Ringhals) for numerous consultations regarding rapid boration dilution;

To David Diamond and H. Nourbakhsh of Brookhaven National Laboratory for discussions and insights regarding both rapid boron dilution and fuel misloading events;

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And to the many and varied plant personnel -- including reactor operators and other members of plant operations staffs, outage planners, and service managers -- that have participated in discussions with us.

CONTENTS

## ILLUSTRATIONS

TABLES

# 1.0 INTRODUCTION AND SUMMARY

## 1.1 Scope

The scope of this report is to assess the probability and consequences of reactivity events during reactor shutdown, to review existing administrative controls in place to prevent such events, and identify other actions which may be helpful in further reducing their possibility.

The following reactivity events were evaluated:

- Gradual boron dilution events, in which the boron concentration in the active portions of the reactor coolant system is nearly uniformly reduced, (section 2)

- Rapid boron dilution events, in which a relatively unborated volume of water is swiftly swept into the core, causing localized reactivity increases, (section 3)

- Core loading errors during refueling (section 4)

Reactivity events that were discarded from consideration were:

- Events with frequencies estimated as less that 1E-7/reactor-year in NUREG/CR-5368, ("Reactivity Accidents", 1990); e.g., discharge of a diluted accumulator

- Reactivity events associated with recovery from an accident condition; e.g., ECC recirculation with diluted sump)

- Reactivity events caused by core design error; e.g., improper core loading specification

- Reactivity events during power operation.

In addition, a refueling practices survey of PWR utilities was completed, and the results are reported (Appendix 4A).

## 1.2 Style

In several areas, there are significant uncertainties in the physical phenomena and course of events. Engineering judgement, based on discussions with knowledgeable individuals, was used where believed appropriate. Where the judgements being expressed are our own, rather than reflecting a consensus, we've used the first person; i.e., "we believe" rather than "it is believed".

## 1.3 Summary of Conclusions and Recommendations

Since soluble boron is used to control reactivity in PWRs, a relatively large reactivity shutdown margin is attainable. The minimum prescribed shutdown in PWRs is typically 5% (k-eff of 0.95) during refueling, and the actual shutdown margin is usually higher. This large shutdown margin provides considerable safety margin against inadvertent criticality as a result of fuel loading errors, control rod misplacements, or temperature changes. On the other hand, PWRs have the potential for reactivity addition due to boron dilution events.

No cases have been found in operating experience of inadvertent criticality in a PWR during shutdown or refueling operations.

### Gradual Boron Dilution

A gradual boron dilution event occurs when the boron concentration in the active portions of the Reactor Coolant System (RCS) is nearly uniformly reduced due to operator action or equipment malfunction. The frequency of gradual boron dilution events in recent years during shutdown operation of U.S. PWRs is now down to no greater th 0.02/reactor year, down by at least a factor of three from pre-1985 experience. Criticality has not occurred from any of the boron dilution events to date, and the estimated frequency of inadvertent criticality due to boron dilution is less than 1E-4/reactor year. Criticality would cause low power generation, too low to cause core damage, so gradual boron dilution events are not considered a significant contributor to core damage.

The gradual boron dilution events that occurred were caused by human performance problems, equipment malfunctions, or steam generator tube sheet decontamination and tube removal/plugging activities. No cases of boron dilution caused by maintenance related valve misalignment have been reported.

### Rapid Boron Dilution

A rapid boron dilution event is defined as a relatively unborated volume of water being swept into the core, causing a local reactivity perturbation. One example would be Reactor Coolant Pump (RCP) startup with stagnant and unborated water in the RCP suction.

Rapid reduction in local core boron concentration is a low frequency event --no such case has been reported. Estimates for important scenarios range from 1E-4 to 1E-7 per reactor-year.

In the so-called Swedish scenario (section 3.3.2), clean water is inadvertently introduced into the RCP suction during a refueling outage (such as by improper steam generator maintenance), and is swept into the reactor core when the RCPs are jogged as part of the filling and venting operation prior to RCS heatup. Reactivity insertion rates in excess of $10 per second have been calculated, and higher rates appear possible. No recent three-dimensional transient nuclear-thermal-hydraulic analyses involving power generation have yet been reported, and such analyses are subject to considerable uncertainty, particularly regarding boron transport and mixing in turbulent flow through complex geometry. Existing analyses are inconclusive as to whether severe core damage (expulsion of molten fuel into the core coolant with attendant pressure waves) could result for limiting cases. In our opinion, this scenario cannot be proven to be generically acceptable (i.e., no severe core damage). Indeed, we believe cases can probably be found that can be proven to have unacceptable results. Therefore, we recommend that precautions be taken to ensure a very low probability of occurrence. Precautions regarding boration and dilution are listed in section 3.6, and do not appear unduly restrictive (e.g., don't dilute unless at least one RCP is running). Most of these precautions are common U. S. practice. Some plants may be implementing all of them. The absence of these dilution events since 1983 in the US suggests that steam generator maintenance practices have improved markedly. We did not review maintenance practices and requirements, and have no conclusions as to whether further improvement would be desirable.

There have been at least eight precursor events in which some volume of clean water has been inadvertently put into the RCS as a result of faulty steam generator maintenance (either secondary water through tube leaks or leaks in nozzle dams) (see sections 3.3.2 and 2.2). None of these precursors have occurred in the U. S. since 1983, and all were detected well before filling and venting. In addition to ingress of secondary water, other potential causes of clean water ingress into the RCP suction include inadvertent or ill-advised dilution during RCS filling; malfunction of the chemical and volume control system during filling; injection of cold and unborated water into either the RCP seals or the crossover leg (between the steam generator and the RCP) during shutdown; inadvertent (or "sneak") draining of clean water into the RCS crossover leg during shutdown; and failure to borate to refueling boron concentration prior to turning off all RCPs. The likelihood of each cause would be strongly dependent upon station practice. No valid estimate was obtained for the total frequency of having a large volume of clean water upstream of the RCP when it's started. Knowledge of the potential hazard, followed by review of each station's practices and revisions as appropriate, appears to be the most effective way to reduce the likelihood of occurrence.

Since the potential consequences could be severe, boron sampling of the crossover leg of the RCS is recommended prior to RCS filling and prior to RCP restart whenever the possibility exists that water of reduced boron concentration may exist there. For mo PWRs, local samples would have to be drawn from drain lines ("grab samples"). Plant personnel should be aware of the various plant specific scenarios for getting a clean water pocket in the RCS, and hazards associated with it. Procedures should exist to replace a potential clean water pocket with borated water, such as draining the region near the pocket while borating to increase core shutdown.

In the so-called French scenario (section 3.3.1), a loss of offsite power occurs during normal dilution toward criticality following refueling. Emergency power restores the diluted charging flow which then is assumed to collect as a near-stagnant pool of unborated water in the cold leg and reactor vessel because of the essentially zero flow circulating in the RCS. Restart of the RCP when power is restored is postulated to sweep the pocket of clean and perhaps cold water into the core. We believe such a scenario cannot cause a rapid boron dilution because inherent natural circulation will prevent formation of a clean water pocket.

Also, rapid boron dilution caused by unborated water in the residual heat removal system appears capable of causing criticality and power generation that could jeopardize low pressure systems, but the relatively slow rate at which this system can push water into the reactor core appears to be incapable of causing a core disruptive reactivity accident.

Loading Errors

Fuel misloadings in PWRs are uncommon, but do occur. As used here, a fuel misloading is defined as loading a fuel assembly into a core location where it should not be, or failure to place a control rod in a fuel assembly prior to loading into the core (if that is specified in the core loading specification). Loading errors have been discovered by independent verification of the assembly prior to insertion of the assembly into the core, mapping the spent fuel pit rack and/or core, or when attempting to retrieve/insert a fuel assembly from/into a spent fuel pit/core location.

A few years ago, when the maximum enrichment was 3.5 w/o U-235, it was widely believed -- probably correctly -- that PWR criticality was impossible as long as the refueling boron concentration was maintained, regardless of the number of refueling errors. With the trend toward higher enrichment fuel (up to 5 w/o U-235), and lower shutdown margin (5% vs 10%), that situation has dramatically changed.

Perhaps as few as three loading errors, placing fresh and unrodded assemblies in a tight cluster, could cause criticality. Therefore, observation of the changing core pattern by Senior Reactor Operators (SROs) or reactor engineers during fuel loading can a' significantly reduce the risk of inadvertent criticality, provided the personnel are aware of the hazards associated with clusters of fresh or unrodded fuel assemblies.

Although there are wide uncertainty bounds on the estimates, refueling errors do not appear to be meaningful contributors to public risk. There are no offsite radiological consequences due to inadvertent criticality (estimated as no greater than 1E-4 per refueling and probably about 1E-6) because containment integrity is assured during fuel movement. Even fuel damage (estimated as less than 1E-6 per refueling, and probably impossible) would have a vanishingly remote likelihood of causing an offsite radiological release in excess of 10 CFR 20 limits, although worker safety could not be assured if significant fuel damage occurred. Although three dimensional neutronic-thermal-hydraulic analyses can be done for refueling errors, none have been performed to date.

# 2.0 GRADUAL BORON DILUTION

## 2.1 Event Definition

Gradual boron dilution events cover those events in which the reactor coolant system boron concentration is nearly uniformly reduced due to some malfunction in the chemical and volume control system or operator error. These events are distinguished from those events discussed in Section 3.0 (Rapid Boron Dilution) by the characteristic that core boron concentration is gradually reduced.

The classic safety analysis boron dilution event postulates opening of the primary water makeup control valve and either a controller or mechanical failure of the blend system. A more complete discussion of boron dilution initiators is contained in section 2.2. The chemical and volume control system and the reactor makeup system are designed to limit the potential rate of dilution. Primary grade water with a boron concentration less than that of the reactor coolant system enters the reactor coolant system through the normal charging system. Flow in the reactor coolant loops is sufficient to ensure uniform mixing of the coolant throughout the system. As a result, core boron concentration begins to decrease. The reduction in core boron concentration results in an increase in neutron count rate as indicated on the source range detectors.

As the source range count rate increases, numerous indications and annunciators are available to alert the operator to the fact that a dilution is in progress. Additionally, the status of the makeup system is readily available from main control board indications. Once alerted to the dilution, the operator can terminate the transient by isolating the source of the dilution from the reactor coolant system. The dilution can also be mitigated by initiation of emergency boration. Should the above actions not be taken, then the dilution has the possibility of continuing until the reactor becomes critical.

Once critical, reactor power will continue to increase. The rate of power increase will depend upon reactivity feedback due to fuel and moderator temperature increases. Reactor power, fuel and coolant temperatures will continue to increase until the coolant begins to boil. At that point, the negative reactivity feedback due to void formation will terminate the power increase. As discussed in NUREG/CR-5368 (ref 2.5), preliminary calculations assuming beginning of cycle parameters (which provide the worst case) indicate that a power level of approximately 3% of rated would be reached due to this event. Therefore, no fuel damage is expected to occur. The event would be terminated by manual boration.

## 2.2 Boron Dilution Initiators

Two approaches were used to identify potential boron dilution initiators. The first approach involved a review of pressurized water reactor operating history (INPO Licensee Event Report and Significant Event Report Data Bases, Nuclear Power Experience Data Base, and AEOD Shutdown and Refueling Event Data Base) to identify past events which have resulted in the dilution of the reactor coolant system (RCS). The results of this review are summarized in the following paragraphs. Events which resulted in the dilution of makeup and storage systems, but did not result in RCS dilutions were not included in the review of operating events.

Secondly, a list of potential dilution flow paths was generated on the basis of detailed analysis of plant system interfaces which could potentially inject unborated water into the RCS. Although many flow paths can be identified, very few are credible (defined as requiring two or less valve mispositionings or failures to result in RCS dilution).

### Chemical & Volume Control System Equipment Malfunction

For this initiator, an equipment malfunction occurs in either the primary water system or the boric acid supply system, resulting in a reduction of the boron concentration in the water supplied to the suction of the charging pumps. Two RCS dilution incidents as cited in NUREG/CR-2798 (ref 2.1) have resulted from some type of chemical and volume control system malfunction. In one event, a boric acid controller setpoint error result in the actual boric acid flow being less than the value demanded by the controller. As a result, the boron concentration in the volume control tank, and subsequently the reactor coolant system was reduced from 1470 ppm to 1435 ppm. In another event, a flow controller for the primary makeup water valve failed. This caused excessive primary water makeup flow during blend operations and a subsequent reduction in RCS boron concentration from 1372 ppm to 1259 ppm over a period of thirteen hours. Another equipment malfunction event, as cited in the INPO Licensee Event Report (LER) Data Base (ref 2.2), occurred due to a flow transmitter that was out of calibration. The inputs to the boric acid integrator caused inaccurate operation of the boric acid controller. In approximately an eight hour period, the RCS boron concentration was reduced from 2217 ppm to 1952 ppm. A further review of the RCS boron dilution events contained in the INPO Licensee Event Report (LER) Data Base (ref 2.2) shows that no boron dilution event since 1985 was caused by an equipment malfunction.

### Steam Generator

Several RCS dilutions have resulted from equipment failures and/or human errors associated with steam generator maintenance. Two events cited in NUREG/CR-2798 (ref 2.1) involved secondary to primary leaks due to cut or unplugged SG tubes. In one event, the RCS boron concentration was reduced from 1720 ppm to 1698 ppm, and in the second event the boron concentration was reduced from 1800 ppm to 1733 ppm. In both of these cases the tubes were unknowingly cut or left unplugged. In another event, cited in LER No. 206-80034 (ref 2), an unexpected source of water was supplied to the secondary side of the steam generator while some steam generator tubes were being removed. RCS boron concentration was reduced by 35 ppm. Another event cited in INPO SER 13-90 (ref 3) involved a foreign reactor. During steam generator tube plugging operations, a miscommunication between maintenance and operations personnel resulted in buttoning up a steam generator with a section of tube removed. When the steam generator was filled, approximately 8000 gallons of water from the secondary side was added to the RCS through the tube. The core cooling circuit was diluted from 2083 ppm to 2001 ppm. This event caused a substantial amount of unborated water to be added to the cold leg of the RCS and is discussed further in Section 3.

Three events involved faulty steam generator isolation devices. In two of the events, LER Nos. 206-80029 and 206-80036 (ref 2.2) a faulty inflatable plug or loop seal resulted in leakage of unborated water into the RCS during steam generator channel head decontamination. However, in both events, the boron concentration was never reduced below its required value and shutdown margin was always greater than 10% dk/k. In a third event, LER No. 244-83015 (ref 2.2), a nozzle dam was being seated using diluted water in conjunction with air pressure at 30 psig. The isolation device leaked, and the complete volume of diluted water in the channel head and some air entered the RCS. In addition to the dilution, the air bubble passed through the RCS and entrained in the RHR pump suction, requiring the RHR pumps to be tripped and the subsequent loss of shutdown cooling.

In another event, LER No. 318-82049 (ref 2.2), water used for hydrolasing filled the steam generators past the nozzle lip because a portable drain pump was incorrectly oriented and not removing any of the water from the channel head. The RCS was diluted by an estimated 107 ppm. Shutdown margin remained greater than 22% dk/k throughout the event. Corrective actions taken to preclude such an event included increasing the frequency of checking the water level in the steam generator channel head and using borated water for the hydrolasing. Another hydrolasing related dilution, P.16.C.1979 (ref 2.4), resulted in an RCS boron reduction to 1993 ppm, 7 ppm below the Technical Specification limit.

Another event related to steam generator decontamination identified in NUREG/CR-2798 (ref 2.1), involved a miscalculation of the amount of excess boric acid required to offset the demineralized water added to the RCS during decontamination of the SG tube sheets. As a result, the boron concentration in the RCS was reduced by from 2000 ppm to 1902 ppm.

It should be noted that, with the exception of the event occurring at the foreign reactor, none of the steam generator maintenance related events cited above occurred after

1983. This indicates an increased level of awareness of the potential for boron dilution during steam generator maintenance operations. There was an LER rule change in 198⁄ that changed the reporting requirements (ref 2.7). We doubt that the rule change cause the absence of reported cases since 1983.

Valve Misposition

Several RCS dilutions have resulted from the operator (either the control room operator or an auxiliary operator) inadvertently mispositioning a valve or a series of valves which establishes a flowpath from the primary water system to the chemical and volume control system, or isolating the boric acid system from the chemical and volume control system.

In one event LER No. 346-82012 (ref 2.2), the operators did not completely close the demineralized water makeup valve. As a result, makeup for the RCS had a lower boron concentration than expected. The RCS was diluted to 1698 ppm, which was well above the 600 ppm required to maintain a 1% shutdown margin. The shutdown margin was at least 14% dk/k throughout the event. Another event, LER No. 361-82003 (ref 2.2), occurred while operators were attempting to re-establish shutdown cooling. One operator opened the RHR suction valve from the refueling water storage tank, while another operator was closing the RHR manual suction valve from the RCS. Water drained from the RWST to the RCS, and since it was at a slightly lower boron concentration, resulted in a slight dilution. The boron concentration remained above th Technical Specification limits. A third more recent event, involving a foreign reacto , SER-13-90 (ref 2.3), occurred when operators attempted to simultaneously fill the reactor coolant system and makeup to the RWST. Since the physical configuration of the boric acid system piping prevented such an alignment, the demineralized water from the makeup system was directed to the chemical and volume control system, while the boric acid was directed to the RWST. As a result, diluted water from the volume control tank was introduced into the reactor coolant system. The RCS was reduced to 1940 ppm, which was below the Technical Specification limit of 2000 ppm.

Lining Up Unsaturated Demineralizers

Mixed bed demineralizers have the potential to remove borate ions, and consequently boron, from the reactor coolant as the coolant passes through them. To prevent this from occurring, the demineralizers are first saturated with boric acid. Thus, when the borate ions pass through the demineralizers, the demineralizers will not "take" the borate ions and "give up" the hydroxide ions, which would later combine with hydrogen ions to form water.

Five demineralizer related boron dilution events obtained from PWR operating experiences are listed in NUREG/CR-2798 (ref 2.1). The most severe event resulted in a dilution of 356 ppm. However, even if the demineralizers were not presaturated with boric acid, operation of the demineralizers will eventually saturate them and eventually terminate the dilution. A review of PWR operating experiences indicated that no demineralizer related boron dilution event has been reported since 1982. Therefore demineralizer related boron dilution events are not considered significant risk contributors to boron dilution events.

Required VCT Boron Concentration Not Maintained

In this event, the operator miscalculates the required boric acid flow rate to maintain the volume control tank at the required boron concentration. As a result, water at a boron concentration less than that of the reactor coolant system is added via normal charging. An example of this type of event is listed in Table 2-1 as Event No. 4. The RCS boron concentration was reduced from 2300 ppm to 1996 ppm. The shutdown margin that was maintained throughout this event was 6.8 percent, which was well within the limits required by the plant's Technical Specifications.

Spraydown During Refueling

In this event, unborated water is used to spray down the reactor internals and the cavity walls. As a results, the boron concentration in the refueling canal in reduced. This event is discussed further in Sections 3 and 4 due to the potential for significant dilution and the mode of operation.

## 2.3 Design Features and Administrative Controls to Preclude or Limit Inadvertent Boron Dilution Events

### Design Features

Some plants have a boron dilution mitigation system that will isolate boron dilution flow paths upon receipt of an increasing source range count rate signal.

### Administrative Controls

During refueling, administrative controls are placed on the position of various valves that could potentially result in dilution of the reactor coolant system. These valves are typically closed and secured in position by mechanical stops or with air or electrical power removed, thus preventing the addition of unborated water to the reactor coolant system. Additionally, for those shutdown modes which require the makeup system to be in operation, the valve position for the makeup system is limited, often by a mechanical block, to limit the rate at which the diluted water can be added to the reactor coolant system.

Other controls are implemented through memoranda that address the correct application of a technical specification and the correct procedure and equipment maintenance ar^d calibration schedules. NUMARC 91-06 (ref 2.8) has provided some guidelines that cou assist in reducing the number of these events from occurring. Those that apply to gradual dilution are:

A. Boron dilution paths should be identified for each planned shutdown configuration. Flow paths that may cause a boron dilution should receive appropriate administrative controls.

B. Simultaneous filling of the RCS and the refueling water storage tank should be carefully controlled to reduce the potential for underborated water injection into the core.

C. Shutdown margin calculations should be verified and any differences should be immediately resolved. As a minimum, an evaluation of the shutdown margin should be performed whenever changes are planned that could affect shutdown margin.

D. The addition of unborated water to the refueling cavity or the primary side of the steam generators should be strictly adminstratively controlled.

E. Source range detectors should be frequently monitored during shutdov conditions, particularly during activities that could result in boron dilutic

F. Redundant boration paths should be available to respond to a boron dilution event.

Technical Specification Limits on Boron Concentration

The shutdown margin requirement for modes 1 through 4 is typically 1.0 to 1.9% dk/k. The shutdown margin requirement for mode 5 is typically 1% dk/k. The basis for the 1% shutdown margin is typically to limit the impact of a steam line break; however, it also provides a margin to criticality for other reactivity events such as a boron dilution event. During refueling, the shutdown margin is typically 5% dk/k. This ensures that the reactor will be maintained sufficiently subcritical during the fuel movement.

Procedural Controls on Placing Demineralizers in Service

Placing the wrong type of demineralizer in service or not saturating the desired demineralizer with boric acid can result in an inadvertant reduction in core boron concentration. Implementation of procedural controls reduces the likelihood of an inadvertent dilution by these means.

## 2.4 Operator Actions to Mitigate Boron Dilution Transient

### Diagnosis of Event

Several main control board indications exist which can assist the operator in identifying if a dilution is in progress. The most significant indication is an increase in source range count rate. The audible source range indication in the control room would alert the operators to an increasing count rate. All PWRs typically have an audible source range indication in the control room during shutdown when in the source range. Long term trending of source range count rate is accomplished by periodically monitoring the nuclear instrument strip chart recorders. Additionally, annunciated alarms, such as a "High Flux at Shutdown" (typically set at five times the background count rate) exist to alert the operator. Various indications are also available on the main control board to inform the operator of the status of the chemical and volume control system and reactor makeup system. These include: indication of boric acid flow rate, blended flow rate, and primary water flow rate, status of chemical and volume control system and reactor makeup system pumps, and deviation alarms if boric acid or primary makeup water flow rate differ from their setpoint flow by more than 10%.

Emergency Boration - Because the control rods are inserted into the core during shutdown conditions, the only means the operator has to control reactivity additions is by boron addition. Emergency boration procedures typically require the operator to initiate emergency boration in response to any unexplained addition of positive reactivity into the core. Typically, emergency boration can be accomplished several ways including: a motor operated or manually operated emergency boration valve at the boric acid storage tanks, the normal boration flowpath, and the refueling water storage tank.

## 2.5 Frequency

Twenty-five inadvertent boron dilution events were reported by the end of 1984 and were included in Brookhaven's Report on "Reactivity Accidents" (ref 2.5). None of these events resulted in criticality. These events were caused by both equipment and human performance problems, with human performance causing 80% of the events. Their estimated frequency of criticality from boron events was determined to be 2.0E-4 (ref 2.5). This number was derived by using the information from the events that occurred and extrapolating to the point where the shutdown margin was reduced to zero. That frequency was then multiplied by 0.1 for the failure of the operator to take action. The human performance factor was based on judgement.

Eight boron dilution events that caused a reduction in the core boron concentration that have occurred since January 1985 in U.S. PWRs have been identified from a review of References 2.2, 2.3, 2.4, and 2.6 . These events are numbers 4, 20, 21, 22, 23, 24, 29, and 31 in Table 2-1. Therefore the frequency of boron dilution events is 2E-2 for approximately four hundred reactor-years of PWR operation, down by a factor of three from pre-1985 experience. Four of these events (20, 21, 23 and 29) caused dilutions that were limited in rate and magnitude with no reasonable possibility of resulting in criticality (i.e., adding RWST water at a concentration of 2069 ppm to the RCS at a concentration of 2084 ppm). Two of the events (4 and 24) could have resulted in criticality if unmitigated. The cause of one of the events (22) is unknown, therefore the potential of resulting in criticality is questionable. However, it was included in the frequency of inadvertent criticality to be conservative. One other event (31), resulted in a dilution of approximately 20 ppm, and since it was within the boron sampling accuracy, was dismissed. Therefore, the frequency of boron dilution events with the potential to cause criticality is about three in four hundred years of PWR operation, or approximately 0.7E-2 per reactor-year. Assuming a probability of 0.1 for the operator not recognizing an increase in the source range count rate and corresponding alarms, and 0.1 for not detecting the dilution by RCS boron sampling or a water balance, the frequency of inadvertent criticality from boron dilution is less than 1E-4.

## 2.6 Conclusions

Gradual boron dilution events are not expected to cause core damage, even if they are unmitigated. No criticalities have resulted from any of the gradual boron dilution events that have occurred to date. In fact, they appear to have been terminated with a small loss of reactivity margin. The frequency of gradual boron dilution events reported at commercial nuclear plants in the United States has declined significantly in recent years. The reported gradual boron dilution events involved operator performance problems, or steam generator tube sheet decontamination and tube removal/plugging activities. Maintenance related valve misalignments did not contribute to the reported gradual dilution events.

# Table 2-1
## Reactor Coolant System Boron Dilution Events

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| EVENT | DATE | PLANT | REFERENCE(S) | EVENT DESCRIPTION |
|---|---|---|---|---|
| 27 | 05/01/74 | POINT BEACH 1 | P.09.C.0020 NPE | RCS 1435 PPM, 43 PPM BELOW EXPECTED 1478. BLENDER DELIVERED WATER 20% LESS CONCENTRATED THAN DESIRED, ERROR BETWEEN ACTUAL AND INDICATED BORIC ACID FLOW TO BLENDER |
| 26 | 10/01/76 | ZION 1 | P.07.D.0014 NPE | RCS DILUTED FROM 1081 TO 964 PPM (T.S 920 PPM) CAUSED BY ISOLATION WATER VALVE OPEN |
| 25 | 05/01/79 | ST. LUCIE 1 | P.05.D.0219 NPE | MODE 6, RCS DILUTED TO 1698 PPM (T.S. 1720 PPM), 2 S/G TUBES CUT AND NOT PLUGGED. |
| 9 | 07/05/80 | SAN ONOFRE 1 | LER 206-80029 | INFLATABLE SEALS ON S/G LEAKED DURING DECONTAMINATION RESULTING IN DILUTION, BORON CONC. >2400 PPM, K ≤ 0.90 |
| 10 | 09/01/80 | SAN ONOFRE 1 | LER 206-80034 | DURING S/G TUBE REMOVAL WATER FROM SECONDARY DRAINED TO PRIMARY RCS DILUTED BY 35 PPM |
| 11 | 09/22/80 | SAN ONOFRE 1 | LER 206-80036 | RCS LOOP SEAL FAILED DURING S/G DECON, RCS BORON >2500 PPM, KEFF ≤ 0.90 |
| 19 | 09/22/80 | BEAVER VALLEY 1 | LER 334-80070 | DILUTION OF RCS DUE TO PROCEDURAL ERROR. BORIC ACID BLENDER DID NOT ACCOUNT FOR MANUAL EMERGENCY BORATION FLOW PATH IN SERVICE. SDM > 5% |
| 32 | 07/01/81 | MAINE YANKEE | P.16.C.1393 NPE | RCS DILUTED BY 100 PPM DUE TO DEMIN STORAGE TANK OUTLET VALVE FAILURE |
| 12 | 08/05/81 | DAVIS-BESSE | LER 346-81041 | DILUTION FROM 1044 TO 970 PPM CAUSED BY PLACING A PREVIOUSLY SATURATED DEMIN IN SERVICE. SDM ≥ 1.0% |
| 18 | 03/12/82 | MCGUIRE 1 | LER 369-82023 | COLD SHUTDOWN, RCS @ 1025 PPM (< T.S. 1030 PPM 1%) PROCEDURAL DEFICIENCY DUE TO OPERATING BORIC ACID BLENDER WITHOUT VERIFYING CONCENTRATION DURING LARGE ADDITIONS |

Table 2-1 (Cont.)
Reactor Coolant System Boron Dilution Events

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| EVENT | DATE | PLANT | REFERENCE(S) | EVENT DESCRIPTION |
|---|---|---|---|---|
| 7 | 03/14/82 | DAVIS-BESSE | LER 346-82012 | DURING COLD SHUTDOWN, 14% SD, FAILURE TO CLOSE DEMIN WATER MAKEUP VALVE RESULTED IN DILUTION TO 1698 PPM, THIS WAS ABOVE THE T.S. LIMIT OF 600 PPM OR 1% SHUTDOWN |
| 16 | 03/14/82 | SAN ONOFRE 2 | LER 361-82003 | DURING MODE 6, RCS WAS DILUTED, RWST SUCTION OPENED AND RCS SUCTION TO CHG PUMP CLOSED, SDM ABOVE T.S. |
| 8 | 04/15/82 | SURRY 1 | LER 280-82048 | FAILURE TO MAINTAIN MAKEUP BORON CONCENTRATION EQUAL TO RCS CONCENTRATION RESULTED IN DILUTION |
| 17 | 05/26/82 | SURRY 2 | LER 281-82030 | COLD SHUTDOWN IN PREP FOR FILLING RCS LOOP A, MALFUNCTION OF THE BORIC ACID FLOW CONTROLLER CAUSED DILUTION. (SDM > T.S.) |
| 15 | 06/09/82 | SAN ONOFRE 1 | LER 206-82016 | DURING COLD SHUTDOWN, DEMIN WAS PLACED IN SERVICE BEFORE IT HAD BEEN BORON SATURATED CAUSING DILUTION OF 211 PPM, SDM > 10% |
| 13 | 10/22/82 | CALVERT CLIFFS 2 | LER 318-82049 | WHILE IN MODE 6, HYDROLAZING WATER FILLED S/G PAST THE NOZZLE LIP AND MAY HAVE DILUTED RCS BY 107 PPM, SDM 22%, PORTABLE DRAIN PUMP WAS IMPROPERLY ORIENTED |
| 28 | 02/01/83 | SALEM 2 | P.16.C.1979 MPE | RCS DILUTED TO 1993 PPM (T.S. 2000) DUE TO S/G HYDROLAZING |
| 14 | 04/12/83 | GINNA | LER 244-83015 | DURING S/G DECON, NOZZLE DAM LEAKED IN COLD LEG, RCS WAS 2400 PPM TO ACCOUNT FOR 2500 GAL OF WATER IN THE S/G CHANNEL HEAD (T.S. 2000 PPM) WATER AND AIR LEAKED INTO COLD LEG. |
| 30 | 11/26/84 | CATAWBA 1 | LER No. 413-84027 | MODE 3, RCS BORON CONCENTRATION DECREASED TO < 2000 PPM - CAUSE UNKNOWN |
| 21 | 04/02/85 | ZION 1 | LER 295-85013 | MODE 6, CAVITY DILUTED DURING ECCS FULL FLOW TESTING, SDM > 10% |
| 20 | 04/29/85 | COOK 1 | LER 315-85021 | MODE 6 TRANSFERRING WATER FROM RWST (2069 PPM) TO RCS (2084 PPM) DILUTION. > TS 2000 PPM |

## Table 2-1 (Cont.)
## Reactor Coolant System Boron Dilution Events



| EVENT | DATE | PLANT | REFERENCE(S) | EVENT DESCRIPTION |
|---|---|---|---|---|
| 22 | 07/29/85 | COOK 1 | LER 315-85035 | MODE 6, RCS WAS > 2000 DECREASED TO 1967 PPM BORATED TO 2012 PPM FELL TO 1972 PPM BORATED TO 2128 (T.S. 2000 PPM) CAUSE UNKNOWN |
| 24 | 10/26/85 | SURRY 2 | LER 281-85012 | COLD SHUTDOWN RCS FILL CAUSED DILUTION FROM 2217 TO 1952 PPM, FLOW TRANSMITTER TO BORIC ACID INTEGRATOR OUT OF CALIBRATION |
| 31 | 01/20/86 | TURKEY POINT 4 | LER NO. 251-86003 | RCS BORON CONCENTRATION WAS REDUCED TO 1930 PPM (TS > 1950) - CAUSE UNKNOWN |
| 29 | 12/01/87 | SEQUOYAH 1 | P.16.C.3191 NPE, LER NO. 327- | COLD SHUTDOWN RCS > 2000 PPM, ADDED 75 GALLONS OF DEMIN WATER, DILUTED BY 4 PPM CAUSED BY INADEQUATE PROCEDURE FOR CHEMICAL ADDITIONS. |
| 3 | 11/09/88 | DOEL 1 | SER 13-90 | PLANNED DILUTION TO REDUCED CB FROM 2425 TO 2000 PPM, MISINTERPRETATION OF SCALE CALIBRATION OF BORIC ACID BLENDER CAUSED DILUTION TO 1% S.D. (1507 PPM) |
| 1 | 12/20/88 | ROBINSON 2 | SER 18-89 | MODE 6 SPRAYING INTERNALS, RWST 1950 PPM, 650 PPM AT TOP OF CAVITY AND 2 FEET THICK |
| 2 | 01/23/89 | SAN ONOFRE 1 | SER 18-89, P.16.C.3337, LER 206-89002 | MODE 6 SPRAYING CAVITY WALLS, SRS INOP, 440 GALS OF DEMIN WATER USED FOR 19 HOURS, RCS 2663 PPM DILUTED TO 2626 PPM (T.S. 1900 PPM, 5%) |
| 23 | 04/23/89 | SURRY 1 | LER 280-89016 | COLD SHUTDOWN, DILUTION FROM RCP STANDPIPE PRIMARY WATER MAKEUP VALVE, DILUTED BY 80 PPM, SDM >9.3% |
| 4 | 10/25/89 | SURRY 2 | SER 13-90, LER 281-89015 | WHILE IN COLD SHUTDOWN AND FILLING THE RCS, ERROR IN BORIC ACID BLENDER DILUTED RCS FROM 2300 TO 1996 PPM, SDM >6.8% T.S LIMIT IS 2000 PPM |
| 6 | 01/04/90 | GOLFECH 1 | SER 13-90 | WHILE COLD SHUTDOWN, FILLED RCS AND RWST SIMULTANOUSLY, RCS DILUTED TO 1940 PPM BELOW T.S. OF 2000 PPM |

Table 2-1 (Cont.)
Reactor Coolant System Boron Dilution Events

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| EVENT | DATE | PLANT | REFERENCE(S) | EVENT DESCRIPTION |
|---|---|---|---|---|
| 5 | 03/03/90 | BLAYAIS 4 | SER 13-90 | WHILE COLD SHUTDOWN, S/G TUBE WAS REMOVED AND NOT PLUGGED IN COLD LEG, 8000 GALS ADDED AT 45 GPM OF SECONDARY WATER DILUTED RCS FROM 2003 TO 2001 PPM - POSSIBLE STAGNANT LOOP RCP PROBLEM |

NOTE: The event listing in this table is not complete prior to 1980. For example, the following additional events have been listed by INPO:

| | | |
|---|---|---|
| Crystal River-3 | 1/26/77 | LER 302-77000(77-8) |
| Crystal River-3 | 2/10/77 | LER 302-77000(77-12) |
| Calvert Cliffs-1 | 1/28/78 | LER 317-78000(78-9) |
| Crystal River-3 | 3/10/78 | LER 3092-78016 |
| Millstone-2 | 4/14/78 | LER 336-78005 |
| Point Beach-1 | 11/7/79 | LER 266-79019 |

## 2.7 References

2.1 NUREG/CR-2798 (ORNL/NSIC-208), "Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants", July, 1982.

2.2 INPO Licensee Event Report Data Base.

2.3 INPO Significant Event Report 13-90, "Unplanned Boron Dilutions from Shutdown Conditions", August, 1990.

2.4 Nuclear Power Experience Data Base, Stoller Corp., Boulder, CO.

2.5 NUREG/CR-5368 (BNL-NUREG-52198), "Reactivity Accidents", January, 1990.

2.6 AEOD Special Evaluation Report, "Review of Operating Events Occurring During Hot and Cold Shutdown and Refueling", Draft, December, 1990.

2.7 NUREG--1022 (DE84 900039), "Licensee Event Report System", September, 1983.

2.8 NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management", December, 1991.

# 3.0 RAPID BORON DILUTION

## 3.1 Definition and Scope

For the purposes of this report, a rapid boron dilution event is defined as the sudden addition of a volume of boron free or nearly boron free water into the core region. The rapid boron dilution event differs from the events discussed in Section 2.0 in the rate at which the unborated water enters the core. For the gradual boron dilution events discussed in the previous section, there is sufficient mixing of diluted water in the reactor coolant system to result in only a slow decrease in core boron concentration. Furthermore, the diluted volume is dispersed uniformly throughout the core region. In contrast, a "rapid boron dilution" event occurs when the flow into the reactor core contains streams of relatively unborated water. As a result, the reactivity addition in the core can be localized and potentially severe.

Historically, analysis of rapid boron dilution events were first reported some 20 years ago (ref 3.1) for a startup of an isolated loop containing cold, clean water (only possible in a plant with loop isolation valves). Point kinetic analyses were found to grossly overpredict the severity of the transient. Transient three-dimensional analyses were completed, and showed about 3% fuel rod failure and less than 0.5% of fuel elements completely melted. Total energy release was determined to be insufficient to breach the Reactor Coolant System (RCS). These results were considered by the reactor designer to be unacceptable despite stringent administrative controls on opening loop isolation valves, and protection-grade interlock circuits were deemed necessary to prevent inadvertent startup of an isolated loop. The event was thus removed from design basis evaluation and further analyses became academic. Therefore, analyses of dilution events in Final Safety Analysis Reports (FSARs) have been limited to the gradual dilution events described in Section 2.0.

During commissioning of the Ringhals 4 PWR in Sweden in 1982, unborated water was used for hot functional testing (prior to core load), and the Reactor Coolant System (RCS) partially drained prior to borating. Subsequent boration of the residual water in the suction piping of the Reactor Coolant Pump (RCP) proved to be extremely difficult; i.e., the water in that location stayed unborated when the rest of the RCS was borated for core loading. Preliminary reactivity calculations indicated that injecting a volume of unborated water equal to the volume of one "crossover" leg (the section of RCS piping that "crosses over" from the steam generator to the RCP suction) could generate a severe reactivity transient if swept into a core with new fuel (ref 3.2). During their annual training in 1983, operators of Ringhals 2, 3, and 4 were alerted to this potential hazard and instructed to delay any startup for boron sampling of the crossover leg if they had reason to believe there had been a previous ingress of boron-free water.

The Chernobyl accident in 1986 heightened interest in reactivity accidents with the potential for severe core damage, including rapid boron dilution (ref 3.3). A great deal of uncertainty exists in the hydraulic and neutronic response for such an event, and existing analyses are inconclusive as to whether severe fuel damage or dispersal can occur for any event with a significant chance of occurrence. As a result, this section of the report summarizes the results and status of other work on this subject, rather than presents detailed analyses of our own.

Two very recent studies, from Sweden and Brookhaven (ref 3.4 and 3.5), became available after the draft of this report was prepared. We consider their results to be significant, and have substantially revised the draft text to incorporate review of them. Based on those results, we draw conclusions (not necessarily those of the authors of those reports).

Several rapid boron dilution events have been analyzed previously (refs 3.3, 3.6, 3.7, and 3.8). In general, these events can be classified into two categories. The first category involves the dilution of a reservoir connected to the Reactor Coolant System (RCS) and the spurious discharge of the contents of the tank into the core region. Two scenarios in this category include: 1) the dilution of an accumulator and the spurious opening of the accumulator discharge valve, and 2) the dilution of the Refueling Water Storage Tank (RWST) and inadvertent actuation of the safety injection system. A detailed analysis of these events is presented in NUREG/CR 5368, "Reactivity Accidents." (ref 3.6). This report estimated the probability of a significant reactivity event from these two scenarios as 6.9E-10/yr and 3.7E-08/yr respectively. Because the probability of these events is so low, they do not require further consideration in this report.

The second category of rapid boron dilution scenarios involves the start of a Reactor Coolant Pump (RCP) and the subsequent injection of a relatively unborated water slug into the core. Various scenarios are discussed in more detail in Section 3.3.

## 3.2 Anatomy of a Rapid Boron Dilution Event

Two things are required for a rapid boron dilution event to occur: the collection of a stagnant volume of relatively unborated water, particularly in the crossover or cold leg of the RCS, and the subsequent rapid transmission of that volume of unborated water into the core region. As discussed in the report "Some Local Dilution Transients in a Pressurized Water Reactor" (ref 3.7), the formation of a stagnant zone of unborated water requires very low or no flow in the reactor coolant loops.

If any RCP is operating, there will be sufficient circulation throughout the RCS to prevent the formation of a stagnant zone. The Residual Heat Removal (RHR) system provides enough circulation in the portions of the RCS being circulated to prevent the formation of a stagnant zone in the reactor vessel, cold leg piping between the RHR discharge and the reactor vessel, and hot leg piping between the reactor vessel and the RHR suction. However, the RHR system may or may not cause circulation through the steam generators. Any of three conditions may block RHR circulation through the steam generators:

1. Steam generator tubes drained. During most of a refueling outage, the steam generator tubes are drained (filled with air or nitrogen).

2. Vapor pocket. If the RCS is totally depressurized and the water level near the flange, a low temperature water vapor pocket is likely to exist at the top of the steam generator tubes (the manometer effect, ref 3.9).

3. Thermal gradients. If water in the steam generator shell is warmer than the active portion of the reactor coolant, RHR flow through the reactor vessel is generally too low to force water through the steam generator tubes as a parallel path. This condition would exist if the RCPs had been shut off prior to completion of the RCS cooldown.

If all forced circulation is shut off, natural circulation through the reactor coolant loops as a result of core decay heat will generally result (unless prevented by one of the above mechanisms). Natural circulation flow through the steam generators is generally much higher than would be forced by RHR flow. However, it's possible for variations in temperatures between the steam generators to prevent natural circulation through one loop, particularly at cold shutdown. We therefore recommend that the reactor coolant in the steam generators and reactor coolant suction piping (the crossover legs) should be considered as stagnant at all times that all RCPs are shut off.

If diluted water is added to a stagnant region, a pocket of less borated water results. The addition of diluted water may occur either deliberately, as in the case of proceeding to a normal reactor startup, or inadvertently.

One French study (ref 3.8) analyzed the potential impact of a clean water slug inserted into the core. Two parametric cases were analyzed, in one case the boron concentration of the core was assumed to decrease by 400 ppm with no change in core inlet temperature, while the other case involved a reduction in core inlet temperature of approximately 110 Deg-F in conjunction with the dilution. For the case of the dilution only, no core damage was predicted; however, for the case of the dilution in conjunction with the cooling, significant core damage was predicted.

In support of this reactivity risk study, scoping calculations were performed by the Westinghouse Commercial Nuclear Fuels Division for a few fuel assemblies, without burnable poisons, in clean and borated water (2000 ppm). The reactivity was calculated with the KENO-PC code, using parameters considered typical of current fuel design. We believe the results, listed in the table below, are indicative of expected reactivity for current enrichment and without burnable poisons. However, the values would have to be verified for a specific fuel design prior to use in design or safety analysis.

| No. of Assemblies | U-235 Enrichment | Boron Concentration | K-eff |
|---|---|---|---|
| 1 | 5 w/o | 0 ppm | 0.95 |
| 2 | 4 w/o | 2000 ppm | 0.76 |
| 2 | 4 w/o | 0 ppm | 1.05 |
| 2 | 5 w/o | 2000 ppm | 0.81 |
| 2 | 5 w/o | 0 ppm | 1.08 |
| 3 | 4 w/o | 2000 ppm | 0.82 |
| 3 | 4 w/o | 0 ppm | 1.12 |
| 3 | 5 w/o | 2000 ppm | 0.87 |
| 3 | 5 w/o | 0 ppm | 1.15 |
| 4 | 4 w/o | 2000 ppm | 0.90 |
| 4 | 4 w/o | 0 ppm | 1.20 |
| 4 | 5 w/o | 2000 ppm | 0.96 |
| 4 | 5 w/o | 0 ppm | 1.24 |

The above results show that two fresh and unpoisoned assemblies with enrichments of 4 w/o U-235 would be critical in unborated water. These results are listed here to illustrate the potential highly localized effects of subjecting assemblies to pure water. It should be noted that typical core designs do not place two fresh unpoisoned assemblies next to one another, and that burnable poisons (to whatever extent is required by the core design) are added before putting the assembly in the reactor core.

In principle, an optimally shaped and placed volume of clean, cold water on the order of 10 cubic feet can produce prompt criticality in a cold core at beginning of cycle. Considering the relatively high velocities at which fluid can be swept into the core (roughly 3 feet/second for a single RCP running on a 4-loop plant, twice that velocity on a 2-loop plant), neither prompt criticality nor high reactivity insertion rates can be ruled out. If the reactivity insertion is rapid enough, severe core damage could result, including dispersal of molten fuel with resultant pressure waves. Fuel dispersal has been conservatively calculated for some postulated rapid boron dilution events, but we know of no realistic calculation which proves such a result is possible for any realistic scenario.

Accurate deterministic analyses of the possible extent of core damage following a rapid boron dilution event are difficult in several areas. First, the size, shape, location, and spatial boron content of the postulated pocket of low boron concentration is speculative. Second, an accurate prediction of the boron transport and turbulent mixing as the pocket is swept to and through the core requires state of the art hydraulics modeling and is dependent upon plant-specific design. We know of only one such analysis (ref 3.4). Third, determination of the reactivity effect of the spatial boron transient requires three-dimensional analysis. The static 3-D analyses are well within the state of the art, but will vary with each core design and perhaps with each reload. Fourth, a detailed transient 3-D analysis is necessary to reasonably predict local conditions near the hot spot. Point kinetics (or 1-D or 2-D) calculations tend to be either indefensible or extremely conservative. Fifth and last, converting localized fuel conditions to core and RCS damage states requires judgement subject to either uncertainty or conservatism.

The failure threshold for unirradiated and intact fuel rods is 210 to 220 cal/gm radially averaged peak fuel enthalpy (ref 3.10). In this range, the failure mechanism is brittle fracture of the cladding caused by severe oxidation. This failure mechanism does not lead to fuel dispersal or severe core damage; i.e., there would be no significant fuel dispersal, the fuel rods would remain in a coolable geometry, and no pressure pulses would occur.

When the fuel enthalpy exceeds about 300 cal/gm for unirradiated fuel (equivalent to gross melting of the UO2 fuel pellet), the fuel failure mode changes. Molten fuel is expelled into water and fragments into small particles. Mechanical energy in pressure waves following fuel dispersal could threaten the integrity of the pressure boundary. The threshold for detecting nuclear to mechanical energy conversion appears to be about 325 cal/gm for unirradiated intact fuel, with a conversion efficiency less than 1% in the range of 325 to 500 cal/gm (ref 3.10).

The above failure thresholds are somewhat lower for irradiated fuel.

## 3.3 Rapid Boron Dilution Scenarios

Various studies have evaluated various boron dilution scenarios, both probabilities and deterministics. In one of the most complete such studies, Sven Jacobson (ref 3.4) evaluated 15 scenarios and was able to dismiss 12 of them on the basis of very low probability. These results are summarized on Table 3-1. Evaluation of one event (steam generator tube rupture with backfill during cooldown), although low frequency, led to

Table 3-1
Summary of Various Boron Dilution Scenarios*

| Event | Probability | Transport mechanism | Actions |
|---|---|---|---|
| Tube rupture + backfill cooldown | 3.0E-7/reactor year | RCP | Procedure modification |
| Malfunction in CVCS during pressurization | 3.1E-10/reactor year | RCP | No actions required |
| Leaking thermal barrier during pressurization | < 1.0E-8/reactor year (estimated) | RCP | No actions required |
| Blackout during start-up after refuelling | 1.5E-4/reactor year | RCP | Further analysis |
| Diluted RWST during reactor cavity filling | Not significant | RHRS (not applicable) | No further analysis |
| Diluted accumulator + inadvertent opening of MOV | 6.9E-9/reactor year | Accumulator nitrogen pressure (not applicable) | No actions required |
| Diluted RWST + inadvertent leakage to reactor | 1.5E-8/reactor year | Gravity, RHRS (not applicable) | No further analysis |
| Diluted accumultor + leaking MOV | < 4.1E-9/reactor year | RCP | No actions required |
| Diluted RWST + spurious SI | 3.7E-8/reactor year | LHSI (not applicable) | No further analysis |
| LOCA + diluted accumulator | 5.4E-8/reactor year | Accumulator nitrogen pressure (not applicable) | No further analysis |
| LOCA + diluted RWST | 2.7E-8/reactor year | LHSI (not applicable) | No further analysis |
| LOCA with sump water diluted | Not significant | LHSI in recirculation (not applicable) | No further analysis |
| LOCA/SGTR | Not available | SG secondary steam pressure (not applicable) | No further analysis |
| Inadvertent dilution at shutdown | 9.0E-9/reactor year | RCP | No actions required |
| Dilution during SG maintenance | > 1.0E-8/reactor year | RCP | Further analysis |

*Jacobson (ref 3.4), reprinted with permission modification of the tube rupture recovery procedure. The other two events (the so-called French and Swedish scenarios) required further analysis. They are discussed in more detail later.

Six illustrative scenarios for rapid boron dilution, drawn from various sources, are described below.

**Scenario A (Dilution During RCS Filling)**

The reactor is shutdown and being cooled by the RHR system. The steam generator tubes are drained, but the RCS is in the process of being filled and pressurized. As a result of a CVCS malfunction or operator error, for some period of time diluted water is supplied to the charging pumps, which in turn supply seal injection water to the RCPs. About half this supply water will run backwards through the RCP (to fill the cold leg side of the steam generator tubes), and half will go into the active portions or the reactor coolant. Normal charging is assumed to be isolated. The existence of the clean water pocket is assumed not to be detected by boron sampling. (This requires a procedural violation.) When the RCP is started, the clean water pocket is swept into the core. Jacobson estimates the frequency of this event as about 1.E-10/yr -- too small for further consideration (ref 3.4). However, this frequency estimate is quite plant dependent. For example, some plants routinely fill via RCP seal injection (an abnormal occurrence with conditional probability of about 1.E-4 in Jacobson's evaluation). Other plants may charge into cold legs which have no RHR discharge. In principle, this scenario could also occur as a result of a deliberate but ill-advised dilution during RCS filling combined with a malfunction of the Chemical and Volume Control System. (Typical U.S. practice calls for boron dilution only after reaching hot zero power conditions.)

**Scenario B (Steam Generator Inleakage - The "Swedish Scenario")**

The reactor is shutdown and being cooled by the RHR system. As a result of a steam generator tube leak (most likely caused by improperly completed steam generator maintenance or inspection), secondary water enters the reactor coolant system, and collects in the RCP suction piping, steam generator outlet plenum, and perhaps in the steam generator tubes. The existence of this pocket is assumed to be undetectable by normal boron sampling of the reactor coolant being circulated, and assumed to be undetected (although detectable) by mass balances. Subsequent start of the RCP sweeps the clean water into the core. This scenario, sometimes referred to as the "Swedish scenario", is discussed in more detail later in this report. Note that, except for the mechanism of introducing unborated water into the RCP suction, it is identical to Scenario A.

**Scenario C (Loss of AC Power during Dilution - The "French Scenario")**

The reactor has just been refueled and is the process of being started up. A boron dilution (toward the critical boron concentration) is in progress when a loss of offsite power occurs, resulting in the trip of all RCPs. Decay heat is low and natural circulation does not occur in the reactor coolant loop(s) receiving the diluted charging flow from the Volume Control Tank (VCT). Emergency power comes on and automatically restores the charging flow. In the absence of alarms drawing attention to the dilution in progress, the operators fail to secure the dilution (as required by plant Technical Specifications), and the entire volume of the VCT is discharged into the RCS. The loss of AC power also may automatically isolate letdown, (with the assumption that it is not manually re-established), such that the incoming charging flow is not heated by the regenerative heat exchanger. The incoming unborated and possibly cold water therefore forms a stagnant and relatively unborated volume in the cold leg and bottom of the reactor vessel. When the VCT level is low, charging pump suction is switched to the borated (and cold) Refueling Water Storage Tank (RWST). But before a significant amount of borated RWST water is added to the RCS (borating the unborated volume), offsite power is restored, the operators restart an RCP, and the clean water is swept into the core. This boron dilution scenario is sometimes referred to as the "French Scenario" since it was first publicized in reference 3.8, and led EdF to install an automatic system to switch the charging pump suction from the VCT to a borated source on loss of power to the reactor coolant pumps. The scenario is also described in references 3.4, 3.5, and 3.11, and is discussed in more detail later.

**Scenario D (Unborated Water in RHR System)**

Startup of the RHR system when it contains totally unborated water was evaluated by the French (ref 3.12) with the extreme assumption that no mixing occurred, such that a clean water front moved through the core at a speed determined by the RHR flow; causing the core average boron concentration to drop linearly from 1500 ppm to 0 ppm in 75 seconds. The interesting result revealed by the analysis was that the core power generation caused RHR system rupture on overpressure after about 8 seconds, stopping the dilution well before fuel damage limits were reached. This result suggests that RHR flow is incapable of causing a core disruptive reactivity accident. The estimated frequency of this scenario was on the order of 1.E-8/year, too low for further consideration in any event.

**Scenario E (Boration after Shutting Off RCPs)**

During RCS cooldown at the beginning of a refueling shutdown, the RCPs are postulated to be are turned off at a relatively high temperature (such as 140 Deg-F) and before borating to refueling boron concentration. With hot water left in the steam generators, RHR flow would be unlikely to force circulation through the steam generator tubes, with formation of a stagnant pocket of low boron concentration. During subsequent draining and refilling, this pocket would remain in the crossover leg and/or steam generator. Common PWR practice is to continue RCP operation until refueling boron concentration is reached. This practice should be considered mandatory.

**Scenario F (Dilution of Refueling Cavity)**

A scenario of interest, not involving RCP restart, involves local dilution of the refueling cavity water. Two recent events (ref 3.13) illustrate the potential for significant local dilution during refueling operations. Specifically, these events involved the formation of a diluted layer of water at the top of the refueling cavity. In one case, unborated water was used to spray down the reactor internals to minimize airborne contamination. Because the unborated water was less dense than the borated water in the refueling cavity, the unborated water remained near the top of the refueling cavity. Additionally, because cavity water level had to be maintained relatively constant to permit continued maintenance work, the more dense borated water was removed from the bottom via the cavity drain system. After the cavity was refilled in preparation for refueling, the boron concentration in the cavity was sampled, revealing a concentration of 650 ppm near the surface, as opposed to 1950 ppm at the bottom of the refueling cavity. The stagnant layer was about two feet in depth. In the second case, demineralized water was used to spray down the cavity walls as the cavity level was being reduced after refueling. A miscommunication resulted in an excess of demineralized water being used to perform the task. As a result, a diluted layer formed at the top of the refueling cavity, reducing the RCS boron concentration from 2663 ppm to 2626 ppm. This event occurred when the source range detectors were inoperable, which would have prevented the operations staff from detecting an increase in count rate if the diluted layer passed through the active fuel region and caused a reactivity increase. With this scenario, the cavity is then postulated to be drained, lowering this diluted layer into the reactor vessel and eventually into the hot leg piping (for mid-loop operation to remove steam generator nozzle dams). The diluted water would then be drawn into the RHR suction, discharged into a cold leg, and forced into the reactor core. (Note, that based on the discussion of Scenario D above, this scenario might cause criticality and significant power generation, but does not appear capable of causing core damage.)

### 3.3.1 Loss of AC Power During Dilution - the French Scenario

As noted above, this scenario postulates a loss of offsite power while diluting toward the critical boron concentration following refueling. Charging flow is re-established on emergency diesel-generator power, and the dilution continues. However, without RCPs running and little decay heat, stagnant reactor coolant is postulated such that a volume of unborated water develops in the cold leg and reactor vessel inlet plenum, which is later swept into the core when AC power is restored and RCPs restarted.

This scenario has received more attention than any other rapid boron dilution scenario, and is the prime subject of the recent Brookhaven scoping study (ref 3.5). That reference addresses the scenario from three aspects: probabilistic, deterministic regarding mixing, and reactivity analysis.

The probabilistic assessment considers three PWRs, one from each U.S. PWR vendor (Calvert Cliffs, Oconee, and Surry). Assuming no operator action that would prevent the event, the frequency of a boron dilution transient was estimated as about 1.E-5/yr, with about a factor of two variation between the three plants, and essentially dependent only upon when RCPs are restarted. (The conditional core damage probability was assumed to vary linearly from zero to unity to zero depending on the time between charging flow restart and RCP restart.) Our following review comments are limited to the Surry station evaluation, although they may be applicable to the other PWRs as well.

The charging flow was assumed to be re-established on emergency power, but letdown flow through the regenerative heat exchanger was assumed not re-established as the most likely case, such that cold charging flow was assumed to enter the RCS. (Letdown is isolated automatically when charging stops, to prevent hot and flashing reactor coolant from entering the Volume Control Tank, or VCT). However, on Surry (as on typical Westinghouse plants), charging pumps are stripped off the bus on loss of voltage, and are not automatically reconnected when the bus is energized by emergency diesel generators. The operating procedure for loss of offsite power calls for manually establishing charging and letdown flow simultaneously. Hence, we think it unlikely (probability in the range of 0.02 to 0.2) that the operator would restore charging but neglect to restore letdown. Continued letdown would mitigate the dilution because of the enhanced mixing between warm charging flow and reactor coolant.

(In the Ringhals 2, 3, and 4 plants, the charging pumps are automatically restarted on emergency power following loss of offsite power. However, letdown is not automatically isolated, so incoming charging flow would be near RCS temperature on those plants as well. Plant designs differ, and some may have both automatic restart of charging and automatic letdown isolation.)

The dilution was assumed to be limited to the working volume of the VCT, with automatic switchover to the RWST as the VCT was emptied, since the primary makeup pump is not on a vital bus. Some Westinghouse plants have the primary makeup pumps on buses that are energized by emergency power. On some other plants, the non-vital buses could be cross-connected to emergency power. Generally, manual action would be necessary to re-energize the make-up pumps. Manual action to restart a primary makeup pump in order to continue dilution in the face of a loss of offsite power is unlikely. Among other reasons, it would be a violation of the plant Technical

Specifications which forbid dilution unless forced circulation exists. Further, operation of the makeup pump would cause audible clicks from the batch integrator, which can be heard by all occupants of the control room as a reminder that dilution is in progress. Thus, we think that continued dilution would be limited to the contents of the VCT even on plants where the makeup pump is on a vital bus.

However, by far the greatest conservatism in the Brookhaven assessment is the assumed conditional probability of core damage if an RCP is restarted just as the VCT is drained to the point that automatic switchover to the RWST occurs. Brookhaven assumed unity probability. We believe the probability is zero due to mixing between the incoming charging flow and the RCS fluid, as discussed below.

The Brookhaven report also provides very useful thermal-hydraulic analyses, based on mixing models developed for Pressurized Thermal Shock studies. Cold (or cool) charging flow is calculated to partially mix with the stagnant water in the RCS, such that the minimum boron concentration in the reactor lower plenum would be 60% to 75% of its initial value, not 0%; i.e., local boron concentration reduced from 1500 ppm to 900-1100, depending on the plant and the case.

However, we believe that even that is unduly conservative. Flow in the RCS will not stagnate following loss of AC power. As part of this ORAM study, we have conducted scoping studies on the expected RCS flow behavior following loss of AC power at hot, zero power with low decay heat. We assumed 0.05% decay heat, roughly equal to decay heat 4 months after shutdown and with replacement of one-third of the core. Reactor coolant pump coastdown takes roughly four minutes. (Large flywheels on the RCP motors provide inertia and slow decrease in flow. Reactor coolant flow decreases to 50% in about 12 seconds, to 10% in about 1-1/2 minutes, to 2% in roughly 3-1/2 minutes, and the pump impeller stops at about 4 minutes due to pump seal and bearing friction.) At the time the pump impeller stops, flow temporarily drops to 0.65%, then recovers to 0.85% (about 800 gpm per loop) within a few minutes as a result of the decay heat (0.05% of rated power). This 800 gpm flow per loop is roughly 7 to 8 times the charging flow. Based on discussions with H. Nourbakhsh (ref 3.14), author of that section of the Brookhaven report, and the equations and figures in that report, perfect mixing of the incoming charging with the RCS flow would be expected even if the charging flow had no heating in the regenerative heat exchanger. Mixing would be further assured by: (a) forceful jet impingement of the charging flow stream (about 6 feet/second) and splashing off the opposite wall of the cold leg pipe; (b) warmer charging flow in the highly likely case that letdown flow is re-established; and (c) further mixing of fluid with natural circulation flow from the other loops among the forest of instrument tubes in the reactor vessel lower plenum.

Natural circulation flow in one loop could be stopped, at least temporarily, if auxiliary feedwater flow to other steam generators subcooled them so much that all natural circulation went to them. Hot water in the reactor core outlet will not tend to rise into a steam generator containing even hotter water. Auxiliary feedwater is automatically actuated to all steam generators on loss of offsite power. An early step in typical loss of offsite power recovery procedures instructs the operator to throttle auxiliary feed flow to prevent excessive cooldown. There's some (fairly small) chance that he would terminate aux feed flow to the loop with the charging line much sooner than to the other steam generators. As mentioned, that would be a temporary condition and would not prevent natural circulation and mixing in the reactor vessel. Any temperature difference between the charging flow and the RCS, of course, will cause convection currents within the cold leg between the charging line and the reactor vessel.

Three-dimensional static reactivity calculations by Brookhaven indicated reactivity insertion rates of 2 to 9 % dk/sec ($3 to $13/sec) for assumed slug boron concentrations of 900 to 0 ppm, respectively (1500 ppm initial boron concentration). Point kinetic analyses were then performed. Most indicated relatively benign results. Some did not. These analyses were useful in suggesting the transient behavior as a first-cut estimate, but suffer from the fundamental limitation that they are point kinetic estimates. Conservative calculations showing an unacceptable result are inherently inconclusive.

Jacobson (ref 3.4) approached the puzzle in an entirely different manner. He assumed that the incoming charging flow was at the same temperature as the RCS, so no convection currents were established. The incoming unborated charging flow was assumed to fill the cold leg piping (displacing borated water) until all the unborated water was transferred from the VCT into the RCS. His estimate of the frequency of the event is about 1.E-4/yr, about a factor of ten higher than Brookhaven's, based primarily on his estimate of when RCPs would be restarted. (He assumed most cases of loss of offsite power were of short duration, and that restart of the RCPs would be controlled by the human reaction time of the operators in going through the loss of offsite procedure.) He conducted state-of-the-art hydraulics calculations involving particle tracking with the PHOENIX computer code, and put the resulting core transient spatial boron concentration into a 3-D static neutron physics code to calculate reactivity. The hydraulics calculation indicated substantial breaking up of the boron-free pocket as it was swept to the reactor core, but with large boron gradients at the core inlet -- from 2000 to 0 ppm at different locations. The reactivity transient was relatively mild for the core under study -- not enough to cause criticality. However, the patterns of boron concentration indicate that the reactivity transient would be strongly sesitive to the core loading patterns; i.e., whether regions of high reactivity coincided with locations of low boron concentrations.

If needed, the probability of the French dilution scenario could be reduced further by routing the dilution flow directly to the suction of the charging pump (termed the "alternate dilution mode"). This is the current practice at Ringhals and some other plants (ref 3.20), and is said to be preferable for other reasons (it gives faster response).

Our preceding comments on the mixing that would be caused by natural circulation apply also to these results; i.e., we don't believe a clean water pocket can be caused by this scenario.

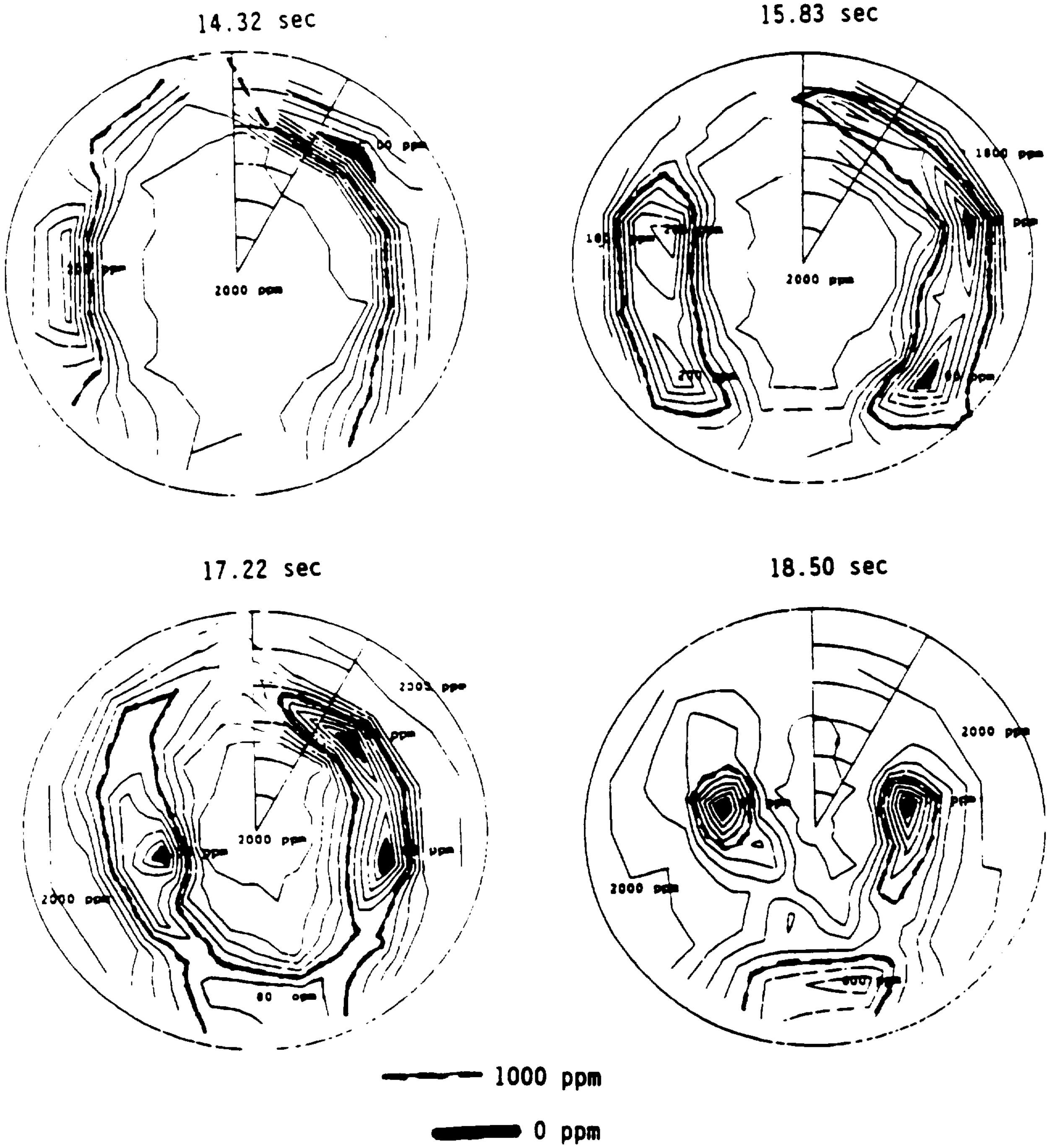
### 3.3.2 Steam Generator Inleakage - the Swedish Scenario

In this scenario, a large volume of undiluted water is postulated to leak into the RCP suction piping and steam generator from the secondary, and be swept into the core when the RCP is started as part of RCS filling and venting. Jacobson lists 5 precursor events extracted from Nuclear Power Experience, all during the period 1976 through 1982, in which inadvertent RCS dilution occurred as a result of steam generator maintenance (either tube leaks or leaks through nozzle dams). These events, plus another two events reported in 1983, are discussed in section 2.2. In addition to these seven precursors, 8000 gallons of secondary water was introduced into the Blayais (French) reactor in March, 1990, through an open steam generator tube (ref 3.15). Post-event analysis of the Blayais event showed that if the operators had not taken action to add boron and stop the dilution, criticality might have occurred in approximately four hours. In all of these precursor events, the clean water inleakage was detected and corrected long before RCS filling and venting. Based on the precursors he listed, Jacobson concluded that the estimated frequency of occurrence was greater than 1.E-8/yr and required further analysis.

Note that this scenario is identical to scenario A (Dilution during RCS Filling), and very similar to scenario E, except for the source of unborated water.

To date, this scenario has been treated extensively only by Jacobson, although Italian researchers have work in progress on it (ref 3.16). Also, analysis of incorrect startup of an isolated loop has recently been reported for a VVER plant (six-loop PWR of Soviet design), using a flux synthesis method to approximate the abnormal flux shape (ref 3.19). Reduction of boron concentration to 600 ppm in one coolant loop resulted in a peak fuel pellet enthalpy of 234 cal/gm, causing limited fuel damage.

For his assessment, Jacobson postulates a small secondary-to-primary leak that takes several days to fill the entire 280 cubic feet (2100 gallons) of the stagnant zone upstream on the RCP. (A larger leak would spill into the active region of the RCS and be detected by boron sampling as well as by mass balances.) Starting the RCP sweeps this water toward the reactor core. Particle tracking analysis with the PHOENIX hydraulics code was used to determine the boron spatial transient into and through the core. The attached Figure 3-1 shows the calculated spatial boron concentration at the core inlet at various times in the transient. Note that considerable breakup of the clean water slug has occurred, yet large variations in boron concentration exist (from 2000 to 0 ppm). The time-varying spatial boron concentration was input into a 3-D static nuclear design code (SIMULATE-3) to determine the time varying reactivity. For the reference core




'Jacobson (ref 3.4), reprinted with permission

Figure 3-1
Transient Boron Concentration at Core Inlet
for RCP Start with Clean Water Pocket*

design, he calculated a 9% k-eff increase, at rates up to 5%/sec (about $8/sec). This reactivity gain was insufficient to overcome the available shutdown margin (in excess of 15% for that core design), so no criticality occurred. A perturbation case assuming a 500 cubic foot (3700 gallons) clean water pocket indicated a 12% increase in k-eff. Hydraulic tests are now being conducted in Sweden to confirm the conservatism of the PHOENIX particle tracking model as used by Jacobson.

However, these results were calculated for a core with a large shutdown margin -- k-eff of 0.83 at the normal refueling boron concentration of 2000 ppm (more than 15% shutdown margin). Jacobson noted that extrapolating these results to a more normal core configuration, with a 10% shutdown margin (typical for Ringhals), would give an "uncomfortably small margin" considering the uncertainties involved.

In the U.S., the industry trend is toward higher enrichment cores with a 5% shutdown margin, for which the extrapolated results would be more severe. Jacobson's results indicate a 10% reactivity gain (reactivity is dk/k), and a change in k-eff from 0.83 to 0.92 is equivalent to a change from 0.95 to 1.05. Further, Jacobson's reference core had only a few isolated fresh fuel assemblies, and those on the core periphery, whereas most current U.S. cores are "low leakage", with the most reactive fuel on the inside. The relatively large regions shown on Figure 3-1 with boron concentrations of 0 to 1000 ppm could well coincide with the most reactive regions of the reactor core. (Note that a core with k-eff of 0.95 at 2000 ppm would be critical between 1600 and 1700 ppm; and that diffusion and slowing down distances in a cold PWR are short enough that criticality can occur in just a few assemblies, as illustrated by Table 3-1.) Finally, cores are being designed that require more than 2000 ppm to obtain a 5% shutdown margin, and the reactivity perturbation is proportional to initial boron concentration.

For these reasons, a straightforward extrapolation of the Jacobson results to a core representative of U.S. trends, suggesting a rapid reactivity transient raising k-eff from 0.95 to 1.05 at rates in the neighborhood of $8 per second, would be crude and not necessarily bounding.

In our opinion, no amount of analyses of this scenario, using rigor and conservatism appropriate for design basis analysis, will conclusively prove acceptable results for this scenario for all core reload designs. In fact, we strongly suspect that accurate, rigorous analysis will prove the opposite --that unacceptable results (i.e., fuel dispersal) can occur for some cases. Therefore, we conclude that this scenario must be prevented by ensuring that a pocket of unborated water does NOT exist prior to turning on RCPs.

### 3.4 Design Features and Administrative Controls to Prevent a Clean Slug Insertion Event

Technical Specification Requirements on Coolant Loop Flow - The Technical Specifications typically require either a reactor coolant loop or a residual heat removal loop to be in operation in modes 3, 4, and 5. Furthermore, if no forced circulation is available in modes 3, 4, and 5, the Technical Specifications require the operator to suspend all operations involving a reduction in RCS boron concentration.

Sample Coolant Loops prior to Reactor Coolant Pump Startup - The normal startup procedure (ref 3.17), requires all reactor coolant loops to be sampled. Additionally, the fill and vent procedure for some plants, particularly with loop stop valves, (ref 3.18) requires the reactor coolant loops to be sampled to ensure that an adequate shutdown margin exists in the reactor prior to starting the reactor coolant pumps. The sample lines for those plants are located in the crossover leg, and therefore may detect a stagnant diluted zone.

Loop Isolation Valve Interlocks - For those plants that have reactor coolant loop isolation valves, administrative controls and reactor protection grade interlocks on the operation of these valves assist in preventing the occurrence of a clean slug insertion event. If the loop has been isolated from the others by closing the cold leg stop valve, flow from the isolated portion of the loop to the remainder of the system must be established for a specified period of time (typically ranging from 90 to 180 minutes) prior to opening the cold leg stop valve. The Technical Specifications typically require the boron concentration of an isolated loop to be greater than or equal to the boron concentration of the operating loops prior to opening either the hot or cold leg stop valves of an isolated loop.

## 3.5 Mitigative Actions

Diagnosis of the Event - There are several control board indications available to the operator to assist in identifying the potential for a clean slug insertion event. As reported in reference 3.6 for scenario A, the diluted seal water will be injected into both the cold leg and the crossover leg, and some of the diluted water will be directed from the cold leg into the core. This water will be sufficiently mixed to prevent a clean slug insertion, but will result in a gradual decrease in core boron concentration. As a result, the operators will see a gradual increase in the source range count rate, thereby alerting them to the possibility of a dilution in progress. Additionally, many other control board indications, such as valve position indications, primary water flow rate, and pump breaker position, exist to assist the operator in diagnosing the ongoing dilution. These indications would allow the operator to take the appropriate corrective actions.

Emergency Boration - The most likely corrective action to be taken by the operator is to initiate emergency boration. This allows the injection of highly borated water into the reactor coolant system to compensate for the dilution already occurring. Typically, operators are instructed to initiate emergency boration if any unexplained or uncontrolled positive reactivity addition occurs.

## 3.6 Conclusions and Recommendations

Rapid reduction in local core boron concentration is a low frequency event -- no such case has been reported. However, there are several potential mechanisms which could cause such an event, and the possibility exists that a reactivity excursion with severe fuel damage could result. No recent (within the last 20 years) three-dimensional transient nuclear-thermal-hydraulic analyses have been reported, and such analyses are subject to considerable uncertainty, particularly regarding boron transport and mixing in turbulent flow through complex geometry. In principle, bounding analyses, in both conservative and optimistic directions, are within the state of the art, but have not yet been reported.

In our opinion, the so-called French scenario (loss of power while diluting) cannot cause a rapid boron dilution event because of inherent natural circulation and the mixing it will cause. Nonetheless, including instructions in procedures and training to terminate dilution following loss of offsite power seem appropriate. Such action would at least avoid a violation of the plant Technical Specifications. Consideration can also be given to use of the alternate dilution mode as the normal dilution mode during plant startup.

Also, core damage as a result of rapid boron dilution caused by the RHR system appears impossible because of the relatively slow rate at which this system can push water into the reactor core.

Only scenarios with a large volume of unborated water upstream of a Reactor Coolant Pump appear capable of causing severe core damage as a result of a rapid boron dilution. The necessary volume of unborated water appears relatively large -- comparable to the volume of the crossover leg piping itself, and should therefore be detectable by appropriate boron sampling. In our opinion, startup of an RCP with a large volume of clean water cannot be proven to be generically acceptable (but cases can probably be found that can be proven to unacceptable). Therefore, we recommend that precautions be taken to ensure a very low probability of occurrence for scenarios A, B, and E. These precautions include:

- Don't dilute unless at least one RCP is running;
- Restrict dilution to conditions under which natural circulation would be expected if the RCPs were tripped; i.e., heat transfer from the RCS to all SGs (avoid dilution when one or more steam generators are hotter than the RCS);
- When borating for a shutdown, keep at least one RCP running until the desired boron concentration is reached; e.g., to refueling concentration if in a refueling shutdown;

- If normal filling of the RCS is through the RCP seals (with the normal charging line isolated), review the administrative controls and design to ensure that the probability of inadvertent filling with clean water is vanishingly remote;

- Boron sampling of the crossover leg of the RCS is recommended prior to RCP restart AND prior to RCS filling whenever the possibility exists that water of reduced boron concentration may exist there. For most PWRs, local samples would have to be drawn from drain lines ("grab samples"). Draining of the crossover leg and refilling with borated water from the active portions of the RCS is an alternative to sampling on plants with loop isolation valves. [NOTE: Unexpected readings demand investigation. Since clean water is lighter than borated water, a clean water pocket could exist in the steam generator outlet plenum for a long time before mixing with the more dense borated water at the bottom of the crossover leg where the drain is located.]

Plant personnel should be aware of the various scenarios for getting a clean water pocket in the RCS and the hazards associated with it. Where symptoms suggest a clean water pocket might exist, procedures should exist to replace it with borated water, such as draining the region near the pocket while borating to increase core shutdown.

3.7 References


3.1 S. Salah, C. E. Rossi, J. M. Geets, "Three-Dimensional Kinetic Analysis of an Asymmetric Boron Dilution in a PWR Core", Westinghouse Nuclear Energy Systems, Transactions of the American Nuclear Society, Vol 15, p. 831, 1972. Also, "Consequences of Asymmetric Cold Water Addition to a PWR Core from an Inactive Loop" by the same authors, Transactions of the American Nuclear Society, Vol 14, p. 755, 1971.

3.2 Sven Jacobson, "Possibility to Get Prompt Criticality in Conjunction with RCP Start", Swedish State Power Board letter to Westinghouse/Brussels, R4-SVWE-739, December 1982.

3.3 NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States", April, 1989.

3.4 Sven Jacobson, "Risk Evaluation of Local Dilution Transients in a Pressurized Water Reactor", Linkoping University, Sweden, March 1992.

3.5 NUREG/CR-5819, "Probability and Consequences of a Rapid Boron Dilution in a PWR: A Scoping Study", D. J. Diamond, et al, Brookhaven National Laboratory, U.S. NRC, June, 1992.

3.6 NUREG/CR-5368 (BNL-NUREG-52198), "Reactivity Accidents", January, 1990.

3.7 Sven Jacobson, "Some Local Dilution Transients in a Pressurized Water Reactor", LIU-TEK-LIC-1989:11, Linkoping University, Sweden, 1989.

3.8 EDF Report, "Probabilistic Safety Assessment of Reactor Unit 3 in the Paluel Nuclear Power Centre", Moroni, et al, May, 1990.

3.9 NSAC-176, "Safety Assessment of PWR Risk During Shutdown Operation", Toby Burnett, et al, Westinghouse Electric Corp., Electric Power Research Institute, June, 1992.

3.10 T. Fujishiro, "Current Status of Research and Regulation Relevant to Reactivity Initiated Accidents in Japan", Japan Atomic Energy Research Institute report to the Technical Committee Meeting on Reactivity Transient Accidents, IAEA, Vienna, Austria, 17-20 Nov 1987.

3.11 NRC Information Notice 91-54, "Foreign Experience Regarding Boron Dilution", Sept 6, 1991.

3.12 A. Chesnel, "Analyses of Reactivity Accidents in PWRs", Commissariat a l'energia atomique (France), report to the Second Technical Committee Meeting on Safety Aspects of Reactivity Initiated Accidents, IAEA, Vienna, Austria, 14-17 Nov 1989.

3.13 INPO Significant Event Report 18-89, "Potential for Reduction in Boron Concentration in the Reactor During Refueling Activities", August, 1989.

3.14 H. Nourbakhsh, BNL, personal communication, August 20, 1992.

3.15 INPO Significant Event Report 13-90, "Unplanned Boron Dilutions from Shutdown Conditions, August, 1990.

3.16 G. Alloggio, et al, "Low Power Reactivity Initiated Accidents in New Generation Light Water Reactors", ENEL report at the IAEA Technical Committee Meeting on Progress in Development and Design Aspects of Advanced Water-Cooled Reactors, Rome, Italy, September, 1991.

3.17 Zion Nuclear Generating Station, GOP-1, "Plant Heatup".

3.18 Zion Nuclear Generating Station, MI-2, "Reactor Coolant System Fill and Vent".

3.19 M. Antila, et al, "Application of the Synthesis Model in an Asymmetric Reactivity Disturbance of the VVER-440 type Lovissa Reactors", ANS Topical Meeting on Safety of Thermal Reactors, Portland, OR, July, 1991.

3.20 Sven Jacobson, personal communication, Aug 27, 1992.

# 4.0 REACTIVITY CONTROL DURING REFUELING OPERATIONS

This section focuses on the risk associated with reactivity control during refuelings. Traditionally, analyses of refueling operations as presented in Final Safety Analysis Reports have focused on events such as dropped fuel assemblies and loss of spent fuel pit cooling, which have the potential for offsite radiation release. NSAC 129, Analysis of Refueling Incidents in Nuclear Power Plants (ref 4.1), provides a detailed discussion of events such as mishandling of fuel assemblies, reactor cavity draining, and excessive personnel exposures during refueling. However, neither FSARs or NSAC 129 provide a detailed discussion of reactivity control during refueling operations. Discussion in this section is limited to reactivity control in the reactor vessel. Reactivity control in the spent fuel pit is not addressed in this report because criticality in the spent fuel building caused by fuel assemblies placed in incorrect locations or by a boron dilution event is not possible. The spacing and content (borated material) of the racks, together with the boron concentration (typically 2000 ppm) in the spent fuel pool will prevent a criticality in the event that fuel assemblies are misplaced in spent fuel pit locations. Also, the spent fuel pool will also remain subcritical (with proper assembly placement) even if the boron concentration were reduced to 0 ppm.

Technical Specifications for each plant require a minimum shutdown margin for refueling operations. Since soluable boron is used to control reactivity in PWRs, larger shutdown margins are attainable in PWRs than in BWRs (Boiling Water Reactors). The minimum prescribed shutdown margin is typically 5% dk/k for PWRs during refueling operations, compared to the typical shutdown margin of 1% dk/k for BWRs. Because of their larger shutdown margin, PWRs have significantly less potential for inadvertent criticality. As discussed in NRC Information Notice 88-21 (ref 4.2), inadvertent criticalities during refueling or with a partially loaded core have occurred at two U.S. BWRs. A few years ago, when maximum PWR fuel enrichments were 3.5 w/o U-235, it was widely believed that PWR loading errors could not possibly cause criticality. That situation has changed with the current industry trend to higher enrichments.

As the trend towards longer fuel cycles and the corresponding higher enrichments for feed assemblies during core reloads continues, the potential for adverse effects of a mislocated fuel assembly or inadvertent dilution of the refueling water will increase. As a result, the need for strict adherence to controls on reactivity additions to the core will be required.

This section of the report examines reactivity control during refueling. The discussions assume that the final core design and manufacturing process ensure a 5% shutdown margin at the completion of the refueling, and that no mistake is made in the specification of the original refueling sequence. That is, only the risk of fuel assembly misloading errors is addressed.

To illustrate precursors to a potential inadvertent criticality, several industry events are examined. Next, typical refueling controls as identified from utility responses to the EPRI Outage Risk Management Survey (summarized in Appendix 4A) and to NRC Bulletin No. 89-03 (ref 4.3) are included. Finally, a probabilistic assessment of reactivity events during refueling is included.

Reactivity can be increased by boron dilution, fuel assembly misplacement, or control rod withdrawal. In most PWRs, control rods cannot be removed from a fuel assembly during refueling while the assembly is in the core, so control rod withdrawal is not considered. Boron dilution is considered briefly, but the causes of boron dilution events discussed in Section 2 are generally not applicable to refueling conditions. Therefore, the only significant contributor to inadvertent criticality during refueling is considered to be fuel loading errors.

## 4.1 Review of Industry Events Pertaining to Reactivity Control During Refueling

### 4.1.1 Boron Dilution Events During Refueling

Two recent events as described in INPO SER 18-89 (ref 4.4) illustrate the potential for significant local dilution during refueling operations. Specifically, these events involved the formation of a diluted layer of water at the top of the refueling cavity. In one event, unborated water was used to spray down the reactor internals to minimize airborne contamination. Because the unborated water was less dense than the borated water in the refueling cavity, the unborated water remained near the top of the refueling cavity. Additionally, because the cavity water level had to be maintained relatively constant to permit continued maintenance work, the more dense borated water was removed from the bottom via the cavity drain system. After the cavity was refilled in preparation for refueling, the boron concentration in the cavity was sampled, revealing a concentration of 650 ppm near the surface as opposed to 1950 ppm at the bottom of the refueling cavity. The stagnant layer was about two feet in depth. In the second event, demineralized water was used to spray down the cavity walls as the cavity level was being reduced after refueling. A miscommunication resulted in an excess of demineralized water being used to perform the task. As a result, a diluted layer formed at the top of the refueling cavity. This event was further complicated by the inoperability of the source range detectors, which would have prevented the operations staff from detecting a reactivity increase if the diluted layer passed through the active fuel region. The RCS boron concentration was reduced from 2663 ppm to 2626 ppm.

Reference 4.4 cited several recommended actions to prevent the occurrence of events similar in nature to those cited above. These recommendations include:

Briefing control room personnel on the potential for positive reactivity insertion from the time that the cavity is drained, until the vessel head is replaced;

Monitoring the boron concentration at the surface of the water in the refueling cavity following any substantial introduction of unborated water;

Ensuring at least two source range nuclear instruments are operable and monitored for reactivity changes, if possible, during refueling cavity draining;

Tightly controlling the addition of unborated water to the refueling cavity.

Passing an unpoisoned, unrodded fresh fuel assembly with an enrichment of 5.0 w/o U-235 through unborated water will not result in criticality. In addition, the manipulator crane interlocks will prevent a fuel assembly from passing through the diluted layer, as long as the diluted layer remains near the top of the refueling cavity. Therefore, a layer of unborated water is not considered a potential mechanism for inadvertent criticality during refueling. The potential for a sudden dilution event (by a layer of clean water being lowered into the reactor vessel and swept through the RHR system), is addressed in Section 3.

NUREG/CR-2798 (ref 4.5) cited three events of reactor coolant system dilution during refueling. All three events involved steam generator related dilutions. Two cases involved secondary to primary leaks due to cut or unplugged SG tubes. In one event, the RCS boron concentration was reduced from 1720 ppm to 1698 ppm, and in the second event, it was reduced from 1800 ppm to 1733 ppm. The third event involved a miscalculation of the amount of excess boric acid required to offset the demineralized water added to the RCS during decontamination of the Steam Generator tube sheets. The boron concentration was reduced from 2000 ppm to 1902 ppm during this event. No core alterations were being performed during any of the three events, and no criticality resulted. A boron dilution in the neighborhood of 300 to 400 ppm would be required to increase reactivity by 5%, which is the typical PWR Technical Specification shutdown margin requirement for refueling.

Boron dilution events are discussed in Section 2. The operating experience reviewed in that section is not indicative of boron dilution events that could potentially occur during refueling.

### 4.1.2 Potential for Reduction in Shutdown Margin (NRC Bulletin No. 89-03)

In 1989, the NRC issued a bulletin to alert PWR licensees of the potential for the loss of required shutdown margin during refueling operations (ref. 4.3). The bulletin was issued as a result of a 10 CFR Part 21 report to the NRC by a utility. The utility reported a potential for reducing the shutdown margin below the Technical Specification limit (5%

dk/k) during refueling caused by placing fresh fuel assemblies in intermediate positions during core alterations. The utility calculated that if several fuel assemblies with enrichments as low as 4.1 w/o U-235 were grouped together, the shutdown margin would be reduced to below the value required by the Technical Specifications.

Furthermore, additional calculations showed that an inadvertent criticality could result under extreme conditions if a number of highly reactive fuel assemblies were grouped together. As a result, utilities were required to review their refueling procedures to ensure that they maintained adequate shutdown margin during all refueling operations, including placing fuel assemblies in temporary core locations. A summary of the controls used by utilities to ensure adequate shutdown margin is included in Section 4.2.2

## 4.2 Typical Refueling Practices and Controls

### 4.2.1 Core Shuffle vs. Full Offload/Reload

The particular method chosen for refueling varies from plant to plant. A large majority of the plants which responded to the EPRI Outage Risk Management Survey (ref 4.12) stated that they preferred the full offload/reload method of refueling (forty-seven of fifty-four refuelings were offload/reloads). The reasons cited (Appendix 4A) by utilities for this preference are as follows (in decreasing order of being cited):

Inspection or maintenance activities which require access to the vessel, (draining of the RCS, or isolation of the residual heat removal system);

Avoiding fuel damage due to snagging grid straps;

To provide for fuel inspections;

To reduce the risk associated with mid-loop operations.

Those plants that preferred the core shuffle stated that shorter refueling times and less wear on the refueling equipment were the reasons for preferring the core shuffle.

Fuel assembly design improvements such as "non-snagging" grids may alleviate one of the more significant concerns about core shuffles. As the concern about snagging grid straps is eliminated and as spent fuel pit inventories increase, more plants may use core shuffles for their refueling. However, full offload/reloads will still be required to facilitate maintenance activities and avoid mid-loop operations. As cited above, the nature of some outage work, such as core barrel inspections, require the fuel to be removed from the core.

One typical refueling sequence for a core shuffle (as described in the Zion FSAR) involves the following:

Fuel scheduled to be discharged is removed from the core and placed in the fuel transfer system for removal to the spent fuel pit;

Partially spent fuel is transferred from the intermediate region of the core to vacated positions in the center region;

Partially spent fuel is transferred from the outer positions of the core to vacated intermediate positions;

New fuel assemblies are brought in from the spent fuel pool by the fuel transfer system and loaded into the required core locations;

Control rod assemblies are changed in the control rod change fixture prior to insertion of the new assemblies into the core.

In most PWRs, control rods can only be changed in the control rod change fixture, out of the core. In some PWRs however, equipment is available that facilitates changing control rods in the core. The capability to move two fuel assemblies at one time in the core is not physically possible in most PWRs.

For complete offload/reloads, the most common sequence (as described in Appendix 4A) is as follows:

A source bearing assembly is placed into the core near one of the source range detectors;

The process is repeated for the second source assembly;

Fuel is added around the source assembly, then the source assembly is moved to its final location, and the resulting vacant position filled;

The gap between the two source bearing assemblies is filled (bridge built) and the remaining core locations filled, row by row.

From a reactivity control viewpoint, each refueling method offers its own set of advantages and disadvantages. For the core shuffle, the majority of the fuel remains in the core throughout the refueling process. As a result, the shutdown margin for the core is closer to its limit for a greater period of time than it is for an offload/ reload. However, since a core shuffle keeps the assemblies vertical, there is less need for intermediate fuel assembly locations in order to box in twisted or bowed assemblies. (Recently, Westinghouse has begun using a "fuel loading guide" to eliminate the need for intermediate locations.) Additionally, full offload/reloads provide a more effective means of monitoring reactivity changes by an inverse count rate ratio (ICRR) plot than do core shuffles.

### 4.2.2 Typical Administrative Controls During Refueling

The following discussion of administrative controls was based on a compilation of utility responses to the EPRI Outage Risk Management Survey and to NRC Bulletin No. 89-03. The controls discussed are not intended to be complete or integrated, but rather represent examples of controls utilities use to ensure that shutdown margin is maintained during refueling.

#### Shutdown Margin and Boron Concentration Requirements

All plants require a minimum shutdown margin during refueling operations. Typically, this value is 5% dk/k, although one plant responding to NRC Bulletin No. 89-03 cited a requirement of 7.5% dk/k. The shutdown margin is maintained by requiring the boron concentration to be the larger of the value specified in Technical Specifications (typically 2000-2400 ppm) or that which will ensure a keff < 0.95. Several utilities responding to the survey indicated that a 50 ppm uncertainty is applied to the calculated boron concentration, and a 1% dk/k uncertainty is applied to the shutdown reactivity.

Response to both the survey and NRC Bulletin No. 89-03 stated that many plants place an additional administrative control that requires the boron concentration be 100 ppm higher than that specified by the Technical Specifications. Additionally, some plants require the RWST boron concentration to be as much as 300 ppm higher than that required by Technical Specifications.

To ensure that an adequate shutdown margin is maintained, the boron concentration is sampled every 72 hours as required by the Technical Specifications. Some plants increase their RCS boron concentration sampling frequency when a source range instrument is inoperable. Additionally, some plants stated in their responses to both the survey and NRC Bulletin No. 89-03 that they administratively require daily sampling. The Technical Specifications typically require the RCS boron concentration to be sampled every 12 hours if both source range instruments are inoperable.

#### Source Range Monitoring of Reactivity Changes

The Technical Specifications typically require two operable source range instruments both with continuous visual indication in the control room, and one with audible indication in containment while fuel movements are in progress. Operability of the source range detectors ensures that control room personnel can adequately monitor and assess reactivity changes during the refueling process. Additionally, ICRR plots are cited as a means of ensuring operability of the source range instruments during fuel movement, as well as a means of monitoring reactivity changes.

Fuel Movement Verification

Several methods of verifying the proper movement of fuel assemblies to and from the core were identified as part of the survey. These methods include:

Local independent verification of proper assembly location prior to the movement of the fuel. Personnel which performed this verification varied from utility to utility and included: Quality Assurance, Reactor Engineering, and Operations.

Signature of supervisory personnel, typically Reactor Engineering, Operations or both, for each completed step.

Communication with the control room during fuel movement to verify proper assembly location as tracked by tag boards. Some plants also stated they tracked fuel assembly ID numbers.

Core maps taken after refueling has been completed. Often, this is accomplished by videotaping and requires the approval of Quality Control, Reactor Engineering, and Operations. Additionally, some utilities map the spent fuel pit after offload and after core re-load to provide independent verification.

One important, but informal, feature tending to prevent criticality due to loading errors was pointed out by station personnel during personal interviews, but was not mentioned in the written survey responses. A large increase in reactivity due to loading errors can occur only if fresh fuel assemblies, or those without control rods, are improperly clustered together. However, fresh assemblies are bright and shiny and easily distinguished from irradiated assemblies which are dull, dark, and emit a blue glow. The presence or absence of control rods is also readily observable from the containment operating deck, looking down into the core. In addition, a TV monitor may visually display the core in the control room as it is being loaded. Therefore, any cluster of fresh fuel assemblies without control rods would be obvious to observers as the cluster was being formed. Senior Reactor Operators, assuming that they are aware of the potential hazard of clusters of fresh, unrodded fuel assemblies, would be expected to observe a cluster being formed and stop the reloading.

Deviations from Specified Refueling Sequence

For the planned refueling sequence, plants evaluate the most reactive condition expected, typically at or near the final core design state, to ensure that the minimum required shutdown margin is maintained. Deviations from the specified refueling sequence are only permitted under special circumstances. These circumstances include temporary storage of an assembly and "boxing" of a twisted or bowed assembly. Deviations from the specified refueling sequence typically require two approvals: a Senior Reactor Operator (SRO) and one other, typically from the plant's reactor engineering staff.

In responses to both the EPRI survey and NRC Bulletin No. 89-03, most utilities stated they use fuel vendor guidelines as a means of ensuring that the shutdown margin is maintained during intermediate fuel loading conditions. In their responses to NRC Bulletin No. 89-03, several utilities also cited specific controls governing the temporary placement of fuel assemblies in core locations. Typical controls cited in the responses to NRC Bulletin No. 89-03 are listed below.

> If assemblies are temporarily placed along the core baffle, no other fuel assemblies are allowed to be placed next to those assemblies. Furthermore, at least one vacant core location (one utility cited two vacant core locations) must be between the assembly and the remainder of the core.
>
> When it is necessary to straighten a twisted or bowed assembly by "boxing" the assembly, the assemblies used in making the "box" should be assemblies in their final location. If that is not possible, several utilities cited guidance for cases in which the assemblies used are not in their final location. These included:
>
> > one of the assemblies making up the "box" must have a face adjacent to a vacant core location or the baffle; or,
> >
> > if it is necessary to use fresh fuel assemblies to make the "box", the fresh fuel assemblies used must have 20 or more standard fresh burnable absorber rodlets, or a control rod insert; or,
> >
> > the reactivity worth of the assembly being temporarily placed in the core location must be less than or equal to the reactivity worth of the final fuel assembly to be placed in that core location.
>
> One utility stated that if the reactivity worth of the assembly temporarily stored in the core location exceeds the reactivity worth of the assembly which will be finally stored in that location, then placing the fuel assembly in that temporary location would only be permitted if the following were met:
>
> > no more than 2 control rods shall be withdrawn from that part of the core containing fuel; and,

fresh assemblies shall not be placed next to one another unless it is specified in the final core design; and

a dummy fuel assembly shall be used on one face of the core location being boxed in.

For core shuffles, some utilities stated in their responses to NRC Bulletin No. 89-03 that assemblies are permitted only in their initial and/or final core location.

Additionally, two utilities stated in their responses to NRC Bulletin No. 89-03 that all burnable poison rods or control rods must be inserted into their respective assemblies prior to insertion of the assembly into the core.

## 4.3 Probabilistic Assessment of Reactivity Events Due to Refueling Errors

### 4.3.1 Sequence of Events

This section considers the scenario, probability, and consequences of a PWR criticality caused by refueling errors. Three inadvertent criticality events have occurred in BWRs during refueling (ref 4.2), but none in PWRs. (PWRs typically have much larger shutdown margin requirements during refueling (5% vs 1%), and involve no more than one control rod movement at a time.)

Only fuel assembly misloading errors are considered here; i.e., departing from the prescribed loading pattern. Evaluating the probability of gross error in the nuclear design, manufacturing process, or in the translation to the loading sequence, is beyond the scope of this study.

The possibility of a single assembly going critical in a local pocket of unborated water during its movement from the spent fuel pool to the core was considered, and rejected as being vanishingly remote. At least two instances of unborated water pockets have been reported (ref 4.4). The unborated water, being less dense, will tend to stay at the top of the refueling cavity, and fuel assemblies are not raised to the surface of the water in the cavity. Also, a single unpoisoned, unrodded fuel assembly with an enrichment of 5.0 w/o U-235 would not become critical in unborated water. (A clean water pocket being swept into the core as the cavity water level is lowered after refueling is addressed in section 3.)

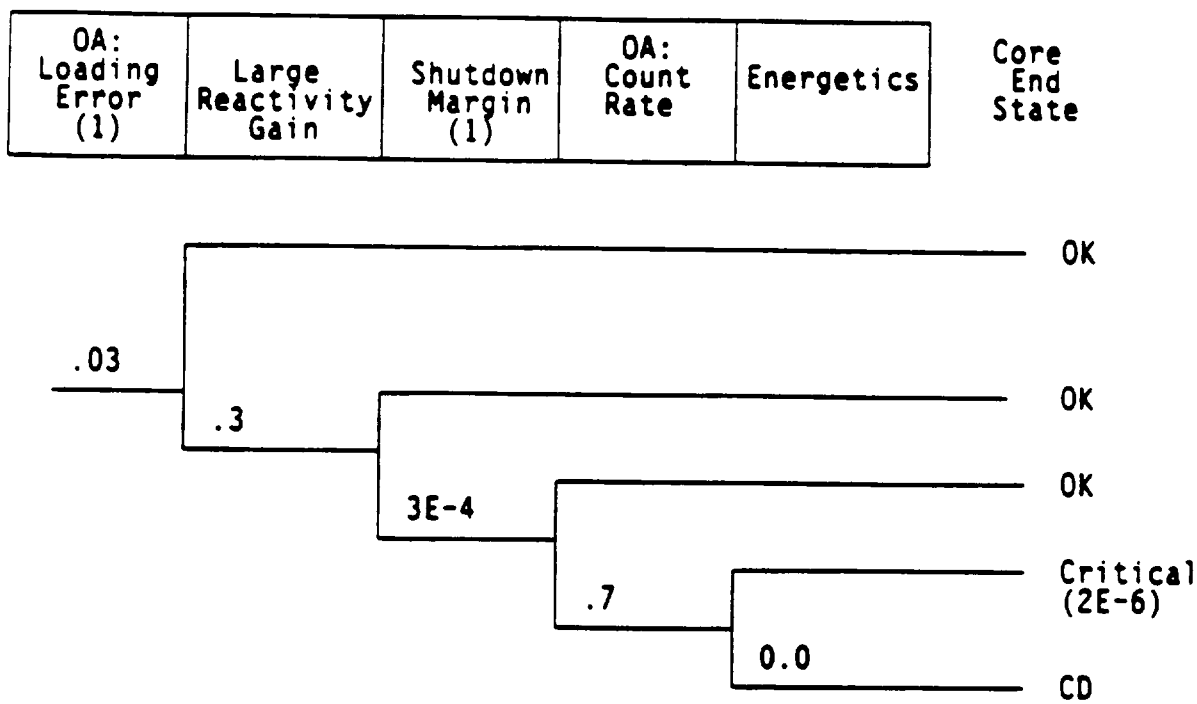
With enrichments less than 3.5 w/o U-235, criticality is not a credible event as long as the refueling boron concentration (typically 2000-2400 ppm) is maintained. With enrichments of 4.5 to 5.0 w/o U-235, criticality is possible with only a few assemblies mislocated. Although only one utility surveyed currently uses fuel in the 4.5 to 5.0 w/o U-235 range, the Westinghouse Commercial Nuclear Fuel Division reports that the current industry trend is toward 4.5 to 5.0 w/o U-235.

Since no single refueling error can cause criticality, one must postulate that prior errors (either refueling or boration) had gone undetected at the time of the refueling error. If the assembly being inserted causes a significant reactivity increase, the source range count rate will increase. An increasing count rate might alert personnel upon approaching criticality, and the fuel assembly would be raised up out of the core region.

If criticality occurs, neutron flux would increase at an exponential rate dependent upon the excess reactivity. Normal procedures require checking the count rate prior to unlatching the crane from the assembly. If the increase is slow, the operators would notice the increasing count rate prior to thermal-power being generated, and raise the assembly. If the increase is rapid, a containment evacuation alarm on high source range count rate might occur before the crane operator had time to respond to the increasing count rate. When the containment alarm occurs, the crane operator, in response to his instructions, would be expected to stop inserting the assembly and evacuate containment. In this case, neutron flux (and power) would continue to increase to the thermal power range. Since a positive moderator temperature of reactivity is likely, heating of the coolant would accelerate the power increase. Fuel temperature increase would cause negative reactivity due to the doppler coefficient, but would not be expected to stop the power increase short of steaming (since the moderator temperature coefficient would most likely be larger in magnitude than the fuel temperature coefficient). Steam formation would temporarily reduce nuclear power generation. As the steam leaves the affected assemblies and is replaced with water, power would again increase. This unstable chugging would continue until the reactor was shutdown, most likely by manual boration from the control room (or possibly by the crane operator raising the assembly). Fuel assembly damage (beyond minor clad cracks caused by rapid fuel heating) may be impossible, but has not yet been demonstrated to be impossible.

### 4.3.2 Event Tree Description

The event tree for refueling criticality is shown on Figure 4-1. The initiating event is a refueling error that places a fuel assembly in an improper location, an event that requires multiple procedural errors. Top events (defining event tree branches) are defined as follows:




NOTES:

1. MULTIPLE PROCEDURAL ERRORS REQUIRED
2. CRITICALITY MIGHT BE
   A. STOPPED BEFORE THERMAL POWER GENERATION (50%)
   B. UNSTABLE STEAMING (CHUGGING) (50%)
3. NEGLIGIBLE PROBABILITY OF RADIOLOGICAL RELEASE

Figure 4-1
PWR Fuel Misloading Event Tree
(Best Estimate Probabilities)

Large Reactivity Gain - Many core reload errors will not significantly increase reactivity. Instead, the error may decrease reactivity or have no net effect on reactivity. For example, errors in placing a new fuel shipment in the spent fuel pool racks may cause an identical fuel assembly to be loaded into a given core location. This was the case in the one instance of multiple errors reported in the PWR refueling survey (Appendix 4A). However, a refueling may involve more than one enrichment, and the burnable poison inserts (BPIs) may differ among fresh fuel assemblies.

Shutdown Margin - No single error (of misloading an assembly) can cause criticality as long as the 5% shutdown margin required by the Technical Specifications has been maintained. Therefore, unless either; a) previously undetected reload errors, or b) boron dilution has occurred, no criticality is possible. (Note that the product of the "Reactivity" and "Shutdown Margin" events represents the conditional probability that criticality can be achieved by the initiating refueling error.)

Count Rate - As the fuel assembly is lowered into the core and criticality is approached or reached, subcritical multiplication will increase, increasing the count rate. The increased count rate may alert the crane operator to stop lowering the fuel assembly prior to reaching criticality.

Energetics - This branch addresses the likelihood that the combination of initial core reactivity, and the rate and magnitude of reactivity being added by the assembly being lowered into the core, will cause a power burst sufficient for disruptive fuel failure. For design basis accident analysis (control rod ejection), the NRC conservatively sets 280 cal/gm (504 BTU/lb) as the threshold below which fuel damage is not expected to be sufficient to jeopardize core cooling (refs 4.6 and 4.7). Above 280 cal/gm, dispersal of molten fuel into the coolant becomes a possibility. For irradiated fuel, 200 cal/gm is commonly taken as the threshold for fuel damage.

The end-states considered are as follows:

> OK - No criticality despite the procedural error (and a possible Technical Specification violation).
>
> Inadvertent criticality - This end-state extends from marginal criticality and a neutron flux that never exceeds the source range, up to and including steaming and power oscillations (chugging). Some cladding failure is conceivable due to strain caused by rapid fuel pellet thermal expansion, but such cladding failure is not considered core damage in the context of PRA assessment.
>
> Fuel assembly damage - Above the 280 cal/gm fuel enthalpy threshold, dispersal of molten UO2 into the coolant becomes possible, with the potential for disassembly of some fuel assemblies. Even with molten fuel dispersal and

mechanical assembly damage (if such a consequence is even possible), the possibility of a significant radiological release is considered too remote to be a concern. Damaged fuel would be predominately unirradiated; more than 35 feet of water covers the core; and containment integrity is maintained. Therefore, the event tree is not developed beyond the possibility of fuel assembly damage.

### 4.3.3 Event Tree Quantification

Frequency of Initiator - Two fuel assembly misloading errors were reported in fifty-four reloads in the PWR survey of outage experience, or an average of 0.04 per reload. The value may be significantly lower in the future as a result of additional controls recently applied (ref 4.3), and is not likely to be substantially higher.

Initiating Event frequency: 0.1 to 0.01/reload (upper & lower bounds)

Best estimate: 0.03/reload

Reactivity - Intuitively, one expects that the majority of reload errors will not cause a significant increase in reactivity. Mislocating fresh fuel assemblies in the spent fuel racks upon initial receipt would cause zero reactivity effect if identical assemblies are switched.

Upper/lower bound: 0.5 to 0.1 (probability that a reload error causes a significant increase in reactivity)

Best estimate: 0.3

Shutdown margin - Insofar as is known to the authors, only one case has been reported in PWR operating experience in which the shutdown margin could have been reduced below the value of 5% required by the Technical Specifications (ref 4.3). However, many of the previous refuelings were with lower enrichments such that much more than a 5% shutdown margin was available. One could argue that the source range count rate would indicate any reduction in shutdown margin so severe that a single fuel assembly misload would cause criticality. Normal trending of source range count rate is a powerful defense against major loss of shutdown margin (e.g., from 5% to 1%), but is not considered a certainty, since the normal count rate may change considerably in the course of refueling. Boron dilution could also cause a loss of shutdown margin. Some cases of boron dilution have been reported during refueling shutdown, but none sufficient to cause a large reduction in shutdown margin. On the basis of the virtual absence of precursor events during approximately one thousand PWR refuelings, the estimated frequency of a loss of shutdown margin so severe that one further misload could cause criticality is taken as one in three thousand refuelings, give or take a factor of 10.

Upper/lower limits: 0.003 - 0.00003/refueling

Best estimate: 0.0003/refueling

Count rate - The Technical Specifications typically require an audible count rate indication inside the containment. The crane operator might be alerted to an increasing count rate and halt assembly lowering prior to criticality. The probability of crane operator success is not considered high for the following reasons. Most of the reactivity added by a fuel assembly would be added in the first few feet of insertion into the core. Typical refueling practice (for offload-reload) is to position the assembly about 2 inches away from other assemblies to prevent grid strap hang-up, and then lower it at a high speed (up to 21 feet/minute), until the assembly is close to the bottom. (The 2" extra water gap is assumed here to have no effect on reactivity.) The neutron source and detectors are at the bottom of the core. Criticality with a very high flux peak at the top of the core might occur within a few seconds of beginning to lower the assembly, before there was a noticeably large increase in detector response. (Flux shape, rather than the speed of neutron diffusion is important. The neutron detector is shielded from the flux increase at the top of the core by a subcritical medium, and could be dominated by subcritical multiplication of the source until there was a very large increase in flux at the top of the core.) The probability is therefore estimated as:

Upper/lower bound - 0.9 to 0.5 (probability of failing to halt assembly travel prior to criticality.)

Best estimate - 0.7

Energetics - No analyses have been found for the potential reactivity transient resulting from insertion of a more reactive assembly into a given location. Three-dimensional dynamic analyses of this nature are well within the state of the art, and much more tractable than the rapid boron dilution cases discussed in Section 3. However, they are beyond the scope of this report. The following discussion, therefore, is neither more nor less than expert speculation as to what the results of such a calculation would be. It is not clear, based on discussions with accident analysts, whether fuel dispersal is physically possible due to misloading fuel. If loading the assembly is hypothetically assumed to increase k-eff from 0.99 to 1.02 (as an extreme case), with half the reactivity gain in the first foot of travel (and 3/4 in the second foot, and so on), and an insertion rate of 21 feet/minute is assumed, the reactivity insertion rate would be about 300 pcm/sec at the time prompt criticality is achieved. Whether such a severe reactivity transient would cause fuel dispersal is uncertain. Fuel dispersal cannot occur without partial fuel melting at the hot spot, or a hot pellet average temperature of about 5000 Deg-F. (For a rapid power burst, fuel heatup is near adiabatic, with little difference between average and centerline temperatures.) However, a fuel temperature increase of only 200 Deg-F on a core-wide basis would cause roughly 300 pcm of negative Doppler feedback. If the conservative approach used for design basis FSAR analyses of rod ejection accidents is applied, a very conservative doppler weighting factor versus peaking factor would be used, and the resultant calculated peak fuel enthalpy might be in excess of 280 cal/gm. Conceptually, however, one is tempted to consider the fuel configuration as a very small core with a large multiplying reflector. If doppler feedback with such a model were assumed to be one pcm per Deg-F increase in hot spot temperature, the calculated peak fuel temperature increase would only be a few hundred degrees F, even with a reactivity transient as severe as 300 pcm/sec, prior to shutdown on steam voids. Positive feedback from moderator heating would increase the reactivity, and could cause a larger fuel temperature increase. We therefore estimate the probabilities as:

> Upper/lower bounds - 0.01 to 0 (conditional probability of rate and magnitude of reactivity insertion being severe enough to cause fuel dispersal)
>
> Best estimate - 0 (We think it is impossible)

Therefore, we estimate the frequency of inadvertent criticality due to refueling errors as between 1.0E-4 and 1.0E-8 per refueling, with a best estimate of 2.0E-6 per refueling. The frequency of fuel damage due to refueling errors is estimated to be between 1.0E-6 and 0 (impossible) per refueling, with our best estimate being zero.

## 4.4 Conclusions

Fuel misloadings in PWRs are uncommon, but do occur. Loading errors have been discovered by independent verification of the assembly prior to insertion of the assembly into the core, mapping the spent fuel pit rack and/or core, or when attempting to retrieve/insert a fuel assembly from/into a spent fuel pit/core location. More consistent and uniform application of the above methods would reduce the likelihood of a fuel misload.

Observation of the changing core pattern by Senior Reactor Operators (SROs) during fuel loading can also significantly reduce the risk of inadvertent criticality, provided the SROs are aware of the hazards associated with clusters of fresh or unrodded fuel assemblies.

Although there are wide uncertainty bounds on the estimates, refueling errors do not appear to be meaningful contributors to public risk. There are no offsite radiological consequences due to inadvertent criticality (estimated as no greater than 1.0E-4 per refueling and probably about 1.0E-6). Even fuel damage (estimated as less than 1.0E-6 per refueling, and probably impossible) would have a vanishingly remote likelihood of an offsite radiological release in excess of 10 CFR 20 limits, although worker safety could not be assured if significant fuel damage occurred. Although three dimensional neutronic-thermal-hydraulic analyses can be done on refueling errors, none have been performed to date.

## 4.5 Comparison With NUREG/CR-5771

Shortly after the original draft of this report was prepared, a Brookhaven report on the same issue became available ("Probability and Consequences of Misloading Fuel in a PWR", NUREG/CR-5771, August 1991, ref 4.8). Despite substantial differences in component probabilities, NUREG/CR-5771 also supports the conclusions that the risk of criticality from PWR misloading is low, and that the associated public risk is negligible.

Salient features of NUREG/CR-5771, and our review comments comparing their work with ours, are discussed below. Table 4-1 compares frequencies estimated in the two works.

NUREG/CR-5771 addresses cycle 9 of Calvert Cliffs 2 (a C-E PWR, using fuel with enrichments up to 4.3 w/o), and evaluates refueling practices and guidelines instituted after NRC Bulletin 89-03 (ref 4.3). The reactivity effects of this core are representative of the current PWR trend toward higher enrichments, except in one important respect -- the shutdown margin is much greater than expected in the majority of plants with higher enrichment fuel. The properly loaded cycle 9 core with all control rods present and 2300 ppm boron (required by the Technical Specifications) has a calculated shutdown margin of 13%, much greater than the 5% required by Tech Specs. For additional safety margin, the shutdown margin is calculated assuming all control rods are out. In contrast, most PWRs base their shutdown margin calculations on all control rods being inserted.

Because of this additional 8% shutdown margin, multiple refueling errors, clustering at least four fresh assemblies without control rods together, would be required to violate the Technical Specification shutdown margin requirement. No such additional safety margin was considered in section 4.3, such that a single loading error could be assumed to cause a Technical Specification violation. (As reported in the PWR refueling survey results in Appendix 4A, PWR stations typically do provide some additional safety margin, even though it is not as large as 8%.)

Brookhaven calculated the change in the shutdown margin for several sets of misloaded fresh, unrodded fuel assemblies clustered together. Their results are shown below:

| Number of refueling errors (Number of fresh, unrodded assemblies clustered together) | Reduction in shutdown margin, % dk/k |
|---|---|
| 1 | 0.5 |
| 3 | 4.0 |
| 5 | 7.8 |
| 9 | 12.6 |

Table 4-1
Comparison of BNL and Oram Frequencies
Events/Reactor-Year[1]

| | NUREG/CR-5771[2] | PWR ORAM (Preliminary) Max/Min | Best Est |
|---|---|---|---|
| Single fuel loading error causing significant increase in reactivity[3] | 0.4[4] | 0.05/0.002[5] | 0.01 |
| Violate Tech Spec shutdown margin | 1E-6[6] | NA | NA |
| Gross reduction in shutdown margin | NA | 3E-3/3E-5 | 3E-4 |
| Criticality | 1E-8 | 1E-4/1E-8 | 2E-6 |
| Core Damage | 1E-8[7] | 1E-6/0 | 0[8] |

Notes:

(1) For the precision in this table, a frequency of one event per reactor-year is equivalent to one event per refueling outage.

(2) NUREG/CR-5771, "Probability and Consequences of Misloading Fuel in a PWR", August, 1991.

(3) A "significant" increase means a few tenths of a percent dk.

(4) Derived from data in Table 3-2 of NUREG/CR-5771, considering 68 type A locations and 89 type B locations.

(5) Based on 0.01 to 0.1 reloading errors per refueling, and a 20% to 50% chance that a refueling error would cause a significant reactivity gain.

(6) Based on a cluster of 4 refueling errors being required to reduce the shutdown margin from its normal refueling value just below the Tech Spec requirement.

(7) Conservatively assuming all criticality events cause fuel damage.

(8) Based on engineering judgement that the reactivity transient, with feedback, cannot be so severe as to cause significant fuel damage.

Allowing for variations from Calvert Cliffs to other cores of other plants, these results indicate that perhaps as few as three refueling errors could cause criticality if the correctly loaded core had only the minimum Technical Specification shutdown margin (5%). We believe this conclusion to be reasonable.

Brookhaven's probability analysis indicates that the expected frequency of reloading errors tending to increase core reactivity is roughly 0.4/refueling. We believe that to be a conservative estimate, since only two errors in fifty-four refuelings are reported in Appendix 4A.

Because of the large shutdown margin compared to the Technical Specification requirement, Brookhaven calculated a frequency of violating the 5% Technical Specification shutdown margin requirement of 1.0E-6/reactor-year (their central estimate). In contrast, we estimated a frequency of 3.0E-4/reactor-year (give or take a factor of ten) for a "gross" reduction in shutdown margin; i.e., to a shutdown margin less than 2%.

Brookhaven also performed k-inf calculations for a 4.08 w/o assembly without burnable poison rods, at a boron concentration of 2000 ppm. These calculations indicated a moderator temperature coefficient of reactivity of +8.6 pcm/Deg-F, a moderator void coefficient of 130 pcm/% over the range of 0 to 40% void, and a doppler coefficient of -1.6 pcm/Deg-F. These results led to Brookhaven's conclusion that the excursion would become more severe as the moderator heated and began voiding. We agree that a positive moderator temperature coefficient may exist, and that this positive feedback could cause a temporarily shorter period if power were increased slowly into the power range. However, the actual moderator feedback would be much less positive than the values cited, since k-inf rather than k-eff was calculated. The difference between k-inf and k-eff is the leakage, and leakage always contributes a negative component to the moderator temperature coefficient. The more peaked the power distribution, the more significant the leakage term is-- and criticality in a misloaded core would be with a very peaked power distribution. If the power is increasing rapidly, the fuel temperature will increase much more rapidly than the moderator, and the negative doppler feedback will dominate. A positive moderator coefficient, we believe, would only reduce the effectiveness of the doppler coefficient for a fast transient, not override it completely.

Brookhaven estimated a human failure probability of 10% for failure to stop lowering an assembly because of an increasing count rate while lowering it. Our estimate of 50%-90% failure probability is based on the expectation that criticality would be achieved soon after the assembly enters the top of the core, and the fact that the neutron source and detector are at the bottom.

Brookhaven estimated a frequency of criticality of 1.0E-8/year, compared to our estimate of 1.0E-6 (give or take two decades). Considering the vast differences in approach and in estimates of various components, that difference is relatively small. Both results support the conclusion that a criticality event during refueling, while possible, is a very low probability event.

But nowhere is the Brookhaven/Westinghouse difference more pronounced than in the conditional probability of fuel damage given criticality. Brookhaven conservatively assumed that probability to be 1.0. Westinghouse judgement (best estimate) is that significant fuel damage is impossible from such an event. The difference stems from differing perspectives (or judgements) regarding the possible reactivity transient. This point has been discussed (ref 4.9) with the principle author of NUREG/CR-5771, and he agrees that our estimate (admittedly speculative) appears credible, and may be accurate. We have also consulted with INEL to see if any of the SPERT or PBF tests were applicable (ref 4.10). No tests were directly applicable, although SPERT-3 (especially SPERT-3E) used rodded, low enrichment uranium dioxide. The SPERT tests generally had much higher reactivity insertion rates than can be achieved while loading fuel, and had no positive moderator coefficients. These SPERT tests demonstrated that oxide-type fuel can take over $2 in reactivity without damage at low pressure and temperature. Analyses of the transient are well within the state of the art for three-dimensional neutronic calculations, but are beyond the scope of this study. Until such analyses are done, we will stick with our prediction of what they will show.

Brookhaven also provided scoping calculations of worker dose, and these results are worth mentioning here. For the gamma and neutron dose to workers in containment due to power generation, Brookhaven calculated that "total dose rate is about 0.03 mrad/hr, which is well below any limit of concern". Worker whole body immersion and inhalation dose rates of 0.1 rem/sec and 0.7 rem/sec, and a thyroid inhalation dose rate of 24 rem/sec, were calculated based on assuming: (1) gap activity released from twelve irradiated assemblies surrounding the improper fresh fuel cluster; (2) activity 4 days after shutdown; and (3) other assumptions recommended by Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident In the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (ref 4.11). The high inhalation dose rate, due entirely to the assumption of substantial fuel damage, would be likely to cause early health effects. Without significant fuel damage, worker dose rates would be negligible.

## 4.6 References


4.1 NSAC 129, "Analysis of Refueling Incidents in Nuclear Power Plants", December 1988.

4.2 NRC Information Notice 88-21, "Inadvertent Criticality Events at Oskarshamn and at U. S. Nuclear Power Plants", May, 1988

4.3 NRC Bulletin No. 89-03, "Potential Loss of Required Shutdown Margin During Refueling Operations", November, 1989.

4.4 INPO Significant Event Report 18-89, "Potential for Reduction in Boron Concentration in the Reactor During Refueling Activities", August, 1989.

4.5 NUREG/CR-2798 (ORNL/NSIC-208), "Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants", July, 1982.

4.6 Regulatory Guideline 1.77, "Assumptions Used for Evaluating A Control Rod Ejection Accident for PWRs", May 1974.

4.7 NUREG-0800, "Standard Review Plan", Revision 2, Section 15.4.8, July 1981.

4.8 NUREG/CR-5771 (BNL-NUREG-52294), "Probability and Consequences of Misloading Fuel in a PWR", August, 1991.

4.9 D. J. Diamond, BNL, personal communication, November 1991.

4.10 Robert Watkins and Jim Crocker, Idaho National Engineering Laboratory, personal communication, Oct 30, 1991.

4.11 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident In the Fuel Handling and Storage Facility for Boliing and Pressurized Water Reactors", U.S. NRC Regulatory Guide 1.25, 1972.

4.12 NSAC-174, "Survay of PWR Plant Personnel on Shutdown Safety Practices and Risk Management Needs", R. Prokopovich et al, Westinghouse Electric, for EPRI ORAM Program, March, 1992.

## APPENDIX 4A

### Outage Risk Management Survey

Part of the EPRI/Westinghouse Program for Outage Risk Assessment and Management involved a survey of PWR utilities regarding their outage planning and operations practices. The responses provide very useful insights into current practices. A portion of that survey dealt with recent refueling experience. Seventeen utilities covering eighteen plants (fifty-four recent refuelings) responded to the refueling questionnaire. These responses are summarized below.

Care must be taken in using quantified results from the survey. Although the experience reported provides valuable insight, the survey was not intended, nor is it suitable, for a rigorous statistical data base.

### Choice of Refueling Method

As part of the survey, the responding utilities indicated the method selected for each of their last three refuelings. The methods used for a total of fifty-four refuelings were discussed in the survey results. Of these fifty-four refuelings, forty-seven were full offload/reloads. The full offload/reload method of refueling was preferred by twelve of the seventeen utilities who stated a preference in their refueling method (one utility cited no preference). The reasons cited (in decreasing order of frequency of being cited) for selecting the offload/reload method include:

> Inspection or maintenance activities which require access to the vessel, (draining of the RCS, or isolation of the residual heat removal system);
>
> Avoiding fuel damage due to snagging grid straps;
>
> To provide for fuel inspections;
>
> To reduce the risk associated with mid-loop operations.

Five utilities expressed a preference in performing a fuel shuffle as opposed to a full offload/reload. Additionally, one utility cited that they would prefer to use a fuel shuffle, but the scope of the outage work usually dictated that a full offload/reload be performed. Utilities cited quicker refueling times (typically about one day less for core shuffles) and reduced wear on the fuel handling equipment as reasons for their preference. Three of the utilities that preferred the fuel shuffle have Westinghouse two loop plants. The other two utilities have Babcock and Wilcox plants. Although their stated preference was to perform a fuel shuffle, they were often forced to perform a full offload/reload to complete required inspections or maintenance.

Of the seventeen utility responses, eight of them indicated that refueling, especially the offload of fuel for those that selected this method of refueling, was typically critical path. However, only five cited this as a consideration in selecting the method for fuel movement. The three utilities that have Westinghouse plants that preferred the fuel shuffle stated that refueling was not on critical path. The two utilities that have Babcock and Wilcox plants stated that refueling was on critical path.

Shutdown Margin Requirements

Fifteen of the seventeen utilities which responded to the survey require a minimum shutdown margin of 5% dk/k during refueling. Nine of the utilities that responded cited that the minimum boron concentration must be at some specified value (typically 2000 ppm) or that concentration which ensures a shutdown margin of 5% dk/k. Eleven of the utilities cited some additional margin for uncertainty applied to the shutdown margin. Typically, a 50 ppm uncertainty is applied to the boron concentration, and a 1% dk/k uncertainty is applied to the shutdown reactivity. Additionally, several utilities cited administrative requirements which require boron concentrations 100 ppm higher than the minimum required to add margin, or account for dilution due to steam generator primary side decontamination.

As a result of the industry trend towards longer fuel cycles and higher enriched fuels, boron concentrations higher than 2000 ppm are required to maintain the minimum shutdown margin. All but six of the utility respondents indicated that they presently use assemblies enriched to greater than 4 w/o U-235. Seventeen utilities responded to the question regarding the maximum fuel enrichment which has been used, or will be used in the next refueling outage. The following table shows a more complete break down of the survey responses.

| Maximum U-235 Enrichment | Number |
|---|---|
| w/o < 4 | 6 |
| 4.0 ≤ w/o < 4.5 | 11 |
| 4.5 ≤ w/o < 5.0 | 1 |
| w/o ≥ 5.0 | 0 |

## Source Range Monitoring

All plants require the operability of the source range instruments to be verified prior to and during refueling operations. The Technical Specifications typically require an analog channel operational test to be performed prior to fuel movement (within eight hours). Additionally, the first assembly inserted into the core is a source bearing assembly and the response of the source range channel is verified. During refueling operations, a channel check is performed every twelve hours. Additionally, all utilities perform a 1/M plot as part of full core reloads and core shuffles, and use the plots to verify the response of the source range instruments. Only one of the seventeen utilities responding to the survey stated that they had used portable source range monitors (for the initial core load). Additionally, one utility stated that although portable source range monitors had never been used during their refuelings, the Technical Specifications allow their use if a permanently installed source range monitor becomes inoperable.

## Typical Refueling Practices

### Reload Pattern (Applies Only to Offload/Reload Method)

Fourteen of the sixteen utilities responding to the survey stated that they bridge the two primary source bearing assemblies first, and then fill in the remaining assemblies row by row. One utility indicated that they formed a crescent shape around the core periphery connecting three excore detectors 90° apart. They then fill in the remaining core locations working towards the opposite side of the core. For the remaining utility, the reload pattern is developed around the secondary sources such that for a tic-tac-toe (i.e., 3x3 grid) pattern is formed. The reload starts at the north, proceeds south, northeast, northwest, southeast, southwest, center, west, and finally east.

### Control Rod Changeout (During Offload/Reload)

Thirteen of the sixteen utilities indicated that control rod changeouts are performed in the spent fuel pit. One of the utilities that does not change the control rods in the spent fuel pit cannot do so because their only control rod assembly change fixture is located in the containment refueling cavity. The control rods are typically moved in the spent fuel pit during the offloading of the core because it has less potential for impacting the schedule.

### Fuel Movement Verification

Several methods of verifying proper movement of fuel assemblies to and from the core were identified as part of the survey. These methods include:

Local independent verification of proper assembly location prior to the movement of the fuel. Personnel which performed this verification varied from utility to utility and included: Quality Assurance, Reactor Engineering, and Operations.

Signature of supervisory personnel, typically Reactor Engineering, Operations or both, for each completed step.

Communication with the control room during fuel movement to verify proper assembly location as tracked by tag boards. Some plants indicated that they tracked fuel assembly ID numbers.

Core maps taken after refueling has been completed. Often, this is accomplished by videotaping and requires the approval of Quality Control, Reactor Engineering, and Operations. Additionally, some utilities map the spent fuel pit after offload and after core reload to provide independent verification.

Deviations from Refueling Sequences

Sixteen of seventeen utilities responding to the survey cited that deviation from the refueling sequence is permitted for cases of bowed or twisted assemblies. These deviations from the original refueling sequence are proceduralized using the equivalent of fuel handling deviation reports. Deviations from the original refueling sequence typically require the approval of a Senior Reactor Operator and a Reactor Engineer. To ensure that no assemblies are misplaced during the process, fuel movements associated with the deviation are tracked and verified in the same manner that normal fuel movements are tracked and verified.

One utility cited an additional event which would require deviation from the reload sequence. If a source range detector were to fail, the secondary neutron source(s) would be temporarily stored 90 degrees offset from their final location with respect to the detector.

Controls to Ensure Shutdown Margin is Maintained During Refueling

All of the utilities responding to the survey cited some form of refueling guideline to ensure that adequate shutdown margin was maintained during refueling. Typically, these controls were provided by the fuel vendor. They also included Technical Specification requirements. The survey cited the following controls used to ensure that shutdown margin is maintained during refueling:

Evaluation of the refueling sequence by Reactor Engineering to ensure that shutdown margin is maintained during the refueling sequence. Additionally, Reactor Engineering evaluates any proposed changes to the refueling sequence;

Verification of refueling boron concentration once every 72 hours (as required by Technical Specifications). One utility verifies the boron concentration once every 24 hours;

Restrictions on the number of assemblies which can be placed in core locations at the core periphery. Additionally, these locations must have open core locations between themselves and the core locations already filled with fuel in the final positions;

Prediction of the approach to criticality using inverse count rate ratio (1/M) plots.

Additionally, the following controls were specifically cited for core shuffles:

All core locations may only contain the original fuel assembly or the final fuel assembly;

No more than three control rods can be removed from the core at any given time.

Fuel Assembly Loading Errors

In the fifty-four refuelings reported in this survey, two of the eighteen plants responding to the survey placed a fuel assembly in an incorrect core location. In one case, the error was detected when the operators attempted to retrieve a fuel assembly from an empty spent fuel pit location. The error was corrected by reversing the fuel handling reload pattern until the misplaced assembly was found. At that time it was relocated to its correct core location. The other case involved the misplacement of nine fresh fuel assemblies in the core. The errors were detected as part of the core map performed following the fuel load. The assemblies had been placed in the wrong locations in the spent fuel pit upon initial receipt at the plant. However, the assemblies were identical and did not require relocation to their original core locations. In neither case did control rod withdrawal occur before detection of the misplaced assemblies.

In addition to the cases cited above, five other utilities cited loading errors which did not occur in the core. Four of the five cases involved the misplacement of fuel assemblies in the spent fuel pit during core offload. In one case, the error was detected by the operator performing the required second verification of the position of the assembly in the spent fuel pit location. The assembly was subsequently placed in its correct location.

The other case, involving the misplacement of three assemblies, was detected during an inventory of the spent fuel pit. These assemblies were also placed in their correct locations.

The remaining loading error cited in the survey involved taking the assembly (which occurred twice) from the wrong spent fuel pit location. However, as the assembly was being removed from the upender, a visual inspection revealed that the wrong assembly was being moved. As a result, the assemblies were returned to their original spent fuel pit locations.

Finally, one utility cited the misorientation of two assemblies from their required orientation. The errors were detected during the core map and the assemblies were properly oriented.