NUREG/CR-5819
BNL-NUREG-52313

Probability and Consequences of Rapid Boron Dilution in a PWR

A Scoping Study

Prepared by
D. J. Diamond, P. Kohut, H. Nourbakhsh, K. Valtonen, P. Secker

Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grant publications, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG/CR-5819
BNL-NUREG-52313

Probability and Consequences of Rapid Boron Dilution in a PWR

A Scoping Study

Prepared by
D. J. Diamond, P. Kohut, H. Nourbakhsh, K. Valtonen, P. Secker

Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grant publications, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG/CR-5819
BNL-NUREG-52313

Probability and Consequences of Rapid Boron Dilution in a PWR

A Scoping Study

Manuscript Completed: March 1992
Date Published: June 1992

Prepared by
D. J. Diamond, P. Kohut, H. Nourbakhsh, K. Valtonen[1], P. Secker[2]

Brookhaven National Laboratory
Upton, NY 11973

Prepared for
Division of Systems Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A3868

[1]Finnish Centre for Radiation and Nuclear Safety, Helsinki, Finland
[2]University of Arizona, Tucson, AZ

Abstract

This report documents the results of a scoping study of rapid dilution events in pressurized water reactors. It reviews the subject in broad terms and focuses on one event of most interest. This event could occur during a restart if there is a loss-of-offsite power when the reactor is being deborated. If the volume control tank is filled with water at a low boron concentration then a slug of this water could accumulate in the lower plenum. This would be the result of the trip of the reactor coolant pumps leading to relatively low flow conditions and the restart of the charging pumps on emergency power. The concern is that this diluted slug will rapidly enter the core after a reactor coolant pump is restarted and this could cause a power excursion leading to fuel damage. This problem was studied probabilistically for three plants and the important design features that affect the core damage frequency were identified. This analysis was augmented by an analysis of the mixing of the diluted water with the borated water already present in the vessel. The mixing was found to be significant so that neglect of this mechanism in the probabilistic analysis leads to very conservative results. Neutronic calculations for one plant were carried out to understand the effect of nuclear design on the consequences of the event.

Table of Contents

Table of Contents (continued)

List of Figures

List of Figures (continued)

List of Tables

Executive Summary

A rapid boron dilution event in a pressurized water reactor is postulated to occur when two requirements are met. The first requirement is that unborated or diluted water enters the reactor cooling system (RCS) during a period when there is very little circulation and the assumption is made that this water collects in a part of the system. The second requirement is that a reactor coolant pump (RCP) is started so that the slug of relatively diluted water passes rapidly through the core with the potential to cause a power excursion and fuel damage.

Although there are several scenarios that qualify as rapid boron dilution events, the one of most concern in this study occurs during a reactor restart. Analysis of this event without accounting for mixing of the diluted water entering the RCS results in a significant core damage frequency. However, if mixing is taken into account it becomes possible to have core damage only under the most extreme set of core initial conditions.

The reactor restart scenario occurs during the period when the reactor is being deborated according to normal procedures so that criticality can be achieved. A loss of offsite power (LOOP) is the initiating event. When this occurs there is reactor trip (the shutdown banks would be withdrawn during deboration) and trip of the charging pumps and the RCPs. When emergency power is brought on line the RCPs are not able to start but the charging pumps will start. It is assumed that the volume control tank (VCT), which supplies the suction for the charging pumps, is filled with highly diluted water. This water continues to be pumped into the RCS if the operator takes no action to switch the suction to a borated source. Since the RCPs are not running, if the natural circulation flow rate is low, the first requirement for a rapid boron dilution is met, i.e., there is the potential for a slug of diluted water to accumulate in the RCS, in this case most likely in the lower plenum.

The second requirement, that an RCP start, is fulfilled after offsite power is restored. It is assumed that the operators will start the RCP in order to resume the restart procedure. When this occurs it is assumed that the slug passing through the core adds sufficient reactivity to overcome the shutdown margin and cause a power excursion. Furthermore, the concern is that the power excursion is sufficient to cause fuel damage.

A probabilistic analysis had been done for this event for a European PWR. The estimated core damage frequency was found to be high enough so that corrective actions were taken. A system was installed so that the suction of the charging pumps would switch to the highly borated refueling water storage tank (RWST) when there was a trip of the RCPs. This was felt to reduce the estimated core damage frequency to an acceptable level.

In order to see if the core damage frequency might be as high in U.S. plants, a probabilistic assessment of this scenario was done for three plants. The plants chosen, Oconee, Calvert Cliffs, and Surry, represent a sample from the three U.S. reactor vendors, Babcock & Wilcox, Combustion Engineering, and Westinghouse. The estimated core damage frequency based on a scoping analysis was 2.8E-5/yr, 2.0E-5/yr, and 9.7E-6/yr for the three plants, respectively. These numbers are relatively high compared to desireable goals, but they are only the result of a scoping analysis and include many assumptions.

Although there were several conservative assumptions made in the analysis it should also be noted that there were several conditions present at these plants which might not be relevant at other plants and would make the core damage frequency *higher* at those other plants. One of these is the initiating frequency for a LOOP which is lower at U.S. plants than at plants in some other countries.

Another condition is related to plant design and the size of the source of unborated water during the event. In this analysis it was assumed that the volume of diluted water available was the volume within the VCT from the normal level to the low level at which point highly borated water from the RWST would automatically start to fill the VCT. In each of the three plants studied, the pump from the source of primary makeup water was tripped after the LOOP and remained tripped until full power was restored. If the design was such that this pump was connected to the emergency bus then the source of unborated water would be greatly increased and the core damage frequency would be increased. Since there are PWRs in Europe where this is the case it may also be true within the U.S. Therefore, some plants may have a higher vulnerability to this event than those chosen for study.

The two most important conservative assumptions in this analysis are:

- The mixing of the injectant is insignificant
- Fuel damage occurs when the slug passes through the core

The first assumption was found to be conservative by performing an analysis using a mixing model that had been developed to treat the mixing of streams of water at different tempertures. The mixing is significant when the injectant first enters the cold leg, when it enters the downcomer, and then as it moves down the downcomer into the lower plenum. The calculations were done for Calvert Cliffs and Surry. The reactor conditions assumed were that the RCS was initially stagnant and that the temperature of the injectant was either 100F° or 290F° lower than the initial temperature of the RCS. If there was significant loop flow due to natural circulation this would enhance the mixing.

The results of the mixing analysis were that the boron concentration in the lower plenum was not expected to be lower than 1080 ppm or 900 ppm for Surry and Calvert Cliffs, respectively, assuming in both cases that the boron concentration in the RCS at the time of the LOOP was 1500 ppm. This means that the reactivity addition would correspond to a change of only 400-600 ppm rather than the 1500 ppm that was theoretically possible.[a]

The second major conservative assumption in the probabilistic analysis is that sending a slug of diluted water through the core will cause fuel damage. In reality the effect of the slug will depend

---

[a] The extent of mixing means that the volume of diluted water created is much larger than the initial volume available from the VCT. Hence when the RCP is restarted the slug will remain in the core for a longer period of time than would be the case with no mixing. This will not have a strong impact on the initial power burst and the potential for immediate fuel damage, but would be important in understanding fuel behavior over a longer period.

on the reactivity addition caused by the diluted slug, the volume and geometry of the slug, the initial shutdown margin, and the thermal-hydraulic feedback.

A scoping analysis of the neutronics was done to show how these factors affect the consequences of the event and in particular whether catastrophic fuel damage might occur. This was defined as damage that could change fuel geometry and was determined by a local fuel enthalpy criterion of 280 cal/g. No consideration was given to other fuel damage mechanisms. A model of the Calvert Cliffs plant was used although some of the results have general applicability.

It was found that the slug boron concentration would have to be less than 430 ppm (i.e., a dilution of 1070 ppm) for catastrophic fuel damage to occur if the shutdown margin was 4% and the Doppler feedback was relatively strong. If the shutdown margin is smaller or if the core has a smaller Doppler feedback, then a smaller dilution would cause a problem. The Doppler feedback plays a very important role and varies significantly during a cycle and for different cycles so that it can have an important effect on the results.

The shutdown margin is made up of the worth of the shutdown bank which enters the core after the LOOP and the shutdown margin that existed prior to that. The shutdown bank worth typically varies from 2% to 5% depending on the plant. The pre-LOOP shutdown margin depends on when during the startup deboration that the LOOP occurs. If it occurs at the start of this period then the shutdown margin will be larger than if it occurs toward the end when the core boron concentration is closer to the value needed for criticality. The probabilistic analysis assumes that the core damage is equally likely anytime during the normal deboration period and, therefore, neglects this effect. For the Calvert Cliffs case the 4% used for the shutdown margin was assumed to be the effect of the shutdown bank only.

Based on the Calvert Cliffs neutronics calculations of shutdown bank worth and Doppler feedback, and mixing calculations indicating a slug boron concentration of 900 ppm, catastrophic fuel damage would not be expected. However, if the magnitude of the Doppler coefficient was half of that used, and the worth of the shutdown banks was only 2% then the result would be close to the fuel damage criterion. Furthermore, if the temperature of the slug was low, this would add to the severity of the excursion due to the positive effect of coolant temperature feedback.

It is important to note that these results will be a function of plant design. Every plant may be vulnerable to some form of rapid dilution event. Plants that use a diluted VCT to deborate may be vulnerable to the reactor restart scenario examined in detail in this study. For those plants the probability that this event leads to fuel damage will be a function of many design factors. Of particular importance are the volume and boron concentration of the VCT, the pumping rate of the charging flow and its orientation at the cold leg, and the reactivity worth of the shutdown banks and Doppler feedback.

Acknowledgements

The authors wish to thank H. Richings and M. Caruso of the U.S. Nuclear Regulatory Commission for their support of this project. The authors also appreciate the help of A. Aronson for generating neutronics data for the NODE-P2 calculations, C-J. Hsu for discussing the relevant probabilistic analysis that had been done for Surry, and S. Perez for producing graphical output for the mixing analysis. Lastly, the authors are indebted to K. Ryan for her help in preparing this report.

1 Introduction

1.1 Objectives

The general objective of this work is to improve the understanding of rapid boron dilution events in pressurized water reactors (PWRs). This is to help the U.S. Nuclear Regulatory Commission (NRC) to determine if any action should be taken to reduce the expected frequency of such events. This objective is met by examining different scenarios and performing scoping probabilistic and deterministic analysis. The probabilistic work quantifies the frequency of occurrence of one sequence which has been of particular interest to the NRC. This is done for plants of each of the U.S. reactor vendors. The deterministic work consists of neutronics and thermal-hydraulics calculations. The neutronics calculations are of the reactivity effect of different dilution slugs and the resulting power excursion. The thermal-hydraulics calculations are of mixing to understand whether unborated water introduced into the reactor coolant system (RCS) can remain as a slug of sufficient dilution to cause a problem.

1.2 Background

Boron dilution events have always been of concern in PWRs. A slow inadvertent dilution due to malfunction of the chemical and volume control system (CVCS) is a design-basis event which satisfies stringent acceptance criteria. The question of whether additional failures beyond the CVCS malfunction might lead to inadvertent criticality and fuel damage has also been addressed in the past. More recently this type of event and many other possible dilution scenarios have been surveyed in a study for the NRC [1.1]. That study noted that more scenarios were being postulated in different countries and that additional work would have to be done in the future to determine the importance of these events.

It is convenient to separate these beyond-design-basis dilution events into three types according to how they cause the power to rise in the core. For one type, the power excursion is caused by a relatively slow, uncontrolled dilution in which the boron concentration in the core changes slowly but steadily throughout the entire core. This type of event requires a large volume of diluted water. It is relatively easy to analyze as the power increase will be determined by a linear reactivity addition, mitigated by feedback effects, until stopped by operator action or the melting of fuel.

A second type of excursion occurs when pumps are off and diluted water accumulates in the lower plenum of the vessel to the extent that the bottom of the core becomes critical and power increases. This power increase causes an increase in the natural circulation flow rate which draws the diluted water up from the bottom of the vessel into the core. Without consideration of thermal-hydraulic feedback, this is an autocatalytic power excursion which is more rapid than the first type above.

The third type of power excursion is caused by a slug of diluted water rapidly entering the core. Because it is a slug, less diluted water is required than in the first type of dilution in which the diluted water mixes uniformly with the water in the RCS. It is this type of event that is currently of interest to the NRC [1.2] and that is the subject of this study. In order to have such an event it is first necessary to introduce diluted water into the RCS during a period when there is minimal circulation so that the water can collect in one place. The slug of diluted water can then be passed rapidly through the core by the startup of one or more reactor coolant pumps (RCPs) or the blowdown of one or more accumulators. This type of event is also different from the other two in that it has the potential of causing catastrophic fuel damage, i.e., rapid changes in fuel geometry, rather than relatively slow fuel melting.

The rapid dilution event (as well as the second, autocatalytic, type of excursion) would occur when the reactor coolant pumps are running as would be the situation during a shutdown period or immediately following reactor trip. The shutdown period is also the time when the core might be most vulnerable to this type of event because control rods are already inserted and, therefore, reactor trip would not be possible to mitigate the power excursion.

1.3 Scope and Organization of this Report

This study is both an overview of rapid dilution events and a detailed analysis of one particular event. In Section 2 a review is given of scenarios which could lead to a slug of water with a low boron concentration passing through the core. One of these events, the reactor restart scenario, is of particular interest because in one country in Europe remedial action has been taken to help prevent the event. Hence, specific probabilistic and deterministic analysis was carried out for this scenario.

The analysis is considered of a scoping nature because it uses simple models and only considers a limited number of plant conditions and designs. The probabilistic analysis is described in Section 3. The analysis is carried out for the Oconee, Calvert Cliffs, and Surry plants, representing each of the U.S. reactor vendors. One of the important assumptions in the analysis is that there is insufficient mixing when a source of diluted water is introduced into the RCS so that that slug has the potential to remain intact and pass through the core when an RCP is started. This assumption is tested by performing an analysis of mixing that is described in Section 4. The analysis is for one particular plant and uses straightforward empirical models.

Section 5 describes the neutronic analysis that was carried out to understand the potential consequences of a diluted slug passing through the core. The analysis includes static calculations of reactivity for different slug geometries and dilutions and dynamic calculations of the power excursion that would be expected if these slugs passed through the core. Although the modeling is based on the reactor restart scenario, much of the analysis would be applicable to other scenarios which lead to a slug of diluted water passing through the core.

A summary of results and conclusions is given in Section 6 and Section 7 contains references.

2 Rapid Dilution Scenarios

2.1 Reactor Restart Scenario

One of the sequences that the industry has been aware of for a long period of time was recently studied in Europe. The result of a preliminary probabilistic analysis indicated that the frequency of occurrence of the event might be large in those plants and hence, the NRC issued an information notice [2.1].

The scenario occurs at the end of a shutdown period when the plant is being brought back to a critical configuration. The normal deboration is done when the reactor coolant system (RCS) is at hot, pressurized conditions and the shutdown banks are withdrawn.

The initiating event for this scenario is a loss of off-site power (LOOP). When that occurs there is a reactor trip and a trip of the reactor coolant pumps. The charging pumps would also stop but it can be assumed that emergency power is brought on line quickly and the charging pumps are energized from the emergency bus. These pumps will continue to pump from the volume control tank (VCT), emptying the diluted water that is present into the RCS. Assumptions are made that the operator takes no action to switch to a boration mode, and that the VCT contains a relatively large volume of water which is at a boron concentration that is much less than that originally in the RCS. This diluted water may be colder than the water in the RCS. When it gets pumped into the cold-leg it is assumed that there is minimal mixing so that the water can collect as a diluted slug at the bottom of the reactor pressure vessel (RPV). The probability of minimal mixing is enhanced if the event takes place after a long refueling when the decay heat level is low and consequently the amount of natural circulation in the RCS is relatively low.

Under these circumstances, if off-site power is recovered, it is likely that the operator will start an RCP in order to continue the process that was interrupted by the LOOP. This will send the slug of diluted water through the core and it is assumed that the reactivity added will be sufficient to overcome the existing shutdown margin and cause a power excursion leading to fuel damage.

When this event was studied in Europe as part of a probabilistic risk assessment (PRA) for operating and shutdown conditions, a relatively high frequency for a LOOP was used and other assumptions regarding inaction of the operator and the mixing of the water were assumed to hold so that the core damage frequency was found to be high. As a result of this rough estimate corrective action was recommended and a program of analysis and experimentation was initiated. The latter is to examine the effect of mixing which if present would eliminate the creation of the slug. The corrective action was a hardware change which would switch the suction of the charging pumps to the refueling water storage tank (RWST) when there was an RCP trip. Since the boron concentration in the RWST is very high this would eliminate the possibility of this accident. Taking into account the reliability of this new system would significantly reduce the expected frequency of occurrence.

2.2 Other Scenarios Involving Startup of an RCP

There are several other sequences that have been postulated which involve the startup of an RCP after a period in which diluted water has accumulated somewhere in the RCS. By necessity these sequences occur during shutdown. They have been studied in varying degrees of detail for one Westinghouse plant [2.2] and the following summary is based on that study.

A sequence studied by the French [2.3, 2.4] starts with a loss of power and failure of equipment involved in the startup procedure. This could lead to boiling in the core. If the auxiliary feedwater is operable this water is assumed to condense in the steam generator. The condensate which is unborated could accumulate in the cross over leg so that when the situation returns to normal and the RCPs are started, a diluted slug could be sent through the core.

Another source of diluted water in the cross over leg is secondary water. If steam generator tubes are cut either on purpose or inadvertently during steam generator modifications, and no repairs are made before the secondary side is brought back into service, then leakage of unborated water into the primary will occur when the secondary is filled. There were two such events [2.5] reported for the period from June 1969 to January 1981 which were found to cause an overall reduction in boron concentration rather than a localized diluted slug. These dilutions were both detected at an early stage and resulted in less than a 100 ppm change in RCS boron concentration.

A sequence studied in great detail by Swedish workers [2.6, 2.7] is one initiated by a steam generator tube rupture (SGTR). The plant is initially at hot zero power or if at power, it is shut down immediately. It is assumed that the RCPs are tripped due to a loss of power or some other cause. If the operators use a backfill cooldown procedure then unborated water from the secondary will enter the RCS. If this water does not mix but is assumed to collect in a stagnant part of the loop then if an RCP is started there is the possibility that the slug will enter the core and cause a power excursion.

Calculations performed in Sweden showed that the boron concentration in the core could go from 850 ppm to a minimum of 163 ppm in 10 seconds. These same calculations did not show any immediate fuel damage due to the energy deposition. However, the calculations are claimed to be inconclusive and further analysis is needed. As a result of this analysis Westinghouse, in 1990, recommended to the Westinghouse Owners Group that changes be made to the Emergency Response Guidelines regarding the procedures after a SGTR.

Other sources of unborated water during shutdown that could cause a problem if a slug collects and an RCP is started include the RCP seal water flow or a leaking thermal barrier or the water used to clean the refueling cavity after refueling. The cavity is hosed down with unborated water to remove radioactivity. An event involving more than 12 hours of inadvertent dilution from an unattended hose has occurred [2.8] causing a change in RCS boron concentration of 340 ppm.

2.3 Opening of Loop Stop Valves

A situation that is similar to a pump restart is the opening of a loop stop valve when pumps are running. Calculations had been done by Westinghouse [2.9] to determine the consequences of a startup of an inactive unborated loop without consideration of how the loop became diluted. All rods were assumed to be initially out of the core and hence, the worth of the scram reactivity would be considerable. In the worst case considered, where they also assume that the temperature of the water in the isolated loop is relatively cold, they calculated that approximately 3% of the fuel experiences clad rupture and <0.5% melt completely. However, insufficient information was presented to know what was the worth of the control rods and it is not possible to say that the calculations bound all possible consequences.

2.4 Blowdown of a Diluted Accumulator

If an accumulator has become diluted then if there is an inadvertent blowdown of that accumulator there is the potential for a diluted slug to pass through the core. The blowdown is postulated to occur during shutdown when the RCS is at atmospheric pressure and the accumulator is at operating pressure (625 psia). There is a motor operated valve that isolates the accumulator during shutdown. If this valve is not deenergized according to procedures, then there is possibility that it can open allowing the accumulator fluid to enter the cold loop and flow through the downcomer and into the core.

There are several ways in which the accumulator can become diluted. In a study at Brookhaven National Laboratory [2.10] it was determined that the most likely cause was back-leakage during operation at end-of-cycle. With a low boron concentration in the RCS and leakage through the check valves, the accumulator boron concentration could change dramatically if monitoring instrumentation was defective or operators did not respond properly.

A detailed probabilistic analysis of this type of event was carried out for a Westinghouse plant and showed that the expected frequency was insignificant. However, since that study an accumulator dilution has occurred which indicates that the most likely source of diluted water might be demineralized water that has been added for testing. This occurred in a French plant in July 1991. The unborated water that had been used for testing was left in the accumulator and eventually 350 ft^3 of this water flowed under gravity into the RCS. Although the discharge of the accumulator did not occur suddenly and the dilution of the RCS did not have any consequences, it was an important precursor and also indicates that the most likely source of diluted water might be due to maintenance rather than back-leakage.

3 Probabilistic Analysis

3.1 Introduction

In this section an estimate is made for the frequency of a rapid dilution event which could lead to core damage. The analysis is for the reactor startup scenario as described in Section 2.1. It is carried out for the Oconee, Calvert Cliffs, and Surry plants which were designed by Babcock & Wilcox, Combustion Engineering, and Westinghouse, respectively. This enables one to understand not only the important systems and operator actions with regard to this scenario, but also to identify any major differences that might exist between plants designed by each of the U.S. PWR reactor vendors. The specific plants selected for study were chosen because of the availability of information.

For each plant a summary description of the important systems for this type of event is first presented. This consists of a section describing the systems through which the unborated or diluted water might enter the reactor coolant system (RCS), and a section describing the relevant electrical systems. After this, the probabilistic analysis for each plant is presented in subsections on timing, modeling, and quantification. The quantification is done separately for refueling and nonrefueling outages. A summary section at the end presents the core damage frequencies for each plant and a discussion of important assumptions used in the analysis.

3.2 System Description - Oconee Station

3.2.1 Makeup and High Pressure Injection System

The makeup and dilution of the RCS is accomplished at Oconee using the High Pressure Injection (HPI) system. The relevant portions of the HPI and related systems are shown in Figures 3.1, 3.2 and 3.3. In normal operation a small amount of coolant is bled off from the RCS through the letdown and is directed to the purification demineralizers. The letdown (upper left on Figure 3.1) is cooled by the letdown coolers and can be isolated using several valves (HP-3[a], HP-4, HP-5 and HP-6). The output from the demineralizers is directed through a three way valve (HP-14, upper right on Figure 3.1) into the letdown storage tank (LDST) or into the deborating system where it is normally collected and stored in one of the RC bleed holdup tanks (upper left on Figure 3.3). Another RC bleed holdup tank holds demineralized water for dilution purposes.

Reactor coolant may directly enter the letdown storage tank through the three way control valve (HP-14) or from the RC bleed holdup tank by operating the RC bleed transfer pump 1A. The other RC bleed transfer pump, 1B, is used to supply fresh demineralized water during the deboration operation, transferring deborated water to the letdown storage tank.

The letdown storage tank serves as a surge tank and normal suction source for the HPI pumps (lower center on Figure 3.2). Another source of suction for the HPI pumps is the borated water storage tank (BWST).

---

[a] Figures show valves without the "HP" designation.

The normal makeup flow is supplied by one of the HPI pumps and controlled automatically by a control valve (HP-120) to maintain the level in the pressurizer. During normal operation the three way valve (HP-14) is in "normal" position directing all letdown flow into the letdown storage tank. The valve may be placed in the "bleed" position to direct the letdown flow to the deborating demineralizer or RC bleed holdup tank. HP-14 is automatically placed in the "normal" position when there is low level in the letdown storage tank.

During startup of the reactor, the operator has to deborate the RCS from the shutdown boration level to achieve criticality. The dilution requires adding a predetermined amount of demineralized water to the RCS through the letdown storage tank. The deboration procedure, given in Table 3.1* directs the operator to determine the amount of demineralized water or batch size that is needed. The operator then sets the totalizer/batch controller (flow meter and integrator) to the desired setting and opens the makeup control valve HP-15 (middle right on Figure 3.1).

The control valve remains open until the totalizer indicates the end of the batch size and an automatic signal closes the valve. Even though the totalizer/batch controller is started and control valve HP-15 is opened, makeup to the letdown storage tank is prevented until the makeup isolation valve HP-16 is opened. Once the transfer path is established, the RC bleed transfer pump is started to add the desired quantity of demineralized water.

The rate of addition of deborated water may be as much as or less than the letdown flow. It could range from 45 to 90 gpm and normally is about 70 gpm. The volume of the batch size is generally larger than the volume of the letdown storage tank which requires the operator to position the HP-14 three way valve to the "bleed" position. Consequently, the boron concentration in the letdown storage tank may decrease as the demineralized water is being added by the RC bleed transfer pump. If the transfer rate is slower than the makeup rate through the HPI pumps then the three way valve has to be in an intermediate position to maintain the letdown storage tank level.

As a result of this process the following system conditions may be obtained:

1. The letdown storage tank volume is diluted to low boron concentrations by adding demineralized water

2. Depending on the demineralized water transfer rate or dilution rate, the boron concentration may be as low as 0-200 ppm (transfer equals makeup rate) or may range to a maximum of about 50% of the RCS boron concentration, i.e., 1000-1200 ppm (transferred demineralized water is mixed with letdown).

3. The water level in the letdown storage tank is maintained at an intermediate position (between high/low) during the deboration operation.

---

* Oconee Nuclear Station Operating Procedure OP/3/A/1103/04, "Soluble Poison Concentration Control," Duke Power Co., approved Feb. 28, 1989.

Table 3.1
Procedure Used to Dilute LDST

NORMAL MAKE-UP TO THE LDST

1.0 Initial Conditions

1.1 Determine the amounts of borated and unborated water to be used in the batch makeup.

2.0 Procedure

2.1 Set the desired batch size.

2.2 Ensure 3HP-15 (Makeup Control) is reset; control knob to OPEN and Toggle Switch to START.

2.3 Open 3HP-16 (Makeup Isolation).

2.4 Start the desired Bleed Transfer pump.

2.5 Open its respective discharge valve:
3CS-46 (Bleed Transfer Pump 3A Discharge).
OR
3CS-56 (Bleed Transfer Pump 3B Discharge).

2.6 If BLEED is required, ensure 3CS-26 (Letdown to BHUTs) and 3CS-41 (Bleed Tank 3A Inlet) are open.

2.7 If more volume is required than available in the LDST, position 3HP-14 (LDST Bypass) to BLEED.

2.8 When Batch size is reached:
2.8.1 Close 3HP-16 (Makeup Isolation).

2.8.2 Stop the selected Bleed Transfer pump and close its respective discharge valve:
3CS-46 (Bleed Transfer Pump 3A Discharge).
OR
3CS-56 (Bleed Transfer Pump 3B Discharge).

2.8.3 Ensure 3HP-14 (LDST Bypass) is in the NORMAL position.

2.8.4 Clear and reset 3HP-15 (Makeup Control).

3.2.2 Electrical System

The boron dilution accident of interest is initiated by a loss of off-site power during the deboration process. For this reason the specific features of the electrical distribution and supply system play an important role and must be discussed in some detail.

The line diagram of the Oconee electrical system is presented in Figure 3.4. The external grid connects to the 230 and 525 kV switchyards which are interconnected by a 230/525 kV autotransformer. One of the two buses (yellow bus) at the 230 kV switchyard plays a fundamental role in supplying power to the plant auxiliaries should the switchyard become isolated from the external grid.

The 230 kV switchyard and the yellow bus are also connected to a two-unit hydro station (Keowee Hydro) through an overhead line that provides emergency backup power. The hydro units perform the role played by diesel-generators at other plants. If there is a switchyard isolation (loss of grid), the 230 kV yellow bus will be reconnected to the hydro stations and be available to supply power to all station startup transformers.

The startup transformers CT-1, 2 and 3 (see Figure 3.4) can supply most of the unit auxiliaries, including the reactor coolant pumps (RCPs) which are connected to the 6.9 kV buses. If the switchyard is unavailable, then emergency power from the hydro station is provided through an underground connection to transformer CT-4, which supplies power to essential auxiliary equipment connected to the 4 kV buses. In this case, the RCPs cannot be utilized and may be restarted only after the switchyard or grid is recovered.

The important features of the electrical system may be summarized as follows:

1. Loss of grid events (not weather related) will not affect the availability of the 230 kV switchyard and the RCPs may be restarted at any time after the hydro units provide backup power.

2. Weather related loss-of-off-site-power events, or switchyard trouble, deenergizes the 6.9 kV buses and the RCPs may be restarted only after the recovery of the off-site grid or switchyard.

3.3 Probabilistic Analysis - Oconee Station

3.3.1 Accident Sequence Timing

The outcome of boron dilution scenario is strongly dependent on the timing of events or the time evolution of the expected responses. In order to properly model and estimate the risk due to the accident scenario, the time behavior of the various events must be established with reasonable certainty.

Probabilistic Analysis

The deboration process itself is rather time consuming due to the small makeup and letdown flow relative to the total RCS volume. For the purpose of this analysis, it is estimated that the RCS contains about 75,000 gallon of water and the average makeup/letdown flow is about 70 gpm. The initial boron concentration at the start of the deboration is on the order of 2000 ppm and the final boron level is assumed to be 1500 ppm. The change in the RCS boron concentration (C) can be calculated using:

$$\frac{dC}{dt} * V_{RCS} = W_M * C_M - W_L * C_L \tag{3.1}$$

where V_{RCS} is the volume of the RCS, C_M and C_L are the boron concentrations of the makeup and letdown, respectively, and W_M and W_L the makeup and letdown flow rates, respectively. The initial boron concentration will exponentially be diluted to the final boron concentration as a function of time. Solving the above equation using $W_M = W_L$, $C_M = 0$, and $C = C_L$ gives a time span of 5 hours for the dilution process from 2000 to 1500 ppm.

The average length of deboration was estimated by the Oconee station operators to be 8-12 hours. This is consistent with the above calculation since the rate of deboration is dependent on the actual actions occurring during the startup process, and these are expected to necessitate a deboration slower than theoretically possible. Hence, the analysis will set the deboration time span as 8 hours.

The maximum amount of diluted primary grade water that can be injected into the RCS when the RCPs are not running (after a loss of off-site power or LOOP) is the available water in the letdown storage tank, which is assumed to be diluted to a very low boron concentration. The total volume between the HI/LO levels is approximately 420 ft^3. The water level is expected to be around midlevel, maintained by regulating the dilution flow and the position of the three-way valve, HP-14. Therefore, after a LOOP event the amount of diluted water volume is assumed to be approximately 250 ft^3 or 1870 gallons. Once the low level is reached, the suction source for the HPI pumps shifts to the BWST and highly borated water is pumped into the RCS. Assuming that the makeup flow is about 70 gpm, the time interval before the switchover to the BWST is about 20-25 minutes once the HPI pump is restarted after the LOOP event.

The probability for conditional core damage P(CCD) is defined in order to determine the time-dependent probability that there is core damage once the charging pumps start to pump diluted water into the RCS while there is no longer forced circulation. Without having the benefit of the mixing and neutronic calculations discussed in Sections 4 and 5, respectively, and in order to complete a scoping analysis, a simplistic approach is taken. For the situation after refueling, it is assumed that P(CCD) varies between zero and one depending on the amount of diluted water that enters the system. The value of zero is expected at the beginning of this time period when no diluted water has entered under the relatively stagnant flow conditions. The value of one is associated with the assumption that if the full diluted volume of the letdown storage tank (between HI/LO levels) is injected into the cold leg, a sufficiently diluted region will accumulate in the lower plenum so that fuel damage with the restart of an RCP is certain. Hence, the probability P(CCD) is assumed to increase linearly with time from zero to one in the time period of 25 minutes during which the diluted water in the letdown storage tank is pumped into the RCS.

After the suction source switches to the BWST the potential for core damage decreases since borated water is injected and presumably mixes with the diluted water. It is assumed that after about an additional 15 minutes P(CCD) decreases (linearly) to zero and there is no longer the possibility of a rapid dilution event occurring if the RCPs are restarted.

The time dependence of P(CCD) is shown on Figure 3.5. The bottom curve on the figure is for startups other than after a refueling. After a refueling (especially if it coincides with a long shutdown), the decay heat is relatively low because of the replacement of so many fuel assemblies (typically one-third of the core). However, after a shutdown without refueling which is most likely to occur in the middle of the cycle, the decay heat is expected to be much larger. It is expected to be sufficient so that the natural circulation flow rate is considerably higher than after a refueling. If the natural circulation flow rate is sufficiently large then after injection of diluted water there may be sufficient mixing to reduce the probability that there will be core damage with the restart of an RCP. This is taken into account by decreasing P(CCD) by a factor of 0.5 as is shown in Figure 3.5.

It is important to recognize in the curves shown in Figure 3.5 that t=0 corresponds to the time when the charging flow is reestablished through any of the power sources available. It does not correspond to the beginning of the LOOP. If the hydro units are available, then t=0 is the same as the initial time of the accident, however, for scenarios when the hydro units fail to provide power initially, t=0 corresponds to the recovery of either the hydro units or off-site power.

The restart of the RCPs after the LOOP is also modelled as a function of time. Once a power source is available either from off-site or from the hydro units, the operators are expected to start the RCPs. The preferred method of operating at this stage of the startup is to keep forced circulation in the RCS. Once the power is available, certain procedures have to be followed before the RCPs can actually be started. According to plant operational personnel it is expected that after about 30 minutes the RCPs would be running.

The model, therefore, assumes that the cumulative probability of restarting the RCPs increases from zero to one in the thirty minute time period after recovery of power. This is shown in Figure 3.6. Again the time t=0 does not correspond to the occurrence of the LOOP but rather, in this case, to the availability of high capacity electrical power, i.e., the off-site grid or the hydro unit through the 230 kV switchyard.

3.3.2 Accident Sequence Modelling

The event tree shown in Figure 3.7 was developed to evaluate the different accident sequences; in particular those leading to core damage (CD) due to rapid dilution as marked on the diagram. The first top event (ILOOP) is the accident initiator, i.e., the loss of off-site power event during the start up period after the plant was placed in a shutdown condition. The shutdown itself may be divided into two different categories: a) shutdown when refueling is done and b) shutdown without refueling. As explained in Section 3.3.1, this is important because of the relationship between decay heat levels and the probability of the diluted water from the letdown storage tank mixing with the higher concentration water in the RCS

The top event SWYRD questions the type of LOOP eve One type (represented by the top branch) is grid related without affecting the availability of the 230 kV switchyard. This is important since the hydro station, through the switchyard, is capable of providing electrical power backup to all emergency sources including the RCPs. However, if the switchyard is affected, or the overhead connection to the hydro station is unavailable (represented by the bottom branch), then the RCPs may not be started, but the HPI pumps (charging flow) may have electrical power through the underground cables from the hydro station.

The top event HYDRO questions the availability of the hydro station given a demand. If the hydro station fails to start (bottom branch), it may be recovered and this is modelled by top event NR-HYDRO or non-recovery of the hydro station. For this top event the success path (top branch) represents the recovery of the hydro station.

The HPI pumps are powered from the emergency buses and their availability (and the expected charging flow) is questioned op event CHG. The top branch signifies that they are available.

The top event NR-SWYRD questions the recovery either of the switchyard or the off-site grid, either of which would reestablish electric power to the RCPs. For this top event the success path represents the recovery of the power supply.

The top event CCD questions whether there is core damage given the amount of diluted water that has entered the RCS. The bottom branch represents the possibility that this occurs.

The last top event questions the status of the RCPs and whether or not the operator started one of them. The top branch represents the successful restart of the RCPs.

The most important top events in the accident event tree are time dependent and a conventional static approach is inappropriate to model the complete sequences. The time dependence may be included in the event tree by assuming that each top event is a time functional and the end-states are also dependent on time. This may be considered a process, where the event tree being asked and evaluated at each time step $[t,t+\Delta t]$ and the final values are summed or integrated over the respective time period. The actual numerical evaluation of these integrals will be discussed in Section 3.3.3.

There are five sequences which involve potential core damage throug a rapid boron dilution scenario. These are shown in Figure 3.7. The other sequences are either safe conditions or other types of accident sequences, such as station-black-out, which are not the subject of this analysis. The following is a short summary and description of each CD sequence:

Sequence 1 Given the LOOP event the switchyard remains operational. The hydro station starts up and provides a backup source of power for the unit, including the RCPs. The charging flow is automatically reestablished by the

HPI pumps. The operator decides to start up the RCPs and depending on the elapsed time, core damage may occur.

Sequence 2 — Again, after a LOOP event the switchyard remains available, but the hydro station fails to start. When the hydro station recovers, the charging flow is immediately started and after a while the operator may start the RCPs leading to a reactivity excursion if a sufficient amount of diluted water is available in the RCS.

Sequence 3 — The switchyard is also affected by the LOOP event, however, the hydro station is able to provide emergency power through the underground connection. The charging flow is started, but the RCPs may be started by the operator only after the switchyard recovers.

Sequence 4 — Both the switchyard and the hydro station are initially unavailable. The charging flow is started after the hydro station recovers and when the switchyard is able to recover, the RCPs may also be started by the operator.

Sequence 5 — This is the same as Sequence 4 except that the switchyard recovers earlier than the hydro station and both the charging flow and the RCPs may be powered from the grid.

3.3.3 Accident Sequence Quantification

3.3.3.1 Refueling Outage

The frequency of the initiating event, ILOOP, is based on plant-specific data available in a recent update of the plant probabilistic risk assessment (PRA) [3.1]. The total rate of loss of off-site power from all causes is 9.0E-2/yr, which consists of two parts. Seventy percent of these events are such that the switchyard is unavailable or the overhead connection to the hydro station is affected. The remaining 30% of the cases are simple grid losses which do not affect the switchyard or the hydro station.

The refueling outage frequency is about 0.6/yr and the average duration of the startup dilution is 8 hours. It is assumed that the core damage frequency will be independent of when during the deboration the LOOP occurs. This is a conservative assumption as during the early phase of the deboration the initial shutdown margin will be large and the probability that the diluted water can cause a power excursion will be reduced. ILOOP is the product of the frequency of a LOOP (per hour), the duration of the deboration (hours), and the frequency of refueling, and hence, ILOOP= 4.93E-5/yr.

The probability of the top event SWYRD is simply the fraction of LOOP events which affect the switchyard and this was established in the plant specific PRA as 70%. Hence, P(SWYRD)=0.7. The probabilities for HYDRO and CHG are also based on the plant PRA and are

Probabilistic Analysis

P(HYDRO)=9.3E-3 and P(CHG)=8.4E-4. Both of these values were derived by exan ining the specific fault-trees representing the various failure modes.

The time dependence is modelled in the event tree by calculating the contribution of each sequence in each time period [t,t+Δ] and then summing to the total time for which there is core damage potential. The probability at a given time P(CCD) is obtained using the distributions shown in Figure 3.5. The probability per unit time for P(RCPRST) is from Figure 3.6. Since Figure 3.6 is a cumulative probability distribution it is the derivative of that curve with respect to time, i.e., (1/30) per minute, which is equal to P(RCPRST).

The evaluation of the time integral of the CD sequences involves a convolution integral (e.g., $\int dt\{P_1(t) \int dt'[P_2(t')...]\}$) with the appropriate probabilities. In the following, for simplicity, the short form $\int dt\{...\}$ will represent this type of integration.

For Sequence 1 the appropriate integral is:

CDF(S1) = $\int$ dt{ILOOP*[1-P(SWYRD)]*[1-P(HYDRO)]*[1-P(CHG)]*P(CCD)*P(RCPRST)}.

In this sequence the integral is over the period 0 to 30 minutes as the RCPs are expected to be running by the end of this period. The integral of the time dependent portion, $\int$ dt[P(CCD)*P(RCPRST)], is evaluated from 0 to 25 minutes using the ascending part of the distribution shown in Figure 3.5 and from 25 to 30 minutes using the descending part of the distribution[a]. The numerical value is 0.56. Hence, the result for Sequence 1 is that CDF(S1)=8.2E-6/yr.

Sequence 2 is similar to the previous one except that the hydro station fails to start, but recovers to supply emergency power. Hence,

CDF(S2) = $\int$ dt{ILOOP*[1-P(SWYRD)]*P(HYDRO)*[1-P(NR-HYDRO)]*[1-P(CHG)]
*P(CCD)*P(RCPRST)}.

In this sequence, there are two time periods to consider. One is the time related to the recovery of the hydro unit and the other is the time associated with the start of the charging flow, RCPs and core damage potential. The latter period does not start until there is recovery and hence, the integral $\int$ dt[P(CCD)*P(RCPRST)] can be evaluated independently of the question of recovery of the hydro station. Using the results for Sequence 1 the value of this integral is 0.56. The integral $\int$ dt[1-P(NR-HYDRO)] can be assumed to be unity as it is expected that over a long period of time there would be recovery. Hence, the result for Sequence 2 is that CDF(S2)=7.0E-7/yr.

Sequence 3 represents the failure of the high capacity electrical power supply either through the loss of the main grid and the failure of the 230 kV switchyard, or the loss of the grid and the overhead supply line from the hydro station. In either case, the power supply to the RCPs are lost and there

[a] The integral is not carried out to 40 minutes because P(RCPRST) is zero after 30 minutes.

is no forced circulation. However, the underground connection to the hydro station remains intact and the emergency power supply is available for all essential equipment including the HPI pumps providing the charging flow to the RCS. The core damage frequency in this case is

CDF(S3) = ∫ dt{ILOOP*P(SWYRD)*[1-P(HYDRO)]*[1-P(CHG)]*[1-P(NR-SWYRD)]
*P(CCD)*P(RCPRST)}.

The period of vulnerability is the first 40 minutes, since after that time sufficient borated water has been taken from the BWST to eliminate the possibility of a core damage event. During this 40-minute period the switchyard or the off-site power connection may recover, and consequently the RCPs may be restarted.

The recovery rate of off-site power, or the switchyard, will be assumed to be the same and constant through the entire period. Based on operating data for the industry [3.2], at the end of 60 minutes the total recovery is about 15%. Thus the recovery rate is [1-P(NR-SWYRD)]=(.15/60) per minute.

The time dependent portion of CDF(S3) may be written as

$$\int_0^{40} dt\{[1\text{-}P(NR\text{-}SWYRD)]^* \int_t^{t+30} dt' P(CCD)^*P(RCPRST)\}$$

This integral is evaluated using the distribution for P(CCD) shown in Figure 3.5 and taking into account that the limit on the second integral (t+30) cannot go beyond 40 minutes. The result for this expression is 0.0415 and hence, CDF(S3)=1.42E-6/yr.

In Sequence 4, both the switc ird and the hydro fail and recover with a constant recovery rate. It is assumed that in this sequence the hydro units will recover before the switchyard or the grid. This implies that the charging flow is established first and then the RCPs may be restarted after the switchyard or the grid recovers.

The core damage frequency in this case is:

CDF(S4) = ∫ dt{ILOOP*P(SWYRD)*P(HYDRO)*[1-P(NR-HYDR)]*[1-P(CHG])*[1-P(NR-SWYRD)]
*P(CCD)*P(RCPRST)}.

Over a long period (e.g., 24 hours) the hydro unit is expected to recover. Hence, the non-recovery probability is neglected to simplify the calculation. Given the recovery of the hydro unit the sequence progresses as in Sequence 3. Once the charging flow is reestablished by the startup of the HPI pumps, there is a period of vulnerability of 40 minutes, if the RCPs are started during this time. Hence, the time dependence of CDF(S4),

∫ dt{[1-P(NR-HYDR)]*[1-P(NR-SWYRD)]*P(CCD)*P(RCPRST)},

has the same numerical value as the integral evaluated for Sequence 3, i.e., 0 This results in CDF(S4)=1.33E-8/yr.

Probabilistic Analysis

In Sequence 5 the switchyard, or the off-site grid, recovers earlier than the hydro station. The charging flow and the RCPs may be restarted as soon as the off-site power becomes available. The sequence is essentially identical with Sequence 2 except that the off-site power used in Sequence 2 is replaced by the hydro units through the overhead lines.

The core damage frequency in this case is:

CDF(S5) = ∫ dt{ILOOP*P(SWYRD)*P(HYDRO)*[1-P(CHG)]* [1-P(NR-SWYRD)] *P(CCD)*P(RCPRST)}.

The time dependent portion appearing under the integral is effectively the same as in Sequence 2 and was evaluated as 0.56. Hence, CDF(S5)=1.80E-7yr.

The CDF for this type of sequence after a refueling is the sum of the CDF for each of the above five sequences and is equal to 1.05E-5/yr.

3.3.3.2 Non-Refueling Outage

If the accident is assumed to occur after a drained or non-drained outage that does not involve refueling the event tree shown in Figure 3.7 is still applicable. The primary difference between the refueling and non-refueling outages is that the latter occurs with a relatively higher frequency since refueling is only done at about 18 months intervals. The operating history of the U.S. PWR population indicates that the average frequency of non-refueling outages is about 2/yr, which includes both drained and non-drained outages.

The plant specific LOOP frequency was found to be 9.02E-2/yr. The corresponding initiating frequency for the boron dilution event during a start up after a non-refueling outage, again assuming a dilution interval of 8 hours, is 1.64E-4/yr.

Another difference is the larger amount of decay heat after a non-refueling outage. This could considerably enhance the natural circulation rate in the RCS thereby increasing the probability that a slug of diluted water will mix with the borated water before the RCPs are turned on again. This has been taken into account (as explained in Section 3.3.1 and shown in Figure 3.5) by reducing the conditional probability for core damage following the restart of an RCP by a factor of 0.5.

The other variables are assumed to be the same as used for the case after refueling. The five sequences leading to core damage were requantified using the relevant data and resulted in the following results: CDF(S1')=1.37E-5/yr, CDF(S2')=1.16E-6/yr, CDF(S3')=2.36E-6/yr, CDF(S4')=2.21E-8/yr, and CDF(S5')=2.99E-7/yr. All the sequence frequencies increased by about 70% relative to the refueling case. The initiating frequency is about a factor of 3 higher, however, P(CCD) was lowered by a factor of 0.5.

The CDF for this type of dilution sequence after a non-refueling outage is the sum of the CDF for each of the above five sequences which is equal to 1.75E-5/yr.

3.4 System Description - Calvert Cliffs Station

3.4.1 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is designed to perform various functions, the most important being: a) control of RCS volume (letdown and makeup), b) removal of corrosion and fission products, and c) boric acid concentration control. The CVCS consists of two major subsystem, the letdown and charging system and the makeup system. Simplified block diagrams of these systems are presented in Figures 3.8 and 3.9.

The normal reactor coolant letdown from one cold leg first passes through two letdown stop valves (upper left in Figure 3.8), then through the regenerative heat exchanger where its temperature is reduced by transferring heat to the makeup flow entering the RCS. Both letdown stop valves fail closed if instrument air pressure is lost (as in a LC[illegible]nt). The letdown flow then passes through the excess flow check valve and flows through [illegible] valve. The control valve is operated by a signal from the pressurizer level [illegible]tain constant level in the pressurizer.

The temperature of the letdown is further reduced in the letdown heat exchanger (cooled by component cooling water) for the proper operation of the ion exchangers. A temperature controller on the outlet of the heat exchanger senses the letdown flow temperature and if it reaches a high level it shifts the three-way ion exchanger bypass valve to the bypass position preventing the hot liquid from entering the ion exchangers. A pressure control valve is also provided on the outlet of the letdown heat exchanger to prevent the fluid from flashing.

The letdown flow then passes through the purification filters, the ion exchangers and the letdown strainer before entering the Volume Control Tank (VCT). There is a three-way inlet valve (CVC-500 in Figure 3.8) that can be operated manually or automatically. In automatic mode the position of the inlet valve is controlled by the level in the VCT and for high level it directs the excess flow to the liquid waste processing system. Normally, the valve is aligned to direct letdown flow into the VCT.

The VCT is used to accumulate letdown water and RCP leak off, to receive makeup water from the makeup system, and to provide positive suction head for the charging pumps. The level in the VCT is controlled by a level controller, which at high level (110") shifts the inlet control valve position to bypass, at 90" starts automatic makeup from the makeup system and at 87.5" alarms as LO level.

If the level in the VCT drops to 3", the suction of the charging pumps is aligned to the Refueling Water Storage Tank (RWST) by closing the outlet valve (CVC-501) and opening the valve connecting to the RWST (CVC-504). The VCT supplies water to the charging pumps, which provide the makeup flow to the RCS. Three charging pumps are provided and normally, one pump is selected for operation. Each pump is capable of supplying 44 gpm makeup flow, which is returned to the RCS through the regenerative heat exchanger.

Probabilistic Analysis

The makeup system (see Figure 3.9) provides a predetermined amount of demineralized water and/or boric acid to the RCS. The system can be put in automatic, borate, dilute, or manual operating mode. For the purpose of this analysis only the dilute mode is discursed.

The dilute mode is used to decrease the boric acid concentration of the RCS. In this mode, the boric acid supply is not used and the makeup consists of demineralized water. The demineralized water is pumped by the reactor coolant makeup pumps (RCMP) through a flow element, a flow control valve and the makeup stop valve before it enters the VCT.

The actual dilution process is accomplished in the following steps:[a]

1. Operator determines that there is space available in the waste processing system for the diverted letdown. Charging and letdown is aligned for normal operation.

2. Total amount of demineralized water to be added is calculated by determining the difference between the desired and existing boron concentration (change in ppm) and relating it to the water volume to be added.

3. Makeup flow controller is set to the desired flow rate consistent with the number of operating charging pumps. The charging or makeup rate (normally 44 gpm - one pump) may be increased, if so desired. The makeup flow controller is shifted to auto position to start the makeup process.

4. Letdown is diverted to the waste processing system, if high level is reached in the VCT.

The main features of the dilution process relevant for the dilution reactivity accident are the following:

1. The VCT is diluted to low boron concentrations by adding demineralized water. The letdown flow is diverted allowing the VCT volume to be replaced by demineralized water.

2. The rate of makeup is matched to the charging rate and consequently the water level is maintained at normal level (95-105") in the VCT.

3. The VCT low level alarm (87.5") is substantially higher than the 3" level where the makeup source shifts to the RWST.

3.4.2 Electrical System

The main features of the electrical system at the Calvert Cliffs power plant are summarized below to the extent which is relevant to the boron dilution accident. The reactivity accident is postulated

---

[a] Calvert Cliffs Nuclear Power Plant Operating Procedure OI-2B, Revision 10, "CVCS Boration, Dilution & Makeup Operations," Baltimore Gas and Electric Co., approved Sept. 24, 1985.

to occur during the RCS dilution when there is a LOOP event followed by the startup of an RCP. The availability of the electrical power supply to the charging pumps and the RCPs is the most important consideration with respect to the electrical system.

The simplified block diagram of the Calvert Cliffs electrical power system is shown in Figure 3.10. The major sources of electrical power are provided by off-site and on-site sources. The normal supply is through the 500 kV main buses, which are connected to the electrical grid by two 500 kV transmission lines. In addition, the two generators of the two units (Unit 1 & 2) are also connected to the main buses.

The 500 kV buses are connected to the 13 kV buses (see Figure 3.10) through the station service transformers which can also be directly supplied from off-site power by connecting them to a 69 kV transmission line. The 13 kV electrical buses are directly connected to the RCP motors and also energize the safety related 4 kV buses. The RCP electrical supply is, therefore, separated from all safety related loads (4 kV buses) and upon loss of the 500 kV/69 kV transmission line connection, the RCPs are without any major source of electrical power.

The 4 kV emergency electrical buses have a second source of power provided by the emergency diesel generators. Given a LOOP event, the breakers connecting the diesel generators to the emergency 4 kV buses close. The emergency diesel generators start and begin accepting loads in a predefined automatic sequence as determined by the load sequencer.

The charging pumps are connected to their electrical supply (which is powered by the diesel generators) 10 seconds after the generator breaker closes (Step 2 of the sequencer) and consequently, the charging pumps will continue to supply the makeup flow. In step 6 or about 30 seconds later, the instrument air compressors are also connected back to the safety buses. The letdown line is isolated upon loss of instrument air pressure, but this is unlikely given the relatively quick (in 30 seconds) restart of the air compressors.

The main features of the electrical system with respect to the boron dilution accident are the following:

1. The RCPs are powered directly from the 13 kV buses which are lost during a LOOP event (loss of electrical grid, two 500 kV and a 69 kV transmission line). There is no additional source of backup power source to the 13 kV buses. The RCPs may be restarted only if off-site power recovers.

2. The charging pumps and the instrument air compressors are sequentially loaded to the diesel generators and after a LOOP event this equipment is restarted in about 30 seconds.

3.5 Probabilistic Analysis - Calvert Cliffs Station

3.5.1 Accident Sequence Timing

In order to assess and adequately model the outcome of a potential dilution accident, the timing associated with the various events and operator actions as well as the corresponding probability values must be estimated and included in the modeling as was done for Oconee in Section 3.3.1. Two of the important considerations are the amount of diluted water available for injection into the RCS and the time period of the dilution process.

The total time for the dilution may be estimated by using Equation 3-1. The normal letdown and makeup rate $W_L = W_M$ is about 40 gpm with one charging pump in operation. Assuming that two pumps are put in operation for a faster dilution rate (84 gpm), the length of time to dilute the RCS from 2000 to 1500 ppm would be about 4.5-5 hours. Since the actual time varies greatly depending on the specific circumstances, it will be assumed (as was done for Oconee) that the dilution time is 8 hours.

The VCT level is maintained between 90-110" (Normal) and the switchover to the RWST occurs at 3" (LO/LO) and the water volume corresponding to 100" (Normal-LO/LO) is about 2900 gallon. Depending on the rate of charging, this volume may be injected into the RCS in about 30-50 minutes.

For Calvert Cliffs the sequence would most likely proceed as follows: As dilution proceeds at an average rate of 84 gpm, a LOOP event occurs. The RCPs coast down and the diesel generators start up establishing charging flow. The operator is likely to reduce the charging rate and tries to recover off-site power. The LOOP procedure[a] directs the operator to borate the RCS if a cooldown is expected. The boration may be accomplished either by using the boric acid addition system or simply supplying makeup water from the RWST.

In order to maintain the possibility of quick recovery and continuation of the start up procedure, rather than borate it is preferable to continue the makeup at a slower rate for a short period of time. Therefore, it is expected that the makeup from the VCT continues and the letdown is diverted to the VCT. The dilution from the makeup system would automatically stop, since the reactor coolant makeup pumps are powered from non-safety buses and are not connected to the diesel generators.

The amount of diluted water contained in the VCT, about 2900 gallons, is expected to be injected into the RCS at a rate which may last 50-70 minutes. It is assumed, based on the results discussed in Section 5, that after about 2000 gallons of diluted water are injected into the RCS, the passage of a diluted slug through the reactor core would lead to core damage. No credit is taken for mixing that might occur as discussed in Section 4.

[a] Calvert Cliffs Nuclear Power Plant Emergency Operating Procedure EOP-2, Revision 1, "Loss of Off-Site Power," Baltimore Gas and Electric Co., approved Feb. 10, 1988.

The conditional core damage probability, P(CCD), is assumed to vary linearly from zero to one in a time period of 40 minutes after injection starts, if the event occurs after a refueling. Since the amount of diluted water is about 2900 gallon, the value would remain at this level (P(CCD)=1) for about 20 more minutes. After the start of the boration or switchover to the RWST the probability of a dilution accident and consequently the value of P(CCD) would decline and reach zero at 80 minutes. Figure 3.11 presents the function P(CCD) as a function of time[a]. As was done for Oconee, a second distribution with half the probability is also shown on the figure to represent the situation when the startup is not after a refueling. This takes into account the higher decay heat and the greater likelihood that the diluted water will mix before the RCPs are restarted.

The cumulative probability distribution for the startup of the RCPs is assumed to be the same as for the Oconee calculation (Section 3.3). After recovery from a LOOP event, the operator will try to restart the RCPs in a 0-30 minute time frame. This is shown in Figure 3.6.

3.5.2 Accident Sequence Modelling

Figure 3.12 shows the boron dilution event tree developed for the Calvert Cliffs Station. The first top event, ILOOP, is the loss of off-site power initiator and represents the loss of the electrical grid and/or the two 500 kV and the 69 kV transmission lines.

The next top event, DSL, questions the availability of the emergency diesel generators which would provide backup power for the safety systems, but not for the RCPs. The diesel generators may fail to start, but could recover after a certain period of time and this is modelled in the top event NR-DSL or non-recovery probability of the diesel generators. Note that the top branch (i.e., success) under this event represents recovery of the diesel generators.

The charging pump availability is examined at the top event CHG. The recovery of the off-site power is an important event and P(NR-LOOP) expresses the probability of non-recovery in a given time interval and is the lower branch (or failure path) on the tree.

The last two top events are related to the condition of the diluted slug and its potential effect on the reactor core. CCD is conditional core damage given that the diluted water has entered the RCS. The RCPRST top event reflects the probability of restarting the RCPs after the LOOP event recovered.

There are three sequences marked in Figure 3.11 involving core damage potential. The other sequences are other unrelated scenarios that are not discussed here, since they do not involve dilution accidents. The three sequences are summarized as follows:

Sequence 1 — After a LOOP event the diesel generators start and the charging flow is automatically reestablished. As soon as off-site power is recovered the operator restarts the RCPs in a time frame of about 30 minutes. The

[a] This is Option A. An Option B will be discussed in Section 3.5.3.1.

charging flow is reduced to 44 gpm to extend the time window for LOOP recovery before borating the RCS.

Sequence 2 — After the LOOP event the diesel generators fail to start, but recover sooner than the off-site power and charging flow is automatically restarted. After off-site power recovers the RCPs may start and core damage may result.

Sequence 3 — This is similar to Sequence 2 except the off-site power recovers earlier than the diesel generators. Both charging and the RCPs may be started leading to the dilution event.

3.5.3 Accident Sequence Quantification

3.5.3.1 Refueling Outage

The initiating frequency, ILOOP, is based on the loss of off-site power frequency data obtained from industrial experience. The loss of off-site power frequency is .11/yr or 1.26E-5/hr. The average yearly frequency of a refueling outage is 0.6/yr and the average length of a dilution during start up is about 8 hours. Hence, ILOOP=6.03E-5/yr.

The time independent probability values P(DSL) and P(CHG) were obtained from Reference 3.3. The value P(DSL)=1.44E-2 represents the unavailability of two diesel generators and P(CHG)=3.3E-3 reflects the unavailability of the three charging pumps.

The calculation of core damage frequency is done similarly to that for Oconee given in Section 3.3.3.1, i.e., with convolution integrals. The core damage frequency for Sequence 1 is:

$$CDF(S1) = \int dt\{ILOOP*[1-P(DSL)]*[1-P(CHG)*[1-P(NR\text{-}LOOP)]*P(CCD)*P(RCPRST)\}$$

The terms ILOOP, P(DSL) and P(CHG) are independent of time but the time dependence of the other terms must be taken into account. The charging flow restarts after the LOOP event and there is a period of vulnerability of 80 minutes (see Figure 3.11). Hence, the portion of the integral where time dependence is considered is:

$$\int_0^{80} dt\{[1-P(NR\text{-}LOOP)]*\int_t^{t+80} dt' P(CCD)*P(RCPRST)\}$$

The above expression is evaluated using the distributions shown in Figures 3.6 and 3.11. P(CCD)=(t'/40) in the interval [0,40], =1 in the interval [40,60], and =(80-t')/20 in the interval [60,80] and P(RCPRST)=(1/30) per minute. The probability of recovering off-site power is [1-P(NR-LOOP)]=(0.15/60) per minute and is based on data for 15% recoveries in one hour. The result of the evaluation is a probability of 0.12. Hence, CDF(S1)=7.11E-6/yr.

In Sequence 2 the diesel generators (DGs) fail initially, but at least one DG recovers before the off-site power recovers. The core damage frequency is:

CDF(S2) = ∫ dt(ILOOP*P(DSL)*[1-P(CHG)]*[1-P(NR-DSL)]*[1-P(NR-LOOP)]
*P(CCD)*P(RCPRST)).

This sequence may be evaluated using the results for Sequence 1 and noting that over a long period (24 hours) the diesel generators would eventually recover. This implies that ∫ dt[1-P(NR-DSL)] is approximately one and the other time dependent terms are independent of the diesel recovery rate. The last three terms in the integral for CDF(S2) are the same as for Sequence 1 and hence, CDF(S2)=1.04E-7/yr.

In Sequence 3 the off-site power recovers sooner than the diesel generators. The charging flow is immediately reestablished, but the RCPs are restarted during a 30-minute time period. At the end of the 30 minutes, the RCPs are running and total mixing is assumed. The core damage frequency is:

CDF(S3) = ∫ dt(ILOOP*P(DSL)*[1-P(NR-LOOP)]*P(CCD)*P(RCPRST)).

As was the case with the recovery of DGs in Sequence 2, in this sequence it can also be assumed that the probability of recovery of off-site power is unity over a 24 hour period so that ∫ dt[1-P(NR-LOOP)] is taken as one. The last two terms are evaluated:

$_0\int^{30} dtP(CCD)P(RCPRST)$

since the RCPs would have started by the end of this 30-minute period. For this integral P(CCD)=./40. This results in CDF(S3)=3.30E-7/yr.

The total core damage frequency for a refueling outage for this type of rapid dilution scenario is the sum of the frequencies for the three different sequences or 7.54E-6/yr.

In the above analysis it was assumed that the makeup rate after a LOOP was reduced to 40 gpm from the normal 84 gpm in order to extend the time period before it might be required to initiate boration of the RCS. The effect of this assumption is examined by assuming that the makeup rate remains at 84 gpm. The effect on P(CCD) is shown as Option B in Figure 3.13. The maximum value is now reached at 25 minutes and the period of vulnerability for the accident is reduced to 60 minutes from the 80 minutes with the smaller makeup flow.

With the different makeup rate the core damage frequencies for the three sequences become 4.74E-6/yr, 6.92E-8/yr, and 5.04E-7/yr, respectively. The total core damage frequency is 5.31E-6/yr which is a reduction of 30% and reflects the decreased period of vulnerability.

3.5.3.2 Non-Refueling Outage

For a non-refueling outage the approach is similar to that explained in Section 3.3.3.2. The initiating frequency is higher than for a refueling outage because there are two of these outages per year. Hence, ILOOP=2.01E-4. The other change is the reduction in the conditional probability for core damage given the injection of the diluted water and the restart of an RCP. This is because of the higher natural circulation flow rate assumed for these outages. This is reflected in the time dependent probability P(CCD) shown in Figure 3.11. The result with the assumption of 40 gpm for the makeup flow is: CDF(S1')=7.11E-6/yr, CDF(S2')=1.04E-7/yr, and CDF(S3')=3.30. This results in a total core damage frequency of 1.25E-5/yr.

If the assumption is made that the makeup flow rate is not changed from the 84 gpm expected before the LOOP (Option B), then the core damage frequencies for the three sequences become 7.87E-6/yr, 1.17E-7/yr, and 8.37E-7/yr, respectively, and the total becomes 8.82E-6/yr.

3.6 System Description - Surry Station

3.6.1 Chemical and Volume Control System

At the Surry plant it is the chemical and volume control system (CVCS) which controls makeup and letdown and boron concentration. Figures 3.14 and 3.15 show the boron control and letdown/makeup control subsystems of the CVCS, respectively.

To carry out the RCS deboration, the quantity of primary grade water required, along with the rate of addition, is first determined from tables and graphs[a] and set on the batch integrator control. A sample sheet from this procedure is shown in Table 3.2. When boron dilution is initiated, both the primary grade water flow control valve (FCV-1114A on Figure 3.14) and the primary grade makeup stop valve (FCV-1114B) open to establish the flow to the Volume Control Tank (VCT), with the primary water supply pump running.

The boric acid flow control valve (FCV-1113A) is closed so that only primary grade water can enter the VCT through the VCT spray nozzle. The primary makeup stop valve (FCV-1113B) is also closed to prevent the primary grade water from going directly into the charging pump suction header. This is a precaution against a sudden increase in reactivity although bypassing is allowed during the early phase of a xenon transient. When the amount of injected primary grade water reaches the value set on the batch integrator, the makeup valves shut automatically. It is important to note that the rate of addition of primary grade water is determined by the operator and is generally less than the charging rate.

The VCT has an internal volume of 300 cubic feet and normal operating pressure and temperature of 15 psig and 105°F, respectively. The spray nozzle flow is normally about 120 gpm. The VCT level control valve (LCV-1115A), located upstream of the VCT, is a solenoid-operated control valve which is positioned by instrument air to maintain the VCT level at less than 85%. When the VCT level reaches a preset value, the VCT level control valve will begin to direct letdown flow to the Boron Recovery System. At a VCT level of 85%, all letdown flow is diverted to the Boron Recovery System. In the event that the VCT level falls to 13%, the charging pump will automatically shift its suction from the VCT to the Refueling Water Storage Tank (RWST).

---

[a] Surry Power Station Operating Procedure 1-OP-8.3.2, "Blender - Dilute," Virginia Electric Power Co., Aug. 19, 1988.

Table 3.2
Extract from Surry Dilution Procedure

INITIALS

5.0 Procedures

_______ 5.1 Initial conditions satisfied.

_______ 5.2 Precautions and limitations noted and satisfied.

_______ 5.3 Determine the rate and magnitude of primary grade water, to get the desired dilution (see nomograph).

_______ 5.4 Place the MAKEUP MODE CONTROL switch to the "STOP" position.

_______ 5.5 Set the primary grade flow controller (FC-114A) for desired flow rate and set primary grade water integrator (YIC-114A) to desired quantity.

_______ 5.6 Place the MAKEUP MODE SELECTOR switch to "DILUTE".

_______ 5.7 Place the MAKEUP MODE CONTROL switch to "START".

_______ 5.8 Verify the following actions taken place:

Dilute	FCV-1113A	FCV-1113B	FCV-1114A	FCV-1114B
Mode	Closed	Closed	Controlling	Open

_______ 5.9 Dilution will be automatically stopped when the integrator (YIC-114A) setpoint is reached.

_______ 5.10 When the dilution operation is complete, refer to 1-OP-8.3.1 for returning blender to automatic.

During the RCS deboration, the primary grade water exiting from the VCT is injected into cold-leg B by a charging pump at a rate of approximately 120 gpm. About 20% of this flow is diverted to the RCPs for use as seal water. The letdown flow from cold-leg A, in the meantime, is partially diverted to the Boron Recovery System. As explained above, the primary grade water addition is usually less than the charging rate and in order to maintain a constant level in the VCT some letdown flow must be diverted to the Boron Recovery System. The water level in the VCT is maintained between 60-85%. This implies that during the period of about eight hours required for the RCS deboration at Surry, about 60-85% (corresponding to approximately 1800 gallons) of the VCT volume is filled with highly diluted primary grade water.

When the charging pump stops due to loss of off-site power, the letdown orifice isolation valve (HCV-1200B) shown in Figure 3.15 will close automatically, isolating the letdown flow. Unless the operator reopens the valve by resetting the handswitch to OPEN position, it will remain closed even after the charging pump is restarted by emergency diesel power. At the moment the off-site power is lost, it can be assumed that the VCT still contains about 1800 gallons of primary grade water. Since power to the primary water supply pump is also lost, no more primary grade water is pumped into the VCT. From this point on, two somewhat different scenarios are conceivable depending upon whether the letdown flow is quickly restored following the restart of the charging pump.

Probabilistic Analysis

If the letdown orifice isolation valve (HCV-1200B) is not quickly reopened, the relatively cold (120°F) primary grade water will be injected into the RCS by the charging pump without being heated properly at the regenerativ. heat exchanger due to the absence of letdown flow on one side of the heat exchanger. As the water level in the VCT decreases, the VCT level control valve (LCV-1115A) will gradually realign to allow any letdown flow to enter the VCT. Since no letdown flow is entering the VCT, the VCT water level will continue to fall until it reaches the level of 13%, at which point the suction of the charging pump will automatically switch from the VCT to the RWST. At a flow rate of 120 gpm, it will take about 15 minutes for the VCT level to drop to the 13% level. The amount of primary grade water discharged from the VCT into the RCS during this period is approximately 1600 gallons.

If the charging pump continues to run, this volume of primary grade water will be followed by intake from the RWST with a boron concentration of 2300 ppm. It is likely, however, that the letdown flow will be reestablished before the VCT level falls to the 13% level. Since the letdown flow is isolated, the charging flow introduced into the RCS will cause the pressurizer level to gradually increase. The charging flow control valve (FCV-1122) is controlled by a signal from the pressurizer level instrumentation to maintain a prescribed pressurizer water level. It will automatically close if the level setpoint is reached. If this happens, the operator is likely to quickly reestablish the letdown flow so that the charging flow can be maintained. The subsequent scenario will be similar to that which will be described below for another series of events.

Assuming that the letdown orifice isolation valve (HCV-1200) is reopened by the operator soon after the restart of the charging pump, the VCT water level will still fall initially, causing the VCT level control valve (LCV-1115A) to adjust its position to admit the letdown flow to the VCT. The boron concentration in the letdown flow at this point could range from 1500 to 2300 ppm, depending on when during the deboration the LOOP occurs. Before this letdown flow gets recirculated into the RCS there will be about 1800 gallons of (almost) primary grade (PG) water injected since all the PG water remaining in the VCT can be drained by the charging pump. It will take about 15 minutes for the charging pump to inject the 1800 gallons of PG water into the RCS.

3.6.2 Electrical System

A simplified block diagram for the Surry emergency electrical system is presented in Figure 3.16. The RCPs are connected to the non-safety related buses 1A, 1B and 1C. These buses are not supported by any secondary backup source upon losing electrical power from a loss of off-site power event. The charging pumps, however, are connected to either safety related bus 1H or 1J and these are powered in a LOOP event by the respective diesel generators DG1 or DG3.

3.7 Probabilistic Analysis - Surry Station

3.7.1 Accident Sequence Timing

It is apparent from the forgoing discussions that, in addition to the accumulation of a slug of PG water in the primary system, the off-site power has to be recovered and one RCP has to be started up in order for this type of reactivity accident to happen. The times at which these causative events occur relative to each other appear to play an important role in determining the probability and severity of such an accident. As noted previously, it takes about 15 minutes to inject all the PG water left in the VCT into the cold leg. If the off-site power is recovered in about 15 minutes after it is lost, and one of the RCPs is restarted immediately, the likelihood that the accident will occur is maximized. Since the injection of PG water is followed by that of either RWST water (which contains 2300 ppm of boron) or letdown water (which contains at least 1500 ppm of boron), the slug of water which has settled at the bottom of the lower plenum will eventually mix with the borated water reducing the probability of a dilution accident.

Assuming that the letdown water contains 1700 ppm of boron when it is injected into the RCS following the exhaustion of PG water in the VCT, and that complete mixing will occur in the lower plenum, about 10 minutes is sufficient to raise the boron concentration in the lower plenum to the level that is no longer a threat to criticality. In other words, if off-site power is not recovered for more than 25 minutes and if during this period, the charging pump (which was restarted by diesel power soon after loss of off-site power) continues to inject water from either the VCT (PG water followed by letdown water) or the RWST, the frequency of occurrence of this accident will become vanishingly small.

The time behavior of the conditional core damage probability or P(CCD) is modelled as indicated in Figure 3.17. It increases linearly to a value of one in 15 minutes. After this point the probability decreases linearly to zero in another 10 minutes. After this time there is no longer the possibility of this sequence occurring.

The RCP may be restarted after recovery from the LOOP event and this is modelled as it was for Oconee and Calvert Cliffs. The time dependence of the cumulative probability is shown in Figure 3.6 and indicates that the RCPs are expected to be started within 30 minutes of recovery from the LOOP.

3.7.2 Accident Sequence Modeling

To perform the probabilistic analysis for this accident scenario, the event tree shown in Figure 3.18 was developed. This event tree was applied for both refueling and nonrefueling outages. As was seen in the analysis for Oconee and Calvert Cliffs, the top events that change with outage are ILOOP and CCD.

The second top event, DSL, in Figure 3.18 questions whether the emergency diesel generators are available and provide back up power to the emergency safety buses. The third top event, NR-DSL, questions the recovery of the diesel generators. If the diesel generators are available, then the charging flow may be started, if at least one of the three charging pumps are available and its respective electrical supply bus is powered. This function is represented by top event CHG.

The recovery of off-site power is questioned next in top event NR-LOOP. This is a prerequisite for restarting the RCPs, since the diesel generators do not have sufficient capacity. Once the RCPs can be started the top event CCD questions the potential for core damage. The restart of the RCPs is questioned in the top event RCPRST.

Probabilistic Analysis

The event tree as indicated in Figure 3.18 is a time dependent tree, since some of the top events strongly depend on the time and this must be taken into account to evaluate the final CDF. There are three sequences marked as leading to core damage (CD) and these indicate where the boron dilution reactivity excursion leads to core damage. The following is a short description of each CD sequence:

Sequence 1 — After a LOOP event the diesel generators start and the charging flow is automatically reestablished. As soon as off-site power is recovered the operator will restart the RCPs in a time frame of 30 minutes.

Sequence 2 — After the LOOP event the diesel generators fail, but recover sooner than the off-site power and charging flow is immediately restarted. After off-site power recovers the RCPs may start and core damage may result.

Sequence 3 — This is similar to Sequence 2 except the off-site power recovers earlier than the diesel generators. Both charging and the RCPs may be started leading to a reactivity accident.

3.7.3 Accident Sequence Quantification

3.7.3.1 Refueling Outage

The initiating event frequencies were calculated based on (Surry) plant specific data. For refueling outages the frequency is 0.6/yr and the failure rate of off-site power is 2.85E-5/hr. Assuming that the RCS deboration requires about eight hours, ILOOP=6.03E-5/yr.

The time independent probabilities P(DSL) and P(CHG) were obtained from Reference 3.3. P(DSL)=3.7E-03 represents the unavailability of two diesel generators and P(CHG)=4.0E-03 reflects the unavailability of the three charging pumps.

The calculation of core damage frequency is done similarly to that for Calvert Cliffs given in Section 3.5.3.1. The core damage frequency for Sequence 1 is:

$$CDF(S1) = \int dt\{ILOOP*[1-P(DSL)]*[1-P(CHG)]*[1-P(NR\text{-}LOOP)]*P(CCD)*P(RCPRST)\}.$$

The first three terms ILOOP, P(DSL) and P(CHG) are independent of time and only the last three terms must be evaluated by considering their time dependence. Effectively, the charging flow starts after the LOOP event and there is a period of accident vulnerability of 25 minutes (see Figure 3.17). The time dependent portion may be written:

$$\int_0^{25} dt(\{1-P(NR\text{-}LOOP)\}, \int_t^{25} dt' P(CCD)*P(RCPRST)\}$$

The above expression is evaluated taking into account the functions given in Figures 3.6 and 3.17. P(RCPRST)=(1/30) per minute. P(CCD)=t'/15 in the interval [0,15] and =(25-t')/10 in the interval [15,25]. The value of [1-P(NR-LOOP)], or the probability of LOOP recovery, is (0.15/50) per minute based on a 15% recovery rate at the end of one hour. Evaluating the integral in the subintervals leads to a probability of 5.92E-2. Hence, CDF(S1)=3.55E-6/yr.

In Sequence 2 the diesel generators fail initially, but at least one DG recovers before the off-site power recovers. The CDF is evaluated from:

CDF(S2) = ∫ dt{ILOOP*P(DSL)*[1-P(CHG)]*[1-P(NR-DSL)]*[1-P(NR-LOOP)]
*P(CCD)*P(RCPRST)}.

This sequence may be evaluated using the results of Sequence 1 by noting that over a long period (24 hours) the diesel generators are expected to recover. This implies that ∫ dt[1-P(NR-DSL)] is approximately one. The other time dependent terms are independent of when the diesels recover. For this sequence the last three terms are the same as for previous sequence and CDF(S2)=1.32E-8/yr.

In Sequence 3, the off-site power recovers sooner than the diesel generators. The charging flow is immediately reestablished, but the RCPs are restarted over a 30-minute period. At the end of the 30 minutes, the RCPs are running and total mixing is assumed. The core damage frequency is:

CDF(S3) = ∫ dt{ILOOP*P(DSL)*[1-P(NR-LOOP)]*P(CCD)*P(RCPRST)}.

The integrated probability for off-site recovery is similar to that for diesel recovery, i.e., eventually all LOOP events recover within 24 hours so that ∫ dt[1-P(NR-LOOP)] is about one. The last two terms are the only time dependent terms and they are evaluated from 0 to 25 minutes using the functions given in Figures 3.6 and 3.18. The final result is that CDF(S3)=9.37E-8/yr.

The total core damage frequency is obtained by summing the frequencies for the three sequences and is 3.66E-6/yr.

3.7.3.2 Non-Refueling Outage

The frequency of the accident during a non-refueling outage is calculated taking into account the different initiating frequency and a different probability for conditional core damage (as seen in Figure 3.17). The frequency of a drained maintenance outage is 1.2/yr and a non-drained maintenance outage is 0.6/yr. Using these numbers and the estimated frequency for a loss of off-site power results in ILOOP=2.01E-4/yr.

The results for the same three sequences are now: CDF(S1')=5.89E-6/yr, CDF(S2')=2.19E-8/yr, and CDF(S3')=1.56E-7/yr. The total core damage frequency for this type of accident in this type of outage is 6.08E-6/yr.

3.8 Summary

The results of the analysis for the three plants are given in Table 3.3 which shows not only the expected core damage frequency (CDF), but also the initiating frequency of these events. The latter was of particular interest because in the analysis done in Europe (cf Section 2.1), the initiating frequency was quoted as being an order of magnitude higher. The CDF is similar for all three plants and in the range considered significant.

Table 3.3
Summary of Important Frequencies

	OCONEE		CALVERT CLIFFS[a]		SURRY	
	INIT FR /YR	CDF /YR	INIT FR /YR	CDF /YR	INIT FR /YR	CDF /YR
REFUELING	4.93E-5	1.05E-5	6.03E-5	7.54E-6	6.03E-5	3.66E-6
NON-REFUELING	1.64E-4	1.75E-5	2.01E-4	1.25E-5	2.01E-4	6.08E-6
TOTAL		2.80E-5		2.00E-5		9.74E-6

[a] Option A

These results are dependent on plant design and various assumptions used in the analysis. The most important assumptions are summarized below. Note that some of them result in overestimating the core damage frequency.

1. The dilution time during startup is 8 hrs. The consequences of the event are independent of when during this period the loss of off-site power occurs. In reality, an event occurring early during this period will have more shutdown margin to overcome and is, therefore, expected to have less of an effect than an event occurring near the end of the normal dilution procedure.

2. No credit is given for the operator to take action and stop the charging flow from the VCT after the LOOP. Although dilution while the shutdown banks are inserted or the RCPs are stopped, (as would be the case after a LOOP) is not a normal procedure, it is assumed that since the operator knows that the flow from the primary water makeup pump has ceased that no other action is deemed warranted. An action that could be taken by the operator would be to switch the charging pump suction to the RWST.

3. For all three plants the dilution is done with flow from the VCT. It should be noted that in some plants the suction for the charging flow comes directly from the primary grade makeup water source and once the PG water pump is tripped there is no longer the potential for adding unborated water to the RCS.

   Oconee. The dilution rate is about the same as the letdown and the volume in the letdown storage tank is diluted to low boron levels (0-200 ppm). The available volume in the letdown storage tank is about 1900 gal.

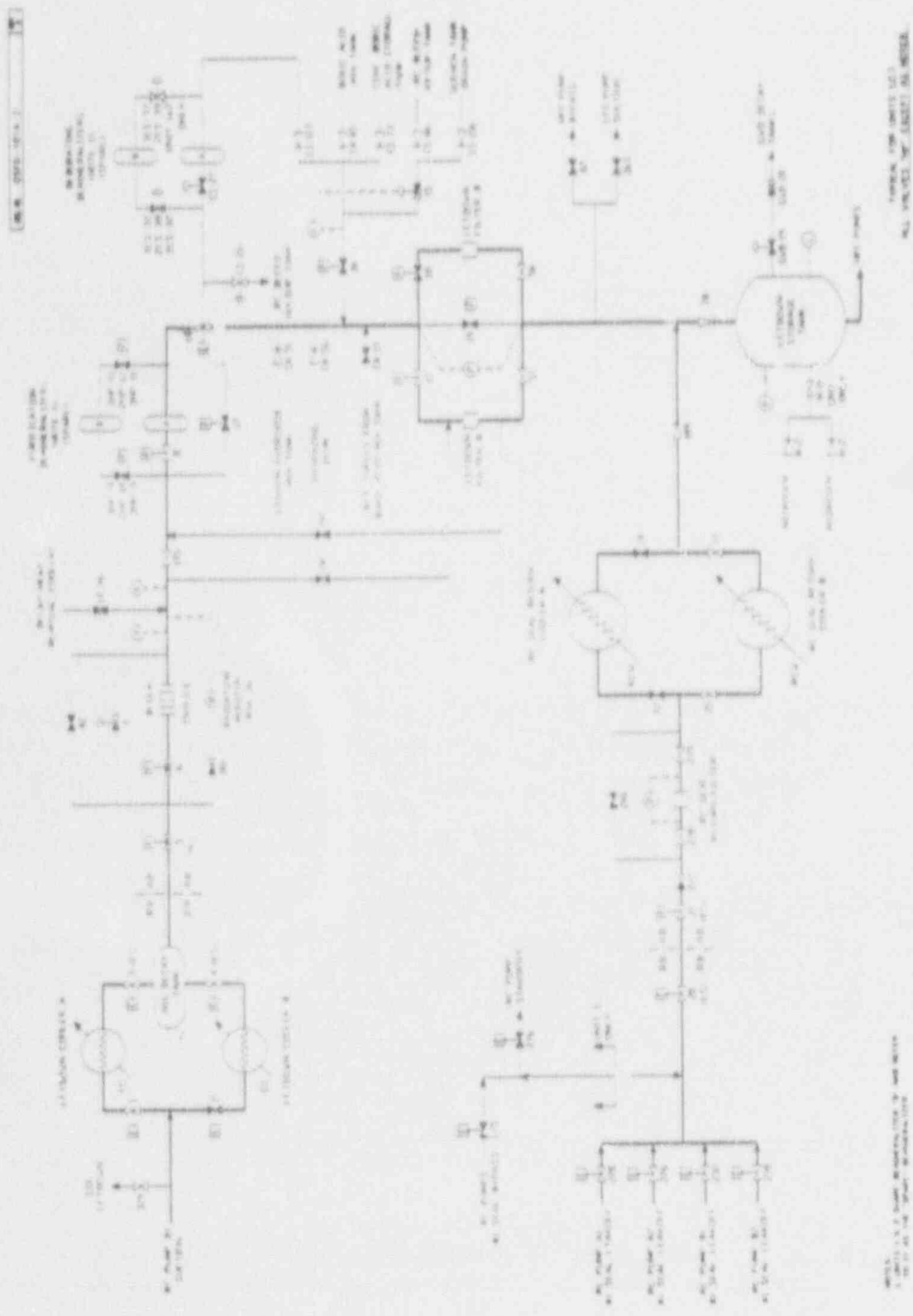

Figure 3.1 High Pressure Injection System - Oconee

Calvert Cliffs: The dilution rate is matched by the letdown flow rate. The volume control tank (VCT) is eventually diluted to a very low boron concentration. The available volume for injection into the RCS is about 2900 gal.

Surry: The dilution rate is generally lower than the charging rate, and the VCT may get diluted to a very low boron concentration (0-100 ppm). The dilution flow is always directed to the VCT and bypassing is allowed only during xenon transients. The available volume for injection into the RCS is about 1600 gal.

4. For all three plants the potential for an accident is limited by the amount of diluted water in the VCT as the supply of primary grade water is stopped by a PG makeup pump. However, there are plants where this pump is connected to the emergency bus and the probability of an accident will be increased if primary grade water continues to be pumped into the VCT. This appears to be the case for some plants in France and Sweden and is another reason why the problem may be more serious there. The question of whether the makeup pump trips or continues to run has to be evaluated on a unit by unit basis.[a]

5. Refueling outage: The conditional core damage probability is linearly changing between zero and one, corresponding to the amount of diluted water injected into the RCS. Mixing and switchover to a borated source reduces the probability of core damage from one to zero over a short period of time.

   Non-refueling outage: The probability for conditional core damage varies between zero and one-half to account for the potentially higher natural circulation rate and mixing.

   For any outage the probability of core damage used is expected to be conservative because it does not account for any mixing that may occur (cf Section 4).

6. If off-site power, or another adequate power source, is available, the reactor coolant pumps (RCPs) will be started over a 30-minute interval.

[a] At the Ringhals plant in Sweden of three units designed by Westinghouse, two have the PG water pump connected to an emergency bus.

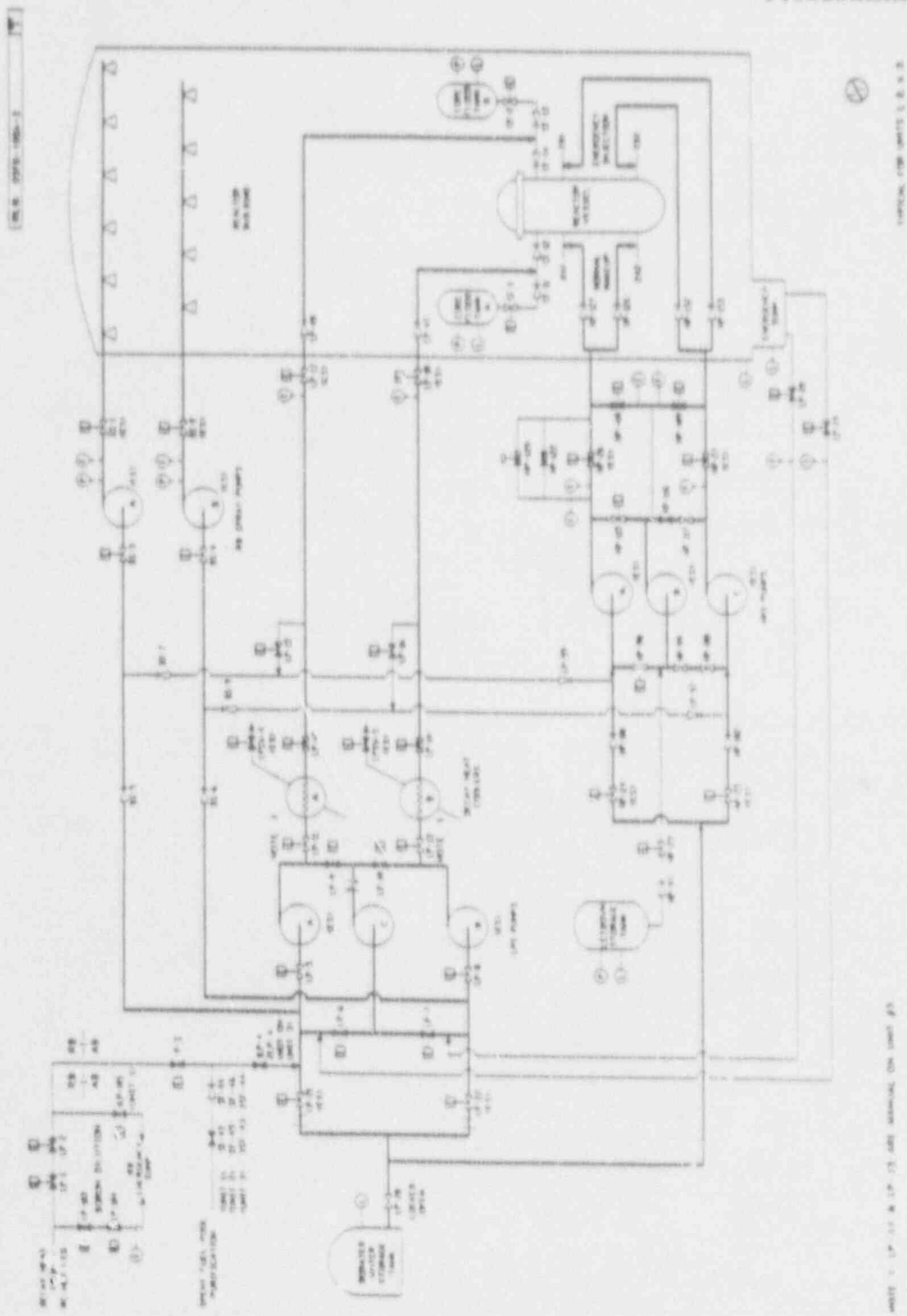

Figure 3.2 ECC and RB Spray Systems - Oconee

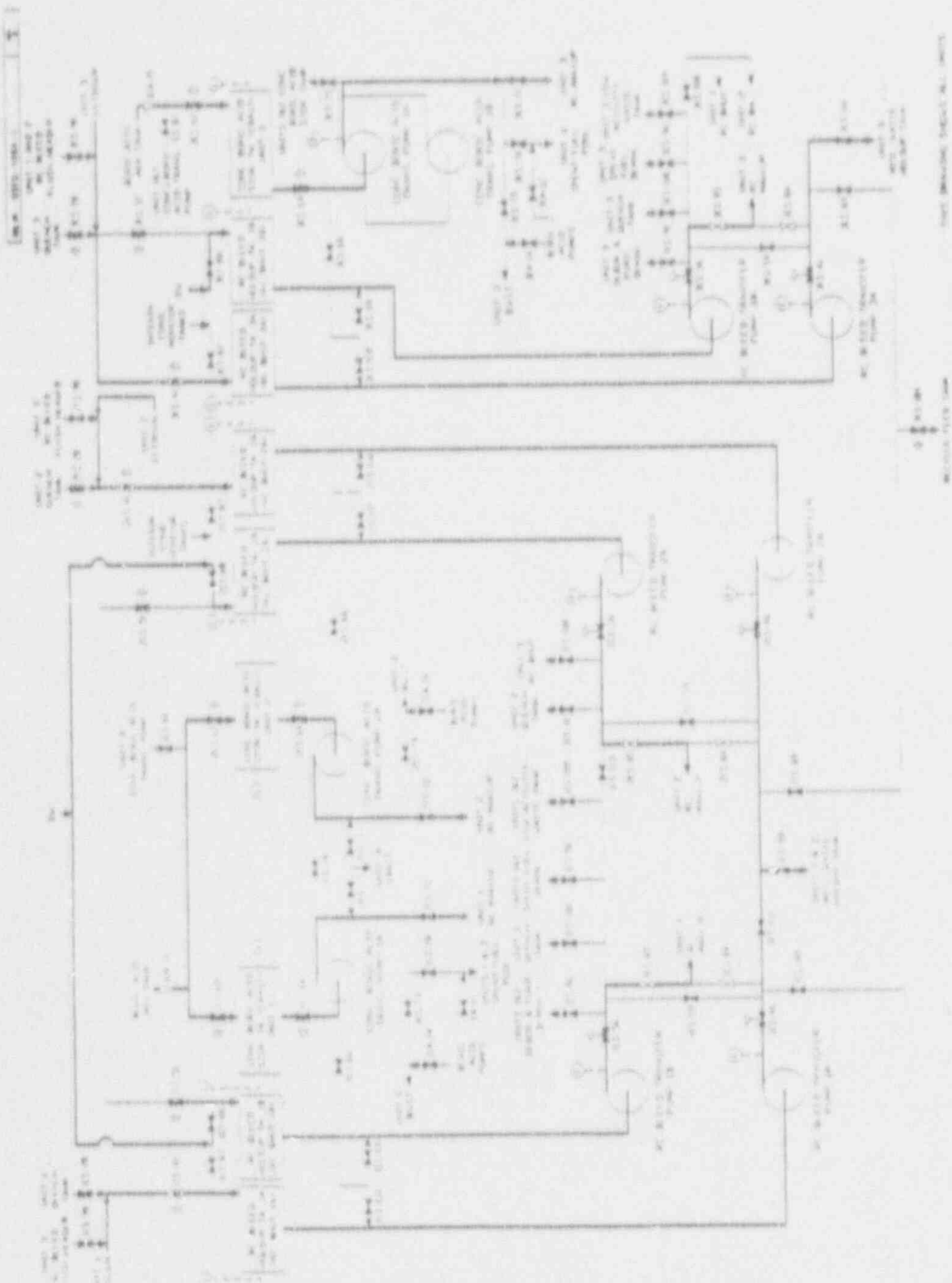

Figure 3.3 Coolant Treatment System - Oconee

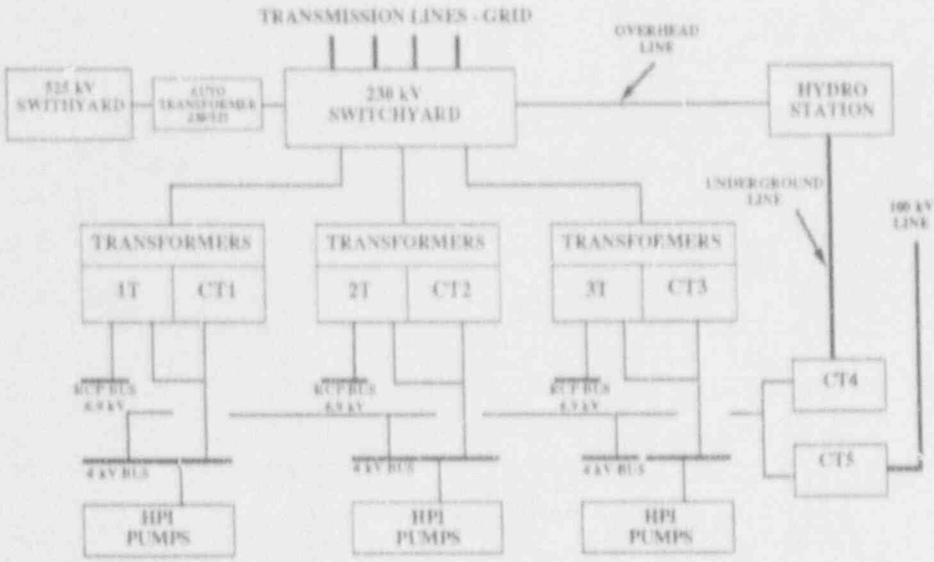

Figure 3.4 Electrical System - Oconee

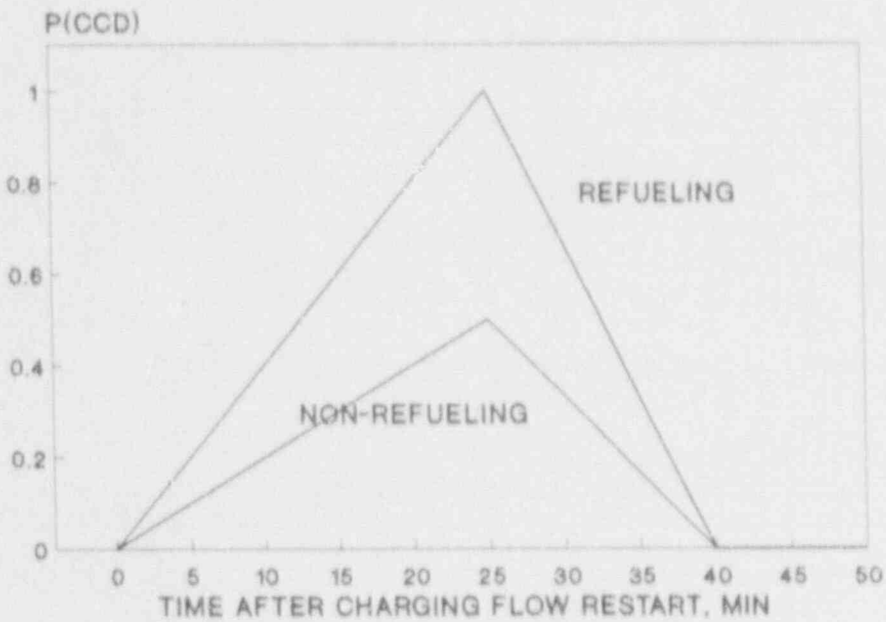

Figure 3.5 Conditional Core Damage Probability - Oconee Plant

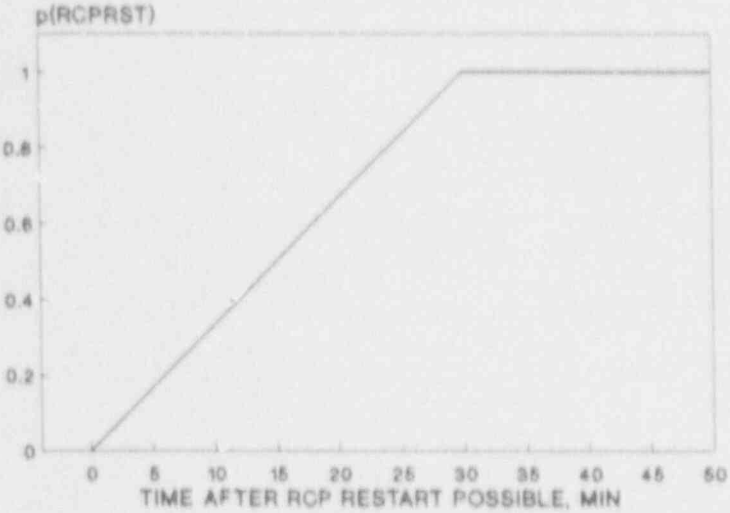

Figure 3.6 Cumulative Probability for RCP Restart

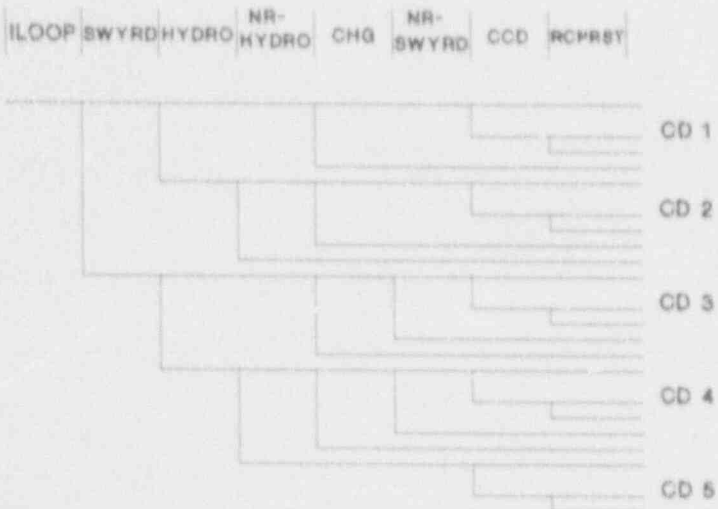

Figure 3.7 Boron Dilution Event Tree - Oconee

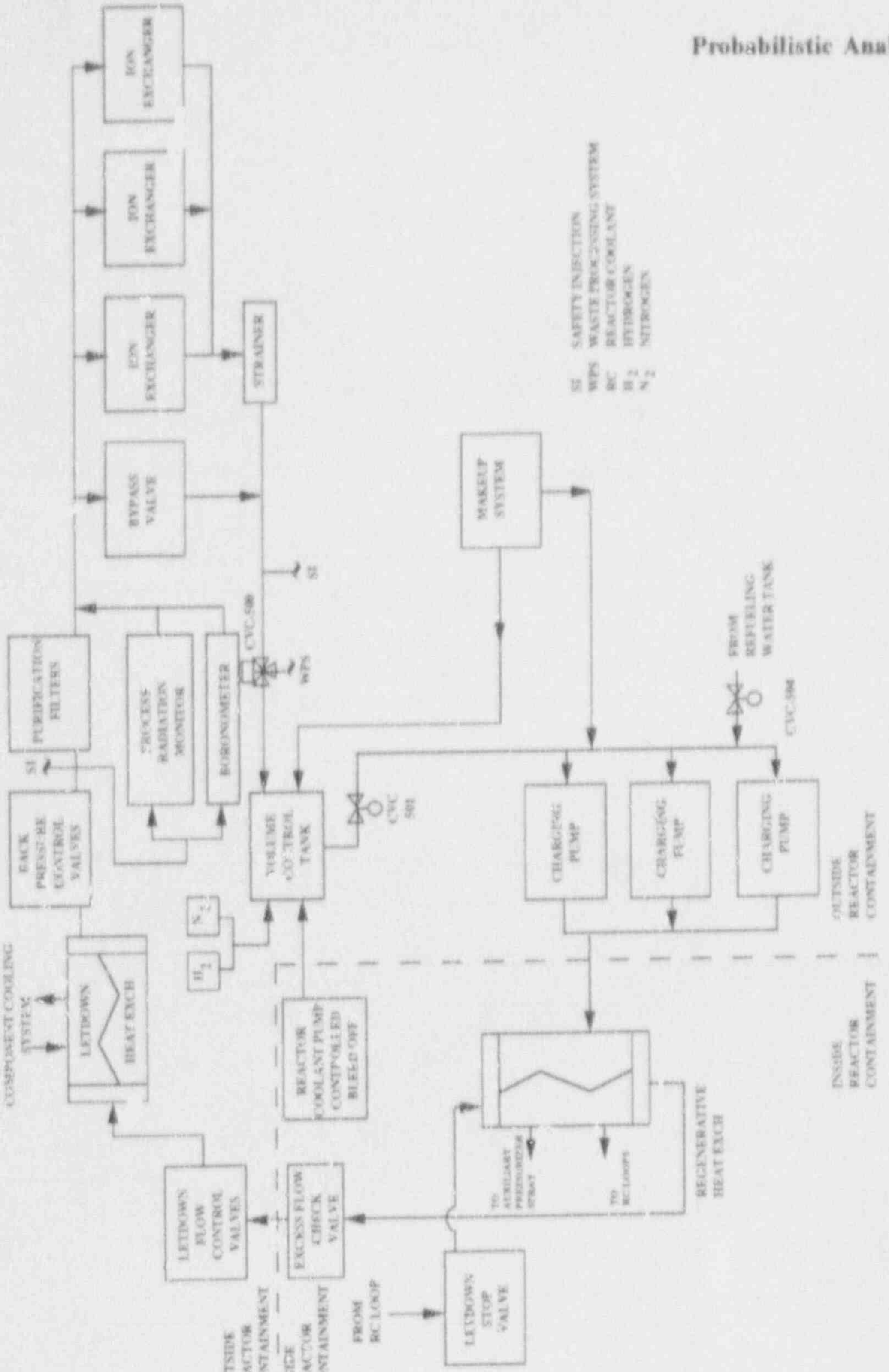

Figure 3.8 Charging and Letdown System - Calvert Cliffs

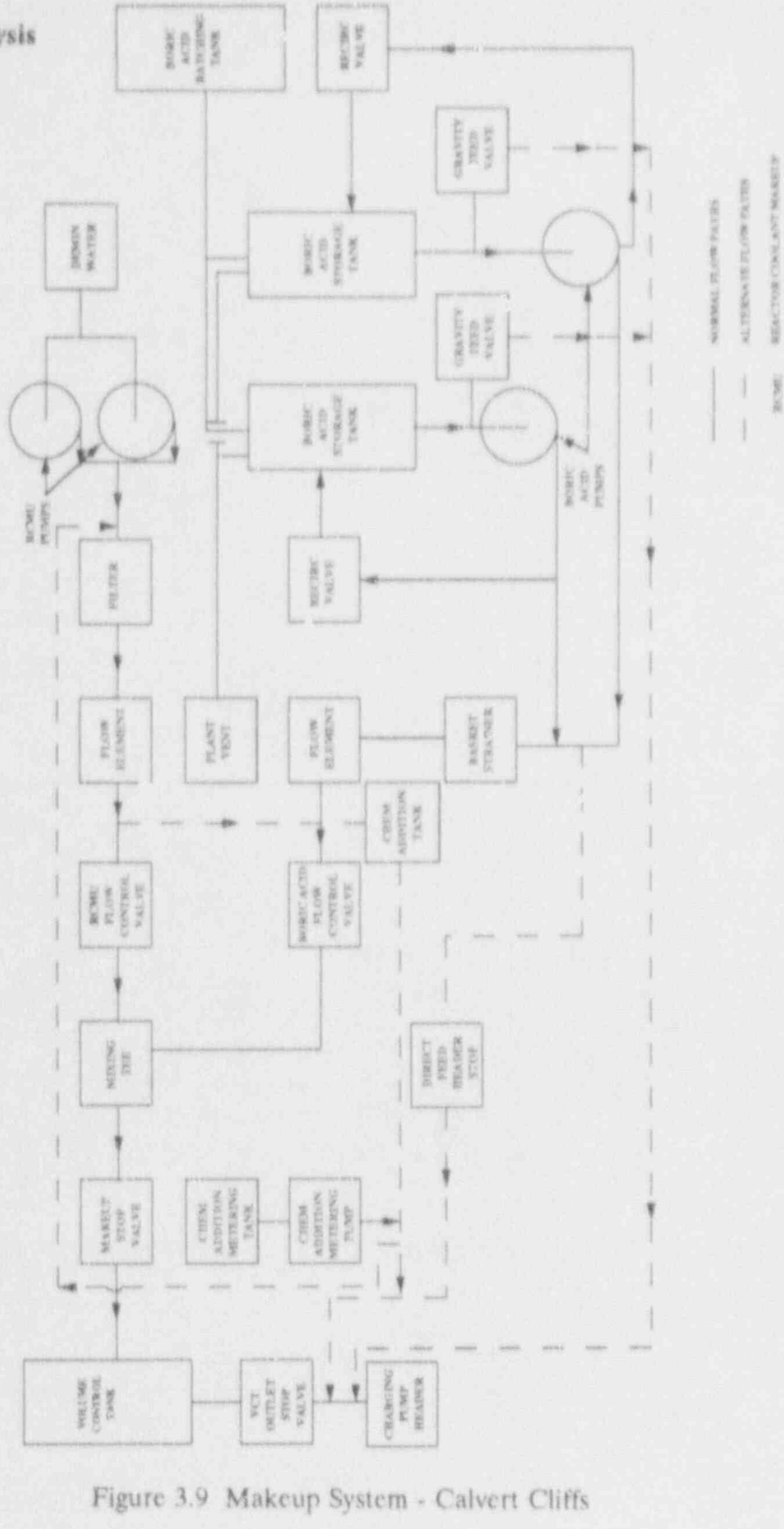

Figure 3.9 Makeup System - Calvert Cliffs

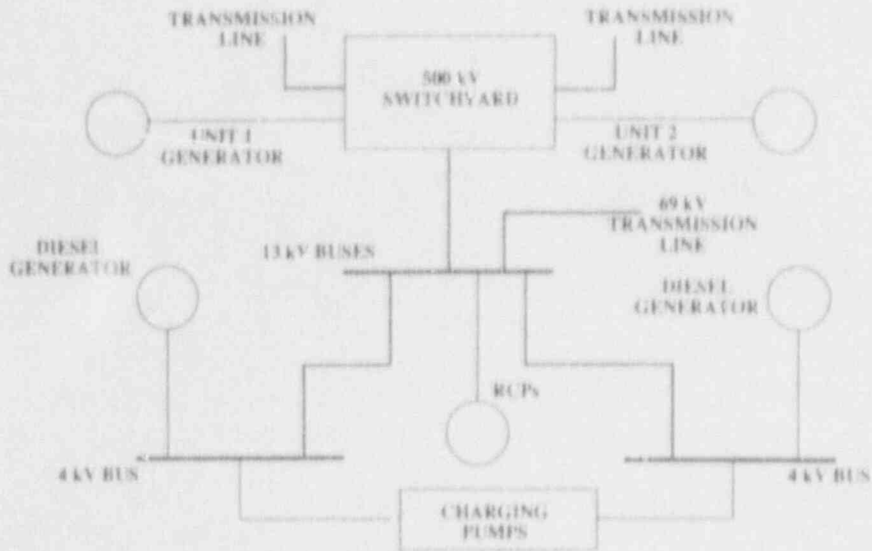

Figure 3.10 Electric System - Calvert Cliffs

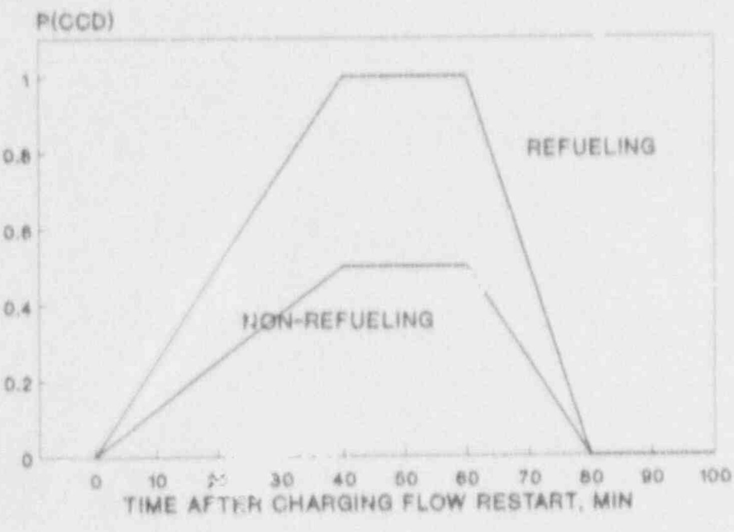

Figure 3.11 Conditional Core Damage Probability Calvert Cliffs Plant - Option A

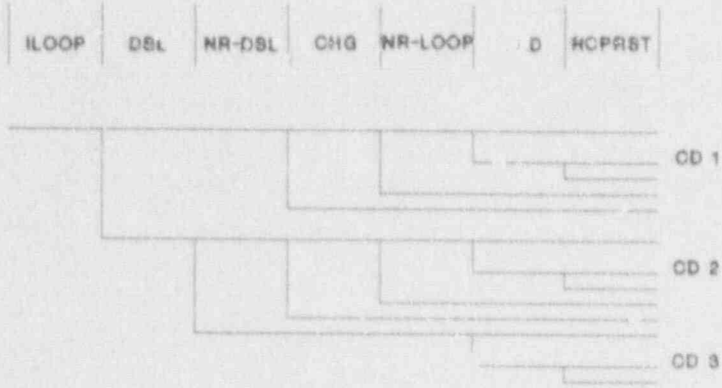

Figure 3.12 Boron Dilution Event Tree - Calvert Cliffs

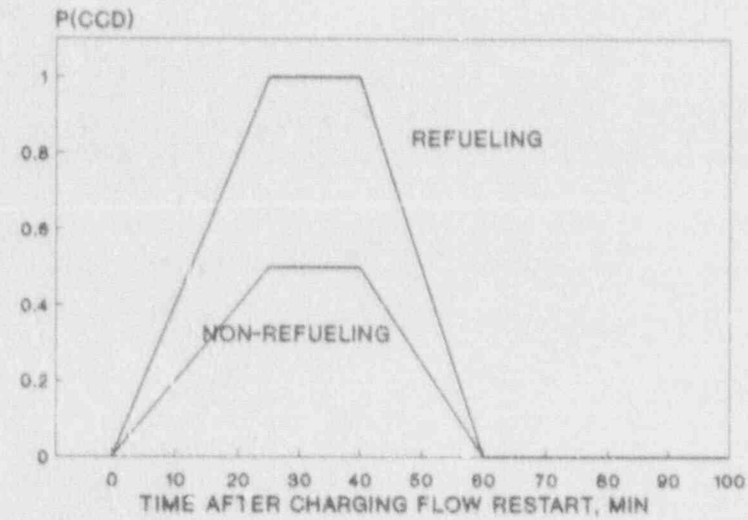

Figure 3.13 Conditional Core Damage Probability Calvert Cliffs Plant - Option B

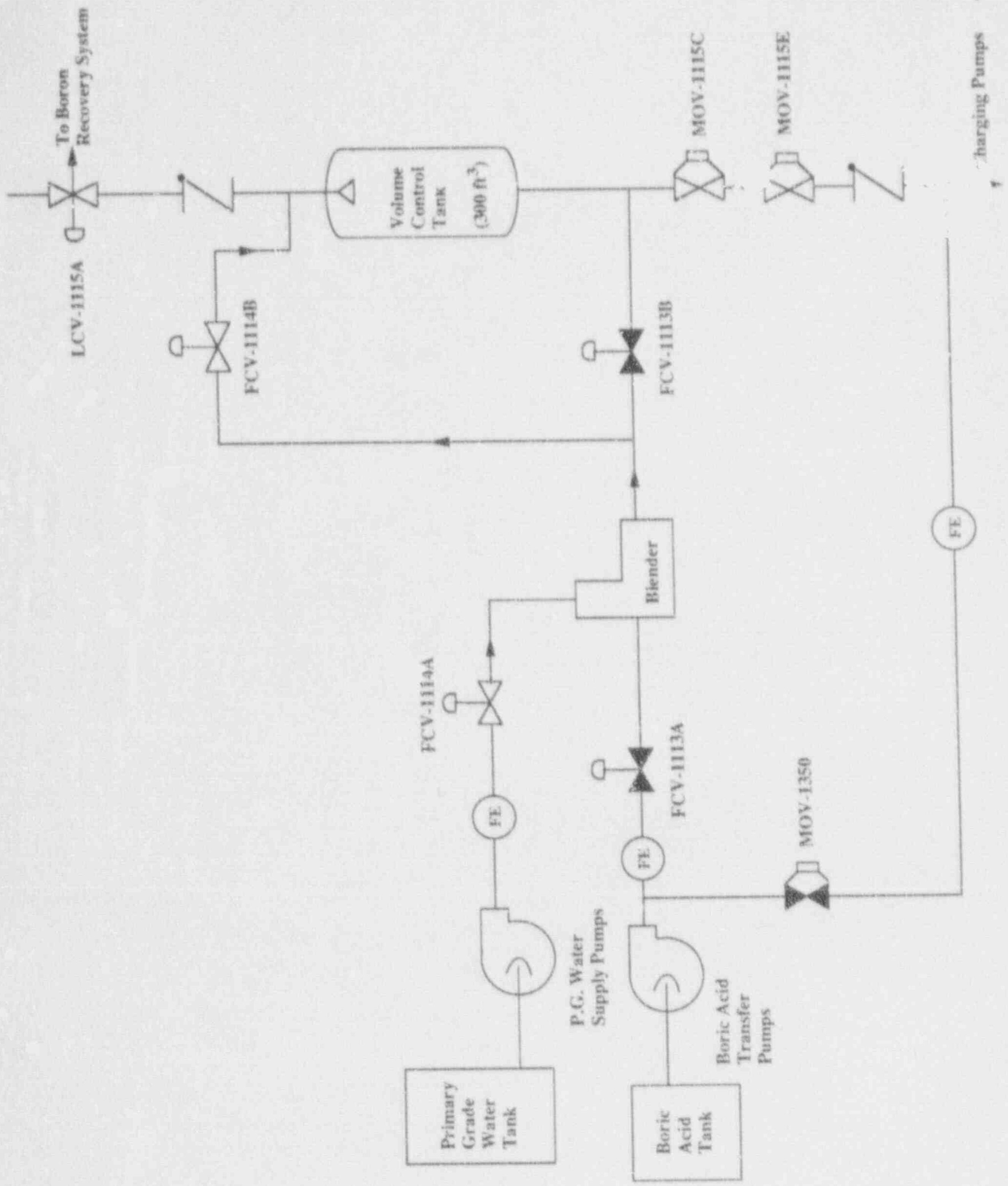

Figure 3.14 Boron Dilution Surry

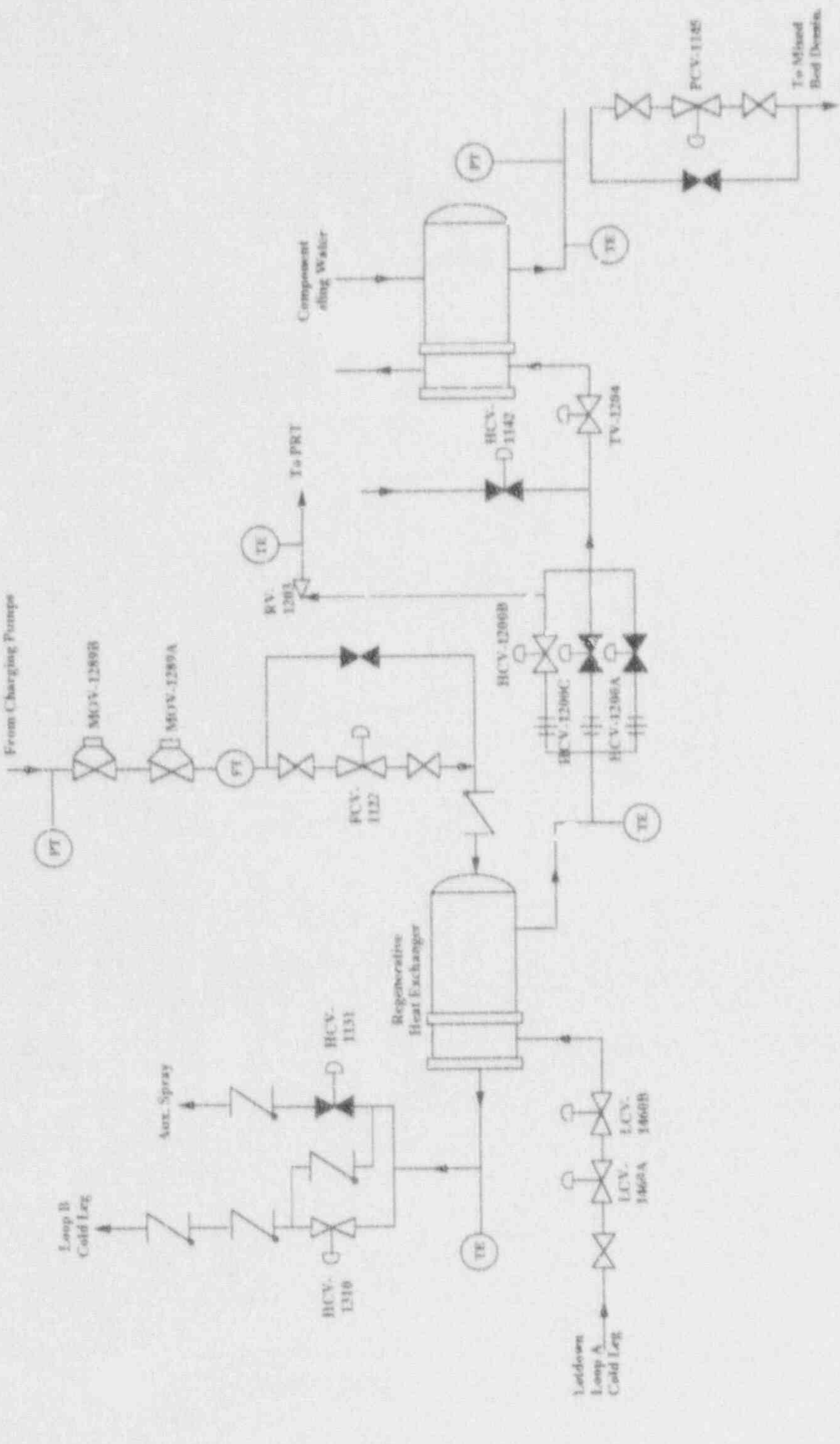

Figure 3.15 Charging and Letdown Subsystem - Surry

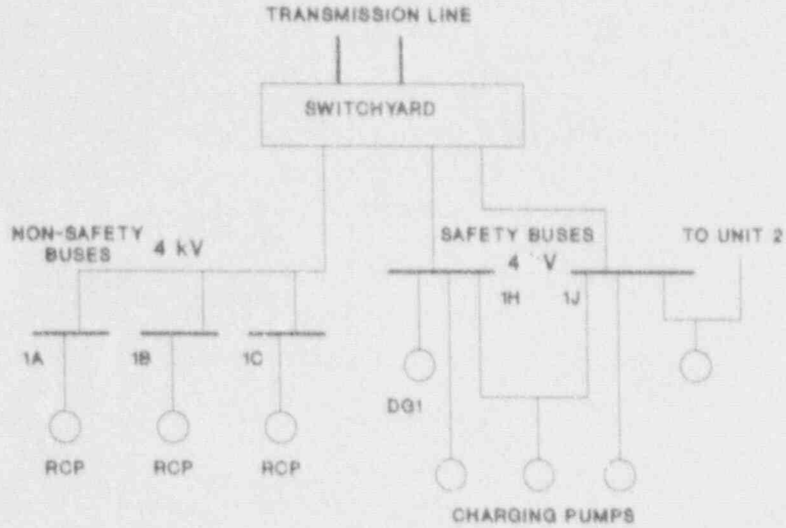

Figure 3.16 Electrical System - Surry

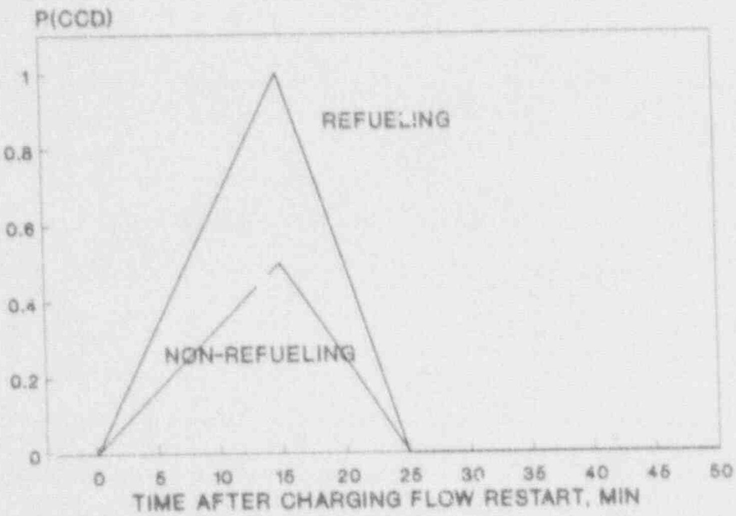

Figure 3.17 Conditional Core Damage Probability - Surry Plant

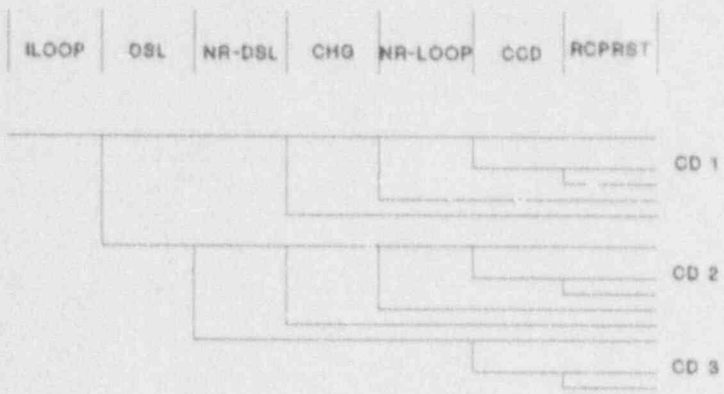

Figure 3.18 Boron Dilution Event Tree - Surry

4 Thermal-Hydraulic Analysis

4.1 Introduction

In Section 3 the conservative assumption was made that the charging flow, consisting of unborated water, does not mix sufficiently with the borated water in the RCS so that a diluted region accumulates in the lower plenum with the potential to cause a power excursion. It is known that there will be some mixing and in this chapter the extent of this mixing is quantified. The analysis assumes that the unborated charging flow is colder than the water in the RCS and it is injected into the cold leg which is otherwise stagnant or at a low natural circulation flow rate.

The modelling approach is similar to that used in the regional mixing model developed by Nourbakhsh and Theofanous [4.1, 4.2]. That work was in support of the NRC Pressurized Thermal Shock (PTS) study to predict the overcooling transients due to high pressure safety injection into a stagnant loop of a PWR. The analysis includes quantification of mixing (entrainment) at locations where the mixing is expected to be intense such as at the connection of the charging line to the reactor coolant system and in the downcomer. These mixing models are then used to determine the dilution boundary as a function of time.

4.2 Thermal Mixing Considerations

Qualitatively, the physical condition may be described with the help of Figure 4.1. In the absence of loop flow, the relevant parts of the system include the loop seal, pump, cold leg, downcomer, and the lower plenum. Initially, this portion of the primary system is filled with borated water with a boron concentration of ~1500 ppm and at a temperature near that of normal operation (~550°F). The dilution transient occurs with charging pump(s) injecting unborated water into the cold leg at a rate of ~45 to 96 gpm. Typical temperatures of the makeup (charging) flow are ~400°F to 500°F, although, depending on the plant and the stoppage of letdown flow during LOOP, lower temperatures, on the order of ~160°F, are also possible.

The ensuing flow regime is schematically illustrated in Figure 4.1. A "cold diluted stream" originates with the charging buoyant jet at the point of injection, continues toward both ends of the cold leg, and decays away as the resulting buoyant jets fall into the downcomer and pump/loop-seal regions. A "hot stream" flows counter to this "cold diluted stream" supplying the flow necessary for mixing (entrainment) at each location. This mixing is most intensive in certain locations identified as mixing regions (MRs). MR1 indicates that mixing associated with the highly buoyant charging jet. MR3 and MR5 are regions where mixing occurs because of the transitions (jumps) from horizontal layers into falling jets. MR4 is the region where the downcomer (planar) buoyant jet finally decays. The cold streams have special significance since they induce a global recirculating flow pattern with flow rates significantly higher than the charging flow. The whole process may be viewed as the quasi-static decay of the cold diluted stream within a slowly varying "ambient" temperature and boron concentration.

4.3 Regional Mixing Model

The quantitative aspects of this physical behavior were incorporated in the Regional Mixing Model [4.1-4.3]. This model accounts for countercurrent flow limitations between the cold and hot streams at the cold leg/downcomer junction and incorporates plume mixing rates which are consistent with data from idealized plume geometries.

The computation proceeds at two levels. The one is global and seeks to establish a "mean system response", referred to as "ambient" in the discussion of the flow pattern above. The other is local and seeks to partition mass and energy into the cold and hot stream consistent with mixing rates and countercurrent flow requirements. The global computation depends on the dilution sequence conditions and is discussed in Section 4.4.

At the local level of the computation the flows, energies, boron concentration and the volumes of the cold and hot streams must be established. The mass, energy, and boron balances for the control volume around MR1 (see Figure 4.1) yield:

$$\rho_{ch} Q_{ch} + \rho_h Q_h = \rho_c Q_c \tag{4.1}$$

$$\rho_{ch} Q_{ch} h_{ch} + \rho_h Q_h h_h = \rho_c Q_c h_c \tag{4.2}$$

$$\rho_{ch} Q_{ch} C_{ch} + \rho_h Q_h C_h = \rho_c Q_c C_c \tag{4.3}$$

where ρ, Q, h, and C represent density, flow rate, enthalpy, and boron concentration, respectively, and the subscripts ch, h, and c refer to the charging, hot, and cold streams, respectively. The hot stream flow rate is equal to that entrained into the jet and depends on the injection Froude number, Fr_{ch}, location of injection (side, top or bottom) into the cold leg, and the path length of the jet before reaching the cold streams, i.e.,

$$Q_h = Q_e(Fr_{ch}, D_{cl}, H_c) \tag{4.4}$$

where D_{cl} and H_c represent the cold leg diameter and the height of cold stream, respectively.

This entrainment function can be obtained by using the analytical or experimental results from the idealized buoyant jet geometries. Note that the Froude number is the ratio of inertial to buoyancy forces. Energy and boron concentration can be partitioned into the hot and cold stream volumes such that the total energy and boron mass remain equal to their corresponding mean values obtained from the global calculations.

The essential control of the overall process is provided by the countercurrent flow requirement as expressed by the condition of stationarity of long, neutrally stable waves at the interface between the cold and hot streams, i.e.,

$$Fr_h^2 + Fr_c^2 = 1 \tag{4.5}$$

The Froude numbers in Equation 4.5 must be based on the actual cold stream and hot stream hydraulic diameters (stream cross-sectional area divided by the width of contact between the two streams, W), and respective flow rates exiting or entering the cold leg. A parameter ß is used to express the fraction of jet entrainment, Q_e, coming from the direction of the vessel, i.e., the hot stream flow for use in Frh is $ßQ_e$. Therefore, the portion arriving for entrainment from the loop seal side would be $(1-ß)Q_e$. Since there is no outflow from the horizontal part of the loop seal, an equal volumetric rate of cold stream must flow in that direction. As a consequence, the net flow to be used in Fr_c of Equation 4.5 should be $Q_{ch} + ßQ_e$.

A symmetric behavior, i.e., ß = 0.5, is appropriate if charging flow is injected into a horizontal cold leg. A ß = 1 is used when the charging flow is injected into an inclined portion of the cold leg (e.g., as in the Oconee injection configuration).

Equation 4.5 can be put in dimensionless form [4.4, 4.5] as

$$Q^{*3} + aQ^{*2} + bQ^{*} + c = 0 \tag{4.6}$$

where

$$a = \frac{1}{\beta\rho^{*}\sigma}\left\{\frac{\rho^{*}}{(1-A^{*})^{3}} + \frac{1+2\rho^{*}}{A^{*3}}\right\}, \quad b = \frac{1}{\beta^{2}\rho^{*}\sigma}\left\{\frac{\rho^{*}+2}{A^{*3}}\right\} \tag{4.7}$$

$$c = \frac{1}{\beta^{3}\rho^{*}\sigma}\left\{\frac{1}{A^{*3}} - \frac{1}{W^{*}Fr_{ch,cl}^{2}}\right\} \tag{4.8}$$

$$\sigma = \frac{1}{(1-A^*)^3} + \frac{1}{A^{*3}} \tag{4.9}$$

$$Q^* = \frac{Q_e}{Q_{ch}}, \quad \rho^* = \frac{\rho_h}{\rho_{ch}} \tag{4.10}$$

$$W^* = \frac{WD_{cl}}{A_{cl}}, A^* = \frac{A_c}{A_{cl}}, H^* = \frac{H_c}{D_{cl}} \tag{4.11}$$

and

$$Fr_{ch,cl} = \frac{(Q_{ch}/A_{cl})}{\left\{gD_{cl}\frac{\rho_{ch} - \rho_h}{\rho_{ch}}\right\}^{1/2}} \tag{4.12}$$

Since W^*, A^*, and H_c^* are all geometrically related, Equation 4.6 provides a simple relationship of the form:

$$Q^* = f_1(H^*_c, \rho^*, Fr_{ch,cl}, \beta) \tag{4.13}$$

Similarly, Equation 4.4 can be put in a dimensionless form:

$$Q^* = f_2(Fr_{ch,cl}, H^*_c, D^*) \tag{4.14}$$

where

$$D^* = \frac{D_{cl}}{D_{ch}} \tag{4.15}$$

In reactor applications the variation of ρ^* during a dilution transient is small and the effect of this variation on the results of Equation 4.13 is negligible. The $Fr_{ch,cl}$ increases gradually during a dilution scenario.

The countercurrent flow limiting condition (Equation 4.13) and jet entrainment (Equation 4.14) can be reduced to two sets of plots such that the stratification (H_c^*) and entrainment Q^* can be determined by a simple superposition procedure (see Section 4.4).

The temperature and boron concentration in the downcomer may be estimated on the basis of mixing of the cold stream spilling out of the cold leg. A highly complicated three dimensional mixing pattern occurs in MR3. In the original formulation of the Regional Mixing Model [4.1], the approach was to conservatively neglect this contribution to the mixing in the downcomer. Rather, the cold stream exiting the cold leg was assumed to form smoothly into the planar plume within the downcomer and to decay according to the K-ϵ-θ turbulence model prediction. A refinement was possible on the basis of Purdue's 1/2 scale data [4.3]. The planar plume is taken to form within a distance of $2D_{cl}$ below the cold leg centerline and to be fed in equal volumetric flow rates by the cold stream and surrounding hot volume fluid. Below this point the decay is approximated by that of a planar plume of initial width equal to D_{cl} and Fr = 1.0 as show in Figure 4.2. The plot shows the temperature function vs distance down the plume. The centerline temperature of the plume, T, temperature of the mixed mean region outside the plume, T_m, and temperature at the jump, T_j, are related to concentrations in the present problem.

It should be noted that such thermal stratification is obtained at low (and zero) loop flow, and it cannot be represented with typical system thermal-hydraulic codes (e.g., TRAC and RELAP5) to simulate rapid boron dilution transients. For a well-mixed condition, when system codes are applicable, there must be sufficient loop flow not only to break up the charging plume (jet) but also to produce stable flow into the downcomer. Nourbakhsh and Theofanous [4.6] used the boundary of stability (Fr_{cl} = 1) and developed a criterion for the existence of perfect mixing in the presence of loop flow. Their stratification/mixing boundary, shown in Figure 4.3, can be expressed by:

$$Fr_{cl} \approx \left[1 + \frac{Q_L}{Q_{ch}}\right]^{-7/5} \tag{4.16}$$

Although Equation 4.16 has been developed for the conditions of high pressure safety injection, it is also valid for the Fr_{cl} range of interest for charging injection. Loop flows of 20 (for Surry) to 45 (for Calvert Cliffs) times the charging flow are required to have perfect mixing in the cold leg and, therefore, to be able to apply the typical system thermal-hydraulic codes.

4.4 Boron Mixing Calculations

The regional mixing model was used to assess the extent of boron mixing during a rapid dilution scenario for the Surry and Calvert Cliffs stations. At Surry, charging pumps can deliver 96 gpm of demineralized water from the volume control tank (VCT). The available volume for injection into the RCS is about 1500 gal. This flow is directed to one of the three cold legs via 3-inch ID piping. The charging injection line connects with a 26-inch ID cold leg at the top.

Two cases with charging flow temperature of 160°F and 450°F were considered. No loop circulation was assumed for the duration of the dilution transient.

The "mean system response" was calculated from the global energy balance. Neglecting the heat released from the walls and assuming ρ_m to be a constant we have:

$$\frac{h_m - h_{ch}}{h_{mo} - h_{ch}} = \exp\left\{\frac{-tQ_{ch}\rho_{ch}}{V_{mix}\rho_m}\right\} \tag{4.17}$$

The mixing volume, V_{mix} = 1156 ft^3, representing the volume of one cold leg, one pump, one loop seal (excluding the upstream vertical leg), lower plenum (up to the lower edge of core barrel) and the portion of downcomer below the cold leg was used in the calculations. Assuming a charging flow temperature of 160°F, h_m is decreasing from its initial value (h_{mo}) of 547 Btu/lb$_m$ to 463 Btu/lb$_m$ at the end of the dilution transient (940 s based on the capacity of the VCT and the flow rate).

This corresponds to a cooldown from 548°F to 478°F. The variation of $Fr_{ch,cl}$ is very small during this dilution transient (0.014-0.016).

The entrainment at the injection point (Q^*=3.5) and stratification (H^*_c=0.22) was obtained by superposition of plots of counter-current flow limited entrainment and jet entrainment as shown in Figure 4.4 The jet entrainment correlation based on the results of turbulence model calculations [4.1, 4.2] was used for the analysis for Surry.

After the mixing patterns were calculated, the results were converted to boron dilution using the equivalence between the dimensionless boron concentration and energy distribution. The boron concentration transients at several important locations in the system are shown in Figure 4.5 The mixed mean boron concentration, C_m, exponentially decreases from its initial value of 1500 ppm to 1193 ppm during the dilution transient. The boron concentration at the cold stream, C_c, was obtained from a boron balance for the control volume around MR1 (Equation 4.3) and by assuming the hot stream boron concentration to be equal to the mixed mean concentration. The boron concentration at the junction of the cold leg and downcomer (~2D_{cl} below the cold leg), C_j, was obtained by assuming the mixing of equal volumetric flow rates by the cold stream and surrounding hot volume fluid as discussed in Section 4.3. Finally the minimum boron concentration reaching the lower plenum, C_{min}, was obtained from the centerline planar plume decay (Figure 4.2) obtained from the results of the K-ϵ-θ turbulence model [4.1, 4.2]. The actual boron concentration in the lower plenum is higher than C_{min} because of mixing in the lower plenum. The result from Figure 4.5 shows that the lowest boron concentration in the lower plenum will be no less than 1080 ppm. This occurs at the time when the VCT would be emptied of diluted water and would start to be replenished with highly borated water from the refueling water storage tank.

Similar calculations were performed assuming the charging flow temperature to be 450°F. The results of the entrainment solution and boron concentration transients at different locations are presented in Figures 4.6 and 4.7. Higher charging flow temperature (higher Froude numbers) slightly decreases the entrainment at the injection point. However, due to a lower mass flow of the makeup water (due to the lower density) the boron concentration transients are slightly higher than for the case with lower temperature. The result is that the minimum boron concentration in the lower plenum is only 1100 ppm in this case.

Boron mixing calculations were also performed for Calvert Cliffs. Under normal boron dilution conditions at Calvert Cliffs three charging pumps are used, each delivering 44 gpm of demineralized water from the volume control tank (VCT) to two of the four cold legs via 2-inch ID piping. The charging injection lines connect with the 30-inch ID cold leg pipes at the side. The available volume for injection into the RCS is about 2900 gal. During a LOOP only two of the three charging pumps are transferred to the emergency electrical bus. Therefore, each cold leg with a charging line will receive 44 gpm. As was done with Surry, two cases with charging temperature of 160°F and 450°F were considered.

The mixing analysis for Calvert Cliffs was similar to that for Surry. However, the jet entrainment correlation used for top injection (as found in Surry) is not applicable to Calvert Cliffs because the charging injection is from the side. Using the correlation obtained by Riester et al., [4.7] for the prediction of the trajectory of horizontal buoyant submerged jets and using a simple entrainment coefficient for the horizontal part of the jet, the entrainment correlation was modified to estimate the mixing due to side injection. The results for the entrainment solution and boron concentration transients are presented in Figures 4.8-4.11. Significant mixing is predicted during the boron dilution transients and the minimum boron concentration in the lower plenum is 900 ppm with the cold water injection (Figure 4.9) and 960 ppm with the hotter water (Figure 4.11). It should be noted that the Froude number of injection (based on the injection nozzle diameter) for both Surry and Calvert Cliffs is approximately 3-7 and thus there is additional mixing due to forceful jet impingement and splashing off the opposite wall in the cold leg which is neglected here.

4.5 Summary and Conclusions

The Regional Mixing Model, which has been developed to study the thermal mixing of interest to pressurized thermal shock, was utilized to assess the extent of boron mixing in the absence of loop flow during a reactor restart scenario. Illustrative reactor predictions for Surry and Calvert Cliffs indicate significant mixing during the boron dilution transients. Indeed, for the cases considered the boron concentration in the lower plenum does not fall below 900 ppm. However, these cases do not encompass all possible physical situations for these plants. It would also be desirable to assess the applicability of the model when the temperature of the charging flow is higher, to improve the understanding of mixing when the injectant enters at the side or bottom of the cold leg piping, and to quantify the additional mixing due to jet impingement for the range of injection Froude numbers of interest to boron dilution.

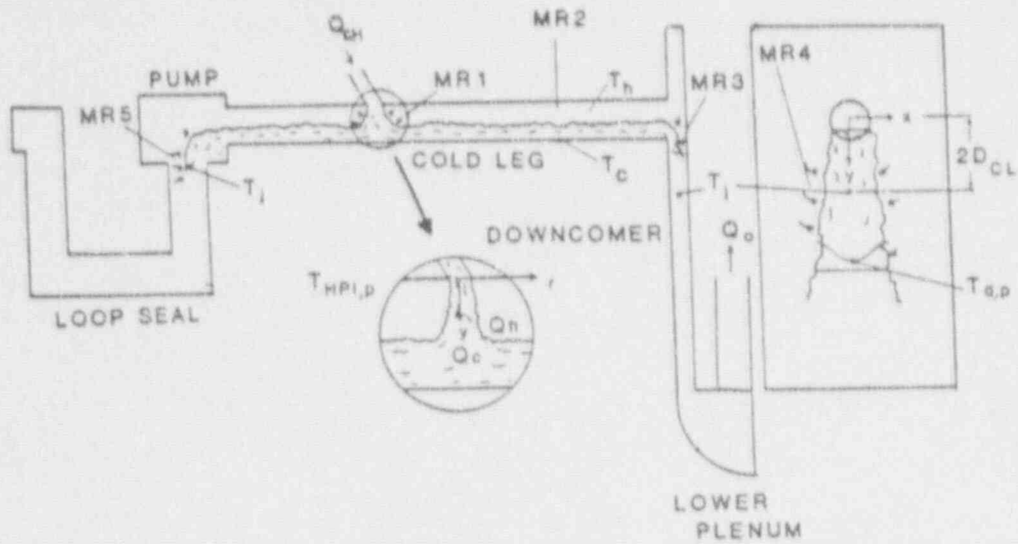

Figure 4.1 Schematic of the Flow Regime and Regional Mixing Model [4.2]

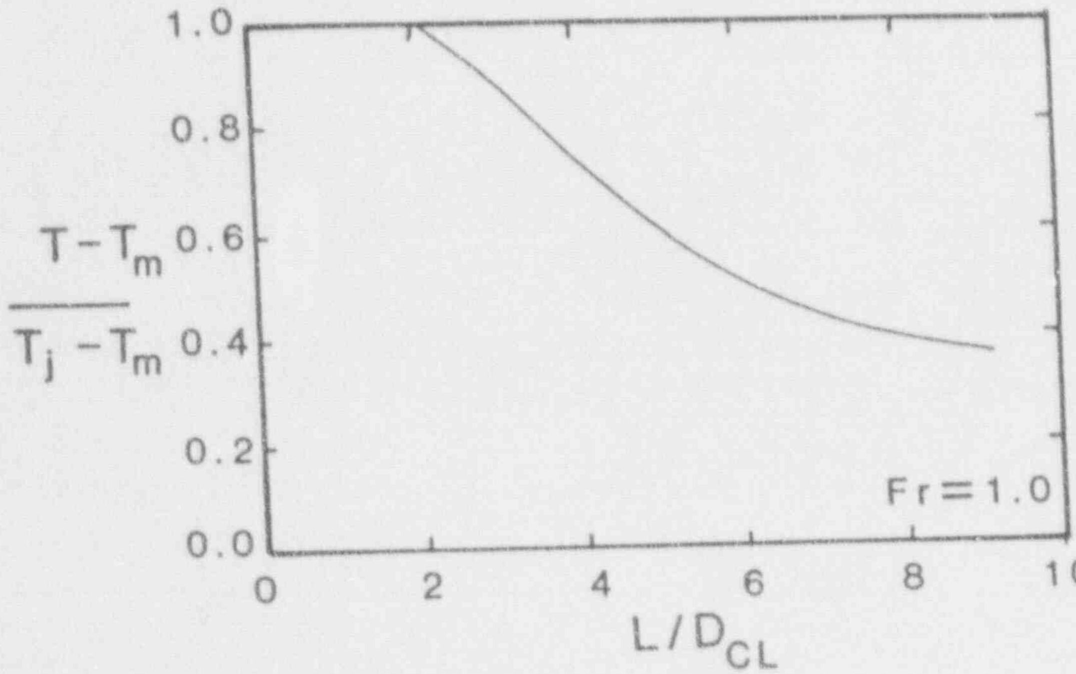

Figure 4.2 Prediction of Centerline Temperature in the Downcomer Planar Plume [4.2, 4.3]

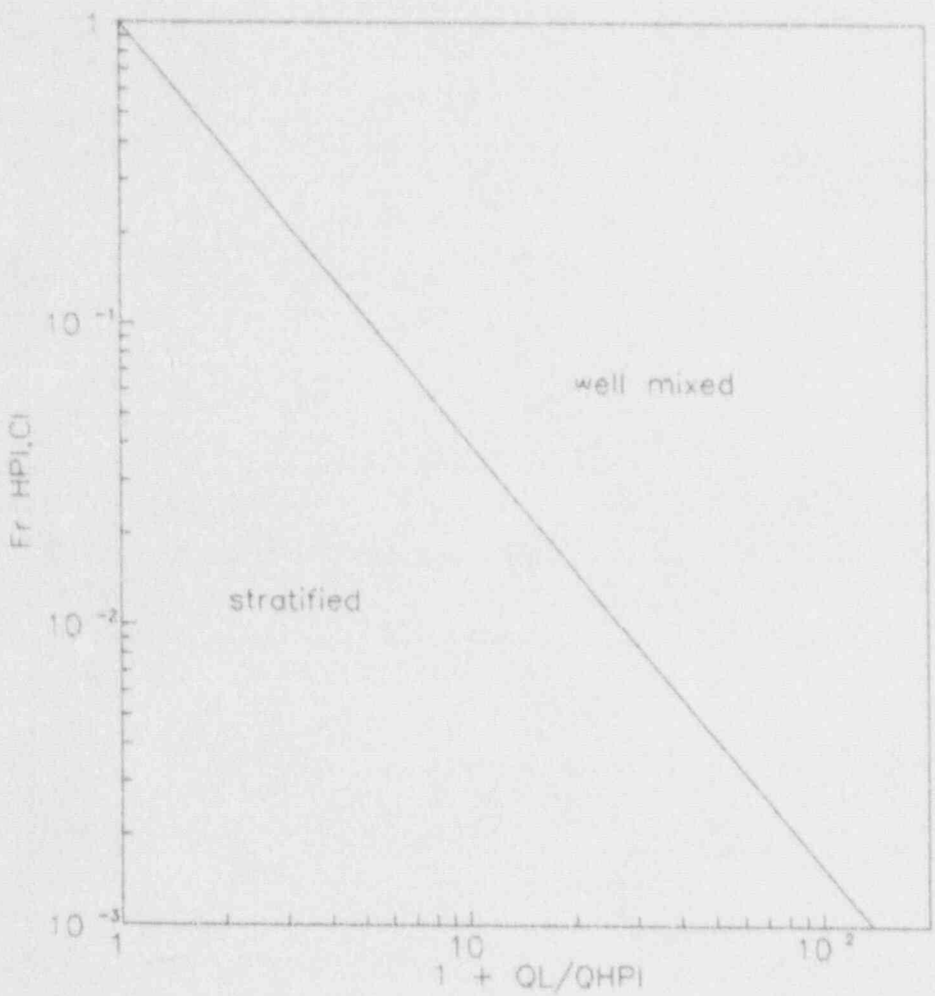

Figure 4.3 Theoretical Stratification Criterion (Equation 4.16)

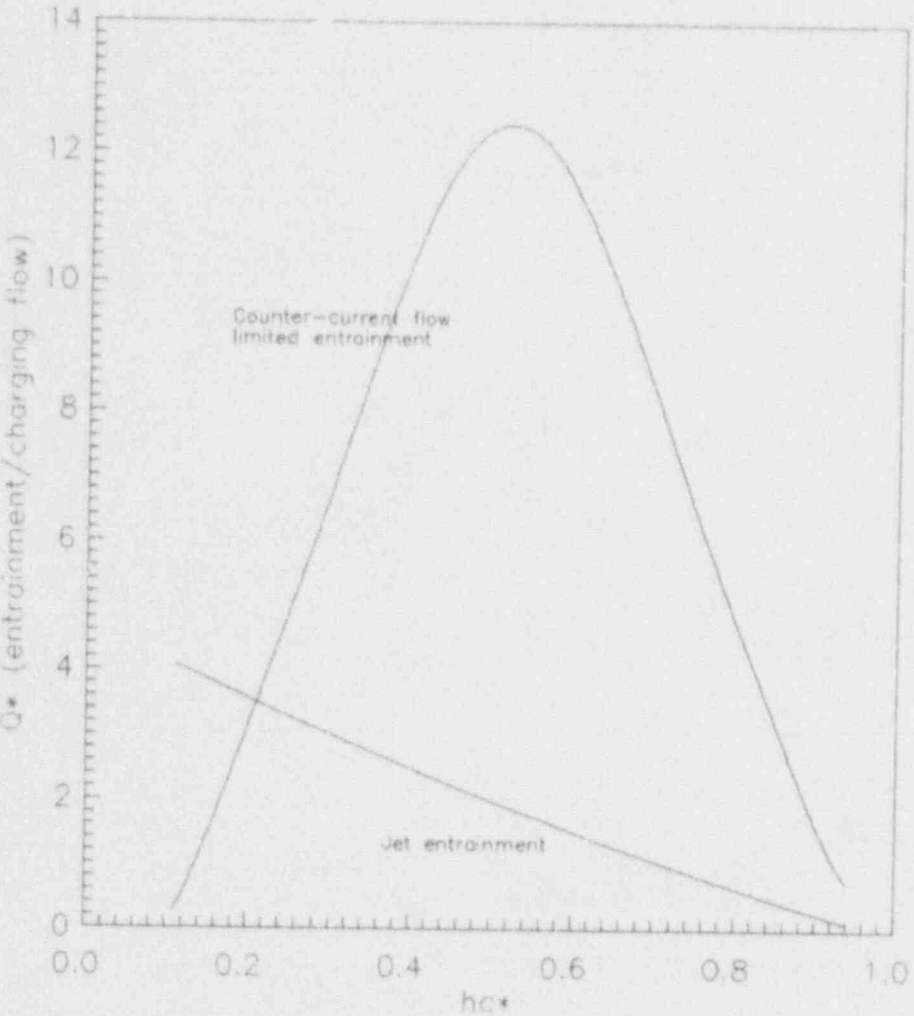

Figure 4.4 Entrainment Solution for Surry with Charging Temperature of 160°F ($Fr_{ch,cl}$=0.014,ρ^*=0.76)

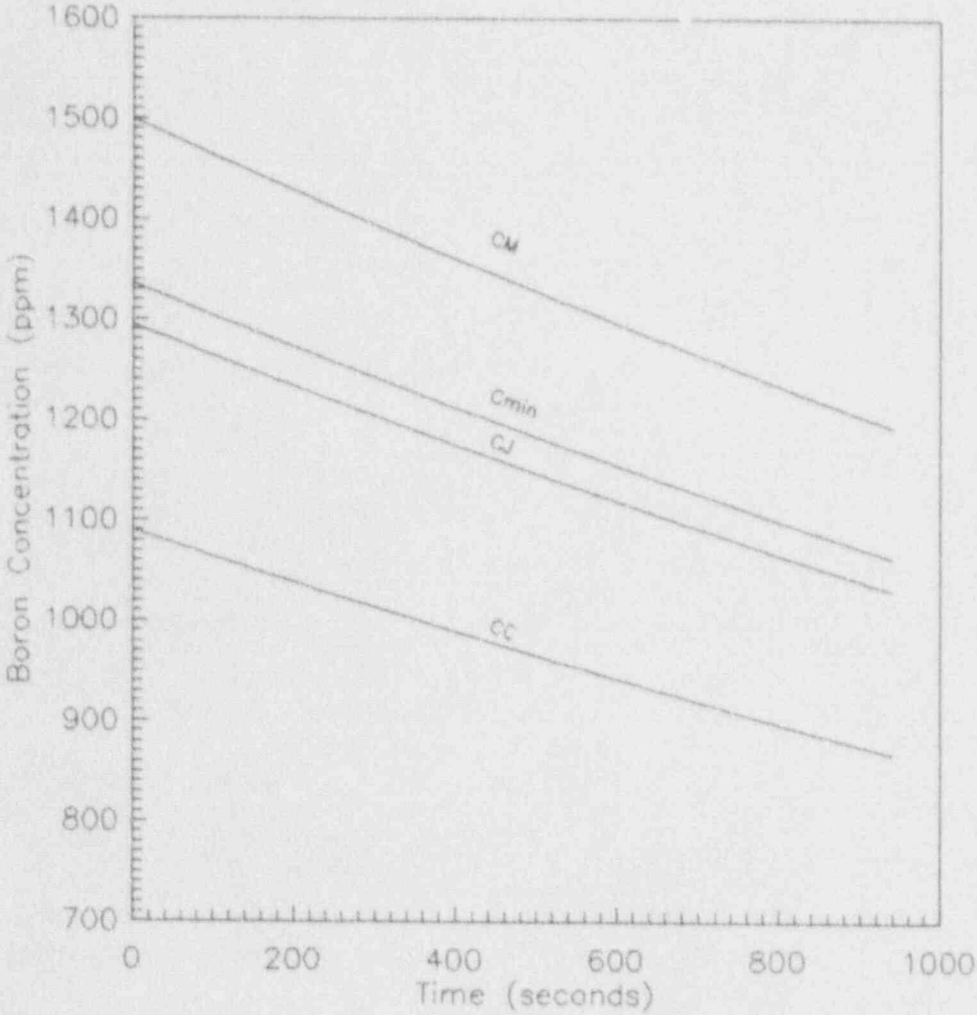

Figure 4.5 Predicted Boron Concentration Transients for Surry with Charging Temperature of 160°F

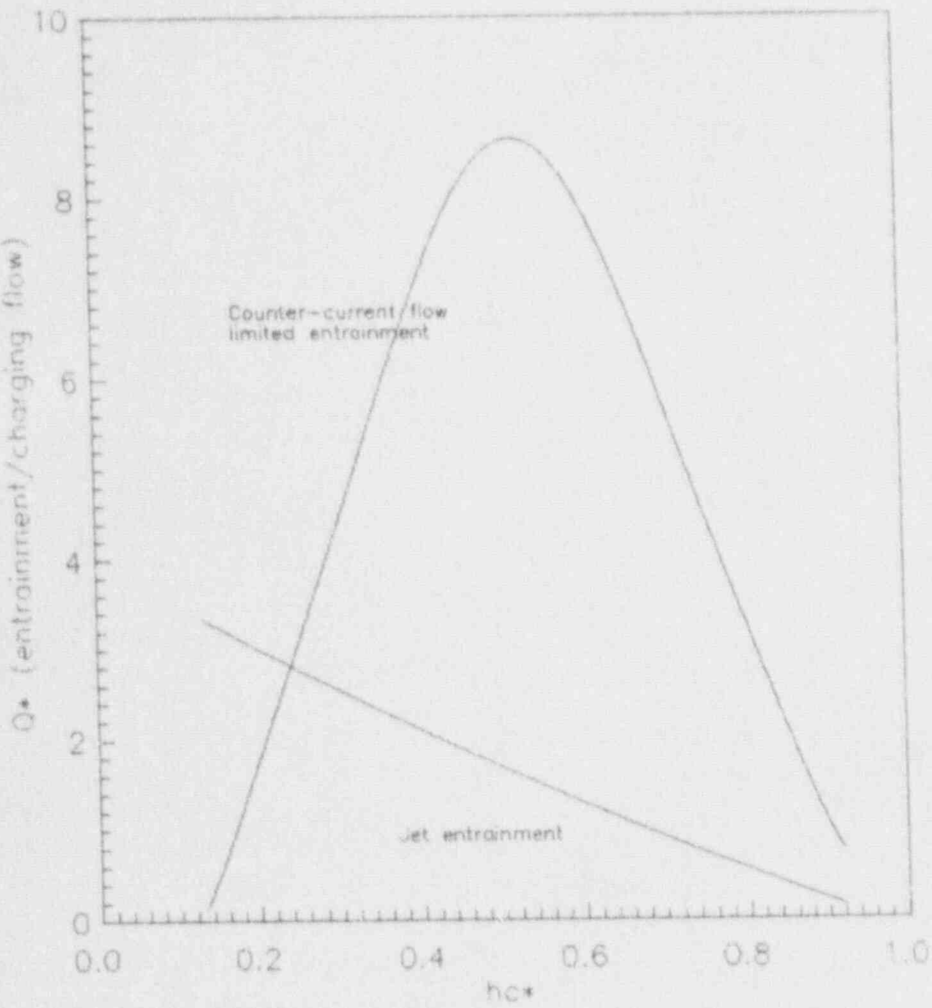

Figure 4.6 Entrainment Solution for Surry with Charging Temperature of 450°F ($Fr_{ch,cl}$=0.021,ρ^*=0.9)

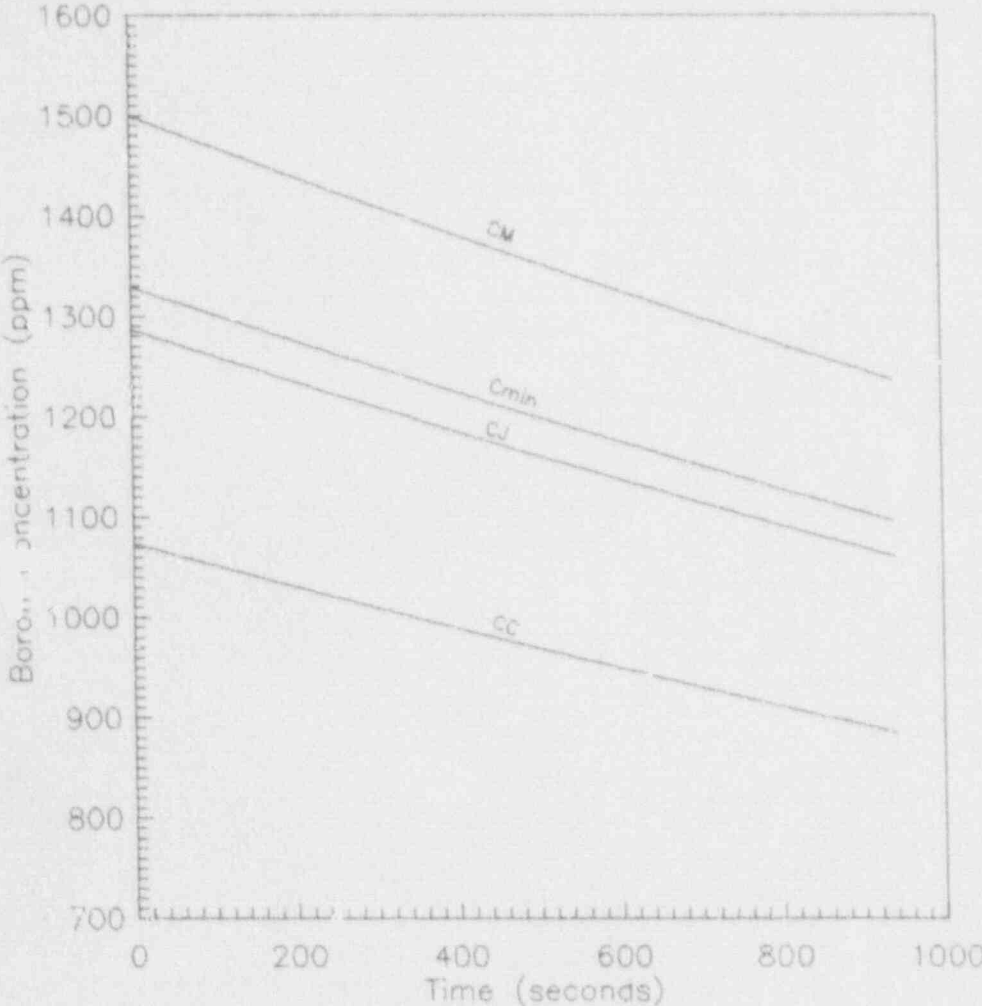

Figure 4.7 Predicted Boron Concentration Transients for Surry with Charging Temperature of 450°F

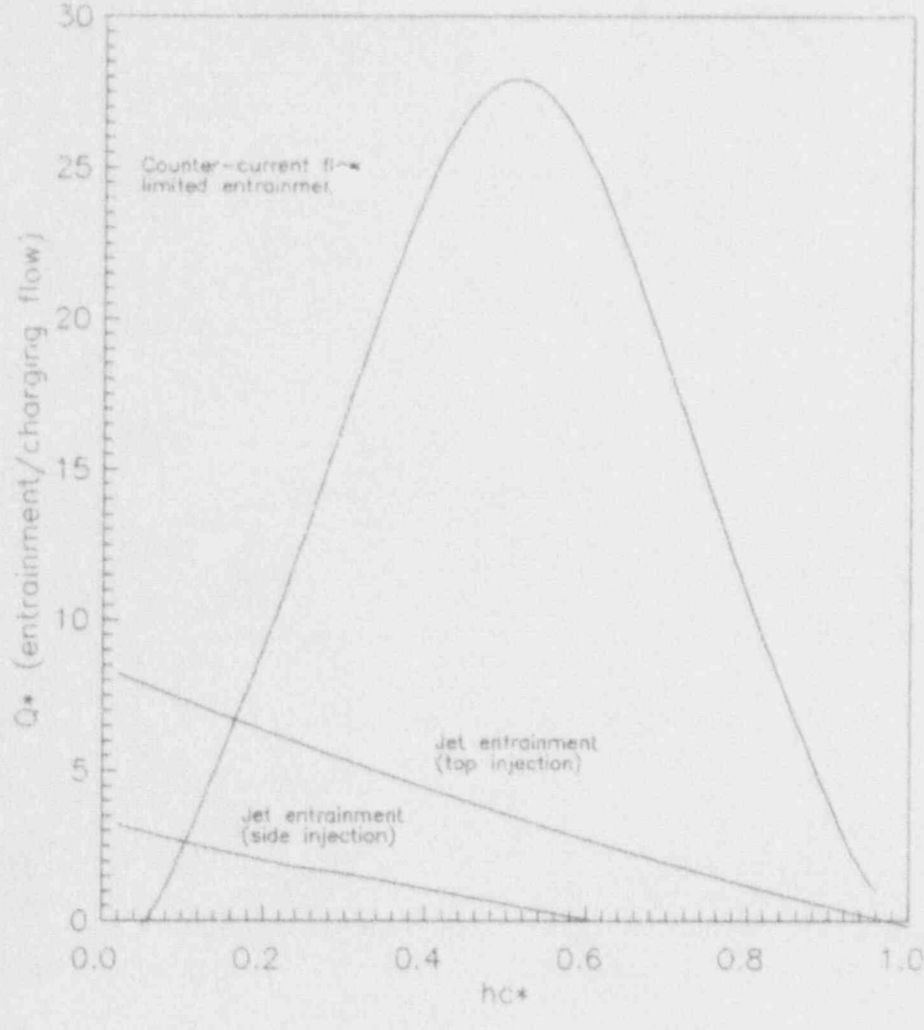

Figure 4.8 Entrainment Solution for Calvert Cliffs with Charging Temperature of 160°F($Fr_{ck,cl}$=0.0045, ρ^*=0.76)

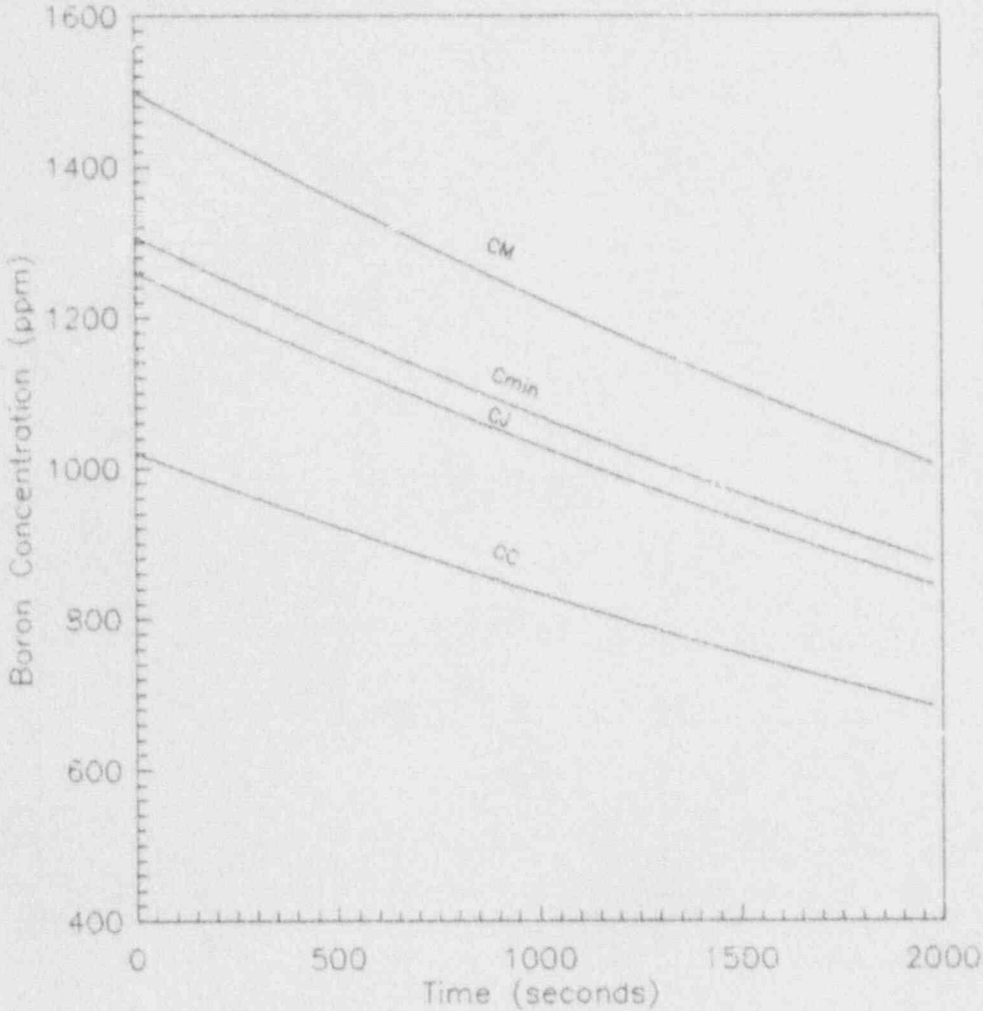

Figure 4.9 Predicted Boron Concentration Transients for Calvert Cliffs with Charging Temperature of 160°F

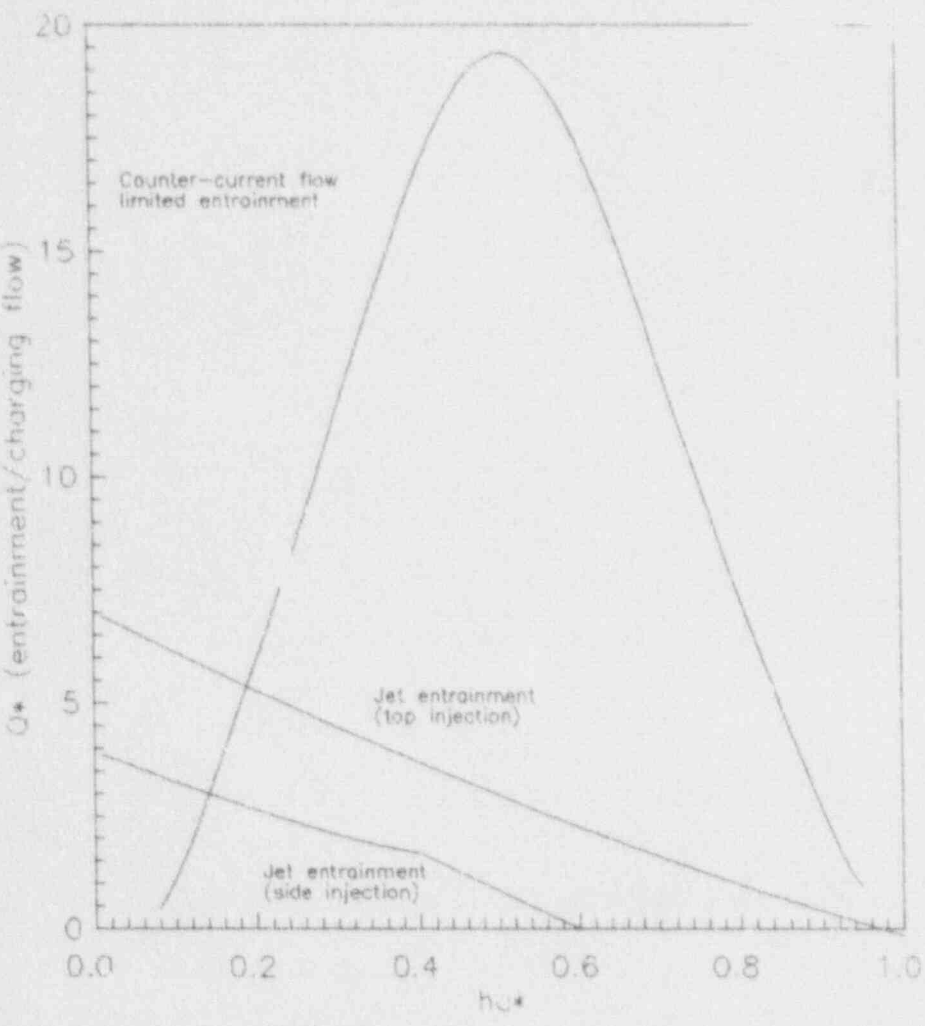

Figure 4.10 Entrainment Solution for Calvert Cliffs with Charging Temperature of 450°F ($Fr_{ch,cl}$=0.007,ρ^*=0.9)

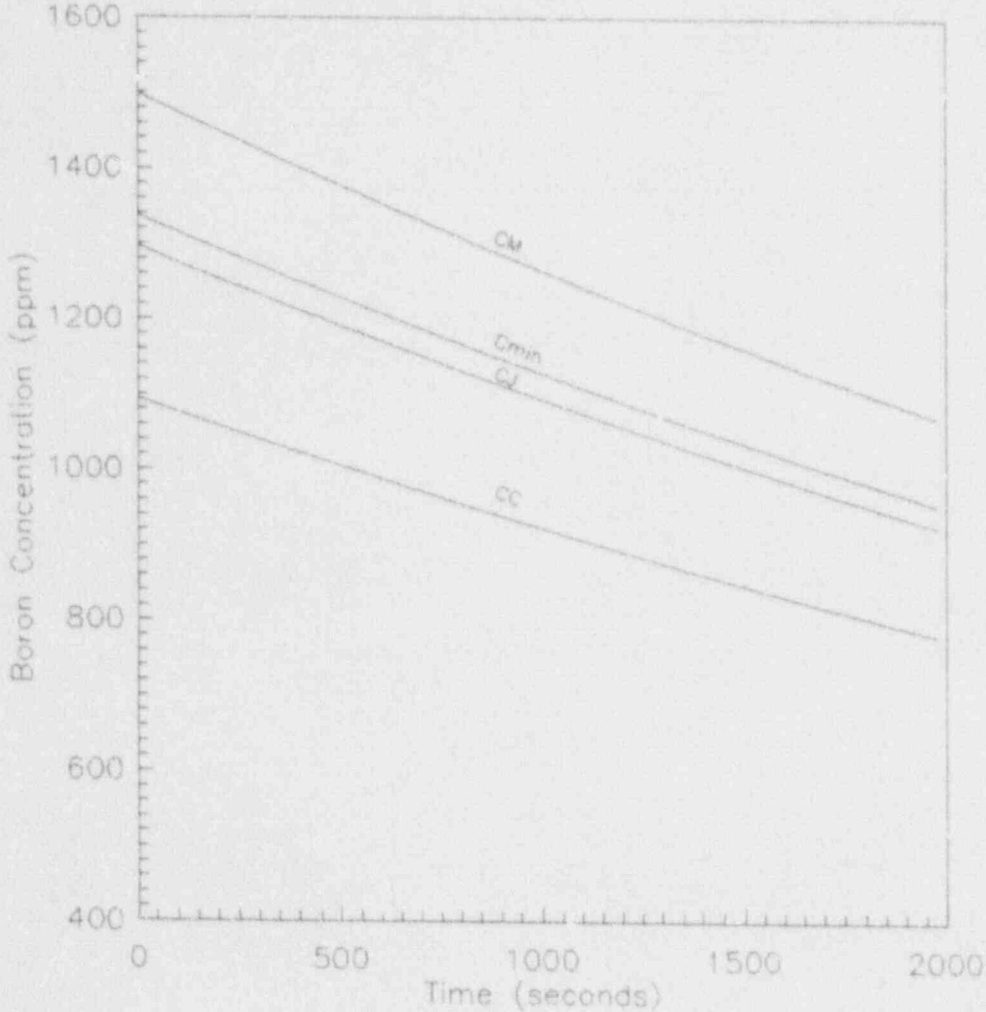

Figure 4.11 Predicted Boron Concentration Transients for Calvert Cliffs with Charging Temperature of 160°F

5 Analysis of Consequences

5.1 General Methodology

In order to rigorously calculate the response of the core to the injection of a slug of diluted water, a three-dimensional dynamic core model is needed. That model should have three-dimensional neutron kinetics and, if not three-dimensional thermal-hydraulics, then at least one-dimensional thermal-hydraulics in multiple channels. It would calculate the transport of boron throughout the core for a given distribution at the core inlet. Since this capability was not available for this project, and since resources were not available to develop a rigorous model from those partial models that do exist, the approach taken was to do a study based on an approximate synthesis method.

In this synthesis method static three-dimensional core calculations are combined with point neutron kinetics calculations to determine the power excursion. The static calculations determine the slug reactivity which is input to the power calculation. In lieu of doing detailed boron transport calculations different slug geometries and boron concentrations are assumed. The slug is assumed to enter the core uniformly across either the entire inlet area, or across only a section of the inlet. The slug reactivity is calculated as a function of the position of the slug front and a constant speed is assumed in order to translate the space dependence into a time dependence.

The neutron kinetics model, with the standard six groups of delayed neutron precursors, is combined with a heat conduction model in order to improve the accuracy of the fuel temperature calculation relative to an adiabatic model. This model also calculates the core average fuel enthalpy. At the time at which the fuel enthalpy is at a maximum the position of the slug front is noted and the corresponding static calculation is used to determine the power peaking factor. The local peak fuel enthalpy is then calculated by adding to the initial core average fuel enthalpy the increase in core average fuel enthalpy multiplied by the power peaking factor. The local peak fuel enthalpy can then be compared to the criterion for catastrophic fuel damage. This is taken to be 280 cal/g (1.2 MJ/kg) because it is equal to the approximate threshold for mechanical energy release, as determined from experiments [5.1], and because fuel fragmentation has been observed [5.2] at this fuel enthalpy. Hence, we equate catastrophic fuel damage with a change in geometry which could in turn lead to other fuel damage mechanisms.

Note that this approach does not take into account other consequences of rapid dilution events. If there is no catastrophic fuel damage there is still the possibility of release of fission products due to cladding damage either because of stresses caused at lower enthalpies or because of dryout on the surface of the clad. There is also the possibility of a pressure increase that could be excessive under shutdown conditions.

5.2 Static Core Model

The three-dimensional core calculations were carried out using NODE-P2 [5.3] This code models the neutronics with one energy group and a nodal method where k_∞ and M^2 are the basic neutronic data for each fuel assembly. NODE-P2 has successfully been applied to core performance problems by many PWR licensees. Cycle 9 of the Calvert Cliffs 2 plant [5.4] was chosen to be modeled with the code because modeling information was available from another recent Brookhaven National Laboratory study at shutdown conditions [5.5].

Analysis of Consequences

The basic neutronics data needed for each fuel assembly found in Cycle 9 were generated using CASMO [5.6]. CASMO is a mult.group, two-dimensional, transport theory code for burnup calculations of light water reactor fuel assemblies. The code has been extensively validated. The data were generated for fresh fuel containing either 0, 4, 8, or 12 burnable poison rods (containing B_4C) and for four types of burned fuel from Cycle 8, each with a different enrichment and/or number of burnable poison rods. Each burned fuel assembly was assumed to be burned t. the average exposure for that fuel type. To simplify the data generation two fuel types representing only five assemblies in Cycle 9 were no' explicitly represented. A single burned assembly from Cycle 7 was represented as one of the bundles from Cycle 8 and 4 erbia bearing demonstration assemblies were represented as fresh fuel with four burnable poison rods.

Table 5.1 gives the enrichment, number of burnable poison ro'', and burnup of each assembly type actually used. The table also indicates whether data was ge. rated with and/or without control element assemblies (CEAs) present. There are two types of CEAs. ach axially zoned differently with rods ..ntaining B_4C, Ag-In-Cd. Al_2O_3, or stainless steel. The assumption was made that all CEAs were identical and contained B_4C rods since more than 85% of the rods are of this composition. Part length assemblies were neglected. For each assembly the data were generated at boron concentrations of 1500, 750, and 0 ppm. At 750 ppm 'he data were generated at the base fuel temperature of 548°F (560 K) and at an elevated temperature of 1200°F (922 K). The moderator temperature was 548°F in all cases.

The core layout for Cycle 9 as modeled is shown in Figure 5.1. There is octant symmetry. In addition to the fuel type, the location of CEAs are noted, with those that are in the shutdown banks and regulating banks identified separately.

Table 5.1
Fuel Assemblies for NODE-P2 Model

Fuel Type	Enrichment w/o	Burnable Poison Rods	Burnup MWd/t	Control Rods
J	4.05	0	30	Yes/No
K	4.08	0	15	No
K/	4.08	8	21	Yes/No
K*	4.08	12	21	No
L	4.30	0	0	Yes/No
LX	4.30	4	0	Yes
L/	4.30	8	0	Yes
L*	4.30	12	0	Yes/No

5.3 Static Calculations

5.3.1 Base Calculations

The NODE-P2 core model with the shutdown banks removed and the boron concentration at 1500 ppm is meant to represent the core during the period of deboration when the final boron concentration has been reached. (The hot, full power (HFP) critical boron concentration with all rods out and equilibrium xenon is 1460 ppm [5.4]; without xenon it would be higher.) At this point the CEAs belonging to the regulating banks would be removed to achieve criticality and then to increase the power to operating conditions. The calculated k_{eff} using NODE-P2 is 1.0083 which is less than 1% too high and within the expected uncertainty. Changes in reactivity are relative to this value.

The worth of the shutdown bank at these conditions is calculated by NODE-P2 to be 4.6%. This is less than the value of 5.9% calculated independently at beginning-of-life, but at full power conditions [5.7], and greater than the worth expected at zero power conditions based on measurements for some of the shutdown banks [5.8].

The worth of a core-wide boron dilution over the range from 1500 to 750 ppm is calculated by NODE-P2 to be 7.8 pcm/ppm. The boron reactivity coefficient quoted for full power conditions is 8.2 pcm/ppm [5.4]. The difference between these boron worths has the correct trend as the removal of control rods at the full power condition is expected to result in a higher value for the coefficient.

At a boron concentration of 1500 ppm a core-average increase in fuel temperature (with the spatial distribution determined by power) from 548°F to 1200°F was calculated by NODE-P2 to give a Doppler coefficient of -2.6 pcm/°F. Again the only comparison that can easily be made is with the coefficient calculated for full power operation. At full power the coefficient is expected to range from -1.0 to -2.4 pcm/°F [5.4]. The magnitude of the Doppler coefficient is expected to be higher at zero power.

5.3.2 Pseudo Time-Dependent Calculations

The worth of a slug of diluted water was calculated by assuming a particular geometry and then doing a sequence of calculations representing the slug as it moved through the core. In these calculations all CEAs were inserted. The 9 cases considered are given in Table 5.2 which lists the change in boron concentration (relative to 1500 ppm) and the slug geometry.

Table 5.2
Diluted Slug Conditions

Case	Change in Boron Concentration, ppm	Slug Geometry
1	600	Semi-infinite step
2	750	Semi-infinite step
3	1000	Semi-infinite step
4	1500	Semi-infinite step
5	750	535 ft^3 rectangular step
6	750	535 ft^3 trapazoidal step
7	750	Semi-infinite step - core center
8	1000	Semi-infinite step - core center
9	1000	Semi-infinite step - core edge

In Cases 1-4 the dilution is assumed to occur uniformly across the core and the change in boron concentration is represented as a sharp wave front (step). The reactivity effect of the dilution in these cases is shown in Figure 5.2 as a function of the position of the front of the dilution boundary. Each node corresponds to almost one foot so that when Node 12 is reached the entire core (136.7 in length) is diluted to the new boron concentration. The curves follow an 'ess' shape with the most rapid changes in reactivity occurring when the slug front is moving through the bottom of the core. (Note that this is the complement to the problem of control rods being worth relatively little until they move *past* the midway point.) An additional non-linearity is the effect of the degree of boron dilution with the reactivity of a 1500 ppm change being more than twice that for a 750 ppm change. As expected, Figure 5.2 shows that the magnitude of the slug reactivity can be very large if the dilution is large.

In Cases 5 and 6 the slug is not semi-infinite but rather corresponds to a fixed volume of 535 ft^3. This corresponds to 2000 gal of water available from the volume control tank (VCT) multiplied by two to account for an amount of VCT water (with an assumed concentration of 0 ppm) mixing with an equal amount of water from the reactor coolant system (RCS) (with an assumed concentration of 1500 ppm) to create the slug of diluted water at a concentration of 750 ppm. The reactivity change for these two cases is shown in Figure 5.3. The rectangular case starts off identical to the semi-infinite step shown in Figure 5.2 and reaches approximately the same peak reactivity since the length of the slug is almost equal to (actually 10/12) the core height. As the wave front approaches the top of the core the boron concentration in the bottom of the core changes back to 1500 ppm and the total reactivity begins to decrease. The volume of the slug is the same in the trapazoidal case but rather than a jump change in concentration it changes over a distance of 5 ft. The total length of the slug in this case is almost 15 ft. As expected, the effect of changing the geometry is a slower rise and a delay in returning to the initial condition.

In Figure 5.4 the effect of different radial geometries is shown by plotting Cases 2, 7, 8, and 9. In Case 7 the dilution is at the center and affects a 7x7 array of fuel assemblies. Since this is only 49/217 (=0.23) of the core the effect of a 750 ppm dilution is less than that obtained when the dilution is uniform (Case 2). This can be seen on Figure 5.4. However, since the slug is in the center of the core, where the neutron importance is relatively high, the effect is greater than 0.23 of that obtained in Case 2. Also shown on the figure is the effect of increasing the dilution in the center from 750 ppm to 1000 ppm (i.e., reducing the boron concentration to 500 ppm). With the larger dilution the figure shows that the reactivity effect is almost equal to that achieved with a uniform dilution.

For the same 1000 ppm dilution Figure 5.4 also shows the effect if only assemblies on the core edge are affected. For Case 9 there are 25 assemblies diluted at either end of one of the core axes. This is shown on Figure 5.5 (which also shows the pattern when the dilution was at the core center). The total of 50 assemblies affected is almost equal to the 49 affected in the center of the core but since the neutron importance at the core periphery is less than at the center, the reactivity effect is much smaller.

5.4 Dynamic Core Model

The dynamic core model consists of the point neutron kinetics equations, including six groups of delayed neutron precursors, and a simple thermal-hydraulic model to obtain the core average fuel enthalpy. The thermal-hydraulic model consists of equations for the average pellet, clad, and coolant temperature. A gap heat conductance (as a function of fuel temperature) is used and a fixed heat transfer coefficient for the clad is determined to yield the proper initial conditions. The model was solved with the DESIRE software package [5.9] on a personal computer.

Boron dilution reactivity as a function of time was taken from the static calculations described in Section 5.3.2 by assuming a constant speed for the slug front. This was taken to be 2.0 ft/s which corresponds to 13% of rated flow. This is an approximation to the flow which would increase from close to zero (assuming little natural circulation) to 20% of full flow in about 20 seconds.

A constant negative reactivity representing the initial shutdown margin was used. This is meant to account for the worth of the shutdown banks in the reactor startup scenario and in general for any other contribution to shutdown margin that might be present before the dilution begins. In the present calculation -4.0% shutdown is assumed to be the base initial condition. In Calvert cliffs this is approximately equal to the shutdown bank worth but in other plants the shutdown worth might be smaller.

Fuel temperature reactivity was calculated during the dynamic simulation using the core average fuel temperature calculated by the model and a Doppler feedback coefficient expressed per unit change in square root of absolute temperature. The Doppler feedback is strongest in the region where the fuel temperature is highest and since the power, and hence the neutron importance, is also highest in this region, it is a gross inaccuracy to represent the Doppler feedback using a core-average fuel temperature. To improve upon the accuracy of the fuel temperature feedback a Doppler weighting factor (DWF) was applied to the reactivity calculated using the core average fuel temperature.

The DWF is defined as the actual change in fuel temperature reactivity during a dilution divided by the change which would occur with the same average temperature change uniformly distributed over the core (or as assumed when generating a core average Doppler coefficient). In practice it is approximated as the change in reactivity if the fuel temperature is increased according to the power distribution expected during the dilution divided by the change in reactivity for the same average temperature change with the power distribution at the initial condition. Using this definition, the DWF is calculated by doing additional static calculations at elevated power (and thus temperature), at each of the slug positions used to determine the boron dilution reactivity. The change in reactivity for this change in temperature at this slug position is then divided by the change in reactivity when the temperature change is made prior to the slug moving into the core. It is then input to the dynamic calculation as a function of time just as the slug reactivity is input.

Some of the nominal initial conditions used for the dynamic calculations are given in Table 5.3. The delayed neutron fraction represents a midpoint in the range of 0.0044 to 0.0070 expected for the cycle [5.4].

The dynamic core model calculates the core average enthalpy. At the time at which this is a maximum the power peaking factor from the static calculation corresponding to this time is extracted and used to correct the enthalpy so that the final result is the local peak enthalpy.

Table 5.3
Nominal Initial Core Conditions

Power (Decay Heat)	10 MW
Fuel Temperature	548°F
Coolant Temperature	548°F
Inlet Flowrate	13% of Rated, 2.0 ft/s
Delayed Neutron Fraction	0.0056
Shutdown Margin	4.0%
Doppler Coefficient	-2.6 pcm/°F

5.5 Dynamic Calculations

The dynamic calculations were carried out to determine the peak fuel enthalpy during a rapid dilution event in order to know if catastrophic fuel damage could occur. No attempt was made to calculate other effects such as the extent of fuel melting in the center of the pellet, clad temperatures if boiling transition is observed, or the pressure rise in the system. A more rigorous calculational model would be necessary to observe this behavior and to understand how the core would respond other than through catastrophic fuel damage.

The power response for the base case described in Section 3.4 is shown in Figure 5.6. In this case a 535 ft^3 slug of diluted water at a boron concentration of 750 ppm passes through the core at a speed of 2.0 ft/s. The power trace is typical for a prompt-critical reactivity excursion in a PWR. When the reactivity addition due to the diluted water exceeds the assumed shutdown margin by the delayed neutron fraction the power rises rapidly. In this case, with an assumed shutdown margin of 4% it can be seen from Figure 5.3 that when the slug-front is between Node 5 and 6 this condition is satisfied. Hence, at approximately 3 seconds into the transient the power rises rapidly until the almost instantaneous fuel temperature response (typical of a PWR) causes sufficient negative feedback to terminate the initial power rise. Although the peak power at 75 GW is very high, what is important in determining the fuel response is the integral of power, i.e., the energy that is deposited in the fuel.

The energy deposition causes an increase in fuel temperature and enthalpy. The radial pellet average enthalpy is the quantity used to determine whether or not catastrophic fuel damage has occurred (cf Section 5.1). The core average fuel enthalpy for this case is shown in Figure 5.7. In order to determine the peak enthalpy in the core a power peaking factor obtained from the steady state calculation is applied as explained in Section 5.1. The power peaking factor is 6.3 and this means that the initial power rise corresponds to a peak fuel enthalpy of 69 cal/g which is much less than the criterion for fuel damage. The power peaking factor is large because the slug is only half-way into the core at this time.

After the initial power rise the power decreases but then, as can be seen on Figure 5.6, the power rises again before it slowly starts to decrease after 5-6 seconds. The increase comes from the fact that reactivity is continuing to be added until that time period. The decrease in power comes from the decrease in slug reactivity but the decrease in core average fuel enthalpy comes primarily from the fact that there is heat transfer from the fuel into the coolant and this becomes appreciable in the period after 6 seconds. Note that when there is a large amount of energy transferred into the coolant the model is no longer applicable. Two-phase flow would be expected and significant negative feedback from the coolant would affect the response of the fuel rods. Although this was not modeled in this calculation the peak fuel enthalpy was calculated during the period of 5-6 seconds when the core average fuel enthalpy reaches a peak. The value was 102 cal/g which again is an indication that no catastrophic fuel damage would occur.

These results will be most sensitive to the initial shutdown margin, the Doppler feedback, and the properties of the slug. Other factors which will have a secondary impact are the delayed neutron fraction, the neutron lifetime, and the speed at which the slug moves through the core.

In order to quantify the effect of initial shutdown margin, several additional calculations were completed. The initial shutdown margin represents what the condition of the core might be by virtue of the operating mode combined with the effect of any control rods that might insert prior to the slug passing through the core. In the reactor restart scenario even if the core was initially at its final boron concentration there would still be some negative reactivity as the reactor is usually brought to critical on the movement of the regulating banks. Assuming that the reactivity hold-down of the regulating banks is small, then the initial shutdown margin is just the worth of the shutdown banks which would scram when there was a loss of off-site power. Although this was assumed to be 4% in the base case, for some plants this migh only be 2%. If there were no rods initially withdrawn then the smallest shutdown margin that could be expected would be the 1% requirement when at hot shutdown conditions. Hence, the calculations were done with an initial shutdown margin down to 1%. At the other end of the scale is the fact that if the shutdown margin were equal to or greater than the amount of reactivity which could be inserted by the diluted slug then no power excursion can take place. In the case being considered with a 750 ppm slug, this is 5.9%.

The results of these calculations are shown in Figure 5.8. Peak fuel enthalpy is plotted for the time corresponding to immediately after the initial power spike and for the time at which the core average enthalpy exhibits a broad maximum (cf Figure 5.7). The times at which these peaks occur become later with an increase in initial shutdown margin as it takes longer to overcome that barrier and become supercritical. As can be seen, when the initial shutdown margin is between 1% and 2% the peak fuel enthalpy can exceed the 280 cal/g criterion for catastrophic fuel damage. This is not the result of the initial power burst but rather the fact that the power remains high due to the continued presence of the diluted slug.

Figure 5.8 also shows the effect of reducing the Doppler feedback by a factor of 0.5. As explained in Section 5.3.1 the reduction in the Doppler coefficient from -2.6/°F to -1.3/°F is consistent with the range of coefficients expected during operation of this cycle of Calvert Cliffs and similar to how other PWRs operate.

An estimate of the boron dilution necessary to cause the fuel enthalpy to exceed 280 cal/g when the initial shutdown margin is 4% can be made by using the results shown in Figure 5.8. Since that figure shows that a reduction of the shutdown margin to approximately 1.5% would cause the peak fuel enthalpy to exceed 280 cal/g it can be estimated that at an initial shutdown margin of 4% one would need additional reactivity worth 2.5%, or approximately an additional dilution of 320 ppm. Hence, if the boron concentration of the slug was approximately 430 ppm and the initial shutdown margin was 4%, catastrophic fuel damage might be likely. Note that with this boron concentration the volume of the diluted slug would be 1.4 times the volume of the unborated water assumed to be available from the VCT or 375 ft^3. The length of the slug would then be approximately 7 ft rather than the 10 ft for the nominal case. This is not expected to alter the behavior of the power excursion during the period where the fuel enthalpy reaches a maximum, however, if the slug was even smaller and more dilute it is not clear that the situation would lead to a higher fuel enthalpy. The competition between a higher positive reactivity insertion and a shorter insertion time in the limit of zero volume leads to a smaller effect.

If the slug were to be located in the center of the core as shown for Cases 7 and 8 in Section 5.3.2 then there are several factors which change the response of the core. It takes a larger dilution to cause the same power excursion with a uniform radial distribution as can be inferred from the curves in Figure 5.4. However, for a given power excursion the peak fuel enthalpy would be higher with the slug localized in the core center because the power peaking factor might be higher. For the cases considered herein the increase in power peaking is approximately 40%. If the situation of concern is a highly dilute slug (with perhaps a high initial shutdown margin) then the power excursion with the slug in the core center will be different from the case with a uniform radial distribution because the smaller inlet area for the slug means that it will be in the core longer. Hence, the severity of the excursion will be a complicated function of slug dilution, geometry, and volume as well as initial shutdown margin and Doppler feedback.

In the above calculations the reactivity feedback from the reactor coolant was neglected. After power rises, the increase in coolant temperature and the possibility of steam voids would tend to reduce the power after sufficient time has elapsed for significant heat transfer. Another, potentially more important effect of coolant temperature occurs at the outset of the event if the diluted slug is significantly cooler than the initial water in the core. This may be the case as one of the important mechanisms for not having mixing of any diluted water entering the system is the relatively high density of the injectant if it is colder than the water in the RCS. The reactivity addition due to the temperature of the slug could be significant, especially at the end of a fuel cycle when the moderator temperature coefficient (MTC) has its largest magnitude. With the same conditions used to generate Figure 5.8 (with shutdown margin of 4%) the peak fuel enthalpy would exceed 280 cal/g if the MTC was greater in magnitude than -20 pcm/°F. This is based on a slug temperature obtained by mixing equal amounts of unborated water at 298°F with 1500 ppm RCS water at 548°F.

The interpretation of the effect of cooler slug temperatures is complicated by the fact that the above neutronics calculations are for a fixed boron density (g/cm^3) and the boron concentrations (ppm) quoted above assume a slug temperature equal to the initial temperature (548°F) in the core. If the coolant density is lower then the boron concentration quoted must be lower. For example, the boron concentration would have to be reported as 675 ppm at 298°F rather than 750 ppm in order to have an equivalent density.

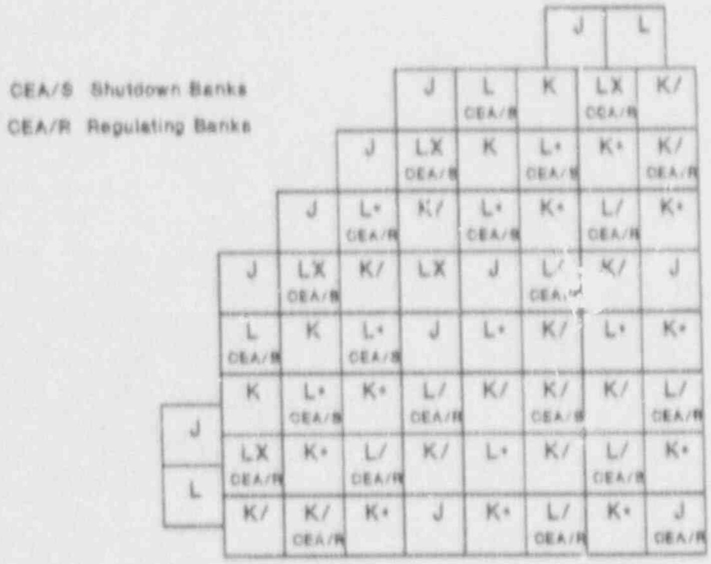

Figure 5.1 Distribution of Fuel Types and CEAs

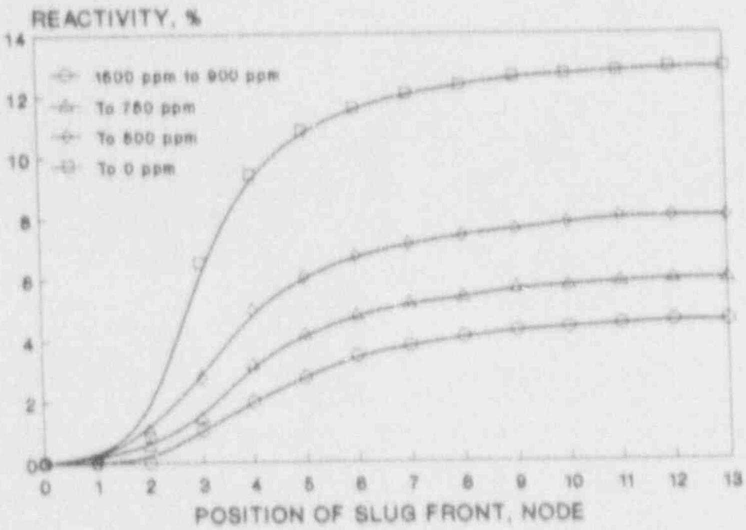

Figure 5.2 Reactivity Effect of Diluted Slug

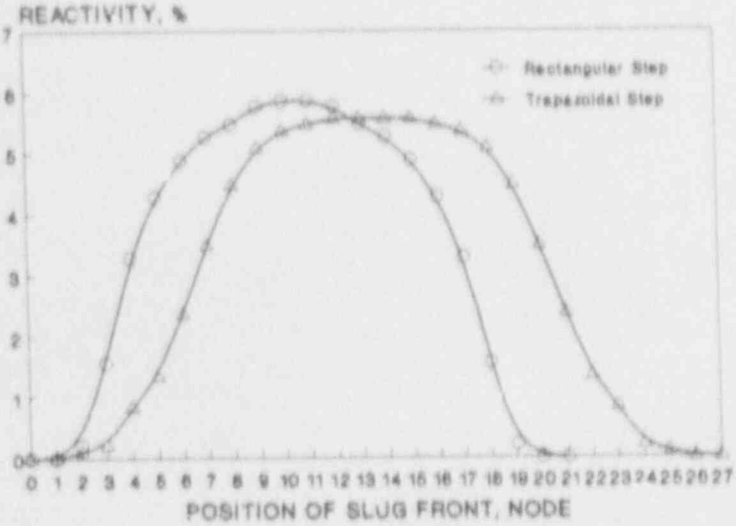

Figure 5.3 Reactivity Effect of Diluted Slug

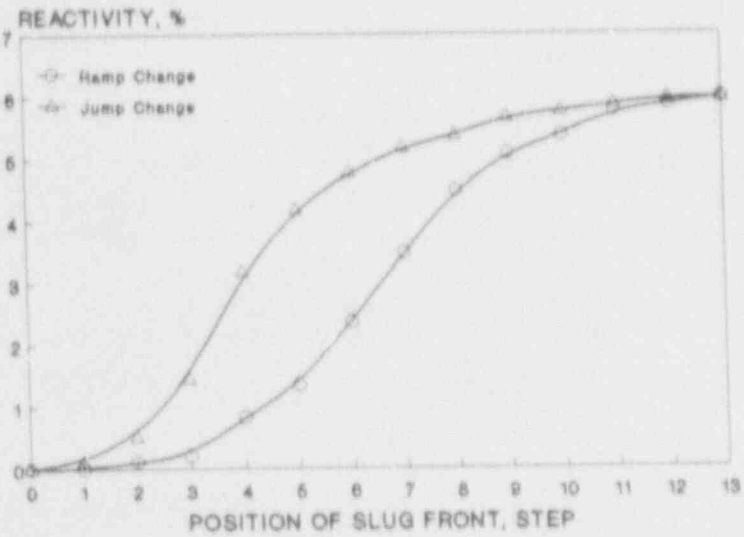

Figure 5.4 Reactivity Effect of Diluted Slug

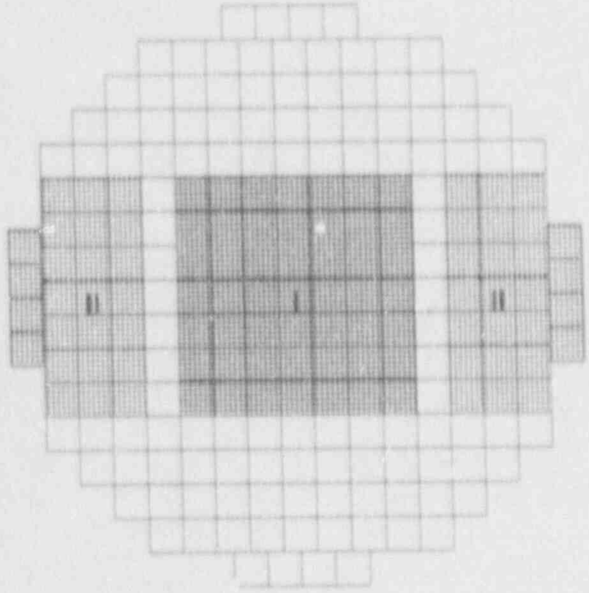

Figure 5.5 Core Center (I) and Core Edge (II) Patterns

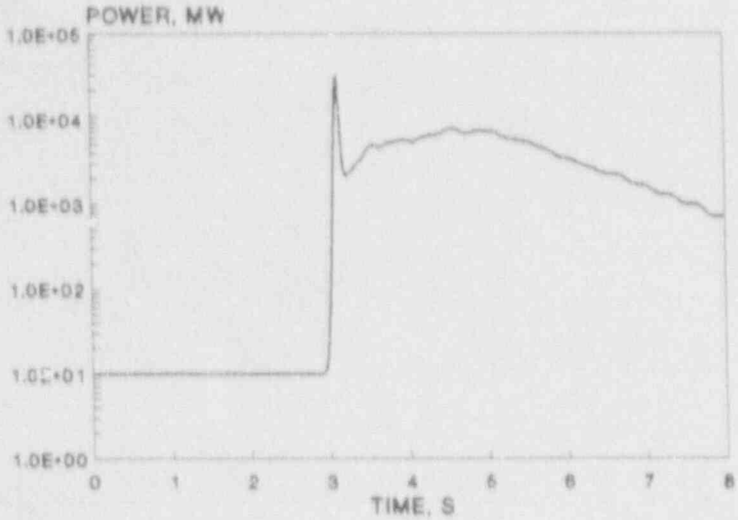

Figure 5.6 Core Power with Finite Slug

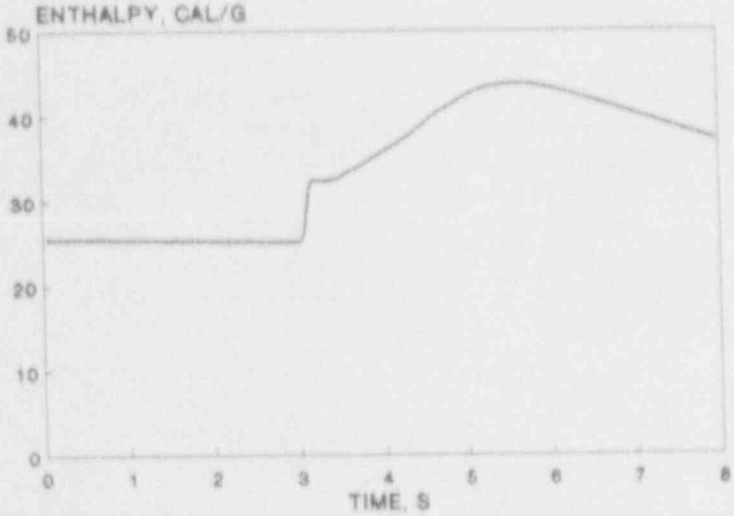

Figure 5.7 Core Average Fuel Enthalpy

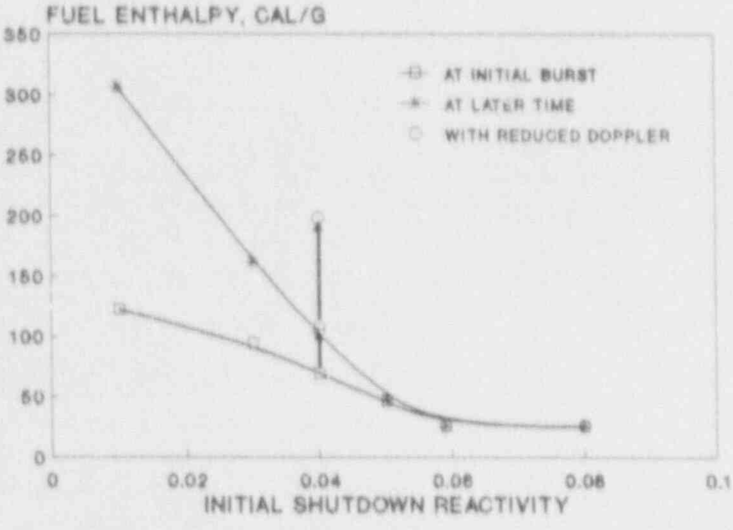

Figure 5.8 Peak Fuel Enthalpy vs Shutdown Margin

6 Summary of Results and Conclusions

The analysis of a rapid boron dilution event has been carried out in three different ways: 1) a probabilistic analysis for the core damage frequency, 2) a mixing analysis to determine the extent of dilution before injection into the core, and 3) neutronics calculations to determine core behavior and the consequences in terms of fuel damage.

The probabilistic analysis was done for reactors from each of the three U.S. reactor vendors. The CDF varies from 9.7E-6/yr to 2.8E-5/yr for the three plants which is what is calculated for other internal events that are considered to be important. It is, however, less than that calculated for the same event in a European reactor. One of the reasons for this is that the initiating frequency is lower in the present analysis. No other conclusions can be reached on the differences because the details of the European analysis were not available for this study.

The analysis shows the importance of the primary grade water pump. For all three plants the potential for an accident is limited by the amount of diluted water in the VCT as the supply of primary grade water is stopped by the trip of the PG makeup pump. However, there are plants where this pump is connected to the emergency bus and the probability of an accident will be increased if primary grade water continues to be pumped into the VCT. This appears to be the case for some plants in France and Sweden and is another reason why the problem may be more serious there. The question of whether the makeup pump trips or continues to run has to be evaluated on a unit by unit basis.

For all three plants the dilution is done with flow from the VCT. It should be noted that in some plants the suction for the charging flow comes directly from the primary grade makeup water source and once the PG water pump is tripped there is no longer the potential for adding unborated water to the RCS.

The results are dependent on assumptions used in the analysis which are summarized below. Note that the assumptions result in overestimating the core damage frequency. More detailed analysis would have to be carried out to quantify the effect of these assumptions.

1. The dilution time during startup is 8 hrs. The consequences of the event are assumed independent of when during this period the loss of off-site power occurs. In reality, an event occurring early during this period will have more shutdown margin to overcome and is, therefore, expected to have less of an effect than an event occurring near the end of the normal dilution procedure.

2. No credit is given for the operator to take action and stop the charging flow from the VCT after the LOOP. Although dilution while the shutdown banks are inserted or the RCPs are stopped, (as would be the case after a LOOP) is not a normal procedure, it is assumed that since the operator knows that the flow from the primary water makeup pump has ceased, that no other action is deemed warranted. An action that could be taken by the operator would be to switch the charging pump suction to the RWST.

3. The analysis does not account for any mixing at the point of injection or mixing of the diluted water due to its flow into the downcomer and down to the lower plenum. It does, however, account for the effect of natural circulation on mixing. The assumption is made that if it is a startup without refueling then there is sufficient decay heat and natural circulation to mix the injectant to the extent that the probability of causing core damage is reduced by a factor of 0.5. Mixing analysis performed for two of the plants shows that this may be very conservative under certain conditions.
4. Another important assumption is that if off-site power, or another adequate power source, is available, the reactor coolant pumps (RCPs) will be started during a 30-minute interval.

The mixing analysis was done with the Regional Mixing Model developed to study the thermal mixing of interest to pressurized thermal shock. Illustrative reactor predictions for a Westinghouse and a Babcock & Wilcox plant show significant mixing during the boron dilution transients. For the cases considered, the boron concentration in the lower plenum does not fall below 900 ppm when the initial boron concentration in the vessel is 1500 ppm. However, these cases do not encompass all possible physical situations for these plants. It would be desirable to assess the extent of mixing when the injectant enters at the side or bottom of the cold leg piping, as is the case in some plants, and to quantify the additional mixing dee to jet impingement for the range of injection conditions of interest.

The neutronic results show that there is the possibility of catastrophic fuel damage depending on a) the initial shutdown margin, b) the Doppler feedback, and c) the properties of the slug, especially the boron concentration. The initial shutdown margin depends on the reactor state at the time of initiation of the event and the reactivity worth of the shutdown bank which will scram before the slug enters the core. The Doppler feedback is responsible for initially terminating the power excursion. This number can vary by a factor of two during a fuel cycle and therefore results will be sensitive to where in the fuel cycle the event takes place. After the initial power excursion the power remains high until the slug has passed sufficiently through the core so that power decreases. Since there may not be much shutdown margin to begin with, after the slug passes through the core the decline in power may not be as rapid as occurs with reactor trip. This also impacts the consequences in terms of fuel damage.

7 References

1.1 D. J. Diamond, C. J. Hsu, and R. Fitzpatrick, "Reactivity Accidents - A Reassessment of the Design-Basis Events," NUREG/CR-5368, Brookhaven National Laboratory, January 1990.

1.2 "Foreign Experience Regarding Boron Dilution," Information Notice 91-54, U.S. Nuclear Regulatory Commission, September 6, 1991.

2.1 "Foreign Experience Regarding Boron Dilution," Information Notice 91-54, U.S. Nuclear Regulatory Commission, September 6, 1991.

2.2 T-L. Chu et al., "PWR Low Power and Shutdown Accident Frequencies Program, Phase 1A - Coarse Screening Analysis," Brookhaven National Laboratory, Nov. 13, 1991.

2.3 L. Bregeon et al., "Considerations on Nuclear Safety in France Two Years After Chernobyl," *Proceedings of the International ANS/ENS Conference on Thermal Reactor Safety*, Avignon, France, October 1988.

2.4 A. Gouffon, "Developments Concerning Reactivity Accidents in PWRs," *Proceedings of the Technical Committee Meeting on Reactivity Transient Accidents*, Vienna, Austria, IAEA-TC-610, International Atomic Energy Agency, November 1987.

2.5 E. W. Hagen, "Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants," NUREG/CR-2798, Oak Ridge National Laboratory, July 1982.

2.6 S. Jacobson, "Reactivity Transients After Pump Start in a Boron Diluted Loop," *Trans. Amer. Nucl. Soc., 61*, June 1990.

2.7 S. Jacobson, "Effects of Local Dilution Transients," *Proceedings of the Second Technical Committee Meeting on Safety Aspects of Reactivity Initiated Accidents*, Vienna, Austria, IAEA-TC-671, International Atomic Energy Agency, November 1989.

2.8 "Reduction of RCS Boron Concentration From 2110 to 1770 ppm Following Refueling," Incident Reporting System No. 0867.00, OECD Nuclear Energy Agency, August 1988.

2.9 S. Salah, C. E. Rossi, and J. M. Geets, "Three-Dimensional Kinetic Analysis of an Asymmetric Boron Dilution in a PWR Core," *Trans. Amer. Nucl. Soc., 15*, November 1972.

2.10 D. J. Diamond, C. J. Hsu, and R. Fitzpatrick, "Reactivity Accidents - A Reassessment of the Design-Basis Events," NUREG/CR-5368, Brookhaven National Laboratory, January 1990.

3.1 "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3," NSAC/60, Electric Power Research Institute, June 1984.

3.2 T-L. Chu et al., "PWR Low Power and Shutdown Accident Frequencies Program, Phase 1A - Coarse Screening Analysis," Brookhaven National Laboratory, Nov. 13, 1991.

3.3 A. C. Payne, Jr. et al., "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," NUREG/CR-3511, Sandia National Laboratory, August 1984.

4.1 T. G. Theofanous and H. P. Nourbakhsh, "PWR Downcomer Fluid Temperature Transients Due to High Pressure Injection at Stagnated Loop Flow," *Proc. Joint NRC/ANS Meeting, Basic Thermal Hydraulic Mechanisms in LWR Analysis*, September 14-15, 1982, Bethesda, Maryland, NUREG/CR-0043, p. 583, U. S. Nuclear Regulatory Commission, April 1983.

4.2 H. P. Nourbakhsh and T. G. Theofanous, "Decay of Buoyancy-Driven Stratified Layers with Applications to Pressurized Thermal Shock, Part I: The Regional Mixing Model," *Nucl. Eng. & Design*, in press, 1992.

4.3 T. G. Theofanous, H. P. Nourbakhsh, P. Gherson and K. Iyer, "Decay of Buoyancy Driven Stratified Layers with Application to Pressurized Thermal Shock," NUREG/CR-3700, U. S. Nuclear Regulatory Commission, May 1984.

4.4 K. Iyer, H. P. Nourbakhsh and T. G. Theofanous, "REMIX: A Computer Program for Temperature Transients Due to High Pressure Injection After Interruption of Natural Circulation," NUREG/CR-3701, U. S. Nuclear Regulatory Commission, May 1986.

4.5 T. G. Theofanous and H. Yan, "A Unified Interpretation of One-Fifth to Full-Scale Thermal Mixing Experiments Related to Pressurized Thermal Shock," NUREG/CR-5677, U.S. Nuclear Regulatory Commission, April, 1991.

4.6 H. P. Nourbakhsh and T. G. Theofanous, "A Criterion for Predicting Thermal Stratification Due to High-Pressure Injection in a Circulating Reactor Loop," *Nucl. Sci. & Engr., 94*, 77-79, September 1986.

4.7 J. B. Riester, R. A. Bayura, and S. H. Schwartz, "Effect of Water Temperature and Salt Concentration on the Characteristics of Horizontal Buoyant Submerged Jets", *J. of Heat Transfer, 102*, 557-562, August 1980.

5.1 K. Yanagisawa, T. Fujishiro, A. Negrini, and F. Franco, "Behavior of PCI-Resistant Additive Fuel for BWR Under Reactivity Initiated Accident Conditions," *J. Nucl. Sci. Technol.*, **27**, 1, January 1990.

5.2 T. Fujishiro, "Current Status of Research and Regulation Relevant to Reactivity Initiated Accident in Japan," *Proc. Tech. Comm. Mtg. Reactivity Transient Accidents*, Vienna, Austria, IAEA-TC-610, International Atomic Energy Agency, November 1987.

5.3 R. D. Mosteller, "ARMP-02 Documentation, Part II, Chapter 11 -NODE-P2 Computer Code Manuals, Volume 1, EPRI NP-4574-CCM, Electric Power Research Institute, October 1988.

5.4 Baltimore Gas and Electric Co., "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket No. 50-318 Request for Amendment Unit 2 Ninth Cycle License Application," Letter to U.S. Nuclear Regulatory Commission, February 7, 1989.

References

5.5 D. J. Diamond, C-J. Hsu, and V. Mubayi, "Probability and Consequences of Misloading Fuel in a PWR," NUREG/CR-5771, Brookhaven National Laboratory, August 1991.

5.6 Å. Ahlin, M. Edenius, and H. Häggblom, "CASMO, A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik Energiteknik AB, June 1978.

5.7 Baltimore Gas and Electric Co., "Calvert Cliffs Nuclear Power Plant Units 1 & 2, Final Safety Analysis Report," January 1971.

5.8 Baltimore Gas and Electric Co., "Calvert Cliffs Nuclear Power Plant Unit 2 Startup Test Report," Docket No. 50-318, May 12, 1977.

5.9 G. A. Korn, "*Interactive Dynamic System Simulation,*" McGraw-Hill Book Co., New York, 1989.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5819
BNL-NUREG-52313

2. TITLE AND SUBTITLE

Probability and Consequences of Rapid Boron Dilution in a PWR

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1992

4. FIN OR GRANT NUMBER

A3868

5. AUTHOR(S)

D.J. Diamond, P. Kohut, H. Nourbakhsh, K. Valtonen[1], P. Secker[2]

6. TYPE OF REPORT

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Brookhaven National Laboratory
Upton, NY 11973

[1] Finnish Centre for Radiation and Nuclear Safety
Helsinki, Finland

[2] University of Arizona
Tucson, AZ

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report documents the results of a scoping study of rapid dilution events in pressurized water reactors. It reviews the subject in broad terms and focuses on one event of most interest. This event could occur during a restart if there is a loss-of-offsite power when the reactor is being deborated. If the volume control tank is filled with water at a low boron concentration then a slug of this water could accumulate in the lower plenum. This would be the result of the trip of the reactor coolant power. The concern is that this diluted slug will rapidly enter the core after a reactor coolant pump is restarted and this could cause a power excursion leading to fuel damage. This problem was studied probabilistically for three plants and the important design features that affect the core damage frequency were identified. This analysis was augmented by an analysis of the mixing of the diluted water with the borated water already present in the vessel. The mixing was found to be significant so that neglect of this mechanism in the probabilistic analysis leads to very conservative results. Neutronic calculations for one plant were carried out to understand the effect of nuclear design on the consequences of the event.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Boron-dilution; boron-primary coolant circuits, boric acid, reactor cooling systems, PWR type reactors-excursions, PWR type reactors-reactor safety; risk assessment; probability, PWR type reactors-reactor accidents, fuel elements-damage, scram, reactor shutdown, neutron transport, reactor cores-damage, PWR type reactors-power losses, blackouts

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67

120555139531 1 1AN19R11M1151
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
P-211
WASHINGTON DC 20555

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67

120555139531 1 1AN19R11M1151
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
P-211
WASHINGTON DC 20555