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# Effects of Control System Failures on Transients, Accidents and Core-Melt Frequencies at a Combustion Engineering Pressurized Water Reactor

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**Prepared for**
**U.S. Nuclear Regulatory Commission**

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# ABSTRACT

Pacific Northwest Laboratory (PNL) performed a probabilistic risk analysis to develop estimates of core-melt frequency and public risk associated with control system failures in a Combustion Engineering pressurized water reactor. Value/impact analyses of possible modifications to prevent control system failures were also conducted. These analyses were based on a failure modes and effects analysis previously conducted at Oak Ridge National Laboratory. The control system failure modes fall into three main scenarios: two scenarios concern overfill of the steam generators, progressing to spillover into the steam lines. The third scenario deals with small-break loss-of-coolant accidents that may require operator action to depressurize the reactor coolant system. The analyses described in this report were performed in support of the U.S. Nuclear Regulatory Commission's program for Unresolved Safety Issue A-47, Safety Implications of Control Systems.

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# EXECUTIVE SUMMARY

Pacific Northwest Laboratory (PNL) has performed a probabilistic risk assessment (PRA) of control related failures in light water reactors (LWRs) for the U.S. Nuclear Regulatory Commission (NRC). This work was performed in support of the NRC's Unresolved Safety Issue A-47 program, Safety Implications of Control Systems. This report focuses on control failures in a representative Combustion Engineering (CE) pressurized water reactor (PWR), Calvert Cliffs 1. The PRA was based on failure modes and frequencies developed for Calvert Cliffs by Oak Ridge National Laboratory (ORNL) (Ball et al. 1985).

In addition, PNL has performed a value/impact analysis of proposed resolutions at Calvert Cliffs to address control system concerns identified by the A-47 program. Value/impact analyses are required by the NRC as input to the regulatory decision making process to insure that the need for and consequences of cost-effective regulatory actions are identified (U.S. NRC 1983a). The purpose here was to assist in screening and evaluating potential resolutions.

## GENERAL OVERVIEW

This report provides the following: 1) a list of control system failures of concern to the A-47 program, 2) a discussion of the safety implications of failures and possible progression to core-melt scenarios, 3) a calculation of risk, 4) a set of candidate resolutions to mitigate or prevent failures, 5) estimates of potential risk reduction achievable by implementation of the candidate resolutions, 6) the cost of implementing the resolutions, and 7) a presentation of resulting value/impact ratios. These topics will be summarized briefly in this section, followed by a more detailed summary of the technical analysis and results.

### Control System Failures of Interest

The A-47 program focused only on those control system failures that could initiate a plant response more severe than previously analyzed in design basis accidents, or failures that could cause plant conditions to exceed operating technical specifications.

Three scenarios identified by ORNL involved control system failures that might progress to more severe failures. Two scenarios center on the feedwater systems, with the potential for overfill of the steam generators (SGs) progressing to spillover into the steam lines. Steam generator overfill is possible due to 1) failure of closure signals to reach the main feedwater regulating valve or 2) failures in the valve itself. (These are subsequently referred to as Overfill Scenarios 1 and 2.) A third scenario deals with the Calvert Cliffs response to a small-break loss-of-coolant accident (SBLOCA), which may require depressurization of the reactor coolant system (RCS) before high pressure injection (HPI) systems can operate.

## Safety Implications

If not terminated by the operator, overfill has the potential to lead to water pouring into the steam lines, possibly resulting in steam line damage that may include major steam line failure. A large uncertainty currently exists concerning this potential, so a high probability of main steam line break (MSLB) given spillover of water into the steam lines was conservatively assumed.

An MSLB can contribute to potential core-damage scenarios because of rapid overcool or the potential for steam generator tube ruptures (SGTRs) induced by the rapid loss of pressure on the steam side of the tubes. If SGTR occurs, this represents a loss-of-coolant accident (LOCA), which is an initiator of core overheating and melting if safety injection of water fails.

In addition, overfill will lead to a transient shutdown in the plant, which can demand safety systems that in turn have the potential for failure. Thus the overfill event identified by ORNL was also examined as a potential transient initiator without the power conversion system available for decay heat removal.

The ORNL analysis identified a range for Calvert Cliffs SBLOCAs of less than 2 in. effective diameter but with a leakage rate greater than 132 gpm that may require operator action to depressurize the RCS before HPI operation is possible. The ORNL analysis estimated this potential at 1 in 10 SBLOCAs. Also identified was a series of events during a LOCA that would likely lead to loss of cooling water to plant air compressors, making loss of instrument air very probable, followed by loss of operation of the turbine bypass or steam atmospheric dump valves. Operator use of the power-operated relief valves (PORVs) would then be required to depressurize the RCS before HPI operation. This action is not called out specifically in current plant emergency procedures. Failure to depressurize would result in eventual core dryout and fuel damage.

## Risk

Event trees were developed by PNL and ORNL for the above scenarios, in order to estimate the conditional probability of core damage given the initiating event. When these conditional probabilities were multiplied by the initiating event frequency, the frequencies of core damage and core melt were obtained. The MSLB analyses developed by ORNL (Minarick and Kukielka 1982) and the Institute for Nuclear Power Operations (INPO 1982) were adapted by PNL to evaluate the potential for core damage given an MSLB. The results of the NRC Steam Generator Tube Integrity program (U.S. NRC 1985) were also used to estimate the probability of SGTR given an MSLB. Recovery from the SBLOCA scenario relied on operator-initiated depressurization and use of the HPI system to cool the core.

For the overfill scenarios, ORNL estimates of initiating frequency were used, with a PNL engineering estimate of the probability of subsequent protective system failure. Based on ORNL initiating frequencies of 9.0E-03/py and

4.4E-04/py for the overfill scenarios, PNL estimated the core-melt frequencies at 3.8E-06/py and 1.8E-07/py, respectively.

Based on ORNL's initiating frequency for the SBLOCA scenario of 1.5E-03/py, PNL estimated the core-melt frequency at 8.25E-06/py. The total predicted core-melt frequency is then 1.2E-05/py.

Representative radioactive release categories for the core-melt scenarios were chosen based on a review of the PRAs, with associated public doses taken from the NRC-sponsored prioritization of safety issues (Heaberlin et al. 1983). The risk was then estimated to be 18, 0.87, and 28.2 man-rem/py for the three scenarios, respectively, for approximately 47.2 man-rem/py. When put in perspective with the risk estimated for other nuclear safety issues (U.S. NRC 1983b), these estimates of core-melt frequency and risk are significant.

Risk Reduction/Cost/Value-Impact

Several plant modifications were examined that might reduce the risks associated with the scenarios defined by ORNL. For Overfill Scenarios 1 and 2, these modifications included such fixes as high water level trips and the use of automatic actuation of isolation valves in the feedwater system. Use of these fixes would reduce the frequency of failures and terminate overfill.

For the SBLOCA scenario, it was estimated that 28.2 man-rem/py, or 8.46E+02 man-rem averted over 30 years, could justify fixes costing a maximum of $8.46E+05 at $1000/man-rem. Several possible fixes are discussed that would reduce a portion of this risk either by reducing the frequency of SBLOCAs or by improving the probability of successful system response.

Given the significant estimate of risk involved, modifications appear feasible and cost-effective for both main feedwater high level trip and better HPI response to SBLOCAs. It should be stressed that this value/impact information is only one of several inputs to the regulatory decision making process. High level feedwater trips in particular may introduce the potential for increased operational transients and plant shutdown. Further scrutiny by NRC, ORNL, and affected utilities of potential modifications can bring additional insight and perspective to these results.

A source of conservatism in this analysis should be noted. In all sequences, a relatively high operator error probability is assumed, because there is considerable uncertainty about this parameter. Operator error probability could be reduced significantly through the use of effective training and emergency procedures, thus lowering the estimates of core-melt frequency and associated risk proportionally. The core-melt potential for these scenarios, as estimated in this report, is thought to be highly conservative.

The analysis is limited in scope, and the conclusions should be interpreted cautiously. First, ORNL did not estimate upper bounds for the initiating frequencies of the scenarios identified. As a result, PNL's estimates of the likely propagation of these failures to core melt do not include upper bounds; only best (or central) estimates are provided. Second, the estimates of core-melt frequency are dependent on several factors that may be plant-specific, including basic hardware reliability, operator response to system failures, and plant response to failures. Care must therefore be taken in applying these results to other CE PWRs. Variations among CE plants can include plant-specific differences in and compensations for factors such as:

- type of feedwater control
- power supplies
- controlling level display
- controlling level record
- back-up or alternate level displays
- annunciators and alarms
- operator training and procedures
- maintenance, general age, and state of equipment.

Control system failures involving steam generator overfills similar to those postulated in this report have occurred in operating PWRs. However, none of these failures has progressed to core damage and subsequent release of radioactive material. The accident sequences developed in this report are therefore speculative and subject to all the uncertainties and limitations surrounding the use of PRAs for predicting nuclear safety.

## TECHNICAL OVERVIEW

In the text below, a more in-depth technical summary of this report is given.

### Overfill Scenarios

The potential control system failures identified by ORNL deal primarily with overfill of the steam generator, with the potential for water entering the steam lines, and the potential for inadequate HPI response to SBLOCAs of a particular size.

The control failure scenarios and propagation to core melt are summarized in Table S.1. For the overfill scenarios, the ORNL failure modes and effects analysis identified a number of failure modes that can result in the feedwater regulating valve failing to close, having received or not received the turbine trip signal. These two cases constituted Overfill Scenarios 1 and 2, with initiating frequencies of 9.0E-03/py and 4.4E-04/py, respectively, including operator failure to terminate the overfill.

For these two scenarios, propagation of damage due to water in the steam lines was considered, including damage to the power conversion system (PCS) function of the feedwater system and condenser, damage to the steam lines causing MSLB, and MSLB propagating to SGTR. Assuming the overfill event constituted a transient shutdown with loss of the PCS, the core-melt frequency and associated risk were minimal: about 5E-09/py for Overfill Scenario 1 and 2E-09/py for Overfill Scenario 2.

TABLE S.1. Summary of ORNL and PNL Estimates of Accident Initiator Frequencies, Core-Melt Frequencies, and Public Risk for the Calvert Cliffs PWR

| Sequence | Initiating Frequency, 1/py | Core-Melt Frequency, 1/py Best Estimate | Public Risk, man-rem/py Best Estimate |
|---|---|---|---|
| Overfill Scenario 1 | | | |
| Transient Shutdown | (0.009)(0.1) | 5.7E-09 | 2.2E-02 |
| Overfill and MSLB | (0.009)(0.5) | 5.0E-08 (core damage) | 1.9E-01 |
| SGTR | (0.009)(0.5) (0.034) = 1.5E-04/py | 3.7E-06 | 1.8E+01 |
| | | 3.8E-06 | 1.8E+01 |
| Overfill Scenario 2 | | | |
| Transient Shutdown | (4.4E-04)(0.1) | 2.8E-10 | 1.1E-03 |
| Overfill and MSLB | (4.4E-04)(0.5) | 2.4E-09 (core damage) | 9.1E-03 |
| SGTR | (4.44E-04)(0.5) (0.034) = 7.5E-06/py | 1.8E-07 | 8.6E-01 |
| | | 1.8E-07 | 8.7E-01 |
| SBLOCA Scenario | | | |
| Inadequate Cooling | 1.5E-02/py(0.1) | 8.25E-06 | 2.82E+01 |
| PTS | 1.5E-02/py(0.01) | 1.5E-08 | 8.1E-02 |
| TOTAL | | 1.22E-05 | 4.72E+01 |

A 0.5 probability of MSLB given overfill was assumed, which could lead to core damage in a PWR. If core damage is equated with core melt (considered to be a conservative assumption), the frequency and associated risk were a factor of 10 higher than for transient shutdown, but still minimal.

Given MSLB, the potential for inducing an SGTR was 0.034, based on the considerations in the Steam Generator Tube Integrity program (NUREG-0844, U.S. NRC 1985). The MSLB was further assumed to occur with 50 percent probability above or below the main steam isolation valve (MSIV), resulting in a conditional probability of recovery from the SGTR of 2.44E-02, again based on NUREG-0844 scenarios. Note that this is only slightly higher than the 9.5E-03/event conditional probability of core melt given a 2 in. SBLOCA, used in the Calvert Cliffs PRA. A higher conditional probability of core melt would be expected given the aggravating MSLB.

The propagation to SGTR was found to constitute the highest estimate of core-melt frequency and risk, at 3.8E-06 core melt/py and 18 man-rem/py or 540 man-rem total (over 30 years) for Overfill Scenario 1. The core-melt frequency for Overfill Scenario 2 was a factor of 20 lower due to the lower assumed initiating frequency. The cost of modifications for both scenarios could then approach approximately $570,000, and still give a value/impact ratio of 1 man-rem/$1000. Many of the contributing failures to these scenarios are thought to cost significantly less than this to fix, thus giving favorable value/impact ratios. These failures and some candidate fixes are summarized in Table S.2.

The Calvert Cliffs plant is apparently lacking a main feedwater (MFW) pump trip with high steam generator water levels. The current design provides a main turbine trip and throttling of the feedwater regulating valve but does not trip the steam driven pump itself or isolate the feedwater delivery lines. The addition of either modification would significantly reduce the probability of an overfill progressing to steam line damage, with implementation costs thought to be under $570,000. Note that if an MSLB probability of 1E-03 given overfill is used, the estimate of core melt and risk drops by several orders of magnitude, making any cost-effective modifications difficult. The significance of the overfill scenarios and any need for plan changes must be evaluated in light of this uncertainty in the potential for MSLB.

<u>SBLOCA Scenario</u>

For SBLOCAs, the ORNL analysis has identified a range of SBLOCAs for the Calvert Cliffs plant of less than 2 in. diameter but with a leakage rate greater than 132 gpm that apparently will require operator action to depressurize the RCS before HPI operation is possible. The effective SBLOCA frequency for such a scenario was put by ORNL at 1.5E-03/py. Also identified was a series of events during a LOCA that lead to loss of cooling water to plant air compressors, making loss of instrument air very probable, followed by loss of operation of the turbine bypass or steam atmospheric dump valves. Operator use of the PORV would then be required to depressurize the RCS before HPI operation, but this is not called out specifically in plant emergency procedures. The probability of failure to depressurize the system given the SBLOCA was given a conditional probability by ORNL of (1.1E-02)(0.5) = 5.5E-03/event.

Using the SBLOCA initiating frequency of 1.5E-03/py, the new ORNL sequence then adds (1.5E-03/py)(5.5E-03 core melt/event) = 8.25E-06/py core-melt frequency. With the same SBLOCA risk categories as used in Calvert Cliffs, this represents an approximate risk of (8.25E-06/py)(0.7)(4.8E+06 man-rem/event) = 27.7 man-rem/py. The total risk considering all release categories was 28.2 man-rem/py, or 846 man-rem over 30 years. This indicates that costs associated with modifications to reduce or eliminate this scenario should be limited to $846,000 to keep a value/impact ratio of 1 man-rem/$1000.

TABLE S.2. Summary of the Value/Impact Analysis for the Calvert Cliffs PWR

| Proposed Fix | Estimated Cost, $ | Estimated Risk Reduction (man-rem) | V/I Ratio (man-rem/$1000) |
|---|---|---|---|
| Overfill Scenario 1 | | | |
| RV Solenoids on Vital Power | 6.0E+04 | 190 | 3.2 |
| Alternate Instrument Air Supply | 3.0E+04 | 36.2 | 1.2 |
| Improved RV Reliability | <1.8E+04 | 18 | 1[a] |
| Additional RV Closure Circuit | <2.6E+05 | 258 | 1[a] |
| Overfill Scenario 2 | | | |
| Additional or Gate in Parallel | 1.6E+04 | 15.3 | 0.9 |
| Additional Relay in Parallel | 1.6E+04 | 1.4 | 0.08 |
| Improved Cable Reliability | <8.7E+03 | 8.7 | 1[a] |
| Overfill Scenarios 1 and 2 | | | |
| Automatic High Level MFW Pump Trip | 1.2E+05 | 570 | 4.75 |
| Automatic Isolation of Feedwater Lines on High Level Plus MFW Pump Trip | 1.6E+05 | 570 | 3.56 |
| Modifications to Reduce Failure of RCS Depressurization | | | |
| Reduced Operator Error in IAS Recovery | <6.15E+05 | 6.15E+02 | 1[a] |
| Reduced Operator Error in PORV RCS Depressurization | <8.44E+05 | 8.44E+02 | 1[a] |
| Implementation of Both IAS Recover and PORV Procedures | <8.46E+05 | 8.46E+02 | 1[a] |
| Elimination of SWS Isolation Given SGIS | <7.6E+05 | 7.6E+02 | 1[a] |

(a) Specifics of a modification or associated costs are uncertain. Calculation of risk reduction in man-rem was used to establish dollar expenditures compatible with a $1000/man-rem cost-benefit guideline.

Note that the potential for vessel failure and core melt due to pressurized thermal shock (PTS) was put at 1.5E-08/py by ORNL, which would contribute insignificantly to the above estimate of core melt. Further consideration of PTS was thus dropped.

The modifications examined to improve the availability of the turbine bypass or steam dump valves, or use of the PORV to depressurize the RCS, all have costs that are probably much less than the amounts above. Incorporation of the PORV operation into the emergency procedures appears to stand out as a likely option because it provides an alternate pathway to depressurize independent of the instrument air system, which impacts the other valves. This does not eliminate, however, the need to examine system changes to keep turbine bypass and steam dump valves available for use.

SBLOCAs Induced by Control System Failure

Finally, ORNL identified a number of control failures that may induce SBLOCAs of less than 2 in. However, no estimate of the frequency for such failures was made. A consideration of risk for such SBLOCAs must include the conditional probability of core melt from the original SBLOCA sequence, as defined in the Calvert Cliffs PRA, as well as the new sequence developed by ORNL. As such, even a small contribution of control system failure to the SBLOCA frequency of 1.2E-02/py could represent significant core-melt frequency and risk. Further consideration may be warranted to identify the frequency of such SBLOCAs.

Conclusions

The core-melt frequency and risk attributed to overfill scenarios in the Calvert Cliffs plant have been estimated in this report. With 570 man-rem over 30 years, modifications for under $570,000 (i.e., 1 man-rem/$1000) may be considered cost-effective ways to reduce the failure frequency of MFW valves to close on demand. Automatic trip of the MFW pumps or closure of feedwater block valves at high steam generator water levels would be the best method of terminating the overfill sequences. Each plant would have to be examined for detrimental effects from such a modification. A less conservative assumption for the potential for MSLB given overfill (i.e., 1E-03 versus 5E-01) would lower the overall estimate of risk by over two orders of magnitude, and the contribution from MSLB and SGTR would then be comparable to transient shutdown with loss of the power conversion system. The overall risk would become insignificant, making modifications difficult to justify.

For SBLOCAs, the potentially inadequate response of Calvert Cliffs to breaks of less than 2 in. and the subsequent risk indicate that modifications are necessary to improve the plant response to such breaks. Any depressurization required could be automatic or manual. The failure probability of both actions would then be estimated at approximately 1E-03/demand; however, the current design and procedures appear to fail the turbine bypass and steam dump valves due to loss of instrument air, and operator procedures do not call for use of the PORV. Improved procedures are needed for maintaining instrumentation and design to insure that the TBVs and ADVs are functional, and for calling for use of the PORV given TBV and ADV failure. Automatic PORV depressurization is a possibility; however, this would introduce the potential for inadvertent valve lifts. Thus the implementation of automatic actuation of a PORV given a SBLOCA may raise additional safety concerns.

# 1.0 INTRODUCTION

The objectives of this report are to examine the potential frequency of core melt and the resultant public risk associated with control system failures in Combustion Engineering (CE) pressurized water reactors (PWRs), and to evaluate the value/impact associated with proposed plant modifications. The scope of this study is limited to failure modes identified by Oak Ridge National Laboratory (ORNL) in their examination of the Calvert Cliffs 1 nuclear power plant (Ball et al. 1985).

Baltimore Gas & Electric Company's Calvert Cliffs 2, an 850 MWe CE PWR unit in Lusby, Maryland, is the twin unit to the reference design in the ORNL investigation. This plant was also the subject of a probabilistic risk assessment (PRA) (Hatch et al. 1982). The PRA study is used in this evaluation of core melt and public risk, where appropriate.

This work is a direct extension of the examination by Pacific Northwest Laboratory (PNL) of the potential core-melt frequency and public risk associated with control system failures in Westinghouse PWRs, Babcock & Wilcox (B&W) PWRs, and General Electric (GE) boiling water reactors (BWRs) (Bickford and Tabatabai 1985a, 1985b, and 1985c). This previous examination of risk was based on failure mechanisms identified by Idaho National Engineering Laboratory (INEL) for the GE Browns Ferry plant (Bruske et al. 1985) and the H. B. Robinson Westinghouse plant (Ransom et al. 1985), and mechanisms identified by ORNL for the Oconee B&W plant (Austin et al. 1985). The approach used in these previous examinations is developed here and applied to the CE Calvert Cliffs PWR.

ORNL reported two major failure modes that impact reactor safety as a result of control system failures:

- insufficient core cooling following small-break loss-of-coolant accidents (SBLOCAs)
- steam generator (SG) overfilling.

Two failure mechanisms were identified for the latter:

- SG overfilling via regulating valve given turbine trip signals to feedwater valve to close
- SG overfilling via failure of turbine trip to signal feedwater valve to close.

Because the plant modifications necessary to reduce the frequency of failure can be different for these two mechanisms, they are developed separately as Overfill Scenarios 1 and 2.

For the two major failure modes, failure mechanisms in the control system have been identified and are developed in the following chapters.

## 2.0 STEAM GENERATOR OVERFILL

The ORNL failure modes and effects analysis (FMEA) of the Calvert Cliffs plant (Ball et al. 1985) identified the safety concerns of steam generator (SG) overfill in a pressurized water reactor (PWR). It was stated that SG overfill could:

- produce secondary side damage that might compromise safety equipment or produce a cascade of events that might have primary side effects, including radiological leakage
- densify primary coolant, which would in turn reduce pressure and possibly cause
  - the loss of pressurizer control
  - vapor lock of the primary flow path
  - excess reactivity from cold flow
- provide excess cooling, which might contribute to pressurized thermal shock (PTS).

Of primary interest here is the initiation of a transient requiring plant shutdown or response of the engineered safety features. PNL believes that the first manifestation of system damage in the overfill scenario is main turbine damage and turbine trip. This in itself is not a serious challenge to plant systems. However, the potential for excessive moisture carryover and even spillover does introduce the potential for loss of the power conversion system (PCS) during transient shutdown and further introduces the potential for water-induced damage in the steam lines, including main steam line break (MSLB).

WASH-1400 (U.S. NRC 1975) considered the consequences of ruptures on the secondary side of a steam generator for a Westinghouse PWR (the Surry plant). Thirty possible accident sequences were identified, all ending in either a rapid cooldown transient or a LOCA. The study concluded that such transients induced by steam generator failures did not lead to core melt but could release activity from the fuel-clad gap due to fuel damage. Thus steam generator rupture was not identified as an important factor in the risks due to transient events (U.S. NRC 1975, p. I-47). More recent considerations indicate that core damage from MSLB can result; however, the damage is expected to be less severe than core melt. This idea is developed below.

To be conservative, the excessive cooldown transient is modeled with an appropriate event tree for MSLB. The potential for core damage as a result of MSLB is then given. In addition, the potential for inducing a steam generator tube rupture (SGTR) or multiple ruptures following MSLB is considered. SGTR scenarios can indeed propagate to core melt in a PWR. The potential for PTS in the Calvert Cliffs plant is also considered.

A discussion of the failure initiators identified by ORNL is given below, followed by a consideration of progression of the accident to transient shutdown, MSLB or SGTR, and core melt.

## 2.1 OVERFILL INITIATING FREQUENCY

ORNL analysis of the Calvert Cliffs main feedwater (MFW) control system indicates that this system has two possible general failure modes that will result in overfill following a plant trip. This then becomes an overfill event given operator failure to terminate the overfeed. The two scenarios identified are as follows:

- Overfill Scenario 1: The regulating valve fails to close given a turbine trip signal following a reactor trip.
- Overfill Scenario 2: The turbine trip fails to signal the feedwater valve to close following a reactor trip.

Note that in both cases the reactor trip could be generated by the turbine trip signal itself, or a reactor trip due to other causes could then generate a turbine trip. The result is the same: the plant is shut down, with steaming rates decreasing from that point in time.

### 2.1.1 Overfill Scenario 1

The following failures and associated frequencies have been identified by ORNL for Overfill Scenario 1:

- loss of instrument air, causing the regulating valve to fail "as is" due to loss of power from a specific circuit (3.5E-02/yr), or failure closed of a solenoid valve (6E-03/yr)
- mechanical failure of the regulating valve (6E-03/yr)
- failure of the closure signal to the regulating valve, due to a transducer failure (3.4E-02/yr), or failure of the hand/auto electric station (8.7E-03/yr).

Thus the estimated total initiation frequency is 9E-02/yr. The probability of operator failure to terminate the overfill was then estimated by ORNL at 0.1/demand, giving an estimate of the frequency of overfill of 9E-03/yr. It is assumed here that the above estimates incorporate a frequency for turbine trip signal generation, with failure on demand of the specific components.

### 2.1.2 Overfill Scenario 2

For Overfill Scenario 2, the following failures have been identified by ORNL:

- failure of the turbine trip signal to close the feedwater valve, caused by an OR gate failure (5.2E-03/demand), failure of a relay to close (5.7E-04/demand), or a cable failure (2.9E-03/demand)

- failure of the regulating valve to close, caused by its failing open at power (5.17E-03/yr), or by its failing to close when required (5E-01/yr).

Thus the estimated total initiation frequency is 4.4E-03/yr. The probability of operator failure to terminate the overfill was again estimated by ORNL at 0.1/demand, giving an estimate of the frequency of overfill of 4.4E-04/yr.

A more detailed discussion of the FMEA used to identify these failure initiators is provided in the ORNL report (Ball et al. 1985). Section 4.5 of the report discusses operator effects, but only estimates operator failure to diagnose SBLOCA scenarios. Overfill scenarios are not presented.

## 2.2 ACCIDENT PROGRESSION ANALYSIS FOR SG OVERFILL

Accident progression to transient shutdown, MSLB, and possible SGTR is proposed here as the scenarios of interest for progression to core melt. The potential for PTS to lead to vessel rupture would require thermal hydraulic simulations of excessive cooldown in the CE design given MSLB. ORNL does indicate, however, that probability estimates of MSLB progression to PTS as calculated by the PTS program did not show MSLB to be a significant contributor to risk. This is therefore assumed to play an insignificant role in progression to core melt, as was the case with the H. B. Robinson Westinghouse PWR (Bickford and Tabatabai 1985a). This assumption can be updated as more information is made available from the PTS program.

### 2.2.1 Transient Shutdown

The SG overfill scenarios examined previously for the A-47 program for the Westinghouse, B&W, and GE plants identified the potential for SG overfill while the reactors are at power (Bickford and Tabatabai 1985a, 1985b, and 1985c). For the CE plant, however, the failure mode identified is for overfill after reactor trip, as discussed in Section 2.1.

Degrading steam quality introduces excessive moisture in the steam flow to the condenser, and eventually actual water flow. The potential then exists for damage to the condenser, resulting in its isolation and loss as a decay heat sink. This would represent a loss of the PCS caused by other than loss of offsite power, or a T2 transient, as developed in the Calvert Cliffs probabilistic risk assessment (PRA) (Hatch et al. 1982). The T3 transient, in which the PCS is initially available and then lost after turbine trip, is also included in the PRA.

The potential for damage in the steam lines beyond simple loss of the decay heat path is developed below. The T2 and T3 transients can represent a significant initiator to core melt, regardless of the potential for MSLB. Achieving cold shutdown under such conditions requires recovery of the PCS, which consists of a feedwater supply and decay heat removal pathway, the latter typically supplied via the condenser. Failure of the condenser results in fuel damage or actual core melt. An alternative PCS function can be supplied by the auxiliary feedwater (AFW), and steam relief via the steam side power-operated relief valves (PORVs) or atmospheric dump valves.

The frequency of T2 transients in the Calvert Cliffs study is 3 per plant-year (py). This assumes a probability of 1.0 for loss of the PCS with the initiating transient. The T3 transient frequency is set at 4 per py, with a 0.01 probability of loss of the PCS after trip. The dominant sequences are as follows:

(T2)(M)(L) = (T2)(PCS failure)(steam relief & AFW failure)(PCS recovery failure)
= (3/py)(1.0)(3E-05)(0.1)
= 9E-06/py

(T2)(K)(M)(L) = (T2)(scram failure)(PCS failure)(steam relief & AFW failure)(PCS recovery failure)
= (3/py)(2.0E-05)(1.0)(3E-05)(1.0)
= 1.8E-09/py

(T2)(M)(Q)(H) = (T2)(PCS failure)(stuck relief valve)(HP recirc. failure)
= (3/py)(1.0)(0.08)(3.54E-05)
= 8.5E-06/py

(T3)(M)(L) = (T3)(PCS failure)(steam relief & AFW failure)(PCS recovery failure)
= (4/py)(0.01)(3E-05)(1.0)
= 1.2E-06/py.

The contribution from the T2 and T3 sequences in the Calvert Cliffs PRA yields a predicted core-melt frequency of 1.9E-05/py of a total plant core-melt frequency of 1.2E-04/py, assuming that the AFW upgrade as discussed in the PRA was implemented. (This compares to the T2 transient frequency in the Oconee B&W PWR of 3/py, with a T2 core-melt frequency of 3.44E-05/py out of a total plant core-melt frequency of 8.2E-05/py [PNL-5544].)

For this program, the initiation frequency is set at 9E-03/py for Overfill Scenario 1 and 4.4E-04/py for Overfill Scenario , assuming that the 0.1 operator error represents the failure to terminate the scenario before overfill.

Although no mechanisms for loss of the PCS have been identified given overfill and no such failures have occurred, the potential exists for water in the steam lines to result in trip of the main feedwater steam-driven turbines or damage to the heat removal path and hence loss of the PCS function. The probability of this action is assumed here to be 0.1, as it was in the risk analyses of other PWRs for the A-47 program (Bickford and Tabatabai 1985a and 1985b).

Recovery would then require reestablishing the MFW or AFW function with a heat removal path. The original AFW system in Calvert Cliffs relied on a manually initiated flow via two steam-driven pumps, which could likewise be impacted by the degraded condition in the steam lines. The upgraded system, however, uses an automatically initiated AFW flow with one steam-driven and one electric-driven pump, and two other pumps available given operator action. If AFW is initiated on MFW trip, delivery via the electric pump would then be assured. If initiated on low SG water level, boiloff of entrained or standing water would occur during the transition from overfill to low level, and thus both pumps would function.

The steam relief valves must also operate to provide the PCS function if the main condenser is damaged and isolated. Experience regarding steam relief valve reliability in two-phase flow indicates that although the valves may chatter, no decrease in the actual reliability of their opening or closing was observed. High pressure recirculation functions should likewise be unaffected by the initial overfill, since they are initiated after a transition from a secondary side overfill to an undercool event. It is thus thought that the failure probability estimates used in the Calvert Cliffs PRA for the AFW with steam relief and high pressure recirculation functions would then be applicable here.

#### 2.2.1.1 Core Melt from SG Overfill and Transient Shutdown

The estimate of core melt due to transient shutdown from SG overfill for Overfill Scenario 1 is:

$$(0.1)[(9E\text{-}03/py)/(3/py)](1.9E\text{-}05/py) = 5.7E\text{-}09/py.$$

For Overfill Scenario 2, the core-melt frequency is:

$$(0.1)[(4.4E\text{-}04/py)/(3/py)](1.9E\text{-}05/py) = 2.8E\text{-}10/py.$$

#### 2.2.1.2 Public Risk for SG Overfill and Transient Shutdown

The public risk associated with the T2 core-melt sequences is divided between several WASH-1400 release categories, as shown below. The man-rem per release is taken from the Value/Impact Handbook (Heaberlin et al. 1983). The public risk is then estimated at 2.2E-02 man-rem/py for Overfill Scenario 1 and 1.1E-03 man-rem/py for Overfill Scenario 2.

### 2.2.2 Main Steam Line Break

The next progression in damage more severe than loss of the PCS function is thought to be MSLB. Again, the WASH-1400 analysis of the Surry plant concluded that MSLBs are not a viable pathway to core melt (U.S. NRC 1975). However, the ORNL precursor study (Minarick and Kukielka 1982) and the updated

INPO precursor study (INPO 1982) did consider MSLB-initiated event trees leading to core damage. There is some difference of opinion as to how to equate the frequency of core damage with core melt, with proposed factors ranging from 1 to 1/30. Here the estimate of core damage is equated with core melt, with these reservations as to the level of conservatism noted.

As discussed in previous PNL examinations of overfill in the Westinghouse PWR, the B&W PWR, and GE BWR (Bickford and Tabatabai 1985a, 1985b, and 1985c) the probability of MSLB is 1.0 for spillover at rated power, and 0.5 for spillover after main turbine failure and plant trip. A value of 0.1 for MSLB was used for cases in which the SG water level stabilized below the point of actual spillover, but with degraded steam quality and the high likelihood of injection of water into the steam lines. The latter estimates would include the potential for continued water buildup after SCRAM and pipe failure due to excessive static load. The level of conservatism for spillover at power is uncertain, given the large possible dynamic loads on the steam lines and support structures. A more comprehensive review of such events to date (Bickford and Tabatabai 1985c) indicates, however, that no such MSLBs have occurred despite several spillover events. The lower probability of 0.5 was proposed for spillover at low power to reflect the degrading steam flow and the likely less severe dynamic forces in the steam lines. A review of static forces caused by dead weight of water in the steam lines if overfill continued to the point of filling the steam lines also indicated a low probability of MSL failure (1E-03) in Browns Ferry (Bickford and Tabatabai 1985c).

The steam generator tube integrity program (U.S. NRC 1985) also uses a 1E-03 probability of MSLB for overfill following an SGTR. Qualitative evaluation of the potential for MSLB damage following overfill in the A-47 program by ORNL (Clark et al. 1985) indicates only that a high potential for damage exists. No specific calculations have been done for Calvert Cliffs.

The ORNL analysis identified overfill scenarios that resulted in overfill 3 to 4.5 minutes after reactor trip unless terminated by the operator. The probability of 0.5 for inducing MSLB is used here to be consistent with the approach used in the previous value/impact analyses.

The frequency of MSLB for Overfill Scenario 1 is estimated to be (9E-03/py)(0.5) = 4.5E-03/py. The frequency of MSLB for Overfill Scenario 2 is estimated to be (4.4E-04/py)(0.5) = 2.2E-04/py.

### 2.2.3 Accident Progression to Core Melt Given MSLB

MSLB is not recognized as a contributor to core melt in the Calvert Cliffs PRA study (Hatch et al. 1982). For this analysis the results of the ORNL precursor study as updated by INPO (INPO 1982) are used to be consistent with the PNL A-47 value/impact analyses of other PWRs (Bickford and Tabatabai 1985a and 1985b). The resulting core-damage event tree given MSLB in a PWR is shown in Figure 2.1. The predicted probability of core damage given MSLB is estimated at 1.1E-05. The failures involved in the overfill scenarios are not thought to impact the response of the engineered safety systems. Location of the MSLBs

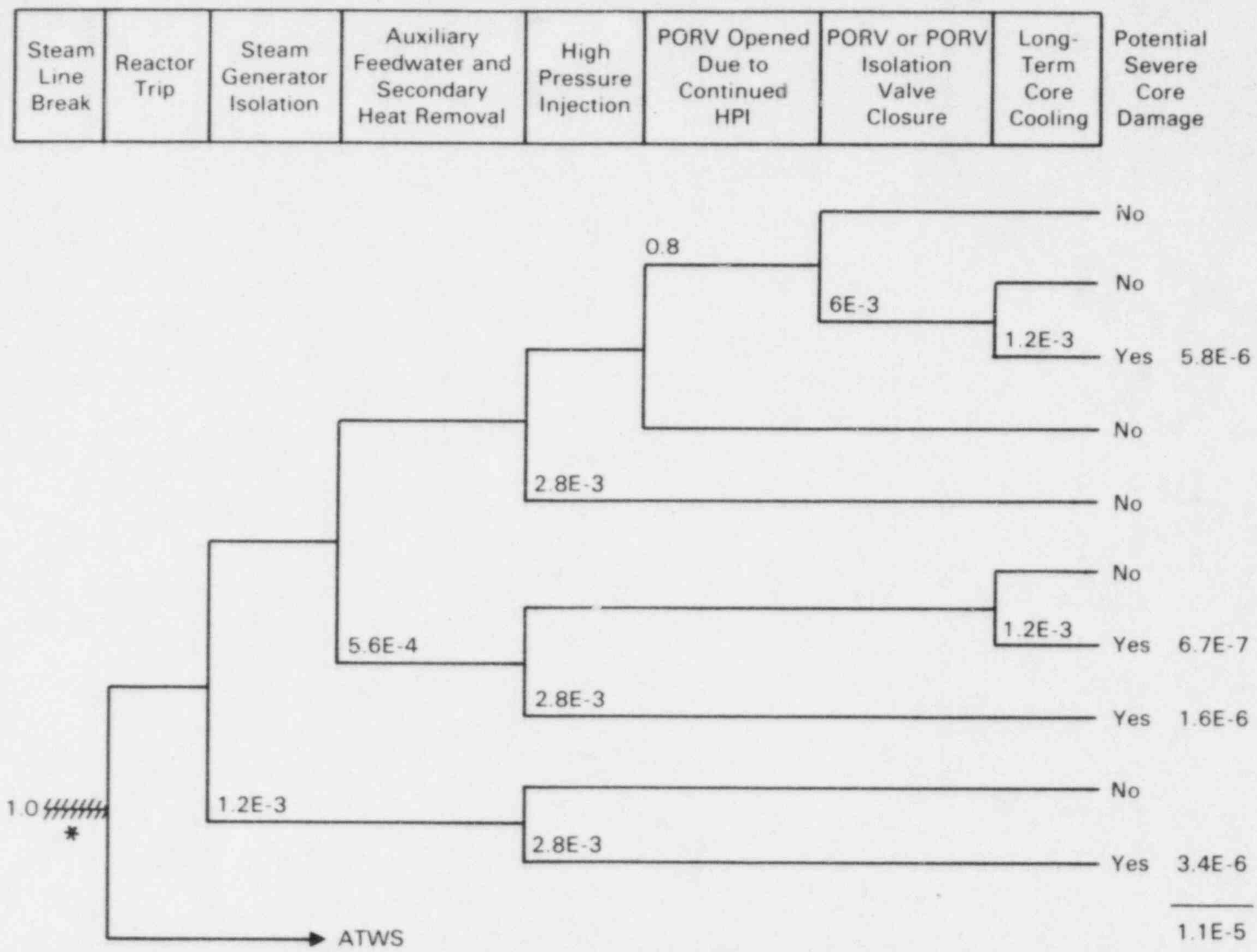


FIGURE 2.1. Modified Overcool Event Tree (INPO 1982)

(i.e., above or below the main steam isolation valves, or MSIVs) is not as critical as it is for tube rupture scenarios (to be developed below), since isolation of feedwater flow can effectively isolate the generator.

#### 2.2.3.1 Total Core Melt from SG Overfill with MSLB

The predicted frequency of core damage due to MSLB for Overfill Scenario 1 is estimated at (4.5E-03/py)(1.1E-05) = 5.0E-08/py. The predicted frequency of core damage due to MSLB for Overfill Scenario 2 is estimated at (2.2E-04/py)(1.1E-05) = 2.4E-09/py.

#### 2.2.3.2 Public Risk from SG Overfill with MSLB

Again, MSLB core-melt scenarios were not developed for the Calvert Cliffs PRA. As shown in Figure 2.1, however, the dominant sequences associated with MSLB and core damage are the result of PORV lift and high pressure injection (HPI) failure. This is similar to many of the transient induced sequences in the Calvert Cliffs PRA in which the risk is primarily associated with PWR release categories 1, 3, 5, and 7, as shown in Table 2.1. This distribution is used here for MSLB scenarios. Again, equating core damage from MSLB to core melt is considered conservative.

The estimate of public risk is 1.9E-01 man-rem/py for Overfill Scenario 1, and 9.1E-03 man-rem/py for Overfill Scenario 2.

## 2.3 PROGRESSION OF SG OVERFILL SCENARIOS TO SGTR AND CORE MELT

The damage caused by MSLB may progress to more serious failures that may impact plant recovery. The one failure directly related to the MSLB that significantly impacts the primary side of the reactor is the potential for SGTR. This failure could be caused by the pressure transient upon blowdown of the secondary side following MSLB. This sequence is developed below.

### 2.3.1 Frequency of SG Overfill, MSLB, and SGTR

Given steam line failure, the accident will most likely progress as a simple cooldown transient. However, the potential exists to induce an SGTR due to the differential pressures generated in the blowdown. The probability of

TABLE 2.1. Public Release Categories Associated with T2 and T3 Transient Core-Melt Sequences for Calvert Cliffs

| Release Category | Probability | Man-rem/Event |
|---|---|---|
| 1 | 0.0001 | 5.4E+06 |
| 3 | 0.7 | 5.4E+06 |
| 5 | 0.007 | 1.0E+06 |
| 7 | 0.3 | 2.3E+03 |

SGTR given MSLB has been addressed by the NRC (U.S. NRC 1985, p. 3-8) as part of its evaluation of Unresolved Safety Issues A-3, A-4, and A-5. Based on experience, this probability (p) is given below:

p (tube rupture following an MSLB) = 0.034.

This was broken down as follows:

p (1 SGTR following MSLB) = 0.017

p (2 to 10 SGTRs following an MSLB) = 0.014

p (more than 10 SGTRs following an MSLB) = 0.003.

The frequency of MSLB and SGTR would then be estimated at (9E-03/py) (0.5)(0.034) = 1.5E-04/py for Overfill Scenario 1, and (4.4E-04/py) (0.5)(0.034) = 7.5E-06/py for Overfill Scenario 2.

### 2.3.2 Progression of SG Overfill, MSLB, and SGTR to Core Melt

The initial plant response to tube ruptures can be modeled as a SBLOCA. However, the long-term system response to an SGTR with an MSLB may differ from that of a LOCA in that water released from the break may not be available for collection at sumps within the reactor building. Long-term recirculation modes also may not be available, as they would be for most LOCAs.

To evaluate the plant response to an SGTR, PNL examined the event tree developed for Calvert Cliffs (Hatch et al. 1982, p. 4-37). In addition, information developed in support of Unresolved Safety Issues A-3, A-4, and A-5 concerning steam generator tube integrity (U.S. NRC 1985) was examined.

Core-melt scenarios in PWRs following an SGTR focus on the potential for failure of the HPI system to successfully provide inventory to the reactor coolant system (RCS). This centers on the early failure of the HPI itself, or exhaustion of the water inventory in the reactor water storage tank (RWST) before depressurization. The latter event is highly dependent on the ability to isolate the affected steam generator. As with the PNL analysis of SBLOCAs in the Westinghouse and B&W PWRs (Bickford and Tabatabai 1985a and 1985b), it is thought that the event trees developed of the Calvert Cliffs PRA do not fully reflect the difficulty of isolating an SGTR under the aggravating conditions of an MSLB.

In the PRAs, many scenarios for single and especially multiple tube rupture events postulate the lifting and sticking open of steam generator relief valves due to the large pressure spike experienced in the secondary side following tube rupture. However, in this case the scenario is driven by an assumed MSLB on the secondary side, making the lift of relief valves unlikely. Here the break location becomes important, since breaks above the MSIVs eliminate any possibility of isolation of the affected generator and significantly increase the potential for failure of the HPI beyond that reflected in the Calvert Cliffs PRA, again due to depletion of the reactor water storage inventory before depressurization can be achieved.

Failure to isolate the steam generator due to rupture of the steam line inboard of an MSIV is considered in NUREG-0844 (U.S. NRC 1985) as a potential contributor to tube rupture. However, this initiating frequency is quite low due to the low random potential for valve or steam line rupture. This then results in a small contribution to the total core-melt frequency for MSLBs inboard of the MSIV predicted in NUREG-0844. This analysis, however, assumes that there is a high probability of steam line break given SG overfill. If a conservative approach further assumes a 50 percent probability of MSLB above or below the MSIV, this scenario may play a dominant role in the resulting conditional probability of progression to core melt.

To determine the potential impact of the MSLB location on core-melt frequency, the appropriate scenarios and failure probabilities from Chapter 3.4 in NUREG-0844 were examined, with the results given below. The scenarios and failure probabilities have also been coupled with the assumed SGTR probabilities (see Table 2.2). This same approach was used to model the Westinghouse and B&W plant response to MSLB and SGTR (Bickford and Tabatabai 1985a and 1985b). Given the level of uncertainty in plant response, this approach is considered appropriate.

TABLE 2.2. Probability of Recovery Given MSLB and SGTR

| Number of SGTRs | Probability of Rupture | Probability of Loss of RWST before RCS Depressurization | Probability of Failure to Isolate SG | Net Core-Melt Probability |
|---|---|---|---|---|
| Case 1: Rupture of MSL Inboard of the MSIV | | | | |
| 1 | 0.017 | 1E-03 | 1 | 1.7E-05 |
| 2 to 10 | 0.014 | 1E-02 | 1 | 1.4E-04 |
| >10 | 0.003 | 5E-01 | 1 | 1.5E-03 |
| Total Probability of Core Melt Given MSLB Inboard of MSIV | | | | 1.66E-03 |
| Conditional Probability of Core Melt Given MSLB and SGTR | | | | 4.87E-02 |
| Case 2: Rupture of MSL Downstream of the MSIV | | | | |
| 1 | 0.017 | 1E-04 | 1E-03 | 1.7E-09 |
| 2 to 10 | 0.014 | 1E-03 | 1E-03 | 1.4E-08 |
| >10 | 0.003 | 1E-03 | 1E-03 | 3.0E-09 |
| Total Probability of Core Melt Given MSLB Downstream of MSIV | | | | 1.87E-08 |
| Net Probability of Core Melt Given MSLB and SGTR | | | | 5.50E-07 |

If a 50 percent probability of MSLB inboard of the MSIVs is used, the conditional probability of core melt given MSLB and SGTR can then be weighted, giving (0.5)(4.87E-02 + 5.50E-07) = 2.44E-02. Using this initiating frequency for overfill, MSLB, and SGTR, the estimated core-melt frequency due to Overfill Scenario 1 is then (9.0E-03/py)(0.5)(0.034)(2.44E-02) = 3.7E-06/py. The estimated core-melt frequency for Overfill Scenario 2 is then (4.4E-04/py)(0.5)(0.034)(2.44E-02) = 1.8E-07/py.

#### 2.3.2.1 Comparison to SBLOCA Response

PNL also examined the possibility of modeling the SGTR event with the SBLOCA event tree from the Calvert Cliffs PRA. With an initiating frequency of 1.0E-03/py for S2 (less than 2-in. break) SBLOCAs, the resulting Calvert Cliffs core-melt frequency due to S2 sequences was 9.5E-06/py, yielding a conditional probability of core melt given SGTR of (9.5E-06/py)/(1.0E-03/py) = 9.5E-03, or approximately 1E-02.

The approach used above to model SGTR with an MSLB then gives roughly a factor of 2.6 increase in the estimated frequency of core melt compared to an analysis based on system response to a SBLOCA only.

#### 2.3.2.2 Pressurized Thermal Shock

The potential for MSLB or SGTR to lead to PTS and possible vessel rupture has not been fully evaluated. The consideration of such events in the Westinghouse PWR (Bickford and Tabatabai 1985a) has put preliminary estimates of vessel failure probability below 1E-06 given the PTS event, indicating that PTS would not contribute significantly to risk for events initiated by the control failures examined. However, the role of PTS should be examined specifically for the CE plant design when the PTS program makes its conclusions.

## 2.4 PUBLIC RISK DUE TO SG OVERFILL, MSLB, AND SGTR

The core-melt sequences above were caused by failure of the water storage tank inventory and by water not being available from the building sumps. As a result, the containment sprays are also assumed to be inoperable. In addition, the release is characterized by a significant leakage of containment, given SGTR and MSLB. With these considerations, only release category 2 at 4.8E+06 man-rem/core melt is used here to estimate the public risk.

The public risk associated with Overfill Scenario 1 leading to MSLB and SGTR is (3.7E-06/py)(4.8E+06 man-rem/event) = 1.8E+01 man-rem/py. The public risk associated with Overfill Scenario 2 leading to MSLB and SGTR is (1.8E-07/py)(4.8E+06 man-rem/event) = 8.6E-01 man-rem/py.

## 2.5 RESULTS OF SG OVERFILL

The results of the consideration of steam generator overfill resulting in a transient shutdown, MSLB, or progressing beyond MSLB to an SGTR are summarized in Table 2.3. Again, the frequencies assume an Overfill Scenario 1

TABLE 2.3. Result of Overfill Scenario for Calvert Cliffs

| Sequence | Frequency, 1/py | Core-Melt Frequency, 1/py (best estimate) | Public Risk, man-rem/py (best estimate) |
|---|---|---|---|
| Overfill Scenario 1: | | | |
| Transient shutdown | (0.009)(0.1) | 5.7E-09 | 2.2E-02 |
| Overfill & MSLB | (0.009)(0.5) | 5.0E-08 | 0.9E-01 |
| SGTR | (0.009)(0.5)(0.034) = 1.5E-04/py | 3.7E-06 | 1.8E+01 |
| | | 3.8E-06 | 1.8E+01 |
| Overfill Scenario 2: | | | |
| Transient shutdown | (4.4E-04)(0.1) | 2.8E-10 | 1.1E-03 |
| Overfill & MSLB | (4.4E-04)(0.5) | 2.4E-09 | 9.1E-03 |
| SGTR | (4.44E-04)(0.5)(0.034) = 7.5E-06/py | 1.8E-07 | 8.6E-01 |
| | | 1.8E-07 | 8.7E-01 |
| TOTAL | | 4.0E-06 | 1.9E+01 |

initiating frequency of 9E-03/py and an Overfill Scenario 2 initiating frequency of 4.4E-04/py, including a 0.1 failure probability of the operator to terminate the overfill and considering both steam generators. The conditional probability of inducing an MSLB is then 0.5, and the conditional probability of progressing to core damage given MSLB is 1.1E-05.

Note also that the predicted core-melt frequency for MSLB is higher than that of the transient by a factor of 10. However, the MSLB frequency actually predicts core damage, not core melt.

The probability of SGTR given MSLB is then 0.034, with a 50% probability, assuming that the break occurs above an MSIV, where water would not be collected by building sumps for recirculation. This break location dominates the risk due to MSLB and SGTR, with the conditional probability of core melt under those circumstances set at 2.44E-02.

The total estimated core-melt frequency for both overfill scenarios is then 4.0E-06/py, with a best estimate of the public risk at 1.9E+01 man-rem/py. The progression of SG overfill to MSLB and SGTR dominates the estimates.

## 3.0 SMALL-BREAK LOCAS WITH LOSS OF INSTRUMENT AIR

The ORNL study of the Calvert Cliffs plant indicates that it may be vulnerable to SBLOCAs (those less than 2 in. diameter) with greater than 132 gpm of coolant loss that do not automatically depressurize the reactor cooling system (RCS) to below the upper pressure limit of the high pressure injection (HPI) system. For these SBLOCAs, manual depressurization would then be required to avoid core melt. The contribution of control failures to this event includes SBLOCA initiators and an aggravating control system failure that may cause the loss of one method of depressurization.

Because the study of the response of the Calvert Cliffs plant to such SBLOCAs is preliminary, the core-melt estimate and risk developed below may require updating as more information becomes available.

### 3.1 INITIATING FREQUENCY FOR THE SBLOCA SCENARIO

The frequency of SBLOCAs of less than a 2 in. effective diameter and leakage greater than 132 gpm is estimated by ORNL at 1.5E-02/py (Ball et al. 1985, p. 5-5). This term includes both control-failure-initiated LOCAs, and LOCAs that are not initiated by control failure, such as those begun by SGTR. Given the current uncertainty in actual plant response to such SBLOCAs, ORNL further reduced the frequency of SBLOCAs by a factor of 10. A frequency of 1.5E-03/py then represents those SBLOCAs that may progress to core melt without operator action.

The Calvert Cliffs PRA (Hatch et al. 1982) uses an estimate of SBLOCAs of less than a 2 in. effective diameter (S2) of 1.0E-03/py. A factor of 10 reduction, such as that used by ORNL to reflect uncertainty in plant response, yields a frequency of interest of 1.0E-04/py. The significant increase in the assumed initiating frequency alone is expected to result in an increased estimate of core melt and risk due to such breaks.

The ORNL failure modes and effects analysis (FMEA) (Ball et al. 1985) identifies a number of control-related failures that may result in SBLOCAs, but no estimate of the frequency of such failures is made. NUREG-0844 (U.S. NRC 1985) sets the frequency of SGTR at 1E-02/py (leaving a remainder of 5E-03/py), indicating that other than control failures will likely dominate the 1.5E-02/py initiating frequency of SBLOCAs used by ORNL. The possibility of a reactor coolant pump seal (RCPS) failure in particular is mentioned, where seal failure could be caused by a normally open valve failing closed in the cooling water supply to the RCPS. If not corrected by operator action, this would lead to seal overheating and failure. No estimate of this frequency is given by ORNL.

Contributors to SBLOCA frequency were not, however, the focus of concern in the ORNL report. Rather, the plant response given a SBLOCA was identified as the potential problem. This analysis therefore focuses on plant response given a SBLOCA.

## 3.2 PROGRESSION TO CORE MELT FOR A SBLOCA

Recovery from a SBLOCA in the Calvert Cliffs plant requires the following system operations: 1) the containment spray system in injection (CSIS) and recirculation (CSRS) modes, 2) the containment air recirculation system (CARCS) to reduce containment pressure, 3) the emergency coolant injection system (ECIS), 4) the emergency coolant recirculation system (ECRS), and 5) the containment heat removal system (CHRS). These functions are shown in the LOCA event tree for Calvert Cliffs (see Figure 3.1) (Hatch et al. 1982, p.A1-15). The dominant SBLOCA sequences calculated in the Calvert Cliffs study are given in Table 3.1.

The frequency of S2 events was 1.0E-03/py in the Calvert Cliffs PRA, resulting in an effective conditional probability of core melt given an S2 SBLOCA of (9.5E-06/py)/(1.0E-03/py) = 9.5E-03/py.

A new scenario defined by ORNL, however, bypasses the above sequences and indicates failure of the HPI function unless depressurization can be achieved. The resulting fault tree is given in Figure 3.2. As shown, the operator has two pathways to depressurize the RCS that must fail in order to block operation of the HPI system. This includes manual operation of the PORVs and the atmospheric steam dump valves (ADVs) or turbine bypass valve. These valves provide increased cooling from the secondary side with a resultant primary side pressure reduction.

The probability of failure of the PORV pathway in Figure 3.2 is driven by the probability of failure of the operator to use the PORVs, estimated by ORNL at 0.5. This is consistent with failure probabilities of operators under stress for actions not called out specifically in emergency procedures. The potential for operator failure to terminate the overfeed is discussed in Section 4.5 of the ORNL report (Ball et al. 1985). ORNL cites an examination by the University of California at Los Angeles of operator response to a SBLOCA or overfill in the Seabrook and Oconee plants. It was recognized that applicability of results from such a study to Calvert Cliffs is tenuous; however, the results were cited as being of possible interest to the A-47 program. For Oconee, the probability for operator misdiagnosis of an overfill was estimated to range from 5E-02 within 15 minutes of the break, dropping to 7E-03 after 1 hour. This compares to ORNL's estimates of 1.0E-03 and 5E-01 operator error probability assumed for TBV and PORV use, respectively. The probability for failure of the ADV pathway is driven by the assumption that loss of instrument air pressure occurs with a probability of 1 during the accident, with only a 0.1 probability of operator recovery. This leads to an estimated probability of loss of this recovery pathway of 1.1E-02.

The estimated frequency of core melt due to a SBLOCA and failure to depressurize is then the initiating event frequency multiplied by the failure probability of the two depressurization pathways, or (1.5E-02/py)(1E-01)(1.1E-02)(5E-02) = 8.25E-06/py. From the ORNL fault tree, the effective conditional probability of core melt given a SBLOCA is then (1.1E-02)(5E-02) = 5.5E-03.

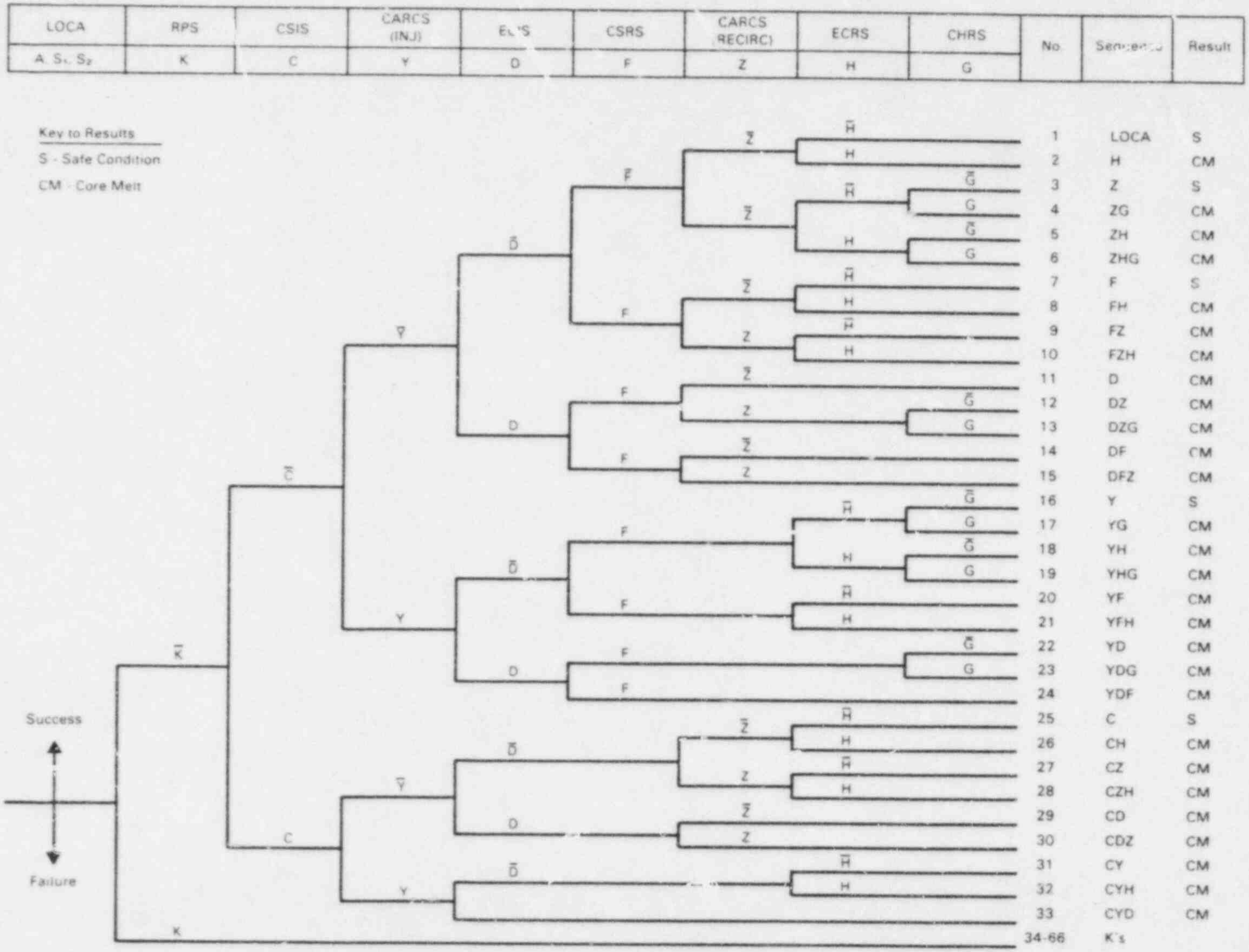


FIGURE 3.1. Calvert Cliffs LOCA Event Tree

TABLE 3.1. SBLOCA Core-Melt Sequences and Release Categories for Calvert Cliffs

| Sequence | Core-Melt Frequency, 1/py | Probability of Release Category | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | 1 | 2 | 3 | 4 | 5 | 6 | 7 |
| S2 H | 5.0E-06 | 1E-04 | | 7E-01 | | 7E-03 | | 3E-01 |
| S2 F H | 3.3E-06 | 1E-04 | 7E-01 | | 7E-03 | | 3E-01 | |
| S2 C Y | 1.0E-06 | 1E-01 | 1E+00 | | 7E-03 | | | |
| S2 Y G | 1.0E-07 | 1E-02 | 1E+00 | | | 7E-03 | | |
| S2 C D | 1.0E-07 | 1E-04 | 7E-01 | | 7E-03 | | 3E-01 | |
| TOTAL | 9.5E-06 | | | | | | | |

For S2 SBLOCAs the ORNL study effectively adds an additional branch to the SBLOCA recovery event tree originally developed in the Calvert Cliffs PRA (Hatch et al. 1982) (see Figure 3.2). The need for depressurization of the RCS precedes the demand of injection systems.

The resulting estimates of core-melt frequency are given in Figure 3.3. The core-melt frequency predicted is 8.25E-06/py. The original PRA SBLOCA recovery branch is shown in Figure 3.2 for comparison to the new ORNL sequence only. As shown, the new ORNL sequence is comparable to the original PRA estimate of risk from SBLOCA recovery, effectively increasing the risk of SBLOCAs by a factor of (9.5E-03 + 5.5E-03)/(9.5E-03) = 1.6.

## 3.3 PUBLIC RISK DUE TO SBLOCAs

This scenario then progresses to core melt given the failure to depressurize, resulting in failure of the HPI systems. This type of core-melt scenario is most closely modeled in the Calvert Cliffs PRA with loss of HPI by the S2CD sequence. Core melts as a result of such system failures are characterized by WASH-1400 release categories 1, 2, 4, and 6 (U.S. NRC 1985), with a probability distribution of 0.0001, 0.7, 0.007, and 0.3, respectively, as given in Table 3.2. Using the associated man-rem/release factors as presented in the Value/Impact Handbook (Heaberlin et al. 1983), the resulting estimate of public risk represented by this scenario is given below for a core-melt frequency of 9.0E-06/py (from Figure 3.3).

Total public risk is estimated at 2.82E+01 man-rem/py. No estimate of the upper bound for the initiating frequency was made by ORNL. This value will be used here as a best estimate. Assuming a 30 year plant life, this becomes 8.46E+02 man-rem over 30 years.

## 3.4 PRESSURIZED THERMAL SHOCK

The potential for PTS was estimated by ORNL as the frequency of a SBLOCA multiplied by the probability of operator failure to open the pressurizer PORVs, or (1.5E-02/py)(1E-02) = 1.5E-04/py. The conditional probability of

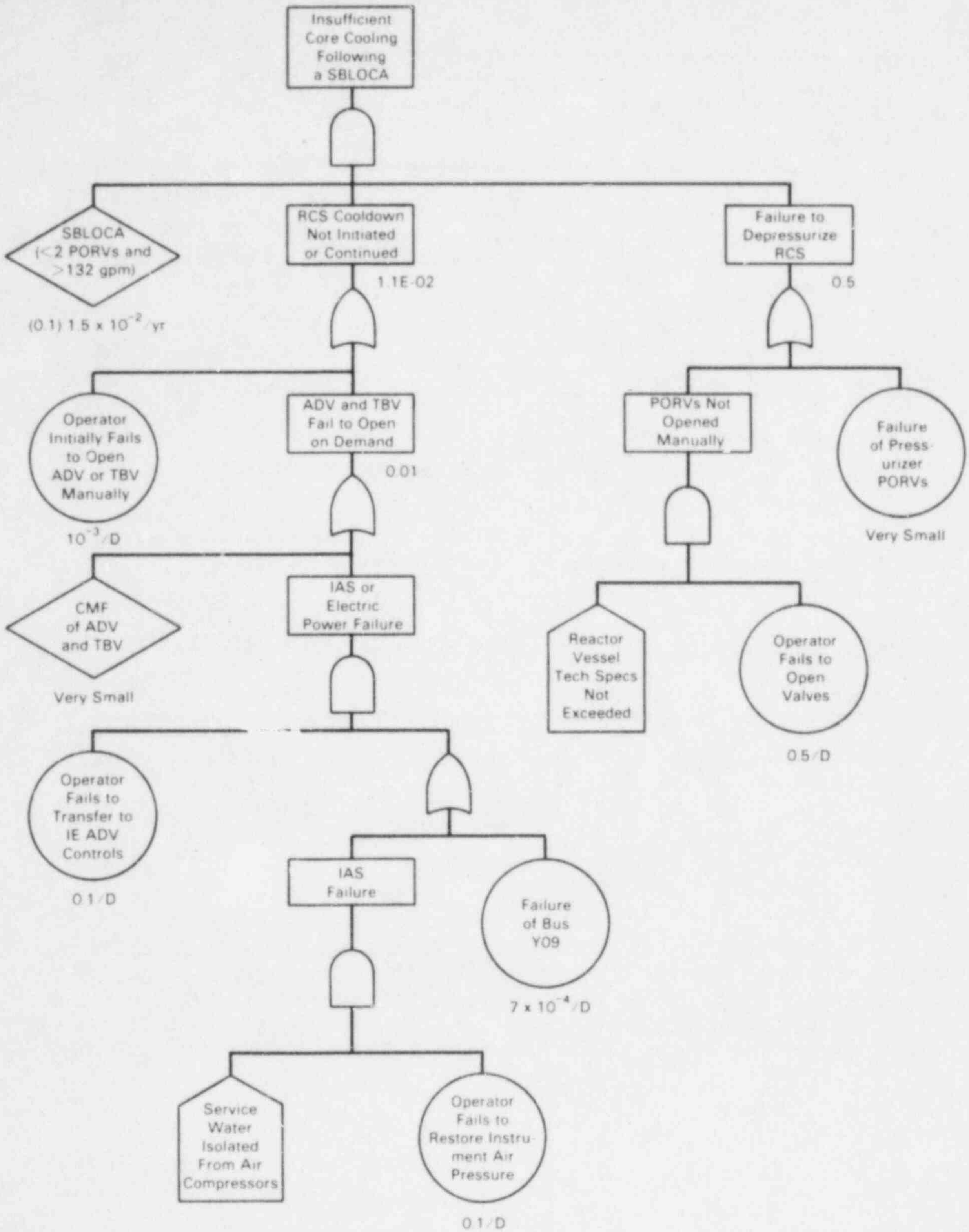


FIGURE 3.2. ORNL SBLOCA Depressurization Fault Tree (NUREG/CR-4265)

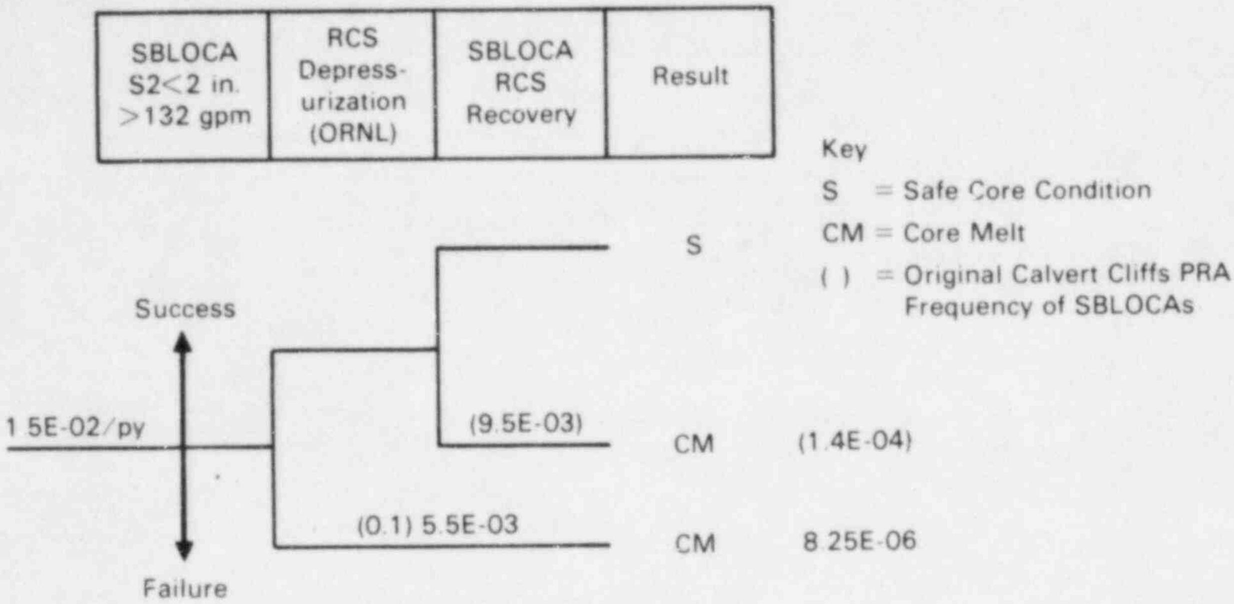


FIGURE 3.3. New Effective SBLOCA Event Tree for Calvert Cliffs

TABLE 3.2. Public Risk Associated with the ORNL-Identified SBLOCA

| Core-Melt Frequency, 1/py | Release Category | Probability | Man-Rem per Release | Man-Rem per py |
|---|---|---|---|---|
| 8.25E-06 | 1 | 0.0001 | 5.4E+06 | 4.46E-03 |
| | 2 | 0.7 | 4.8E+06 | 2.77E+01 |
| | 4 | 0.007 | 2.6E+06 | 1.50E-01 |
| | 6 | 0.3 | 1.5E+05 | 3.71E-01 |
| TOTAL | | | | 2.82E+01 |

vessel failure given the PTS sequency is then approximately 1E-04. The overall frequency of vessel failure and core melt is then 1.5E-08/py. Using PWR release category 1 at 5.4E+06 man-rem/event, this represents 8.10E-02 man-rem/py. This is considered an insignificant contributor to the total core-melt frequency and is not discussed further here.

# 4.0 VALUE/IMPACT ANALYSIS

This chapter presents several modifications to the plant, postulated to evaluate the potential cost and associated reduction in risk.

## 4.1 MODIFICATIONS TO REDUCE OVERFILL SCENARIO 1 FREQUENCY

For Overfill Scenario 1, ORNL identified failures (a) through (e) for the feedwater regulating valve (RV):

| | |
|---|---|
| a. RV fails as is with loss of instrument air, due to loss of power from 1Y09 | 3.5E-02/py |
| b. RV fails as is with loss of instrument air, due to a solenoid failing closed | 6.0E-03/py |
| c. RV fails mechanically | 6.0E-03/py |
| d. closure signal to RV fails due to I/P transducer failure | 3.4E-02/py |
| e. closure signal to RV fails due to a hand/auto station failure | 8.7E-03/py |
| | 8.97E-02/py |

Failures (a) through (e) are discussed in the sections that follow.

The additional factor of 0.1 for operator failure to terminate the overfill brought the sequence frequency to approximately 9E-03/py. Possible fixes to reduce the failure frequency are developed here. Table 2.3 shows that the risk estimates for Overfill Scenarios 1 and 2 are 18 and 0.87 man-rem/py, respectively, for a total of approximately 19 man-rem/py, or 570 man-rem over a 30-year plant life. This indicates that modifications would have to be limited to approximately $570,000 to achieve a value/impact ratio of 1 man-rem/$1000. Lowering the potential for MSLB given overfill from 5E-01 to 1E-03 lowers the estimated risk by a factor of (5E-01/1E-03) = 500, lowering the MSLB and SGTR contribution to that estimated for transient shutdown. The net effect is to lower the risk estimate by over 2 orders of magnitude. With this in mind, specific fixes addressing the above failures were developed and are presented below.

### 4.1.1 Vital Electric Power for Feedwater RV Air Solenoids

Loss of the 1Y09 power bus results in loss of power to the air control solenoid valve for the RV, causing the valve to fail as is. As it is now designed, the electrical circuit transfers vital load to 1Y09 on failure of the vital 1Y01 bus, but 1Y01 does not pick up non-vital loads on failure of the non-vital 1Y09 or 1Y10 buses. It is proposed here that the air control solenoids be added to the vital load of 1Y01, requiring the loss of both vital and non-vital buses for a power failure to impact the air solenoids.

This will be assumed to reduce failure (a) to the square of the reliability of one supply alone, or $(3.5E\text{-}02)^2 = 1.23E\text{-}03$, reducing the total failure rate to 5.59E-02/py for a reduction of 1 - (0.0559/0.0897) = 34.2% in the initiating frequency.

This would reduce core melt for Overfill Scenario 1 by (0.342)(3.8E-06/py) = 1.30E-06/py, and reduce public risk by (0.342)(1.8E+01 man-rem/py) = 6.2 man-rem/py, or 190 man-rem over 30 years.

The cost of such a modification depends greatly on whether the existing electrical equipment for the 1Y01 bus accepts the additional wiring and load represented by adding the air solenoids. Being a vital 1A power bus now, it is likely that a safety evaluation would also be required. The cost is then estimated at the following:

- \$25,000 for a safety evaluation
- 1 man-month or 4 man-weeks of additional QA at \$2270/week
- \$20,000 for supplies
- 1 man-week of engineering support at \$2270/week
- 1 man-week of craft services for installation at \$2270/week.

This gives a total of \$58,600, for a value/impact ratio of 190 man-rem/\$58,600 = 3.2 man-rem/\$1000.

### 4.1.2 Alternate Instrument Air

Failure of the air solenoid above causes the RV to again fail as is. This failure frequency could be reduced by adding an additional solenoid and air line to the valve to open on failure of the original solenoid. This air would be supplied by the existing instrument air supply. This could also be plumbed to an alternate air supply, but non-instrument air supplies tend to contain entrained oils that degrade air valve performance. The plant air (PA) supply would be a possible source, since it already is used as a backup to the 1A air supply.

Note that the failure of electric bus 1Y09 would again cause loss of air if the new solenoid is connected to the same 1Y09 bus as before. As a result, failure (a) would still exist. This modification is then assumed to reduce failure (b) to the square of the reliability of one solenoid alone, or $(6E\text{-}03)^2 = 3.6E\text{-}05$, reducing the total failure rate to 8.37E-02/py, for a reduction of 1 - (0.0837/0.0897) = 6.7% in the initiating frequency.

This would reduce core melt for Overfill Scenario 1 by (0.067)(3.8E-06/py) = 2.6E-07/py, and reduce public risk by (0.067)(1.8E+01 man-rem/py) = 1.2 man-rem/py, or 36.2 man-rem over 30 years.

The addition of one solenoid and logic to open the solenoid on loss of air pressure is assumed to be comparable to the above electric modification without the safety study, reducing costs by about half, to \$30,000. The value/impact ratio is then estimated at 36.2 man-rem/\$30,000 = 1.2 man-rem/\$1000.

### 4.1.3 Vital Electric Power to RV Plus Additional Air Solenoid

A modification for an additional air solenoid powered by the alternate vital electric bus is assumed to reduce both failure (a) and (b) above to the square of the reliability of one supply alone, or $(3.5E-02 + 6E-03)^2$ = 1.68E-03, reducing the total failure rate to 5.0E-02/py for a reduction of 1 - (0.05/0.09) = 44.4% in the initiating frequency.

This reduces core melt for Overfill Scenario 1 by (0.44)(3.8E-06/py) = 1.7E-06/py, and reduces public risk by (0.44)(1.8E+01 man-rem/py) = 7.9 man-rem/py, or 240 man-rem over 30 years.

If it is simply assumed that the cost of both modifications is the sum of each, or \$60,000 + \$30,000, the value/impact ratio is 240 man-rem/\$90,000 = 2.7 man-rem/\$1000.

### 4.1.4 Reliability of RV

The failure rate used by ORNL for the regulating valve is 6E-03/py, which corresponds to an hourly failure rate of approximately 1E-06/hr over 7008 hours per year at an 80% plant capacity. This is within the range used in WASH-1400 for air operated valves. Reliability improvement programs for the RV may include new equipment and testing programs. However, without data the new valves would be predicted to fail with essentially the current failure rate.

It is assumed that the reliability of the feedwater regulating valve can be improved to reduce the mechanical failure rate of the RV by at most 50%, using a combination of new equipment and testing. The failure rate would then be reduced to 3E-03/py, reducing the total failure rate to 8.67E-02/py, for a reduction of 1 - (0.0867/0.0897) = 3.3% in the initiating frequency.

This would reduce core melt for Overfill Scenario 1 by (0.033)(3.8E-06/py) = 1.3E-07/py, and reduce public risk by (0.033)(1.8E+01 man-rem/py) = 0.6 man-rem/py, or 18 man-rem over 30 years.

The cost of such a modification would have to be at or under \$18,000 to fall within the 1 man-rem/\$1000 range.

### 4.1.5 Alternate Circuit for Closure Signal to RV

This would require the addition of another I/P transducer where the operation of either transducer would propagate the RV closure signal. In addition, some modification to the hand/auto station would be required, the nature of which is uncertain at this time. The maximum possible reduction in the failure rate given a "perfect" station is calculated along with the risk reduction to estimate the maximum allowable costs of any modification while still giving appropriate value/impact ratios.

Modifications for failures (d) and (e) would be assumed to correct the failure of the signal to reach the RV, reducing the failure rate by (3.4E-02/py + 8.7E-03/py) = 4.27E-02/py and reducing the total failure rate to 4.7E-02, for a reduction of 1 - (0.047/0.09) = 47.8%.

This would reduce core melt for Overfill Scenario 1 by (0.478)(3.8E-06/py) = 1.8E-06/py and reduce public risk by (0.478)(1.8E+01 man-rem/py) = 8.6 man-rem/py, or 258 man-rem over 30 years.

The cost of such modifications to totally eliminate the I/P transducer failure and hand/auto station failure contribution to the scenario would then have to be limited to about $258,000 to achieve a value/impact ratio of 1 man-rem/$1000, with smaller costs giving a higher ratio.

## 4.2 MODIFICATIONS TO REDUCE THE FREQUENCY OF OVERFILL SCENARIO 2

For Overfill Scenario 2, ORNL identified the following failures, which result in the feedwater valve (FWV) not closing on turbine trip, with the regulating valve also open:

| | | |
|---|---|---|
| a. | closure signal to FWV fails due to OR gate failure | 5.2E-03/demand |
| b. | closure signal to FWV fails due to relay failure to close | 5.7E-04/demand |
| c. | closure signal to FWV fails due to cable failure | 2.9E-03/demand |
| | | 8.67E-03/demand |

The failure of the RG valve to close on demand or fail at power is 0.505/yr and the operator failure probability was again 0.1, for an initiating frequency of (8.67E-03/demand)(0.1/demand)(0.505/yr) = 4.38E-04/py, or approximately 4.4E-04/yr.

### 4.2.1 Additional OR Gate in Parallel

The addition of another OR gate to the circuit is assumed to reduce failure (a) to the square of the reliability of one gate alone, or (5.2E-03) = 2.70E-05, reducing the total failure rate to 3.59E-03/demand, for a reduction of 1 - (0.00359/0.00867) = 58.6% in the initiating frequency.

This would reduce core melt for Overfill Scenario 2 by (0.586)(1.8E-07/py) = 1.05E-07/py, and reduce public risk by (0.586)(0.87 man-rem/py) = 0.51 man-rem/py, or 15.3 man-rem over 30 years.

The cost of such a modification is expected to be minimal. The addition of one more logic element is estimated at $10,000 for equipment and materials, with the total costs as follows:

- 1 man-week for additional QA at $2270/week
- $10,000 for supplies
- 1 man-week of engineering support at $2270/week
- 1 man-week of craft services for installation at $2270/week.

This yields a total of $16,800, for a value/impact ratio of 15.3 man-rem/$16,800 = 0.9 man-rem/$1000.

### 4.2.2 Additional Relay in Parallel

The addition of another relay to the circuit is assumed to reduce failure (b) to the square of the reliability of one relay alone, or (5.7E-04) = 3.25E-07, reducing the failure rate to 8.19E-03/demand, for a reduction of 1 - (0.00819/0.00867) = 5.5% in the initiating frequency.

This would reduce core melt for Overfill Scenario 2 by (0.055)(1.8E-07/py) = 9.90E-09/py, and reduce public risk by (0.055)(0.87 man-rem/py) = 4.8E-02 man-rem/py, or 1.4 man-rem over 30 years.

The cost of such a modification is estimated to be about the same as that of the additional OR gate, or $16,800, for a value/impact ratio of 1.4 man-rem/$16,800 = 0.08 man-rem/$1000.

### 4.2.3 Modifications to Reduce Cable Failure

Modifications to reduce the contribution to cable failure identified by ORNL would contribute at most a reduction of 2.9E-03 in failure on demand, reducing the failure rate to 5.77E-03/demand for a reduction of (1 - 0.00577/0.00867) = 33.4% in the initiating frequency.

This would reduce core melt for Overfill Scenario 2 by (0.334)(1.8E-07/py) = 6.01E-08/py, and reduce public risk by (0.334)(0.87 man-rem/py) = 2.9E-02 man-rem/py, or 8.7 man-rem over 30 years.

The cost of such modifications to totally eliminate the cable failure contribution to the scenario would then have to be limited to about $8700 to achieve a value/impact ratio of 1 man-rem/$1000. It is highly unlikely that additional cable runs of any appreciable length could be installed for costs at or under $8700.

## 4.3 MODIFICATIONS TO TERMINATE OVERFILL

The other approach to accident recovery from overfill is to implement modifications to terminate the overfill after its initiation. This effectively terminates both Overfill Scenarios 1 and 2, and can thus consider the risk contribution from both. The 570 man-rem total over 30 years again indicates that modifications at or below a cost of $570,000 will have value/impact ratios larger than 1 man-rem/$1000. Considerations for manual or automatic termination are developed below.

### 4.3.1 Manual Termination of Main Feedwater Flow

The ORNL FMEA indicates that the operator may take corrective action to trip the MFW pumps and/or close isolation valves from the MFW system to the steam generator. This would effectively reduce or prevent any flow through the failed regulating valve. Note, however, that the estimate of propagation to core melt calculated in a previous chapter already took into account an operator probability of terminating the overfill, using the above possible actions with the current design. Any reductions in risk associated with operator action would therefore require some measure of improved performance by the operator in terminating MFW flow. Performance could be improved to some degree with new emergency procedures and operator training. However, operator performance in transients is already the subject of specific post-Three Mile Island action items and almost certainly would include overfill transients in general. The A-47 program would then be expected to insure that proper recognition of the unique aspects of the overfills identified by ORNL, if any, would be included in the appropriate operator training program already under way. Therefore, no additional risk reductions or costs associated with this option are estimated here.

### 4.3.2 Automatic Main Feedwater Pump Trip

Another option is to modify the Calvert Cliffs design to allow for automatic isolation of the MFW flow on high SG level. Many plants now have MFW trip and turbine trip on a high SG water level. This could easily be added to the Calvert Cliffs design using the existing instrumentation to generate the high level signal, requiring only the addition of a pump trip relay and signal cables. This cost is expected to be minimal. If the trip function is required to be safety grade, additional expenditures would be required for safety studies, QA records and equipment, and possibly technical specification licensing amendments.

The basic installation is estimated to have a minimal cost of:

- 1 man-week of engineering support at \$2270/week (Heaberlin et al. 1983)
- 1 man-week of craft services for installation at \$2270/week
- \$10,000 for the relay and supplies, for a total of \$14,540.

If a safety grade modification is considered, as it is with many plant feedwater systems, additional costs associated with a license amendment and safety studies would likely be incurred, in addition to the added QA and possibly more expensive components. The NRC staff effort associated with a "typical," uncomplicated technical specification change is characterized as:

- 2 man-weeks of technical staff and 1 man-week of management and legal review to issue a generic letter and model technical specification changes to all impacted licensees

* 1.5 man-weeks of technical staff and 1 man-week of management and legal review for review of licensee's response and preparation of a preliminary significance and hazard analysis report, for a total of 2.5 man-weeks, plus $600 to publish in the Federal Register

* 2.5 man-weeks of technical staff and 1 man-week of management and legal review after 30 day comment period for preparation of final license amendment, plus $200 to publish in the Federal Register.

Thus the total NRC licensing amendment cost of 9 man-weeks at $2270/week, or $20,430 + $800 in publishing costs, is a total of $21,230.

Utility costs are estimated to be about:

* $50,000 for a safety evaluation
* 2 man-months or 8 man-weeks of additional QA at $2270/week
* $20,000 for the relay and supplies
* 2 man-weeks of engineering support at $2270/week
* 2 man-weeks of craft services for installation at $2270/week.

This totals $97,240, for a total NRC and utility cost of $118,470. Additional penetrations, cable runs, and cabinet space could dramatically increase the costs.

The failure probability of such an automatic trip function can be estimated at 1E-03/demand, effectively eliminating the core-melt frequency and risk represented by the overfill scenarios calculated earlier. From Table 2.3, this provides a core-melt frequency reduction of 4.0E-06/py and a risk reduction of 1.9E+01 man-rem/py, for a total of 570 man-rem over 30 years.

The value/impact ratio is then 570 man-rem/$120,000 = 4.75/$1000.

#### 4.3.3 Automatic Isolation of Feedwater Delivery Lines Plus MFW Pump Trip

The other method of preventing MFW flow from progressing to spillover after turbine trip is to use the existing motor-operated isolation valves on the feedwater delivery lines. This modification can be installed separately or included with the above MFW turbine trip modification. If installed separately, the cost would be similar to the MFW trip cost of $120,000. If included in the modification with the MFW trip, this could "piggyback" on all safety studies and license amendment fees with minimal additional cost. Adding an additional $20,000 to the safety study cost and $20,000 for a safety grade relay for isolation valve closure yields a cost of $160,000.

The risk reduction would again be 1.9E+01 man-rem/py, for a total of 570 man-rem over 30 years, with the cost increased to $160,000, giving a value/impact ratio of 570 man-rem/$160,000 = 3.56 man-rem/$1000.

#### 4.3.4 Limitations

It must be pointed out that any comparison between the current Calvert Cliffs design and proposed modifications is highly dependent on a number of factors, including basic hardware reliability and operator response to system failures. For the operator, this can include plant-specific differences in and compensations for a number of factors, including:

- type of level control (three element, one element)
- power supplies
- backup or alternate level displays
- instrument line plumbing configuration
- controlling level display
- controlling level record
- annunciators and alarms
- operator training and procedures
- maintenance, general age, and state of equipment.

A probability of operator failure to terminate the overfill is estimated at $\leq$0.1. Systems that rely more heavily on the operator for detection and correction of failures may also emphasize level display and operator training and procedures. Many of the variables influencing operator performance of high water level trip actions in the Calvert Cliffs plant are not fully defined at this time.

The potential for MSLB (0.5) and subsequent SGTR leading to core melt may also introduce significant conservatism into the above estimates. If a 1E-03 factor for MSLB is used, as it is in some studies, the risk estimated for MSLB and SGTR drops to a level comparable to that of transient shutdown without the PCS. The net effect is a lowering of risk by over 2 orders of magnitude, effecting eliminating any public risk due to overfill. Therefore, the results above must be taken as only a preliminary review of the potential impact of other feedwater control configurations on the A-47 issue, and these recommendations should undergo detailed evaluation.

## 4.4 MODIFICATIONS TO REDUCE THE PROBABILITY OF OPERATOR FAILURE TO DEPRESSURIZE THE RCS

ORNL did not specify the frequency of any possible contributing control system failures in the (0.1)(1.5E-02/py) figure for SBLOCAs. As a result, the (0.1)(1.5E-02/py) frequency is interpreted as a simple initiator frequency and is used below to evaluate modification to the new ORNL RCS depressurization sequence. The entire sequence was estimated to contribute a risk of 8.46E+02 man-rem per plant over 30 years. As a result, an upper bound of $846,000 could be put on expenditures to reduce or eliminate the sequence to keep the value/impact ratio under 1 man-rem/$1000.

Given the occurrence of a SBLOCA, the ORNL FMEA indicates that a steam generator isolation signal will isolate the service water supply from the instrument air and plant air compressors. Restoration of the service water supply is currently called out in the plant's emergency procedures. However, significant delay may occur before these procedures are initiated; ORNL considers the loss of the instrument air system (IAS) very likely during this delay period. A high probability (1.0) was thus assumed for failure of the IAS and hence loss of control of the TBV and ADVs during the early stages of the accident. Potential modifications are developed here.

### 4.4.1 Reduced Operator Error in IAS Recovery

Modifications can focus on decreasing the failure probability of operator restoration of the IAS, currently at 0.1 in Figure 3.2. ORNL indicates that operator actions to initiate RCS cooldown begin approximately 1 hour after the SBLOCA. Restoration of service water to the air compressors, however, is not addressed until Step 20 of the procedure. It is suggested that the procedure be rewritten to address the need for a functioning air supply for valve control earlier in the procedure. If insufficient air supply remains in the accumulators, the service water flow and air compressor operation would be accomplished before valve operation is attempted.

This is estimated to reduce to 0.01 the operator error probability of failure to restore instrument air pressure (see Figure 3.2). The conditional probability of core melt via the ORNL path then becomes (3.0E-05)(0.5) = 1.5E-3. From Figure 3.3, the estimated core melt is then 2.25E-06/py, for a reduction of 6.0E-06/py, or 72.7%. The risk reduction is then (0.727)(2.82E+01 man-rem/py) = 2.05E+01 man-rem/py, or 6.15E+02 man-rem over 30 years.

The potential risk reduction thus indicates that up to $6.15E+05 could be spent implementing the new emergency procedure, with actual costs estimated to be significantly less than this amount.

### 4.4.2 Operator Procedures to Depressurize the RCS with the PORVs

The PORVs can also be used to depressurize the RCS. The probability of operator failure to do this is 0.5 in the ORNL fault tree, due to the lack of specific procedures. If PORVs were used, the operator error could drop to 0.01 or 0.001, on par with the assumed failure probability to use the ADV or TBV. Using 0.001, the new conditional probability of core melt from Figure 3.2 would be 1.1E-5, for a core-melt frequency in Figure 3.3 of (1.5E-03/py)(1.1E-05) = 1.65E-08/py, for a reduction of 8.23E-06/py, or 99.8%. The risk reduction would then be (0.998)(2.82E+01 man-rem/py) = 2.81E+01 man-rem/py, or 8.44E+02 man-rem over 30 years.

Costs could approach $844,000 and still fall under the 1 man-rem/$1000 criteria.

### 4.4.3 Implementation of IAS Recovery and PORV Procedures

If both procedural changes are implemented at once, the new ORNL branch in Figure 3.3 would essentially be eliminated, for a core-melt reduction of 8.25E-06/py and a risk reduction of 8.46E+02 man-rem over 30 years. This again indicates that costs of up to $8.46E+05 could be used during implementation. Again, costs for procedural changes would be expected to be significantly below this value.

### 4.4.4 Removal of SWS Isolation on SGIS Signal

The service water system (SWS) is presently isolated given a SBLOCA via a steam generator isolation system (SGIS) signal. It is proposed that this action be eliminated to keep the SWS on-line and delivering cooling water to critical components such as air compressors. The failure probability of the ADV or TBV to function on demand is then thought to drop to approximately 1E-04/demand. The RCS cooldown failure probability would then become 1.1E-03/demand, and the conditional core-melt probability would become (2.1E-03)(0.5) = 1.05E-03. This represents a core-melt reduction of 90%. The risk reduction then is (0.90)(2.82E+01 man-rem/py) = 2.54E+01 man-rem/py, or 7.6E+02 man-rem over 30 years. Costs could then approach $760,000.

### 4.4.5 Modifications to SWS After Implementation of PORV Procedures

If the operator emergency procedure for use of the PORV in case of TBV or ADV failure is considered to already be implemented, the effectiveness of implementing a PORV procedure change afterward would be reduced. The conditional probability of core melt for the ORNL branch would then be (1.1E-02) (0.001) = 1.1E-05/py. This leaves only 2% of the original risk, or 2 man-rem over 30 years. The new ORNL conditional probability of core melt after implementing the service water isolation change would then be (1.1E-03)(0.001) = 1.10E- 06/py, for a further reduction of 90%, or (0.9)(2 man-rem) = 1.8 man-rem.

This indicates that if the PORV emergency procedures are instituted first, a large contribution of the estimated risk due to this new scenario would be eliminated. The same argument could be applied, however, to using PORV changes after SWS changes. The analysis above indicates that both modifications need to be considered.

## 4.5 MODIFICATION TO REDUCE CORE MELT FROM SBLOCA CONTRIBUTIONS

The original S2 SBLOCA frequency in the Calvert Cliffs PRA is 1.0E-03/py, compared to the 1.5E-02/py assumed by ORNL. Again, a frequency of (0.1) (1.5E-02/py) = 1.5E-03/py was assumed for the SBLOCA of interest for the undercooling scenario. The contribution of this initiating frequency attributable to control failures was not estimated by ORNL. However, in Chapter 3 an estimate of the potential frequency of RCP seal failure due to control failure was made. In an attempt to bound the contribution of control failures to SBLOCAs, the frequency of SGTR events is typically 1E-02/py alone, indicating that new control-related SBLOCAs would likely be bounded by 1.5E-02/py - 1E-02/py = 5E-03/py.

Again, the conditional probability of core melt given a SBLOCA in the Calvert Cliffs PRA was 9.5E-03/event, with the risk primarily associated with PWR release categories 2 (70%) and 6 (30%). The new ORNL sequence requiring depressurization before HPI response gives another conditional probability of core melt given the SBLOCA of (0.1)(1.1E-02)(0.5) = 5.5E-04/event, with the

0.1 factor again reflecting the percentage of SBLOCAs likely to elicit this plant response. This gave a total conditional probability of core melt of 1.01E-02/event.

The core-melt frequency due to control-failure-induced SBLOCAs would then be estimated at (5E-03/py)(1.01E-02 core-melt/event) = 5.05E-05/py. The risk would then be approximately (5.05E-05/py)(0.7)(4.8E+06 man-rem/event) = 1.7E+02 man-rem/py, or 5.1E+03 man-rem over 30 years. This indicates that costs associated with control-failure-induced SBLOCAs would be limited to $5,100,000 to have a value/impact ratio of 1 man-rem/$1000 or higher.

If the modifications discussed in Section 4.4 are implemented first, the contribution to risk of the new ORNL depressurization sequence is significantly reduced. The core-melt frequency due to SBLOCAs with a frequency of 5E-03/py would then be reduced to (5E-03/py)(9.5E-03 core melt/event) = 4.75E-05/py. The risk would then be (4.75E-05/py)(0.7) (4.8E+06 man-rem/event) = 1.6E+02 man-rem/py, or 4.8E+03 man-rem over 30 years. Eliminating the new ORNL sequence from consideration would lower the costs to reduce control-induced SBLOCAs slightly, to $4.8E+06, to have a value/impact ratio of 1 man-rem/$1000 or higher. This again assumes a control-failure-induced SBLOCA frequency of 5E-03/py for control-failure-induced SBLOCAs.

Preliminary indications are that the risks associated with control-failure-induced SBLOCAS could be significant if they represent a sizable fraction of the SBLOCA initiating frequency assumed by ORNL. Further work defining this potential source of SBLOCAs may be warranted.

## 4.6 SUMMARY OF VALUE/IMPACT

The proposed modifications to the Calvert Cliffs plant and the associated reduction in risk, estimated cost, and value/impact are summarized in Table 4.1. This is not a complete list of the failure modes and possible fixes resulting from the ORNL examination of control system failures in the Calvert Cliffs CE plant. The ORNL study is quite extensive, with a number of failure mechanisms identified that may contribute additional safety issues. However, the scenarios examined here are thought to represent those failures of greatest safety concern, and the fixes summarized below address those scenarios directly.

TABLE 4.1. Summary of the Value/Impact Analysis for the Calvert Cliffs CE Plant

| Proposed Fix | Estimated Cost, $ | Estimated Risk Reduction (man-rem) | V/I Ratio (man-rem/$1000) |
|---|---|---|---|
| Overfill Scenario 1: | | | |
| RV Solenoids on vital power | 6.0E+04 | 190 | 3.2 |
| Alternate instrument air supply | 3.0E+04 | 36.2 | 1.2 |
| Improved RV reliability | <1.8E+04 | 18 | 1[a] |
| Additional RV closure circuit | <2.58E+05 | 258 | 1[a] |
| Overfill Scenario 2: | | | |
| Additional OR gate in parallel | 1.6E+04 | 15.3 | 0.9 |
| Additional relay in parallel | 1.6E+04 | 1.4 | 0.08 |
| Improved cable reliability | <8.7E+03 | 8.7 | 1[a] |
| Overfill Scenarios 1&2: | | | |
| Automatic high level MFW pump trip | 1.2E+05 | 570 | 4.75 |
| Automatic isolation of feedwater lines on high level plus MFW pump trip | 1.6E+05 | 570 | 3.56 |
| Modifications to Reduce Failure of RCS Depressurization: | | | |
| Reduced operator error in IAS recovery | <6.15E+05 | 6.15E+02 | 1[a] |
| Reduced operator error in PORV RCS depressurization | <8.44E+05 | 8.44E+02 | 1[a] |
| Implementation of both IAS recover and PORV procedures | <8.46E+05 | 8.46E+02 | 1[a] |
| Elimination of SWS isolation given SGIS | <7.6E+05 | 7.6E+02 | 1[a] |

(a) Specifics of a modification or associated costs are uncertain. Calculation of risk reduction in man-rem was used to estimate maximum dollar expenditure compatible with a $1000/man-rem criteria.

## 5.0 CONCLUSIONS

The results of the consideration of core-melt potential for control system failures in the Calvert Cliffs CE PWR are summarized in Table 5.1. The failure scenarios examined were those identified by ORNL (Ball et al. 1985) as being of principal importance. The subjective judgment of which sequences to analyze was made from an extensive review of control system failures and possible interactions identified by ORNL. This examination is thus not meant to represent an exhaustive study of all failure modes and associated risks in the Calvert Cliffs plant, but does represent a risk study of those failures thought to present the most serious safety concern to the A-47 program at this time.

TABLE 5.1. Summary of ORNL and PNL Estimates of Accident Initiator Frequencies, Core-Melt Frequencies, and Public Risk for the CE Calvert Cliffs PWR

| Sequence | Frequency, 1/py | Core-Melt Frequency, 1/py (best estimate) | Public Risk, man-rem/py (best estimate) |
|---|---|---|---|
| Overfill Scenario 1: | | | |
| Transient shutdown | (0.009)(0.1) | 5.7E-09 | 2.2E-02 |
| Overfill & MSLB | (0.009)(0.5) | 5.0E-08 (core damage) | 1.9E-01 |
| SGTR | (0.009)(0.5)(0.034) = 1.5E-04/py | 3.7E-06 | 1.8E+01 |
| | | 3.8E-06 | 1.8E+01 |
| Overfill Scenario 2: | | | |
| Transient shutdown | (4.4E-04)(0.1) | 2.8E-10 | 1.1E-03 |
| Overfill & MSLB | (4.4E-04)(0.5) | 2.4E-09 (core damage) | 9.1E-03 |
| SGTR | (4.44E-04)(0.5)(0.034) = 7.5E-06/py | 1.8E-07 | 8.6E-01 |
| | | 1.8E-07 | 8.7E-01 |
| SBLOCA Scenario: | | | |
| Inadequate cooling | 1.5E-02/py(0.1) | 8.25E-06 | 2.82E+01 |
| PTS | 1.5E-02/py(0.01) | 1.5E-08 | 8.1E-02 |
| TOTAL | | 1.22E-05 | 4.72E+01 |

## 5.1 CALVERT CLIFFS RESPONSE TO OVERFILL

The ORNL FMEA identified a number of failure modes that can result in the feedwater regulating valve failing to close, having received or not received the turbine trip signal. These two cases constituted Overfill Scenarios 1 and 2, with initiating frequencies of 9.0E-03/py and 4.4E-04/py, respectively, including operator failure to terminate the overfill.

For these two scenarios, propagation of damage due to water in the steam lines was considered, including damage to the power conversion system (PCS) function of the feedwater system and condenser; damage to the steam lines, causing main steam line break (MSLB); and MSLB propagating to steam generator tube rupture (SGTR). Assuming the overfill event constitutes transient shutdown with loss of the PCS, the core-melt frequency and associated risk are minimal, about 5E-09/py for Overfill Scenario 1 and 2E-09/py for Overfill Scenario 2. Assuming a 0.5 probability of MSLB given overfill, the frequency of core damage when equated with core melt is a factor of 10 higher than that for transient shutdown, but is still minimal. Note that MSLB was not equated with core melt in the WASH-1400 Reactor Safety Study (U.S. NRC 1975), indicating that such an assumption gives a highly conservative measure of the risks associated with MSLB.

Given MSLB, the potential for inducing an SGTR is 0.034, based on the considerations in the steam generator tube integrity program (NUREG-0844; U.S. NRC 1985). The MSLB was further assumed to occur with a 50% probability above or below the main steam isolation valve (MSIV), resulting in a conditional probability of recovery from the SGTR of 2.44E-02, again based on NUREG-0844 scenarios. Note that this is only slightly higher than the 9.5E-03/event conditional probability of core melt given a 2 in. diameter SBLOCA used in the Calvert Cliffs PRA. A higher conditional probability of core melt would be expected, given the aggravating MSLB.

The propagation to SGTR was found to constitute the highest estimate of core-melt frequency and risk, at 3.8E+06 core melt/py and 18 man-rem/py, or 540 man-rem over 30 years for Overfill Scenario 1. The core-melt estimate for Overfill Scenario 2 was lower by more than a factor of 10 due to the assumed initiating frequency. The cost of modifications could then approach approximately $570,000, considering both scenarios, and still give a value/impact ratio of 1 man-rem/$1000. Many of the contributing failures to these scenarios are thought to cost significantly less to implement, thus giving favorable value/impact ratios.

Note, however, that this includes the conservative potential for MSLB and a 0.1 operator error factor. More realistic estimates could easily reduce the core-melt and risk estimates by several orders of magnitude, thus essentially eliminating any serious public risk.

The risk presented by the overfill scenarios, given the above conservative analysis, is sufficient to consider the need for modifications. The value/impact indicates that modifications to reduce the feedwater regulating valve failure would be cost-effective, but that modifications to provide another means of isolating feedwater flow are more effective in terminating the scenario and are probably at least as cost-effective. The Calvert Cliffs plant is apparently lacking an MFW pump trip on high steam generator water level. The current design provides a main turbine trip and throttling of the feedwater regulating valve, but does not trip the steam driven pump itself or isolate the feedwater delivery lines. The addition of either modification would significantly reduce the probability of an overfill progressing to steam line damage, with implementation costs thought to be under $570,000.

## 5.2 CALVERT CLIFFS RESPONSE TO SBLOCAs

The ORNL failure modes and effects analysis (FMEA) of SBLOCAs in the Calvert Cliffs plant (Ball et al. 1985) has identified control failure contributions to SBLOCAs as well as a possible new pathway to core melt given a SBLOCA. The frequency of SBLOCA initiators was not estimated by ORNL. PNL has, however, made an initial estimate of such failures based on a review of the ORNL FMEA and other relevant sources.

For SBLOCAs in the Calvert Cliffs plant, the ORNL analysis has identified a range of SBLOCAs less than 2 in. diameter but with a leakage rate greater than 132 gpm (designated S2) that apparently requires operator action to depressurize the reactor coolant system (RCS) before high pressure injection (HPI) operation is possible. Operator recovery is further hindered by the identification of a likely loss of instrument air leading to the inoperability of turbine bypass and steam dump valves normally used for depressurization. The PORV is still available but is not currently addressed in emergency procedures. A frequency of 1.5E-02/py for SBLOCAs was used, which was further reduced by a factor of 10 to reflect those initiators that would actually lead to this scenario.

The new ORNL sequence of operator failure to use the turbine bypass valves (TBVs) or atmospheric dump valves (ADVs) and the power-operated relief valves (PORVs) yields a conditional core melt probability, given the SBLOCA, of (1.2E-02)(0.5) = 6.0E-03/event.

The PNL study then examined the risk contribution from all SBLOCAs leading to core melt via the new ORNL sequence, noting that the risk associated with the original core-melt scenario in the Calvert Cliffs PRA has already been recognized. The initiating frequency is (0.1)(1.5E-02/py) by ORNL, giving a core-melt frequency of (0.1)(1.5E-02/py)(5.5E-03) = 8.25E-06/py.

The dominant release category for SBLOCA core-melt scenarios in the Calvert Cliffs PRA is release category 2 (70%), with risk at 4.8E+06 man-rem/event. Using these same release categories, the risk represented by the new ORNL sequence is (8.25E-06/py)(0.7)(4.8E+06 man-rem) = 2.77E+01 man-rem/py. Considering other release categories brings the total to 2.82E+01 man-rem/py, or 8.46E+02 man-rem over 30 years. This indicates that costs associated with modifications to reduce or eliminate this scenario may approach $846,000 and may maintain a value/impact ratio of 1 man-rem/$1000 or higher.

The value/impact analysis indicates that one of the most effective fixes is to simply include PORV operation in the emergency procedures, reducing the assumed operator error from 5E-01 to 1E-03. Other fixes can reduce the probability of loss of instrument air and hence increase the availability of the turbine bypass or steam dump valves, bringing the availability of these valves up to that of the PORV.

The other option would be to automatically depressurize the RCS to below the HPI upper head limit given a SBLOCA. However, the potential for inadvertent PORV lift and sticking requires operator attention in any case. Because both manual or automatic actuation would still benefit from close operator supervision, it is thought that simple manual PORV operation in the emergency procedures is the best option at this time.

However, the risk associated with this scenario is probably sufficient that modifications to improve the availability of the turbine bypass and steam dump valves could be justified, in addition to inclusion of the PORV to emergency procedures.

## 5.3 CONTROL FAILURE CONTRIBUTION TO SBLOCAs

The A-47 program may also want to consider that fraction of SBLOCAs induced by control-related failures. Both the new ORNL sequence and original core-melt pathway defined in the Calvert Cliffs PRA would be of interest in measuring the risk represented by these SBLOCAs.

Failure modes leading to these SBLOCAs were identified by ORNL for Calvert Cliffs; however, no estimate of the frequency of such events was given. An example included control failures resulting in valve closure and loss of component cooling water to the RCP seals, identified in the ORNL FMEA (Ball et al. 1985). The contribution to risk from control-system-induced failures could be significant, however, even with only a small fraction of the assumed SBLOCA frequency of 1.5E-02/py. The ORNL report did not associate control failures contributing to SBLOCAs with a new major concern. The need to examine SBLOCA contributors may still be of interest to the A-47 program, however.

## 5.4 SUMMARY

The current estimates of core-melt frequency and public risk associated with the control-related failures identified by ORNL are 1.2E-05/py and approximately 47 man-rem/py, respectively. These estimates are dominated by the SBLOCA contribution estimated here, with risk associated with steam generator overfill of less concern at this time.

These results compare to the overall core-melt frequency for the Calvert Cliffs plant of 2.0E-03/py (Hatch et al. 1982), which is primarily associated with transient sequences. The overfill scenario leading to spillover and the SBLOCA scenarios thus represent a relatively small additional fraction to risk. The risk is still significant, however.

The risk represented by SBLOCA response in the Calvert Cliffs plant warrants serious consideration of plant modifications to assume adequate reliability of depressurization options and HPI function.

The need for high water level feedwater trips in the steam generator is less certain. The analysis hinges on the assumed high probability of steam line damage and rupture given overfill, which is justified given the current uncertainty involving this potential. Unless the integrity of the steam lines can be assured to a greater level, main feedwater trip may also be necessary in Calvert Cliffs.

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Pacific Northwest Laboratory (PNL) performed a probabilistic risk analysis to develop estimates of core-melt frequency and public risk associated with control system failures in a Combustion Engineering pressurized water reactor. Value/impact analyses of possible modifications to prevent control system failures were also conducted. These analyses were based on a failure modes and effects analysis previously conducted at Oak Ridge National Laboratory. The control system failure modes fall into three main scenarios: two scenarios concern overfill of the steam generators, progressing to spillover into the steam lines. The third scenario deals with small-break loss-of-coolant accidents that may require operator action to depressurize the reactor coolant system. The analyses described in this report were performed in support of the U.S. Nuclear Regulatory Commission's program for Unresolved Safety Issue A-47, Safety Implications of Control Systems.

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