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Estimation of Risk Reduction From Improved PORV Reliability in PWRs

Final Report

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Prepared for
U.S. Nuclear Regulatory
Commission

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ABSTRACT

An analysis was performed to explore the risk reduction potential of improving the pressurizer PORV and block valve reliability for two representative PWR plants, Indian Point 3, and Oconee 3. Attention was focused upon particular transient scenarios including steam generator tube rupture (SGTR), a stuck-open PORV, cooldown to cold shutdown, and the use of PORV in low-temperature overpressurization (LTOP) protection. The feasibility of using the PORV as a high point vent to supplement the function of Reactor Vessel Head Vent System (RVHVS) was also studied.

The pertinent event trees, fault trees, and the basic data presented in the Indian Point Probabilistic Safety Study (IPPSS) and the Oconee PRA were utilized to quantify the benefit of an improved PORV and block valve reliability in terms of potential reduction in core-melt frequencies. For Indian Point 3, independent core-melt frequency calculations were made based on the Boolean expressions derived for various plant damage states. For Oconee 3, the components of the dominant cut sets and the detailed fault trees shown in the Oconee PRA were given thorough scrutiny to determine their relevance to the hardware or operational failures of the PORV and its block valve.

With the exception of LTOP, the core-melt frequencies attributable to PORV or block valve failures were found to be relatively insignificant, only a very small fraction of the total core-melt frequency due to internal events.

For the case of LTOP, the core melt frequency and associated risk appear to be small for Indian Point 3 and Oconee 3 since the vessels have not had a substantial fraction of their estimated lifetime irradiation. The results of a conservative estimation of health effect, however, indicate that the public risk from LTOP events may become more significant late in plant life when the aging effects of the vessel contribute to increased vulnerability. These effects are being studied in more depth in the NRC's studies for Generic Issue Number 94.

TABLE OF CONTENTS

Page

ABSTRACT.......... iii
LIST OF FIGURES.......... vi
LIST OF TABLES.......... vii
ACKNOWLEDGEMENT.......... viii
EXECUTIVE SUMMARY.......... ix

1. INTRODUCTION.......... 1-1
   1.1 Scope of This Report.......... 1-1
   1.2 Summary of Findings.......... 1-2

2. CORE DAMAGE SCENARIOS AND ESTIMATION OF THE BENEFITS OF AN IMPROVED PORV AND BLOCK VALVE RELIABILITY.......... 2-1
   2.1 Steam Generator Tube Rupture.......... 2-1
      2.1.1 General Characteristics of SGTR Scenarios for Indian Point 3.......... 2-1
      2.1.2 SGTR Core-Melt Frequency Calculations for Indian Point 3.......... 2-6
      2.1.3 General Characteristics of SGTR Scenarios for Oconee 3.......... 2-7
      2.1.4 Estimation of Oconee 3 SGTR Core-Melt Frequency Reduction.......... 2-9
   2.2 Stuck-Open PORV.......... 2-12
      2.2.1 A General Description of Indian Point 3 Stuck-Open PORV Event.......... 2-12
      2.2.2 Computation of Indian Point 3 Stuck-Open PORV Core-Melt Frequency.......... 2-12
      2.2.3 A General Description of Oconee 3 Stuck-Open PORV Event.......... 2-14
      2.2.4 Estimation of Oconee 3 Stuck-Open PORV Core-Melt Frequency Reduction.......... 2-16
   2.3 Use of PORVs in Cooldown to Cold Shutdown.......... 2-19
      2.3.1 A General Description of Cooldown to Cold Shutdown for Indian Point 3 Transient Initiating Events.......... 2-19
      2.3.2 Core-Melt Frequency Calculations for Indian Point 3 Transient Initiating Events.......... 2-21
      2.3.3 A General Description of Cooldown to Cold Shutdown for Oconee 3 Transient Initiating Events.......... 2-22
      2.3.4 Estimation of Core-Melt Frequency Reduction for Oconee 3 Transient Initiating Events.......... 2-23
   2.4 Low Temperature Overpressurization Protection Using PORVs.......... 2-24
      2.4.1 General Descriptions of Low Temperature Overpressurization Events.......... 2-24
      2.4.2 Calculation of Core-Melt Frequencies Resulting from Low Temperature Overpressurization Events.......... 2-26
      2.4.3 Estimated Consequences for Low Temperature Overpressurization.......... 2-28
   2.5 Use of PORV as a High Point Vent to Supplement the Function of RVHVS.......... 2-29

REFERENCES

LIST OF FIGURES

Figures	Title	Page
1	SGTR event tree (Indian Point 3)	2-31
2	Event tree for SGTR event, SL (Indian Point 3)	2-33
3	Fault tree for SGTR event, failure of OP-41	2-34
4	Event tree for SGTR initiating events (Oconee 3)	2-36
5	Fault tree for top event, X_R, failure to achieve long-term cooling at cold shutdown (Oconee PRA)	2-37
6	Fault tree for top event, X_R, continued	2-38
7	Fault tree for top event, X_s, failure to maintain long-term cooling at hot conditions (Oconee PRA)	2-39
8	Small LOCA event tree (Indian Point 3)	2-40
9	Fault tree for the event, PORV path fails to close (Indian Point 3)	2-41
10	Event tree for small-break LOCA events (Oconee 3)	2-42
11	Event tree for transient initiating events (Oconee 3)	2-43
12	Fault tree for top event, Q, failure of RCS integriy (Oconee PRA)	2-44
13	Fault tree for the primary pressure control system (Oconee PRA)	2-45
14	Fault tree for the event, failure of primary feed and bleed cooling (IPPSS)	2-49
15	Fault tree for top event, U_T, failure of core heat removal by HPI (Oconee PRA)	2-50
16	Fault tree for top event, U_T, failure of core heat removal by HPI (Oconee PRA, continued)	2-51
17	Fault tree for top event, W, failure to restore RCS integrity (Oconee PRA)	2-52
18	Event tree for low temperature overpressurization protection using PORVs	2-53

LIST OF TABLES

Table	Title	Page
A	Indian Point 3 Core-Melt Frequencies Attributable to PORV and Block Valve Failures (Per Reactor Year)	1-3
B	Oconee 3 Core-Melt Frequencies Attributable to PORV and Block Valve Failures (Per Reactor Year)	1-4
C	Percentage Reductions in Total Core Melt Frequency Achievable by Improving the Reliability of PORV and Block Valves	1-5
D	Plant Risk Values Attributable to PORV and Block Valve Failures (Per Reactor Year)	1-5
1	Summary of SGTR Core Melt Sequences and Frequencies (Oconee 3)	2-54
2	A Brief Summary of Bin I Core-Melt Sequences (Oconee PRA)	2-55
3	A Brief Summary of Bin II Core-Melt Sequences (Oconee PRA)	2-56
4	Calculated Core-Melt Frequencies for Transient Initiating Events (Indian Point 3)	2-57
5	A Brief Summary of Bin III Core-Melt Sequences (Oconee PRA)	2-58
6	A Brief Summary of Bin IV Core-Melt Sequences (Oconee PRA)	2-60
7	Estimated Environmental Release Fractions for a Core Melt Accident Resulting From a Low Temperature Overpressurization Event With An Open Containment	2-61
8	Summary of Input Assumptions for the CRAC-2 Consequence Calculations for a Core Melt Accident Resulting from A Low Temperature Overpressurization	2-61
9	Estimated Reactor Vessel Failure Probability for a Low Temperature Overpressure Event ($T<100C$ and $P\sim2485$ psia) Based on ORNL Structural Analyses	2-61
10	Total Estimated Risk for the Low Temperature Overpressurization	2-62

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EXECUTIVE SUMMARY

A brief summary of the principal results and the conclusions obtained from this study is outlined in the following.

(1) For the events involving a steam generator tube rupture (SGTR), a stuck-open PORV, and use of PORV for cooldown to cold shut-down, the structure and logic of the relevant event trees and fault trees presented in both the Indian Point Probabilistic Safety Study (IPPSS) and the Oconee Probabilistic Risk Assessment (PRA) were scrutinized closely and found to be generally consistent with the block- diagrams developed and furnished by the NRC staff for depicting the scenarios of these transients. These event trees and fault trees were considered adequate for use as the basis for estimating the reduction in the core-melt frequencies achievable by improving the reliability of both PORVs and the associated block valves for these particular transient sequences.

(2) To conform to the distinct PRA methodologies employed respectively in the IPPSS and the Oconee PRA, the following different approaches were taken to assess the benefit of improving the PORV reliability.

(a) For Indian Point 3, Boolean expressions useful for computing the core-melt frequencies corresponding to different plant damage states were first derived based on each of the transient initiating event-trees presented in the IPPSS. Independent calculations of the core-melt frequencies for the particular transient sequences were then made by using these Boolean expressions together with the event-tree top events branch-point split fraction data shown in the IPPSS for the eight different electric power states. To quantify the potential benefit of improving the PORV reliability, those event-tree top events pertinent to the operations of PORVs and the associated block valves were examined in detail, by constructing fault trees if necessary, to determine how much reduction in the unavailabilities of these top events can be obtained by improving the reliability of PORV and the block valves. The newly estimated unavailability data for those top events were then substituted into the Boolean expressions for recalculating the core-melt frequencies resulting from the improved PORV and block valve reliability.

(b) For Oconee 3, it was considered unnecessary to carry out independent calculations of the core-melt frequencies by using computer codes such as the SETS code. Instead, the dominant minimal cut sets listed in the Oconee PRA for different core-melt bins were given close scrutiny by reference to the elaborate fault trees presented in the same report to explore whether they contain any fault-tree primary (or basic) event relevant to the operations of the PORV and its block valve. If any dominant minimal cut set was found to comprise such primary events, the failure probabilities assigned to those primary events were adjusted to reflect the improvement in the reliability of the PORV and the block valve and the core-melt frequency associated with the minimal cut set was recalculated. In this manner, the reduction in core-melt frequency due to improved PORV and block valve reliability could be quantified.

(3) The core-melt frequency resulting from a SGTR initiating event was calculated to be 1.6×10^{-6}/reactor year for Indian Point 3. It was revealed that the probability assigned to the event-tree top event OP-41, which bears a

close relationship with the use of PORV in controlling the break flow, consists almost entirely of the probability of operator errors in recognizing the situation and taking appropriate actions. The contribution to the probability of event OP-41 due to hardware failures of PORVs, block valves, and pressurizer spray is minimal.

If the reliabilities of PORVs and the block valves were improved to such an extent that no hardware failure of these valves would occur, the SGTR core-melt frequency could be reduced by roughly 6.9×10^{-8}/reactor year. In addition, if operator's errors in controlling the break flow and depressurizing the RCS were eliminated entirely, the core-melt frequency could further be reduced by a total of 9.4×10^{-7}/reactor year. These represent reductions of 4.3% and 58% respectively of the total predicted SGTR core-melt frequency.

For Oconee 3, the components of the sixteen dominant minimal cut sets listed in the Oconee PRA for SGTR core-melt sequences were carefully examined and found to contain none of the primary events designating operational failures of PORV and its block valve. The core-melt frequencies of these sixteen dominant cut sets add up to 2.4×10^{-6}/reactor year. There are additional core-melt frequency contributions from all other non-dominant cut sets amounting to 2.7×10^{-7}/reactor year, thus making the total SGTR core-melt frequency 2.7×10^{-6}/reactor year. The logic of the bin-level fault tree suggests that any cutset which might comprise primary events related to the operation of PORV and the block valve can only appear as a non-dominant cut set. According to a conservative estimate based on the results of an independent calculation performed at BNL,[4] no more than half of all the non-dominant cut sets actually contain primary events related to the operation of PORV. A rough estimate shows that the magnitude of core-melt frequency reduction achievable by improving the PORV reliability is no larger than 1.4×10^{-7}/reactor year, about 5% of the total SGTR core-melt frequency.

Although the BNL calculation[4] yielded one dominant cut set comprising a PORV-related primary event, it actually represents an operator error that can not be rectified by improving the PORV reliability.

(4) A stuck-open PORV is treated in the IPPSS as a small-LOCA initiating event. In this study, the total core-melt frequency for the small-LOCA initiating events was calculated to be 8.6×10^{-5}/reactor year, based on the initiating frequency of 0.0201 per reactor year, which includes all the small-LOCA initiating events such as a small pipe break, a stuck-open PORV or safety valve, etc. If the frequency of a stuck-open PORV can be specified exactly, its core-melt frequency can be calculated proportionally. By a fault-tree analysis, the probability that the PORV path will fail to close after automatic or manual openings was found to be roughly 1.6×10^{-4}. Based on the assumption that the PORV challenge frequency is 0.3 challenge/reactor year, the frequency of a stuck-open PORV then becomes 4.8×10^{-5}/year. The core-melt frequency due to a stuck-open PORV can thus be calculated as $8.6\times10^{-5} \times (4.8\times10^{-5}/0.0201) = 2.0\times10^{-7}$/reactor year. It is apparent that if the PORV reliability can be improved to such an extent that no PORV will stick open, this core-melt frequency can be eliminated.

Depending upon whether it is self-initiated or induced by a transient, a stuck-open PORV is treated in the Oconee PRA by means of a small-LOCA event tree or a combination of the event-trees for the small-LOCA and transient initiating events. The dominant minimal cut sets listed under these categories were, therefore, thoroughly inspected and the following findings were made.

(a) Of all the dominant cut sets scrutinized under these categories, only four dominant cut sets were found to contain a primary event directly connected to the operation of a PORV and its block valve. These cut sets represent an event tree sequence type that involves challenging of the safety relief valve (SRV) caused by actuation of high pressure injection (HPI) due to overcooling or a spurious actuation signal. Common to all of these four cut sets is the inclusion of a primary event representing the closure of the PORV block valve prior to the demand, i.e., operating with the block valve closed. The core melt frequency of these four dominant cut sets totals 1.3×10^{-7}/reactor year, which can be essentially eliminated if the PORV reliability is improved so that there is no need to prevent PORV leakage by closing the block valve.

(b) There are several types of event-tree sequences (with more than 70 dominant cut sets) that can be classified as transient-induced small-LOCAs leading to a stuck-open SRV discharging liquid. A majority of these sequences involve failure of the feedwater system. A close inspection of the components of all of the dominant cut-sets belonging to these sequences revealed that none of them contain primary events related to failures of the PORV. Improving the PORV reliability, therefore, has little effect in preventing this type of a small LOCA. Any cut set which might contain a PORV-related primary event can only appear within a group of non-dominant cut sets having a total core melt frequency of 6.8×10^{-8}/reactor year. This value can thus be considered as the upper bound for the magnitude of core-melt frequency reduction attainable by improving the PORV reliability for these sequences.

(c) There are three types of small-LOCA event-tree sequences (with about 22 dominant cut sets) that are not transient-induced, but correspond to a small pipe break or an inadvertent opening of a PORV. Based on the initiating frequency of 0.003/reactor year, the core-melt frequencies resulting from these sequences were found to add up to 6.1×10^{-6}/reactor year. If the frequency of an inadvertent opening of a PORV is known, the associated core-melt frequency can be computed proportionally. For instance, if its frequency is taken to be 6×10^{-5}/reactor year, then the core-melt frequency due to an inadvertent opening of a PORV becomes 1.2×10^{-7}/reactor year. If the reliability of PORV can be enhanced so that no PORV will open inadvertently, the core-melt frequencies belonging to this category can be eliminated.

(d) Among the transient-induced small-LOCA event-tree sequences, the sequence-type G in Bin I and sequence-type F in Bin II (each with 3 dominant cut sets) involve a stuck-open SRV discharging steam. The initiator of these core-melt sequences is a spurious low pressurizer-pressure signal which results in the actuation of pressurizer heaters and the blocking of PORV from opening. The SRV is challenged mainly because of the unavailability of the PORV, which, in this case, is caused by a control action taken in response to the low pressurizer-pressure signal. Consequently, unless the control logic is changed, the core-melt frequencies associated with these sequences can not be reduced by improving the reliability of the PORV.

To sum up, if the PORV and its block valve can be made perfectly reliable, the core melt frequency (CMF) due to a stuck-open PORV (roughly 3.3×10^{-7}/ry for Oconee 3) can be eliminated. In the Oconee PRA, the CMF due to all small LOCA events (including small RCS pipe break, reactor coolant pump seal failure, control rod drive seal leakage, letdown exceeding charging, and inadvertent opening of the PORV or safety valves etc.) was estimated to be 7.6×10^{-6}/ry. A stuck-open PORV, thus, accounts for roughly 4.2% of the CMF due to all small-LOCA events.

(5) For attainment of cold or hot shutdown conditions following a reactor trip caused by transient-initiated events, a careful inspection of the logic of event-trees and fault-trees presented in the IPPSS and the Oconee PRA for the various transient initiated events revealed that the PORV is primarily used in conjunction with the HPI system for the HPI cooling (feed and bleed) in case of a failure of core-heat removal by the steam generators. In such a case, core uncovery can be prevented if long-term heat removal can be maintained by recirculation cooling.

For Indian Point 3, generalized Boolean expressions describing the core-melt frequencies corresponding to eight different plant damage states were derived based on the nine event trees presented in the IPPSS for transient initiating events. These expressions were used to compute the core melt frequencies resulting from the eleven types of transients considered in the IPPSS, and the results are tabulated in Table 4. For most of these transients, it was found possible to reduce their core-melt frequencies by about 40% through improving the reliability of the PORVs and the associated block valves.

For Oconee 3, all the ten types of Bin III event-tree sequences (with more than 50 dominant cut sets) that involve use of the HPI cooling were found to be of type TBU, i.e., failures of both steam generator cooling and HPI cooling. For the majority of these sequences, the leading cause for the failure of HPI cooling is an operator's failure to make a decision to initiate the HPI cooling. The total core-melt frequency for these sequences was found to be approximately 2.7×10^{-5}/reactor year. Improving the reliability of the PORV and its block valve can only reduce this core-melt frequency by about 9×10^{-8}/reactor year. Unlike Indian Point 3, the HPI cooling can be achieved in Oconee 3 not only by using the PORV, but alternatively by using the two safety valves if the PORV fails to open. This is the main reason why improving the reliability of PORV in Oconee 3 has a relatively small benefit in reducing the core-melt frequencies resulting from HPI cooling failure.

(6) In connection with the utilization of PORVs for the protection of the reactor vessel from low temperature overpressurization, the core-melt frequencies due to low temperature overpressurization events were conservatively estimated to be 4.8×10^{-6}/reactor year for Oconee 3 and 2.6×10^{-6}/reactor year for Indian Point 3, based on an initiating frequency of 0.13 events/reactor year. These core-melt frequencies can be lowered by 1.8×10^{-6}/reactor year and 6.8×10^{-9}/reactor year respectively if the PORVs and the associated block valves can be made to operate on demand without flaw. Based on these calculated core melt frequencies, the offsite radiological consequences for a low temperature overpressurization event were determined using the CRAC-2 code;[16] the total estimated risks are summarized in Table 10.

It should be remarked, however, that some preliminary information obtained from recent Generic Issue 94 studies suggests that the BNL sutdy is overly conservative in implicitly assuming that all the LTOP events result in pressure high enough to threaten the integrity of the reactor vessel. The most recent operating experience, gained since LTOP protections measures have been implemented, indicates that, given a LTOP event, there is only about 10% chance that the vessel pressure will reach such a critical condition. The aforesaid conservatism, however, tends to be offset by the vessel rupture probability data used in the BNL study, which were found to be nonconservative also based on recent preliminary GI-94 studies.

It can thus be concluded that, due to the complexity of the problem and the many uncertainties involved, these results can only be considered as rough estimates based on several assumptions which remain to be justified.

(7) For the possible use of PORV as a high point vent to supplement the function of the Reactor Vessel Head Vent System (RVHVS), the unavailability of RVHVS was first estimated to be roughly $4x10^{-4}$, by constructing fault trees based on the schematic diagram of RVHVS shown in Millstone Nuclear Power Station Unit 3 FSAR (Figure 5.4-17). The probability of failure to vent noncondensable gases using either the RVHVS or the PORV was found to be less than 10^{-5}.

These fault trees, however, were found to be incompatible with those presented in the Oconee PRA or the IPPSS due to discrepancy in the timing of relevant events. It was, therefore, concluded that, although the RVHVS may be effective in mitigating the consequence of core melt at its early stage, we were unable to determine its impact on the core melt frequency by the use of the currently available PRA information.

In summary, with the exception of low temperature overpressurization (LTOP) events, the risk contribution from PORV or block valve failures is judged to be insignificant. The greatest potential reduction in core damage frequency was calculated for Oconee 3 where it was estimated that the total core damage frequency (due to internal events) could be reduced by about 3% if the PORV and the associated block valve could be improved to operate effectively 100% of the time during postulated LTOP events.

Public health consequences were also calculated for LTOP since for this event there is the possibility for a relatively significant release of fission products directly to the environment. The results of a conservative estimation of health effects indicate that the public risk from LTOP events is very small until late in plant life when the aging effects of the vessel contribute to increased vulnerability. These effects are being studied in more depth in the NRC's studies for Generic Issue Number 94.

1. INTRODUCTION

1.1 Scope of this Report

This report presents results of a study conducted by BNL for the Nuclear Regulatory Commission staff. The study has been conducted in support of the resolution of Generic Issue CI-70, to provide input to NRC's decision whether improvements are required to increase the reliability of pressurizer power operated relief valves and their associated block valves.

The information presented here illustrates how particular severe accident sequence frequencies could be reduced by improving PORV reliability. In part, this information has been gleaned by carefully examining the Probabilistic Risk Assessments (PRAs) for two plants, Indian Point 3 and Oconee 3.

Indian Point 3 is a Westinghouse plant with two PORVs; Oconee 3 is a Babcock and Wilcox plant with a single PORV. These two plants therefore span a useful range of design characteristics, and the availability of PRAs for them ensures a useful starting point.

The scenarios involving PORVs which are studied here are the following: steam generator tube ruptures (SGTR), stuck-open PORV, use of the PORV in reaching cold shutdown conditions, and a failure of low-temperature overpressurization (LTOP) protection. The practicality of using the PORV as a high point vent is also studied. Not studied here are ATWS sequences (Anticipated Transient Without Scram). This omission does not reflect on the importance of this issue; rather, BNL was directed by NRC staff not to pursue it, as it is being studied under other auspices. The subject of HPI cooling (feed and bleed) was briefly examined because this mode of cooling was found to be essential in reaching cold shutdown conditions if core heat removal by the steam generators fails.

For each family of scenarios studied, an estimate of the contribution to core damage frequency involving PORV failure is presented; this contribution would vanish for perfect PORVs, and accordingly represents an estimate of "benefits" to be gained by substantial improvements in PORV reliability. Because PORVs will never be "perfect," these estimates may overstate the benefits to be gained from PORV improvements; note, however, that they are based on truncated PRA results, and it is formally possible that residual contributions to core damage frequency have been missed. However, care has been taken to understand the plant models, and the results are believed to overstate the benefits as implied above, owing to the unattainability of perfection in PORV operation.

1.2 Summary of Findings

The core-melt frequencies attributable to failures of PORVs and the block valves are summarized in the following Tables A and B for Indian Point 3 and Oconee 3 respectively. The values shown in these tables represent the magnitude of core-melt frequency reduction attainable by improving the reliability of both PORV and the block valve. The corresponding percentage reductions in the total core melt frequencies for various accident scenario types are shown in Table C.

Since the low-temperature overpressurization (LTOP) event has the potential for contributing the greatest amount to plant risk from PORV failures, public health consequences were estimated for LTOP and are reported in Table D.

In summary, with the exception of LTOP, the risk contribution from PORV or block valve failures is judged to be insignificant. The greatest potential reduction in core damage frequency was calculated for Oconee 3 where it was estimated that the total core damage frequency (due to internal events) could be reduced by about 3% if PORV and the associated block valve could be improved to operate effectively 100% of the time during postulated LTOP events.

One of the major uncertainties in the above estimations of core damage frequency is the assumed PORV and block valve failure rate. For this study, these values were taken from the individual plant PRAs and these values were:

Plant	PORV Failure to Open	PORV Failure to Close	Block Valve Failure to Open
Indian Point 3	5×10^{-4}	5×10^{-4} (assumed)	1.51×10^{-3}
Oconee 3	4.9×10^{-3}	1.1×10^{-2}	6.0×10^{-3}

The use of the large reported range of valve failure rates indicated in the table lead to a similarly broad range of estimated potential reductions in core damage frequency. It is believed that the use of the Oconee 3 failure rates results in estimates of potential benefits to be gained through PORV improvements for PWRs, in general that are somewhat conservative (optimistic). Note, however, that one other major reason why the potential benefit that can be gained through PORV improvements was estimated to be larger for Oconee 3 in this study is that Oconee 3 has only one PORV, as opposed to two PORVs possessed by Indian Point 3.

In addition to potential reductions in core damage frequency brought about by improved PORV performance, potential reductions in public health consequences were also evaluated. LTOP has the potential for combining worst case conditions for health consequences - mainly due to the possibility that the containment may be open for refueling or testing/maintenance, etc. thus leading to a direct release of fission products to the environment from a failed reactor vessel. It was conservatively estimated in this study that for the Oconee plant at vessel present conditions, the public risk from LTOP would be about 1.2 person-rem per reactor year. Accounting for increased vessel vulnerability as the vessel ages, it was estimated that the Oconee public risk from LTOP would increase to a value of about 40 person-rem per year at the end of life of the plant (and vessel). These values of public risk were derived by combining the estimated health consequence values with the previously calculated values of core damage frequency.

Table A
Indian Point 3
Core-Melt Frequencies Attributable to PORV and Block Valve Failures
(Per Reactor Year)

Plant Damage State	SGTR	Stuck-Open* PORV	Cooldown to Cold Shutdown	Low Temp** Overpress. Protection	Total
V2E1	$1.\ x10^{-12}$				$1.\ x10^{-12}$
V2E2	$2.2x10^{-13}$				$2.2x10^{-13}$
V2L	$6.9x10^{-8}$				$6.9x10^{-8}$
SLFC		$1.93x10^{-7}$			$1.93x10^{-7}$
SLF		$7.0x10^{-12}$			$7.0x10^{-12}$
SLC		$1.7x10^{-12}$			$1.7x10^{-12}$
SL		$1.2x10^{-16}$			$1.2x10^{-16}$
SEFC		$6.6x10^{-9}$			$6.6x10^{-9}$
SEF		$1.4x10^{-12}$			$1.4x10^{-12}$
SEC		$3.6x10^{-10}$			$3.6x10^{-10}$
SE		$2.4x10^{-11}$			$2.4x10^{-11}$
TEFC			$8.5x10^{-7}$		$8.5x10^{-7}$
TEF			$3.1x10^{-11}$		$3.1x10^{-11}$
AE				$6.8x10^{-9}$	$6.8x10^{-9}$
Total	$6.9x10^{-8}$	$2.0x10^{-7}$	$8.5x10^{-7}$	$6.8x10^{-9}$	$1.1x10^{-6}$

*Based on PORV challenge frequency of 0.3 challenge/reactor year.
**Based on initiating frequency of 0.13 events/reactor year.

E = Early
L = Late
S = Small LOCA
T = Transient
F = Fan cooler operating
C = Containment spray operating
V = Interfacing system LOCA
A = Large LOCA

V2E1 (early melt)
V2E2 (early melt)
V2L (late melt)

Table B
Oconee 3
Core-Melt Frequencies Attributable to PORV and Block Valve Failures
(Per Reactor Year)

Core-Melt Bin	SGTR	Stuck-Open* PORV	Cooldown to Cold Shutdown	Low Temp** Overpress. Protection	Total
Bin IIR	1.4×10^{-7}				1.4×10^{-7}
Bin I		1.5×10^{-7}		1.8×10^{-6}	2.0×10^{-6}
Bin II		1.8×10^{-7}			1.8×10^{-7}
Bin III			$9.\times10^{-8}$		$9.\times10^{-8}$
Total	1.4×10^{-7}	3.3×10^{-7}	$9.\times10^{-8}$	1.8×10^{-6}	2.4×10^{-6}

*For a small-LOCA due to inadvertent opening of PORV, the initiating frequency was taken to be $6.\times10^{-5}$/reactor year.
**Based on initiating frequency of 0.13 events/reactor year.

Bin I = RCS pressure and leakage rates associated with small-break LOCAs, with early core melt (i.e., within about 2 hours after the break occurs).

Bin II = RCS pressure and leakage rates associated with small-break LOCAs, with late core melt (after about 12 hours from the time of break).

Bin III = High RCS pressure and leakage rates associated with boiloff of the reactor coolant through cycling pressurizer relief valves, with early core melt (within about 2 hours).

Bin IV = High RCS pressure and leakage rates associated with boiloff of the reactor coolant through cycling relief valves, with late core melt.

Bin IIR = The SGTR core-melt sequences were kept separate from others to permit the consideration of any unique effects related to the consequence analysis.

Table C
Percentage Reduction in Total Core Melt Frequency
Achievable by Improving the Reliability of PORVs
and Block Valves

	Indian Point 3				Oconee 3			
Accident Scenario Grouping	Before PORV Improvement	After PORV Improvement	ΔCMF	ΔCMF/CMF (% Reduction)	Before PORV Improvement	After PORV Improvement	ΔCMF	ΔCMF/CMF (% Reduction)
Steam Generator Tube Rupture	1.63E-6	1.56E-6	6.9E-8	.05	2.7E-6	2.56E-6	1.4E-7	.3
Stuck Open PORV	8.56E-5*	8.54E-5*	2E-7	.15	7.6E-6*	7.27E-6*	3.3E-7	.6
Cooldown to Cold Shutdown	7.93E-6	7.08E-6	8.5E-7	.65	2.65E-5	2.64E-5	9E-8	.2
Low Temperature Overpressurization Protection	2.6E-6	2.6E-6	6.8E-9	.005	4.8E-6	3.0E-6	1.8E-6	3.3

Notes: 7E-8 = 7×10^{-8}

*The IPPSS or the Oconee PRA does not specifically estimate the CMF due to a stuck-open PORV. The numbers shown here represent the CMF due to all small LOCAs.

ΔCMF = Estimated reduction in core melt frequency (CMF) possible for various accident scenario types if PORVs and block valves could be improved to operate perfectly.

CMF = Estimates of total core melt frequency due to internal events taken from plant PRAs (1.3E-4 for Indian Point 3, 5.4E-5 for Oconee 3).

Table D
Plant Risk Values Attributable to PORV
and Block Value Failures
(Per Reactor Year)

Vessel Age	Oconee 3	Indian Point 3
Present condition	1.2 person-rem	0.2 person-rem
End-of-life	40 person-rem	21 person-rem

2. CORE DAMAGE SCENARIOS AND ESTIMATION OF THE BENEFITS OF AN IMPROVED PORV AND BLOCK VALVE RELIABILITY

2.1 Steam Generator Tube Rupture

2.1.1 General Characteristics of SGTR Scenarios for Indian Point 3

The steam generator tube rupture (SGTR) model developed in the IPPSS (Indian Point Probabilistic Safety Study) has undergone a revision[5] to provide more elaborate and accurate delineations of these scenarios. A notable change includes the addition of an event tree top event, SL, no secondary-side leakage to atmosphere, to the SGTR event tree and the development of a separate event tree for this top event. Presentation of the quantitative results of PRA in the IPPSS differs considerably from that in the Oconee PRA. Instead of presenting detailed fault trees for supporting the logic of each of the event tree top events, the probabilities of each of the top events are quantified and tabulated according to the eight electric power states which are Indian Point 3 plant specifics.

One major difference between a SGTR and an ordinary small break LOCA is that the reactor coolant is leaked to the secondary side of steam generator which provides several potential paths for the release of radioactivity to the outside of containment, via the main steam-line, turbine, condenser, condenser exhaust, the SG safety and atmospheric relief valves and the SG blowdown-line etc. Moreover, since the water will be lost to the outside of containment, it can not be used for recirculation cooling.

If the operator can identify and isolate the faulty steam generator, and if the RCS can be depressurized to below the SG safety valve set point, the leakage flow can be effectively stopped. The success of these operations, however, relies heavily on the operator's realizing the situation and taking correct actions.

As will be discussed in greater detail later, the PORV is of important use in connection with controlling the break flow following the initiation of the SGTR event. Usually, the operator will have at least 15 minutes to more than one hour to control the break flow to avoid challenging of a SG safety valve. He then will have many hours (8-24 hours) to secure the leak, depressurize, and cooldown the RCS.

The fault trees for both the SGTR and the event tree top event, SL, presented in the IPPSS are shown in Figures 1 and 2 respectively. Although the SGTR analysis performed in the IPPSS applies to a single, double-ended rupture, it is claimed to bound the case of multiple tube ruptures. The SGTR event tree uses the following symbols to designate the event tree top events.

ET-4	Steam Generator Tube Rupture Initiating Event
S	Reactor Trip and Safety Injection Signals
K-2	Reactor Trip
HH-*	High Head Injection
L-3	AFWS Actuation and Secondary Cooling
OP-4	Operator Controls Break Flow
CF	Fan Coolers
CS	Containment Spray

SL No Secondary Side Leakage to Atmosphere
OP-5 Depressurization and Makeup
R-4 Long Term Cooling

The event tree top event, SL, no secondary leak to atmosphere, is a complicated event. Leakage from the ruptured steam generator to the atmosphere occurs if (1) the faulty SG is not isolated, (2) the SG atmospheric relief valve or safety valve stick-open, and (3) the steam-line fails outside containment. Because of the complicated nature of the event, SL, it was modeled in terms of an event-tree of its own. As can be seen from Figure 2, the event-tree SL has two end states, "success" and "leak." By taking the union of all sequences ending in "leak," the conditional relative frequency of failure for SL can be obtained. The SL event-tree contains the following event-tree top events.

SGTR: Initiating Event
SD: Steam Dump Available
OS: Operator Decides to Isolate Steam Generator
IV: MSIV and Bypass Isolated
SG: Isolate Steaming Steam Generator
OP-4*: Operator Controls the Break Flow
MS: Main Steam Line Intact
AO: ARV (Atmospheric Relief Valve) Opens
AC: ARV Closes
AI: ARV Isolated
SN: Safety Valve Not Demanded
SO: Safety Valve Opens and Closes

Detailed description and definition of these top-events can be found in the IPPSS (Amendment 2).

A Brief Description of SGTR Scenarios and Use of PORV During an SGTR Accident

A general description of the SGTR scenarios is given below in relation to the event-tree top events of the SGTR event-tree shown in Figure 1.

Following the initiation of an SGTR, reactor coolant system (RCS) will depressurize due to the depletion of primary coolant inventory by the leakage flow, causing the reactor to trip (top event K-2) as RCS pressure drops to the low pressurizer pressure reactor trip signal set point (1885 psia) generating the reactor trip signal (top event S). The turbine automatically trips as the reactor trip breakers open. After the reactor trip, RCS will continue to depressurize causing the safety injection signal to be activated (top event S) when the low pressurizer pressure safety injection signal set point (1785 psia) is reached. For the high-head safety injection (top event HH-*) to be successful, one out of three safety injection pumps must deliver water to RCS. Actuation of the safety injection automatically causes termination of normal feedwater flow and actuation of auxiliary feedwater flow. The success of auxiliary feedwater system (AFWS) requires starting of one motor-driven pump or the turbine-driven pump. Secondary cooling is accomplished by removing heat from the SG by automatically or manually opening a relief valve in the SG receiving auxiliary feedwater. Heat can also be removed by safety valves, steam dump to the main condenser or blowdown. Since the reactor power is reduced to decay heat level, heat removal can be achieved by one SG under forced convection or natural circulation conditions.[9,10] These events, actuation of AFWS and secondary cooling, are represented by the top-event L-3 in the SGTR event tree.

Once the faulty SG is identified, it is manually isolated (isolate MSIV and its bypass, blowdown, and the steam-line to the turbine-driven AFW pump) to allow termination of break flow and to minimize release of radioactivity. Isolation of auxiliary feedwater flow will help reduce the chance of flooding the steam-line. Operator's action is then initiated to reduce RCS temperature to 50°F below the saturation temperature corresponding to the ruptured SG pressure by means of steam dump from the intact SGs to the condenser utilizing the steam dump system or the atmospheric relief valve. This is to prevent voiding of RCS when it is depressurized to the pressure of ruptured SG.

To terminate the break flow, RCS is then depressurized to the pressure of ruptured SG by using the normal pressurizer spray or opening one PORV pressurizer PORV. Generally, depressurization by pressurizer spray is preferred to PORV because the latter will cause release to the pressurizer relief tank (PRT) which may lead to failure of the rupture disk. These operator's actions to control the break flow is denoted by the event-tree top event, OP-4*. In reality, OP-4* is further split into two operations, OP-41 and OP-42, depending upon whether the top event L-3 (AFWS actuation and secondary cooling) is successful. They are defined as follows.

(a) OP-41 Operator controls break flow given success of L-3.
Success of OP-41 event requires the operator to control RCS temperature and pressure using the AFWS and SG steam reliefs or dumps along with pressurizer spray or opening of one PORV.

(b) OP-42 Operator controls break flow given failure of L-3.
Success requires primary "feed and bleed" with two PORVs open and one high pressure safety injection pump operating.

If OP-41 is successful, the PORV (or pressurizer spray valve) is closed when the RCS pressure equalizes the ruptured steam generator pressure. Both RCS pressure and pressurizer level continue to increase until safety injection is terminated after primary pressure increases 200 psi. This 200 psi termination criterion is imposed for the purpose of collapsing any voids in the RCS which may have been created during the depressurization process and to confirm the integrity of the pressurizer vapor space.

Cooldown of the RCS is continued by dumping steam from the intact SGs until the RHR (Residual Heat Removal) system can be placed in service. In the meantime, the faulty SG is gradually depressurized via steam dump to condenser or through a steam generator power operated relief valve (PORV). The operator has to simultaneously depressurize RCS by using pressurizer spray or cycling PORV to maintain pressure equilibrium between RCS and the ruptured steam generator. This operator's action is represented by OP-50, which is a subdivision of OP-5*, a top event in the SGTR event-tree. Depending upon the success or failure of the preceding sequence of events, HH-*, L-3 and SL, the top event OP-5* (Depressurization and Makeup) is subdivided into the following four events.

(a) OP-50: Success of both HH-* and L-3
Success of this event requires that the operator provides long-term stability until the rupture is repaired. Note from the SGTR event-tree that long-term cooling using the RHR system is not required. Depressurization can be achieved by using pressurizer spray or one pressurizer PORV.

(b) OP-51: Success of HH-* and failure of L-3
Success requires continued primary "feed and bleed" cooldown using one high pressure safety injection pump and two PORVs until RHR system can be activated (below 350°F and 450 psig).

(c) OP-52: Failure of HH-* and success of both L-3 and SL
Success requires establishment of a source or RCS makeup and cooldown to the RHR entry conditions.

(d) OP-53: Failure of HH-* and success of L-3 and failure of SL
If high head safety injection (HH-*) has failed and secondary side leak from the faulty SG is present, some means of replenishing RCS inventory must be established. The operator is advised to blowdown the intact SGs to atmospheric pressure. With auxiliary feedwater available, the SGs can be blown down through the condenser or through the steam generator PORVs to the atmosphere. With no high head safety injection and a steam generator tube rupture taking place, this will cause rapid depressurization of the primary system. In addition to the secondary depressurization, low head safety injection is needed to maintain RCS inventory and core coolability.

For the top-event OP-5* (Depressurization and Makeup), therefore, the pressurizer PORVs are used mainly in conjunction with the events, OP-50 and OP-51 (feed and bleed). The event-tree top events CF (containment fan coolers) and CS (containment spray) are needed only when primary "feed and bleed" cooling is employed. The top-event, SL, analyzes various plausible modes of leakage to the atmosphere through the ruptured steam generator such as failures of steam dump valves along with main steam isolation valve failure, steam-line failures, and failures of atmospheric relief valve and safety valves to open and close etc. As explained earlier, SL is treated by a separate event-tree of its own. The last event, R4 (Long Term Cooling) in the sequence of SGTR event-tree top events represents long-term cooling using the normal closed-loop mode of RHR (Residual Heat Removal) system. For this purpose, one RHR pump is required along with component cooling to one RHR heat exchanger. If RHR has failed, long-term cooling can be supplied by long-term primary "feed and bleed" with a bleed path provided by a pressurizer PORV.

To sum up, the pressurizer PORVs are primarily used in connection with the SGTR event-tree top events, OP-41, OP-42 (feed and bleed), OP-50 and OP-51 (feed and bleed). It can also be used in a long-term primary "feed and bleed" in case of a failure in RHR, and may also be useful, if needs occur, in the depressurization process required in OP-52.

2.1.2 SGTR Core-Melt Frequency Calculations for Indian Point 3

A thorough examination of both the SGTR event-tree and the event-tree for the top-event, SL, shown in Figures 1 and 2 and discussed above has been done in the light of the SGTR transient scenarios and block diagrams developed by the NRC staff, and found to be reasonably correct and complete. These event-trees, therefore, were taken to be the basis for estimating the frequencies of core-melt resulting from a SGTR initiating event.

To derive Boolean expressions for computing the core-melt frequencies corresponding to different plant damage states, the success and failure branches of the top-event, OP-4*, were first subdivided into OP-41 and OP-42 depending on the outcome of the top-event, L-3. Similarly, the top-event, OP-5* was subdivided into OP-50, OP-51, OP-52, and OP-53, according to the conditional outcomes of the top-events, HH-*, L-3 and SL, as described earlier. By taking the union of event-tree sequences belonging to the same plant damage state, followed by simplifications utilizing the rules of Boolean algebra, yielded the following Boolean expressions for the three plant damage states, V2L, V2E2, and V2E1.

$$\mathrm{V2L} = \overline{\mathrm{S}}\ \overline{\mathrm{HH}}\ \overline{\mathrm{L3}} \ast (\overline{\mathrm{OP41}}\ast\mathrm{OP50}+\mathrm{OP41}\ast\mathrm{R4})+\overline{\mathrm{S}}\ \overline{\mathrm{HH}\ast}(\overline{\mathrm{CF}}+\mathrm{CS})\ast\mathrm{L3}\ast(\mathrm{OP51}+\mathrm{R4})+\overline{\mathrm{S}}\ \overline{\mathrm{K2}}\ \overline{\mathrm{L3}\ast}$$

$$(\overline{\mathrm{OP41}\ast\mathrm{SL1}} + \mathrm{OP41}\ast\overline{\mathrm{SL2}})\ast\ \mathrm{HH}\ast\ (\mathrm{OP52} + \mathrm{R4}) \qquad (1)$$

$$\mathrm{V2E2} = \overline{\mathrm{S}}\ \overline{\mathrm{HH}}\ \overline{\mathrm{CS}}\ast\mathrm{L3}\ast\mathrm{CF}\ast(\mathrm{OP51}+\mathrm{R4}) + \overline{\mathrm{S}}\ \overline{\mathrm{K2}}\ \overline{\mathrm{L3}} \ast (\overline{\mathrm{OP41}}\ast\mathrm{SL1} +\mathrm{OP41}\ast\mathrm{SL2})\ast\ \mathrm{HH} \ast$$

$$(\mathrm{OP53} + \mathrm{R4}) \qquad (2)$$

$$\mathrm{V2E1} = \overline{\mathrm{S}}\ast\mathrm{HH}\ast(\mathrm{K2} + \mathrm{L3}) + \mathrm{S} \qquad (3)$$

In the above expressions and in all the event-tree sequences expressions to be used throughout this report, the top-event symbol with a bar above it denotes the success state of the event. The branch-point split fraction data for each of the top-events in the SGTR event tree (conditional on the SGTR initiating event and electric power state) are tabulated in the IPPSS. The data used to quantify the split fractions for the event SL are also tabulated. In this study, the following Boolean expression for the top-event SL was also derived by taking the union of all the event-tree sequences leading to "leak" states.

$$\mathrm{SL} = \mathrm{OS}+\mathrm{IV}\ast\mathrm{SG} + \overline{\mathrm{OS}}\ast(\overline{\mathrm{IV}} + \overline{\mathrm{SG}})\ast\ \lbrack\overline{\mathrm{AO}}(\mathrm{AC})\ \{\overline{\mathrm{MS}}\ast(\mathrm{SD}+\mathrm{OP4})+\ \mathrm{SD}\ast\overline{\mathrm{OP4}}\}\ast\ (\mathrm{AI}+\mathrm{SN}\ast\mathrm{SO})$$

$$+\ \mathrm{OP4}\ast\mathrm{MS} + (\mathrm{OP4}+\mathrm{SD})\ast\mathrm{AO}\ast\mathrm{SO}\rbrack \qquad (4)$$

By using the above Eq. (4), and the data presented in the IPPSS for each of the SL event-tree top events, the following results on the split fraction were obtained for SL.

(i) ac power available at all buses.
   For OP4 = 0.0 $\quad$ $\mathrm{SL} = 10^{-4}$
   For OP4 = 1.0 $\quad$ $\mathrm{SL} = 5.24\times10^{-3}$

(ii) Degraded power state.
For OP4 = 0.0   SL = 3.88×10^{-4}
For OP4 = 1.0   SL = 5.24×10^{-3}

These calculated values for the split fractions of SL were found to be in very good agreement with those presented in the IPPSS. They were used, together with the branch-point split fraction data shown in the IPPSS, to calculate the core-melt frequencies corresponding to each of the plant damage states using Eqs. (1), (2), and (3), and the following results on the core melt frequencies were obtained.

Plant Damage State	Core Melt Frequency	Time of Release (hours after SGTR)	Duration of Release (hr)
V2E1 (early melt)	1.66×10^{-9}	2	1
V2E2 (early melt)	7.44×10^{-11}	2	1
V2L (late melt)	1.624×10^{-6}	24	1
Total =	1.625×10^{-6}		

The total core-melt frequency for the SGTR initiating events is thus found to be 1.625×10^{-6}. This is in good agreement with the value, 1.6×10^{-6}, listed in the IPPSS. For reference, the branch point split fractions for each of the SGTR event-tree top events corresponding to the electric power state No. 1 (i.e., ac power available at all buses) used in this study are listed below.

S = 2.7×10^{-9}	SL1 = 10^{-4}
HH = 1.5×10^{-4}	SL2 = 5.2×10^{-3}
L3 = 1.1×10^{-4}	OP50 = 10^{-6}
OP41 = 0.05	R4 = 5.3×10^{-4}
CF = 3.2×10^{-5}	OP51 = 0.05
CS = 3.6×10^{-5}	OP52 = 0.09
K2 = 3.9×10^{-5}	OP53 = 10^{-3}

It is evident that the core-melt frequencies computed based on the Boolean expressions will decrease if the probabilities assigned to each of the top-events shown above can be reduced. The event-tree top events pertinent to the operations of PORVs or the associated block valves, such as OP40, OP50, OP51, and OP52, etc., can be examined with the aid of fault trees to determine whether improving the PORV reliability can help reduce the failure probabilities assigned to them. In this connection, a fault tree as shown in Figure 3 was constructed to analyze the failure of the top-event, OP41, which is closely related to the use of pressurizer spray or PORV in controlling the SGTR break flow. Note that all human errors related to lowering RCS pressure and controlling the secondary-side steam pressure are lumped together as a fault-tree event. As can be observed, the probability of the event, failure to lower RCS pressure (by using pressurizer spray or one of the PORVs), is about 2×10^{-6}, a rather small number. Since there are two PORVs arranged in parallel, failure of the PORV path occurs only if both of the PORVs fail to open on demand. Moreover, the occurrence of the event, failure to lower RCS pressure, requires additional concurrent failure of the pressurizer spray. The probability of failure to control secondary-side steam pressure due to hardware failures of the valves is also extremely small largely because of the availability of redundant components such as the atmospheric relief valve and the safety valves. This implies that the failure probability of 0.05 assigned to the top-event, OP-40, is essentially composed of that due to operator's errors in taking proper actions to control the break flow by depressurizing RCS and simultaneously controlling the secondary side steam pressure. The failure probabilities allotted to OP-51 (feed and bleed) and OP-52 are also chiefly due to that of operator's errors. Improving the reliability of PORVs can contribute very little to lowering the probabilities of these human errors.

Assuming that the reliability of PORVs is improved in such a way that no failure of PORV or the associated block valves can occur due to their mechanical, electrical or pneumatic systems, the core-melt frequency resulting from the SGTR events can be reduced from 1.63×10^{-6}/ry to 1.56×10^{-6}/ry. Addi-tionally, if the operator's errors in manipulating the break flow (OP-41) and depressurizing the RCS can be totally eliminated, the core-melt frequency can be reduced to 6.91×10^{-7}/ry.

2.1.3 General Characteristics of SGTR Scenarios for Oconee 3

Since a steam generator tube rupture (SGTR) differs from other small LOCA initiating events in several important respects, it was accorded a separate event tree evaluation in the Oconee PRA. The event tree analysis applies to a typical leak rate of approximately 400 gpm at normal RCS and secondary-system conditions resulting from a complete severance of a single SG tube. To facilitate further discussions, the event tree for SGTR initiating events and some of the relevant fault trees developed in the Oconee PRA are shown in Figure 4 through Figure 7. The approaches taken in the Oconee PRA differ from those in the IPPSS in that although the event tree is relatively simple, elaborate fault trees are developed to delineate the supporting logic for each of the event tree top event. One advantage of this type of approach is that it is easier to comprehend the essence of the PRA and to gain a deep insight into the details of the logical basis of the analysis. As compared with the Indian Point 3 PWR, the Oconee-3 PWR has only two steam generators of the once-through type (as opposed to four U-tube type SGs for Indian Point 3), one pressurizer PORV (instead of 2), and two pressurizer code safety relief valves (instead of 3). These and other unmentioned differences in the plant-specifics between the two PWR systems can be expected to exert some influence on the outcome of the SGTR event as well as the measures that can be taken to alleviate its severity. They are properly reflected in the structure of both the event and fault trees developed to assess the probabilistic risk of the relevant events. For example, the failure probability for the PORV path to open to depressurize RCS during an SGTR accident can be expected to be larger for the Oconee PWR (if human errors are not taken into consideration) because it has only one PORV. The event tree shown in Figure 4 also indicates that, in the Oconee PRA, core melt (of bin IR type) is always assumed to take place whenever the HPI fails regardless of whether or not the RCS heat removal by steam generators is successful. Such is not the case with the IPPSS, as can be observed from the event tree shown in Figure 1. In contrast to the IPPSS, the Oconee PRA attaches relatively little weight to the effectiveness of the HPI cooling (feed and bleed) using the PORV, in case of a failure of RCS heat removal by steam generators. The HPI cooling alone is considered to be insufficient to cool the RCS down to the DHR (decay heat removal) entry conditions. For such a case, emphasis is placed upon opening the boron precipitation line (valves LP-103 and LP-104) to depressurize the RCS to within the capacity of the LPI (low pressure injection) pumps. The functions that can be exercised by the PORV in mitigating the consequence of an SGTR accident are logically illustrated by the system-level fault tree TXR07 shown in Figure 6. Normally, a SGTR will cause a slow RCS depressurization, resulting in a reactor trip at its low pressure set point (1815 psia). If feed water flow is available, the RCS depressurization will continue until the set point for the HPI system (1565 psia) is reached. If MFW (main feed water) flow is lost subsequently to the reactor trip, and if the EFW (emergency feed water) system fails to respond properly, manual actuation of the HPI within 30 minutes may be necessary to avoid core uncovery. The manual actuation is required mainly because the RCS will repressurize as a result of the loss of feedwater, and automatic actuation through the ES (engineered safeguard) system will not occur. One important difference between a SGTR and most other LOCAs is that the leakage flow is lost to the secondary system rather than the reactor building. Initiation of HPI on high containment pressure, therefore, will not occur. Moreover, since no water will accumulate in the containment sump, HPR (high pressure recirculation) mode of cooling can not be depended upon, and usually, some other mode of cooling must be established before the inventory of BWST (borated water storage tank) is exhausted.

If both HPI actuation and RCS heat removal by steam generators are successful, the most desirable mode of long-term cooling is to cool and depressurize RCS to the point (250°F, 350 psia) at which the DHR system can be activated for an eventual cold shutdown. Generally speaking, the following three conditions must be met in order to attain the DHR entry conditions.

(a) Feedwater must be available for RCS heat removal.

(b) RCS pressure must be controlled by using pressurizer spray or the PORV, and lowered while maintaining enough subcooled margin.

(c) The operator must be able to control and lower the secondary side pressure by manipulating the turbine bypass valves or the atmospheric dump valves so that heat transfer can be maintained as the RCS is depressurized.

The PORV is used mainly in connection with the operation (b) described above. To depressurize RCS, manual operation of the pressurizer spray is generally preferable to opening the PORV since no steam will be discharged to the PRT (pressurizer relief tank) so as to possibly cause the rupture of PRT rupture disk. The pressurizer spray, however, may not be available if RCPs (reactor coolant pumps) are tripped by the operator, following the HPI actuation, to comply with the instruction given in the emergency procedure guidelines. If the faulty steam generator can be identified and isolated within reasonable time, the operator can probably restart the RCPs provided that enough subcooled margin is maintained to prevent possible cavitation of the RCPs during the subsequent depressurization. If the RCPs are not restarted, manual operation of the PORV becomes necessary in order to achieve the desired reduction in RCS pressure to attain the DHR entry condition. It should be remarked that there is an alternative way of achieving the DHR entry conditions in case the more conventional way of RCS cooldown described above fails. This is to open the boron precipitation line to blow down the RCS to within the capacity of the LPI pumps as mentioned earlier.

The PORV can also be useful if timely cooldown to the DHR entry conditions becomes impracticable and, consequently, the need to maintain a long-term cooling at hot conditions occurs. As logically elucidated by the fault tree shown in Figure 7, there are two ways of achieving this. If feedwater is available, the RCS can be cooled and depressurized to below the MSRV (main steam relief valve) set point so that the faulty steam generator can be effectively isolated by closing the valve off the main steam line. The PORV can be used in combination with the heat transfer provided by the intact steam generator to further depressurize the RCS to the point where HPR can be activated. The RCS inventory that has been blown down through the PORV and collected in the containment sump can then be recirculated for the cooling. It is obvious that this mode of cooling cannot be utilized if one or more MSRVs on the faulty steam generator stick open.

An alternative means of maintaining a long-term hot shutdown condition is to replenish the BWST inventory (in 14 hours) so that the HPI can be continually put to use. This will, however, cause greater contamination of the secondary system and may lead to excessive release of radioactive nuclides.

2.1.4 Estimation of Oconee 3 SGTR Core-Melt Frequency Reduction

The event tree shown in Figure 4 suggests that the core melt sequences originating from SGTR events can lead to two types of plant damage states defined by the core melt bins IR (early core melt) and IIR (late core melt). To quantify the frequencies of the core melt sequences, all the sequences belonging to the same bin can first be combined to obtain the minimal cut sets at the bin level. The Boolean expressions for each of these bins can thus be written as:

$$\text{Bin IR} = R\,\overline{B}_R U_R + R\,B_R U_R \tag{5}$$

$$\text{Bin IIR} = R\,\overline{B}_R \overline{U}_R X_R O + R\,B_R \overline{U}_R X_R O \tag{6}$$

It should be remarked that no further simplification by applying the rules of Boolean algebra is made because the meanings for the events U_R, X_R, and O vary depending on the status of event B_R.

The above Boolean expressions form the basis of bin-level fault trees that can be solved and quantified using computer codes such as the SETS code[11] to obtain the core melt frequencies. In the Oconee PRA, the quantitative results of the analysis performed to obtain core melt frequencies are presented by listing only the dominant cut sets comprising the sequence type assigned to the core melt bin and their frequencies of occurrence.

As pointed out earlier, the primary objective of this study was to evaluate the reduction in core melt frequencies that can be obtained from improving the reliability of PORV and its associated block valve, by making use of the quantitative results presented in the Oconee PRA or the IPPSS. No attempt, therefore, was made to carry out independent calculations by using the SETS code[11] for checking the correctness or accuracy of the quantitative results presented in the Oconee PRA. It should be mentioned, however, that there is a separate task group at BNL currently being engaged in the review of the Oconee

PRA including independent calculations using the SETS code. Consultations with the colleagues in this group were held whenever questions arose.

In view of the way the quantitative results are presented in the Oconee PRA, a logical approach that can be taken to evaluate the possible benefit of an improved PORV reliability is to scrutinize the dominant minimal cut sets in the light of the relevant fault trees to determine whether or not they comprise the primary events (or basic events) related to the hardware failure or the operational failure of both the PORV and the block valve. If they do, the frequencies of the minimal cut sets can be recalculated by changing the probabilities or unavailabilities assigned to those primary events to reflect a hypothetical improvement in the PORV reliability. Any possible reduction in the core melt frequencies can thus be quantified.

A thorough examination of the dominant minimal cut sets belonging to core melt bin IR and IIR, however, revealed that none of the events represented by their components corresponds to the primary (or basic) events related to hardware or other operational failures of PORV and its block valve. Improving the reliability of PORV and its block valve, therefore, has no effect on changing the core melt frequencies derived from those cut sets. The event tree sequences and their occurring frequencies for the SGTR initiating events are summarized in Table 1, together with a brief description of the core melt sequence of each type.

It should be pointed out that in the Oconee PRA, the cut-off value for the core melt frequency calculation using the SETS code was 10^{-8}. In other words, any cut set with a value smaller than 10^{-8} is omitted in obtaining the summation of the core melt frequencies.

From Table 1, it can be observed that the contributory cause for bin IR core melt sequence is the failure to initiate the HPI system following the SGTR due to various reasons such as failure of the BWST to provide suction, failure of suction valve to open or loss of low pressure service water, etc. For core melt bin IIR, type A, a main steam relief valve on the faulty SG stuck open following the SGTR so that HPR can not be chosen as an option for maintaining long-term cooling. The BWST fails to be refilled in 14 hours and DHR with the low pressure system also fails, leading to an eventual core melt. The core melt in bin IIR, type B, occurs in consequence of failure of low pressure injection system to function for HPR or DHR, with an additional failure to refill the BWST.

It is thus evident that none of the dominant cut sets contains any primary event related to failures of PORV or its block valve. The Boolean expression for the bin-level fault tree suggests that any cut set contribution to the core melt frequencies which might have resulted from failures in the operation or hardware of PORV or block valve is either too small to be retained in the SETS calculations or can only appear within the sum of other bin IIR cut sets (i.e., 2.7×10^{-7}) shown in Table 1. The latter amounts to roughly 10% of the total SGTR core melt frequency, and can be considered as the upper bound for the magnitude of core melt frequency reduction attainable by improving the reliability of PORV and its block valve. A conservative but closer estimate based on the results of an independent calculation recently performed at BNL using the SETS code[11] revealed that less than a half of this core melt frequency (i.e., 2.7×10^{-7}) actually receive contributions from cut sets comprising primary events related to PORV or block valve failures. This implies that, for the SGTR initiating events, the actual reduction in core melt frequency obtainable by improving the reliability of PORV and its associated block valve is approximately 1.4×10^{-7}, about five per cent of the total SGTR core melt frequency. This relatively minor benefit in the core melt frequency can be attributed to alternative success paths such as pressurizer spray or the boron precipitation line, which are potentially available to attain the DHR entry condition or hot shutdown condition in case of a failure of the PORV.

Before concluding discussions in this section, it should be remarked that, in contrast to the Oconee PRA, the independent BNL calculation[4] yielded one dominant cut set containing a primary event related to the operation of the PORV. This cut set is listed below together with a brief definition of its primary events.

Frequency = 4.3×10^{-8}/RY

$R(8.6 \times 10^{-3})$ * OBWSTH(0.5) * XRRCPH(0.01) * XOLP1034H(0.1) * RC660SVH(0.01)

R:	SGTR initiating frequency
OBWSTH:	Failure to initiate BWST refill in 14 hours (Figure 7)
XRRCPH:	Operator fails to restart RCPs (Figure 6)
XOLP1034H:	Operator fails to open LP-103 and LP-104 (Figure 5)
RC660SVH:	Operator does not open valve (PORV) (Figure 16)

The primary events constituting the components of the above dominant cut set can be found in the respective fault tree (the figure number of which is shown inside the parentheses). For each of the components, the numeral shown inside the parentheses denotes the probability assigned to the event used in the BNL calculation. The reason why this cut set did not appear as a dominant cut set in the Oconee PRA is probably that smaller probabilities were assigned to the primary events, XRRCPH, XOLP1034H, and RC660SVH so that the frequency became smaller than the cutoff value of 10^{-8}.

Although the primary event, RC660SVH, appearing in the above dominant cut set is relevant to the operation of PORV, it actually represents the operator's failure to take proper action to open the PORV, a human error. Improving the reliability of the hardware of PORV has virtually no effect on reducing the probability of this event. It is obvious, however, that if this human error can be made much smaller than 0.01, this cut set will become negligibly small. Another significant input data change made in the BNL's SETS code calculation was that the unavailability of the primary event, RC4MVOCM (i.e., RC-4 block valve was closed prior to demand), was changed from 0.033 (the value used in the Oconee PRA) to 0.8 in view of the fact that the PORV block valve is closed 75-80% of the time during normal operation.[8] This change, however, was found to have negligible effect on the results of SGTR core melt frequency calculations.[4]

2.2 Stuck-Open PORV

2.2.1 A General Description of Indian Point 3 Stuck-Open PORV Event

In the IPPSS, a stuck-open PORV is treated under the small LOCA event-tree as shown in Figure 8. This event-tree applies to all RCS (reactor coolant system) ruptures initiated by random pipe breaks or valve failures, with blowdown rates equivalent to double-ended circumferential breaks of a pipe less than two inches in diameter. The structure of the event-tree was examined and found to be consistent with the block diagrams prepared by the NRC staff for depicting the transient scenarios of a stuck-open PORV. The system and operator functions are represented by the following event-tree top events.

ET-3: Small LOCA initiating event
TK: Refueling water storage tank
K-3: Reactor trip
SA-1: Safety injection actuation signal
HH-2: High head pumps
L-1: AFWS actuation and secondary cooling
OP-1: Primary cooling feed and bleed
CF-1: Fan coolers
R-2: Recirculation cooling
CS: Containment spray
NA: Sodium hydroxide addition
RS: Recirculation spray

Detailed explanations of each of the top-event are available in the IPPSS.

2.2.2 Computation of Indian Point 3 Stuck-Open PORV Core-Melt Frequency

In this study, the following Boolean expressions were derived, based on this event-tree, for computing the core-melt frequencies corresponding to eight different plant damage states.

$$\mathrm{SLFC} = (\mathrm{const})*\overline{\mathrm{CF1}}*\overline{\mathrm{CS}}*\mathrm{R2} \tag{7}$$

$$\mathrm{SLF} = (\mathrm{const})*\overline{\mathrm{CF1}}*\mathrm{CS}*\mathrm{R2} \tag{8}$$

$$\mathrm{SLC} = (\mathrm{const})*\mathrm{CF1}*\overline{\mathrm{CS}}*\mathrm{R2} \tag{9}$$

$$\mathrm{SL} = (\mathrm{const})*\mathrm{CF1}*\mathrm{CS}*\mathrm{R2} \tag{10}$$

$$\mathrm{SEFC} = \overline{\mathrm{TK}}\ \overline{\mathrm{K3}}\ \overline{\mathrm{SA1}}\ \overline{\mathrm{CF1}}\ \overline{\mathrm{CS}}*(\mathrm{H2} + \mathrm{L1}*\mathrm{OP1}) \tag{11}$$

$$\mathrm{SEF} = \overline{\mathrm{SA1}}\ \overline{\mathrm{CF1}}*\mathrm{TK} + \overline{\mathrm{K3}}\ \overline{\mathrm{SA1}}\ \overline{\mathrm{CF1}}\mathrm{CS}*(\mathrm{H2}+\mathrm{L1}*\mathrm{OP1}) \tag{12}$$

$$\mathrm{SEC} = \overline{\mathrm{TK}}\ \overline{\mathrm{K3}}\ \overline{\mathrm{CS}}*\ \{\mathrm{SA1} + \mathrm{CF1}*(\mathrm{H2} + \mathrm{L1}*\mathrm{OP1})\} \tag{13}$$

$$\mathrm{SE} = \overline{\mathrm{TK}}\ \overline{\mathrm{K3}}\mathrm{CS}*\ \{\mathrm{SA1} + \mathrm{CF1}*(\mathrm{H2} + \mathrm{L1}*\mathrm{OP1})\} + \mathrm{TK}*(\mathrm{SA1}+\mathrm{CF1}) \tag{14}$$

where const = $\overline{\mathrm{TK}}\ \overline{\mathrm{K3}}\ \overline{\mathrm{SA1}}\ \overline{\mathrm{H2}}\ \overline{\mathrm{L1}*\mathrm{OP1}}$

By substituting the data shown in the IPPSS for the frequency of the small LOCA initiating event and other branch point split fraction for each of the event-tree top events (corresponding to eight electric power states), the following results were obtained.

Plant Damage State	Core-Melt Frequency
SLFC	8.26-5
SLF	2.99-9
SLC	7.41-10
SL	5.13-14
SEFC	2.82-6
SEF	5.85-10
SEC	1.53-7
SE	1.01-8
Total =	8.56×10^{-5}/RY

This is in very good agreement with the value, 8.6×10^{-5} listed in the IPPSS. In the above calculations, the frequency of the small LOCA initiating event was taken to be 0.0201 per reactor year, which includes all small LOCA initiating events such as small pipe breaks and stuck-open PORVs or safety valves etc. To estimate the frequency of a small LOCA caused solely by a stuck-open PORV, a fault-tree as shown in Figure 9 was constructed to analyze the logic for the failure of the PORV path to close following openings due to an automatic actuation or a manual action. Since there are two parallel trains each containing a PORV and a block valve, failure of valves to close in any single train can result in a stuck-open PORV. Also, within a single train, the block valve has to fail to close concurrently with the PORV in order for the PORV path to fail to close. The probability for the PORV path to fail to close after automatic or manual openings was estimated to be 1.6×10^{-4}. If the PORV challenging frequency for all the transients is known, the frequency of a stuck-open PORV can be calculated by multiplying the challenging frequency by this probability. For example, if the PORV challenging frequency is known to be 0.3 challenge/year, the frequency of a stuck-open PORV will be 4.8×10^{-5}/year. The core-melt frequency due to a stuck-open PORV then becomes 8.56×10^{-5} $(4.8\times10^{-5}/0.0201) = 2.0\times10^{-7}$/ry. If the reliability of PORVs and the associated block valves can be improved so that there will be no possibility of a stuck-open PORV, this core-melt frequency can be totally eliminated.

It should be remarked that, according to a rough estimate, 55% of all Westinghouse block valves are currently said to be gagged to circumvent PORV leakages. Leakages from PORVs, however, are usually not severe enough to be regarded as a LOCA. Detection of such leakage is feasible and any ensuing loss of RCS inventory can be compensated by adjusting the charging flow. Inadvertent opening of the block valve, followed by failure to reclose it, therefore, is not treated as a PORV LOCA in this report.

There is also a concern from a different aspect - a concern regarding the possibility of increasing the frequency of small LOCAs due to a stuck-open safety valve in consequence of gagging the block valves. The safety valves are customarily sized to protect the RCS (reactor coolant system) without necessitating the PORVs to be operational during plant operation. The small

LOCA initiating frequency of 0.021/reactor year cited earlier includes the frequencies of both stuck-open PORV and stuck-open safety valves. By virtue of the comparatively higher pressure set point for the safety valves, the challenging frequency of PORVs and safety valves, as a whole, tends to decrease if the block valves are gagged. The occurring frequency of a stuck-open safety valve, nevertheless, may increase due to the higher likelihood for a safety valve to fail open after discharging water rather than steam.

Besides ATWS, there are a few other accident sequences that can conceivably lead to a situation in which pressurizer becomes solid, i.e., filled with water. Included in this category are loss of all feedwater accident (with no recovery in 10 minutes), and steamline break or other overcooling transients which can cause primary system depressurization that leads to actuation of HPI (high pressure injection). As will be more fully discussed later in Section 2.2.4 for Oconee 3, the availability of PORVs has a tendency to prevent the occurrence of a stuck-open safety valve for transients involving actuation of HPI without LOCA. This is primarily owing to the fact that, in this case, safety valves are challenged to relieve water only if the PORVs fail to open. For loss of all feedwater transients (with no recovery in 10 minutes), on the other hand, safety valves are challenged to relieve water irrespective of the opening of PORVs, due to the severity of the accident and the relatively small relief capacity of the PORVs. The availability of PORVs, therefore, has less significance in preventing a stuck-open safety valve for the latter type of transients. Although there is no feasible means of estimating the possible increase in the frequency of a stuck-open safety valve attributable to the unavailability of PORVs, such an increase is believed to have negligible impact on the overall core damage frequency for Indian Point 3.

2.2.3 A General Description of Oconee 3 Stuck-Open PORV Event

The failure of the pressurizer PORV or SRVs to reclose after their openings during a transient is treated in the Oconee PRA as a transient-induced small LOCA under the top event, Q, loss of RCS integrity, in the generalized transient event tree having 14 different transient initiating events, denoted by T. The TQ sequences are then transferred through a branch point to a separate small LOCA event tree that has its own initiating events, S, as shown in Figures 10 and 11. For convenience of ready reference, part of the fault trees developed in the Oconee PRA for supporting the logic of the event tree top event, Q, is illustrated in Figure 12.

The PORV or SRVs may be challenged by an increase in RCS pressure due to: (a) interruption of feedwater to steam generators resulting in the loss of SG heat transfer; (b) actuation of HPI without LOCA resulting in the addition of mass to RCS; and (c) spurious low RCS pressure signal resulting in the energizing of the pressurizer heater banks and addition of energy to RCS. Numerous causes for the interruption of feedwater to SG are identified in the Oconee PRA through system level fault tree, TQ08.[6] They include feedwater line break, failure of emergency feedwater system, and loss of main feedwater due to losses of condenser vacuum, instrument air, and ICS (integrated control system), etc. Similarly, the event (b), actuation of HPI without LOCA, is further developed through a system level fault tree[7] to adduce its various causes including steam line break, overcooling due to excessive MFW (main feedwater) or EFW (emergency feedwater), and the stuck-opening of more than two MSRVs (main steam line relief valves) or turbine bypass valves (TBV), etc. These events are normally capable of cooling the RCS to the HPI set point to actuate the HPI flow. The event (c), spurious low RCS pressure signal, is by itself considered as an initiating event.

The PORV is used to relieve RCS pressure mainly in connection with the events related to (a) and (b) described above. For event (c), the spurious low RCS pressure signal automatically causes PORV to be fail-closed, and thus PORV will not open. There are two system level top events which bear relationship to the use or failure of PORV. These are TRC1 (failure to terminate PORV relief) and TRC401 (PORV fails to open on demand) as can be seen in the fault trees shown in Figure 12. The supporting logic for these events are elucidated through system level fault trees in the Oconee PRA (see Figure 13) including that for human errors, PORV hardware failure or the failure of the block valve to open or close, etc.

<u>The Conditions Required to Challenge PORV or SRVs as Assumed in the Oconee PRA</u>

The conditions under which PORV or SRVs can be challenged differ slightly depending on whether RCS pressurization is caused by interruption of feedwater to steam generators or by actuation of HPI without LOCA.

The Oconee PRA assumes that the PORV is challenged if RCS heat removal fails due to interruption of feedwater to steam generators for a sufficient period of time. The SRVs are assumed to be challenged for <u>steam relief</u> only if the PORV fails to open on demand. If, however, the total feedwater loss persists for more than ten minutes, the SRVs are assumed to be challenged to discharge <u>liquid</u> regardless of whether the PORV is open.

For HPI actuation without LOCA, the SRVs can be challenged only if the PORV fails to open on demand. In other words, the SRVs are not required to open if the PORV opens successfully on demand. If challenged, however, the SRVs will first open to discharge steam, and if the operator fails to throttle HPI before the RCS becomes water-solid, will be required to relieve liquid eventually. These logics are delineated in the system-level fault trees, TQ06 and TQ07, as shown in Figure 12.

<u>Boolean Expressions for Core Melt Bins Resulting from Small Break LOCA (Including a Stuck Open PORV)</u>

As illustrated by the small break LOCA event tree shown in Figure 10, small break LOCA events, either transient-induced or as initiating events, can lead to core melt sequences denoted by bin I (early core melt) or bin II (late core melt) depending upon the success or failure of the preceding event tree top events. The Boolean expressions for these core melt bins can be derived by taking the union of the core melt sequences belonging to the same bin, followed by simplifications using the rules of Boolean algebra. They can be shown to be:

$$\text{Bin I} = (TB\overline{PL}W\overline{U}_T + TQ + S) * Y_s X_s + (TQ + S) * U_s \qquad (15)$$

$$\text{Bin II} = (T\overline{Q}B\overline{PL}W + TQ + S) * \overline{U}_s \overline{Y}_s X_s \qquad (16)$$

The bin-level fault trees can be constructed based on these expressions and then solved and quantified to obtain the core melt frequencies for each of the core melt sequences. This was done in the Oconee PRA and the results were presented in terms of the dominant minimal cut sets and their occurring frequencies. For quick reference, the number of dominant cut sets and their contributions to the core melt frequencies are summarized according to the sequence type in Tables 2 and 3 for core melt bin I and bin II respectively. The column allocated for the sum of all other cut sets lists the contribution to the core melt frequencies by all those cut sets which are not dominant. As pointed out earlier, the Oconee PRA ignores any cut set smaller than 10^{-8}.

Glancing through the list of these tables, it can be noted that the dominant cut sets for bin I consists mainly of the event tree sequences TQU_s, SU_s, SY_sX_s, and TQY_sX_s, whereas those for bin II are composed of the sequences $TQ\overline{UY}X_s$ and $\overline{SUY}X_s$. This is consistent with the Boolean expressions given by Eq. (15) and Eq. (16).

2.2.4 Estimation of Oconee 3 Stuck-Open PORV Core-Melt Frequency Reduction

To explore possible merit of improving the reliability of PORV, the primary events constituting the components of each of the dominant minimal cut sets are scrutinized by reference to the relevant fault trees to determine whether they are related to the operation or hardware of the PORV and its block valve. The results of this scrutiny are summarized and discussed in the following:

(1) The sequence types $B(TQU_s)$ and $D(T_6QU)$ of core melt bin I involve transient-induced small break LOCAs resulting in RCP (reactor coolant pump) seal failures. They are virtually unrelated to the operation of PORV, and, hence, can be excluded from further considerations.

(2) The three sequence types starting with the letter S, i.e., SY_sX_s and SU_s in bin I and sequence $\overline{SUY}X_s$ in bin II, are those caused by a small break LOCA initiating events such as a small pipe break and an inadvertent opening of PORV or SRVs. The latter does not include a small valve leak which can be compensated by charging flow. The core melt frequencies of these sequences add up to 6.13×10^{-6}/ry based on the initiating frequency of 0.003/ry. If the frequency of an inadvertent opening of PORV as an initiating event is known, its core melt frequency can be computed proportionally. For example, if the initiating frequency is known to be 6×10^{-5}/ry, then the core-melt frequency due to an inadvertent opening of a PORV becomes $6.13\times10^{-6}\ (6\times10^{-5}/0.003) = 1.23\times10^{-7}$/ry. If the PORV reliability can be improved so that no inadvertent opening of PORV can occur, the core melt frequency due to this cause can be eliminated.

(3) There are seven types of event tree sequences (i.e., Types E, F, H, and I in bin I and Types A, B, and D in bin II) which are transient-induced small break LOCAs resulting in a stuck-open SRV discharging liquid. The core melt frequencies for these seven sequences add up to 2.84×10^{-7}/ry, with additional contributions from all other cut sets amounting to 6.8×10^{-8}/ry. These are transient sequences involving the failure of the feedwater system and, as such, are governed by the system level fault tree, TQ06, shown in Figure 12. All of the primary events constituting

the components of these dominant cut sets belonging to these event tree sequences were examined closely and found to be unrelated to the operation or hardware of PORV. The system level fault tree, TQ06, suggests that any PORV related primary events can only appear under the system level fault trees, TRC1 (failure to terminate PORV relief) and TRC401 (PORV fails to open on demand). This implies that contribution to the core melt frequencies from any cut set which might contain primary events related to PORV operations can only be included within the sum of all other cut sets, i.e., 6.8×10^{-8}/ry. For those seven event tree sequences, therefore, this value represents the maximum reduction in core melt frequency attainable by improving the PORV reliability. One basic reason for this relatively insignificant benefit is that improving the PORV reliability can produce larger effect on preventing SRVs from being challenged to relieve <u>steam</u> but not liquid. As mentioned earlier, for RCS pressurization caused by loss of feedwater, challenging of the SRVs is contingent upon failure of PORV (to open on demand) for discharging steam only. Since the liquid relieving capacity of PORV is relatively small compared to that of SRVs, the SRVs can be challenged to relieve liquid regardless of whether or not the PORV is open. If the PORV can be made completely reliable, SRVs will not be challenged to relieve steam, and, hence, there will be no possibility of a stuck open SRV discharging steam.

Observation of the results shown in Tables 2 and 3, however, revealed that essentially all of the small break LOCA transients involving failure of the feedwater system end up with a stuck-open SRV discharging liquid rather than steam. This is mainly due to the more dominant nature of the cut sets containing the primary event, RCSRVLC (either SRV fails to close after liquid relief) shown under gate Q12 in Figure 12 and QFDWR (failure to recover feedwater in 10 minutes) also shown under gate Q12 in the same figure. In the Oconee PRA, the probability for these events was taken to be 0.1 and 0.7 (or 1.0) respectively. The probability for the event Q12, SRV fails to close after liquid relief, therefore, is 0.07 (or 0.1). This is much larger than that for Q11 (= 1.2×10^{-4}, SRV fails to close after steam relief) which can be obtained by multiplying the probability for RCSRVSC (= 9.6×10^{-3}, either SRV fails to close after steam relief) by the probability for TRC401 (= 0.012, PORV fails to open on demand). The cut sets going through the gate Q12 is thus much more dominant as compared with those going through the gate Q11. Even if the unavailability of PORV is increased to 0.812 by reason that the PORV block valve is closed 80% of the time, the probability for Q11 is still only about 12% of that for Q12. Since most of the core melt sequences belonging to core melt bin I and bin II involve a stuck open SRV discharging liquid for which PORV reliability is less important, the risk reduction achievable by improving the PORV reliability can be expected to be relatively insignificant.

(4) All the three dominant cut sets belonging to the sequence type G in bin I and three out of the four dominant cut sets belonging to the sequence type F in bin II involve a stuck open SRV discharging steam. The initiating event for these sequences is T_{13}, a spurious low pressurizer pressure signal which could result from failure within the integrated control system (ICS) and the auxiliary control system. It results in the actuation of pressurizer heaters, closure of the pressurizer spray valve, and

the blocking of PORV from opening. There exists, therefore, the potential for the RCS pressure to rise and challenge the SRVs, with the possibility of a stuck open SRV. If this stuck open SRV is followed respectively by a HPI failure and a failure in high pressure and low pressure recirculations, the core melt sequences of bin I, type G and that for bin II, type F result.

The unavailability of a PORV to relieve RCS pressure, in these cases, is caused by the control action taken in response to the spurious low pressurizer-pressure signal, and not by the failure of PORV. Improving the PORV reliability, therefore, cannot contribute to reduction of core melt frequencies for these sequences.

(5) In scrutinizing all the dominant cut sets belonging to bin I and bin II, it was revealed that only all of the three cut sets in type E and one cut set in type F of bin II contain a primary event relevant to the operation of PORV and its block valve. For reference, these cut sets are listed in the following:

(a) Sequence type E, bin II

T(7.0) * RC4MVOCM(3.3×10^{-2}) * RCSRVLC(0.1) * XHPLPR12H($3. \times 10^{-4}$) * MSRVC($3. \times 10^{-3}$)

T(7.0) * RC4MVOCM(3.3×10^{-2}) * RCSRVLC(0.1) * XHPLPR12H($3. \times 10^{-4}$) * IM41($2. \times 10^{-3}$)

T(7.0) * RC4MVOCM(3.3×10^{-2}) * RCSRVLC(0.1) * XHPLPR12H($3. \times 10^{-4}$) * ICRDRTVO($2. \times 10^{-3}$)

Approximate frequency = 5.8×10^{-8}/ry

(b) Sequence type F, bin II

T8(0.01) * RC4MVOCM(3.3×10^{-2}) * RCSRVLC(0.1) * XHPLPR12H($3. \times 10^{-4}$)

Approximate frequency = 1.0×10^{-8}/ry.

The number shown inside the parentheses denotes the frequency of initiating event for T and T8, and for the rest, the unavailability of the individual event.

The initiating event for sequence type F (i.e., T8) is a spurious actuation of HPI, whereas those for the sequence type E sequences are all the transients that lead to overcooling events directly or with additional failures such as MSRVC (two or more MSRVs stuck open) and IM41 (common mode faults affecting all turbine bypass valves), etc. Overcooling normally results in RCS depressurization that actuate HPI. For these event tree sequences, therefore, the supporting logic for the event tree top event, Q, failure of RCS integrity, is governed by the system level fault tree, TQ07, as shown in Figure 12.

All of the above cut sets contain a fault tree primary event, RC4MVOCM (RC-4 block valve closed prior to demand) as one of their components. In the Oconee PRA, this primary event was assigned a probability of 0.033. Also, the primary event QHPIH (operator fails to throttle HPI) was implicitly given a value of unity. This is the reason why QHPIH does not appear explicitly in the above dominant cut sets. Since the PORV block valve is actually 80% of the time closed, a more proper value for RC4MVOCM is 0.8. Furthermore, thermal hydraulic analysis performed for transients involving overcooling such as the main steam line break[1] shows that there is at least 10-15 minutes of time available for the operator to throttle HPI before the RCS pressure reaches the PORV set point. A more reasonable value for QHPIH is, therefore, 0.05 rather than 1.0.[4] Moreover, if a more up-to-date value (8.0E-3) is used for MSRVC, the core melt frequency for type E sequences becomes 1.1×10^{-7}/ry rather than the previously shown value of 5.8×10^{-8}/ry. Similar modifications can be made on the dominant cut set listed under sequence type F, bin II, involving a spurious actuation of HPI at full power. In this case, the operator will have less time to throttle HPI so that a more proper value for QHPIH is 0.1.[4] Together with the change of RC4MVOCM to 0.8, the core melt frequency from this cut set becomes 2.4×10^{-8}/ry rather than the value 1.0×10^{-8}/ry shown previously. The core melt frequencies due to these four cut sets, therefore, total 1.34×10^{-7}/ry.

As pointed out earlier, these four cut sets are the only dominant cut sets that bear direct relation to the operation of PORV and its block valve. The primary event, RC4MVOCM, however, is dominated by human actions. Although the PORV is not required to be operational during normal plant operation, closing the block valve 80% of the time is a precautionary measure taken to avoid loss of RCS inventory due to possible leakage of the PORV. If the reliability of PORV can be improved so that the block valves need not to be closed during plant operation, the core-melt frequencies associated with these four cut sets can be essentially eradicated.

2.3 Use of PORVs in Cooldown to Cold Shutdown

2.3.1 A General Description of Cooldown to Cold Shutdown for Indian Point 3 Transient Initiating Events

Since virtually all of the transient initiating events require reactor and turbine trips and the subsequent cooldown to hot or cold shutdown conditions, all of the transient event-trees presented in the IPPSS were examined individually, and used as the basis for deriving Boolean expressions for computing the core-melt frequencies corresponding to various plant damage states. The event-trees for the following nine transient initiating events were scrutinized.

Event-Tree No.	Description of Transient
5	Steam-line break inside containment
6	Steam-line break outside containment
7	Loss of main feedwater
8	Closure of main steam isolation valve (MSIV)
9	Loss of reactor coolant system flow
10	Core power excursion
11a	Basic turbine trip

11b	Modified turbine trip due to loss of offsite power
12	Reactor trip

The individual event-tree as well as detailed descriptions of its top-events are presented in the IPPSS for each of the above transients. The nine event-trees employ the following symbols to identify system and operator functions.

SA-2:	Safety injection signal and high head pumps
K3:	Reactor trip
MS-1 or MS-2:	MSIV trip
L-1:	AFWS actuation and secondary cooling
OP-1 or OP-2:	Feed and bleed cooling
R3:	Recirculation cooling
CF-2:	Fan coolers
CS:	Containment spray
NA:	Sodium hydroxide addition
TT or TT1:	Turbine trip
OP-3:	Operator stabilizes transient (Event-tree 8) Operator terminates excursion (Event-tree 10)
K-5:	Turbine runback
LS:	Reactor coolant pump seal LOCA (Event-tree 11b)
EP0:	ac power state split fraction at time = 0 (Event-tree 11b)
EP1:	ac power state split fraction at time = 30 minutes (Event-tree 11b)
EP2:	ac power restored in less than 60 minutes (Event-tree 11b)
EP3:	ac power restored in less than 3 hours (Event-tree 11b)

A thorough examination of these event-trees revealed that during the process of cooldown subsequent to the reactor and turbine trips, PORVs are used mainly in connection with "feed and bleed" cooling necessitated by loss of SG secondary cooling due to failures in actuating AFWS (auxiliary feedwater system).

The event trees for these transient initiating sequences suggest that as long as AFWS actuation and steam generator secondary cooling are successful, no core damage will ensue. Depressurization of RCS (reactor coolant system) by means of PORVs, in this case, is not required to avoid core damage. This is partly due to the fact that the AFWS is provided with sufficient water supply to remove the decay heat for at least 24 hours under hot shutdown condition. Moreover, the primary water supply from the condensate storage tank is augmented by a secondary water supply from the 1.5 million gallon city water storage tank shared by Indian Point Units 2 and 3. Redundant level indicators and control room alarms are also provided for the condensate storage tank. If cooldown to RHR entry conditions is desired, RCS depressurization can further be achieved by manually controlling the AFW flow and discharging steam from steam generator reliefs or dumps. In case of the failure of AFWS actuation and steam generator secondary cooling, however, core heat removal must be accomplished by "feed and bleed" cooling using high head injection pumps and the PORVs.

2.3.2 Core-Melt Frequency Calculations for Indian Point 3 Transient Initiating Events

To enable quantification of event-tree sequences for each of these transients, Boolean expressions were derived for each of the event-trees, by taking the union of all of the event-tree sequences belonging to the same plant damage state, followed by simplifications utilizing the rules of Boolean algebra. It was found that these Boolean expressions, each representing a particular plant damage state for the nine different transients, can be grouped together and expressed in the following compact form.

$$\mathrm{SLFC} = \overline{\mathrm{CF2}}*\overline{\mathrm{CS}}*\mathrm{Const1} \tag{17}$$

$$\mathrm{SLF} = \overline{\mathrm{CF2}}*\mathrm{CS}*\mathrm{Const1} \tag{18}$$

$$\mathrm{SLC} = \mathrm{CF2}*\overline{\mathrm{CS}*}\mathrm{Const1} \tag{19}$$

$$\mathrm{SL} = \mathrm{CF2}*\mathrm{CS}*\mathrm{Const1} \tag{20}$$

$$\mathrm{TEFC} = \overline{\mathrm{CF2}}*\overline{\mathrm{CS}}*\mathrm{Const2} \tag{21}$$

$$\mathrm{TEF} = \overline{\mathrm{CF2}}*\mathrm{CS}*\mathrm{Const2} \tag{22}$$

$$\mathrm{TEC} = \mathrm{CF2}*\overline{\mathrm{CS}}*\mathrm{Const2} \tag{23}$$

$$\mathrm{TE} = \mathrm{CF2}*\mathrm{CS}*\mathrm{Const2} \tag{24}$$

The symbols, Const1 and Const2, appearing in the above expressions are summarized in the following for the nine different event-trees.

Event-tree No.	Const1	Const2
ET-5	$\overline{\mathrm{SA}}\ \overline{\mathrm{K3}}\ \overline{\mathrm{OP1}}*\mathrm{R3}*(\mathrm{L1}+\mathrm{MS1})$	$\overline{\mathrm{K3}}*(\mathrm{SA2}+\mathrm{OP1})*(\mathrm{L1}+\mathrm{MS1})$
ET-6	$\overline{\mathrm{SA}}\ \overline{\mathrm{K3}}\ \overline{\mathrm{OP1}}*\mathrm{R3}*(\mathrm{L1}+\mathrm{MS1})$	$\overline{\mathrm{K3}}*(\mathrm{SA2}+\mathrm{OP1})*(\mathrm{L1}+\mathrm{MS1})$
ET-7	$\overline{\mathrm{K3}}\ \overline{\mathrm{OP2}}\ \overline{\mathrm{R3}}*(\mathrm{L1}+\mathrm{T1}*\mathrm{MS2})$	$\overline{\mathrm{K3}}\mathrm{OP2}*(\mathrm{L1}+\mathrm{T1}*\mathrm{MS2})$
ET-8	$\overline{\mathrm{K3}}\ \overline{\mathrm{OP2}}*\mathrm{R3}*(\mathrm{L1}+\mathrm{TT1}*\mathrm{MS2})$	$\overline{\mathrm{K3}}*\mathrm{OP2}*(\mathrm{L1}+\mathrm{TT1}*\mathrm{MS2})$
ET-9	$\overline{\mathrm{K3}}\ \overline{\mathrm{OP2}}*\mathrm{R3}*(\mathrm{L1}+\mathrm{TT}*\mathrm{MS1})$	$\overline{\mathrm{K3}}*\mathrm{OP2}*(\mathrm{L1}+\mathrm{TT}*\mathrm{MS1})$
ET-10	$\overline{\mathrm{K3}}\ \overline{\mathrm{OP2}}*\mathrm{R3}*\mathrm{K5}*\mathrm{L1}*\mathrm{OP3}$	$\overline{\mathrm{K3}}*\mathrm{OP2}*\mathrm{OP3}*\mathrm{K5}*\mathrm{L1}$
ET-11a	$\overline{\mathrm{K3}}\ \overline{\mathrm{OP2}}*\mathrm{R3}*\mathrm{L1}$	$\overline{\mathrm{K3}}*\mathrm{OP2}*\mathrm{L1}$
ET-11b	$\overline{\mathrm{EPO}}\ \overline{\mathrm{EP1}}\ \overline{\mathrm{LS}}\ \overline{\mathrm{K3}}\ \overline{\mathrm{OP2}}*\mathrm{R3}*\mathrm{L1}$	$\overline{\mathrm{EPO}}\ \overline{\mathrm{EP1}}\ \overline{\mathrm{LS}}\ \overline{\mathrm{K3}}*\mathrm{OP2}*\mathrm{L1}$
ET-12	$\overline{\mathrm{OP2}}*\mathrm{R3}*(\mathrm{L1}+\mathrm{TT1}*\mathrm{MS2})$	$\mathrm{OP2}*(\mathrm{L1}+\mathrm{TT1}*\mathrm{MS2})$

The core-melt frequencies computed based on these Boolean expressions and by adopting the branch-point split fraction data shown in the IPPSS are listed in Table 4. In general, these results are seen to agree well with those presented in the IPPSS (values shown inside brackets). It must be remarked that a typical failure probability assigned to the top events, OP-1 or OP-2 (feed and bleed cooling), in these calculations is 4.5×10^{-3}, of which 4.0×10^{-3} is due to hardware failures of PORVs and block valves as depicted by the fault tree shown in Figure 14. The remainder accounts for that due to operator's errors in making a correct decision to initiate the "feed and bleed" cooling. If the reliabilities of PORVs and block valves can be improved so that they will always function perfectly on demand (i.e., if the probability of OP1 or OP2 is lowered from 4.5×10^{-3} to 5×10^{-4}), the core-melt frequencies can be reduced to those shown in the last column of the table. With the exception of turbine trip (loss of offsite power or loss of service water) and reactor trip (loss of component cooling), the core melt frequencies are seen to be reduced by about 40%. One main reason for this relatively significant risk reduction is that the reliability of both the PORVs and the block valves in this case, is relatively important as compared to human errors. For a successful operation of "feed and bleed" cooling, the block valves must open on demand and that the two PORVs must be both operative at least in the first few hours of their operation. In addition, unlike the case with Oconee 3, the safety valves can not be used for the "feed and bleed" cooling. Consequently, no credit can be given to the availability of the safety valves.

2.3.3 A General Description of Cooldown to Cold Shutdown for Oconee 3 Transient Initiating Events

The Oconee PRA event tree for transient initiating events shown in Figure 11 indicates that no core melt will occur as a consequence of the transient initiating events so long as RCS heat removal by steam generators is successful following the reactor trip and that there is no loss of RCS integrity such as that caused by a stuck-open SRV. The Oconee 3 emergency feedwater (EFW) system is designed to provide a sufficient steam generator secondary-side heat sink for cooling down the RCS from a reactor trip at power operation to the DHR (decay heat removal) entry conditions, i.e., 350 psia and 250°F. For cooldown to cold shutdown conditions, therefore, no PORV is required to depressurize the RCS so long as the EFW system is functioning and that the steam generator secondary cooling is successful. The EFW system normally takes suction from the upper surge tanks. Once the EFW is automatically initiated due to loss of feedwater, the operator is required to begin makeup to the upper surge tanks from the demineralized water system, or the condensate storage tank, or the hotwell via the hotwell pump.

In case of a failure of RCS heat removal by steam generators, an event tree top event denoted by B, the PORV and SRVs are required to relieve the RCS pressure. The failure to provide RCS pressure relief by the PORV and SRVs is represented by an event tree top event, P, the occurrence of which is conservatively assumed to lead to a high pressure core melt. The core melt sequence TBP, however, has very low probability of occurrence, as substantiated by the fact that none of the dominant minimal cut sets shown in the Oconee PRA belongs to this sequence. This is primarily due to the low probability for the event, P (estimated to be roughly $9. \times 10^{-10}$), the occurrence of which requires failure of not only the PORV, but also both of the SRVs.

One important utilization of PORV during the process of cooldown to cold or hot shutdown condition is its use in conjunction with the HPI cooling (feed and bleed). In case of a failure of SG heat removal, prevention of core uncovery can be achieved if the PORV can be opened and full flow from one of the three HPI pumps can be initiated within about 30 minutes after the loss of all feedwater. Even if the PORV fails to open, this HPI cooling can be carried out by the cycling of SRVs at their set points of about 2500 psig, provided that full flow from two HPI pumps can be effectively actuated. This logic is delineated by the fault tree for the event tree top event, U_T, as shown in Figure 15.

2.3.4 Estimation of Core-Melt Frequency Reduction for Oconee 3 Transient Initiating Events

By taking the union of the event tree sequences belonging to the same core melt bin in the transient initiating event trees, the following Boolean expressions for core melt bin III and bin IV can be readily derived.

$$\text{Bin III} = T\overline{Q}B * (Y_T L X_T + U_T + P) \tag{25}$$

$$\text{Bin IV} = T\overline{Q}B\overline{P}\overline{U}_T\overline{Y}_T L X_T \tag{26}$$

To facilitate discussions, the dominant minimal cut sets listed in the Oconee PRA for these core melt bins are summarized in a compact form and shown respectively in Tables 5 and 6. It is noteworthy that all of the dominant minimal cut sets in bin III belong to the event tree sequence TBU, i.e., failure of both the SG cooling and HPI cooling. A thorough examination of all the dominant cutsets belonging to bin III further revealed that, for event tree sequence types A through F, they all comprise a fault-tree primary event UTHPIH (operator's failure to make decision to initiate HPI cooling). In the Oconee PRA, this event is assigned a probability of 0.01.

The role played by the PORV in the HPI cooling can be better understood by reference to the fault tree shown in Figure 15. The probability for the event UT02 (HPIS fails with PORV unavailable) is estimated to be roughly 7 x 10^{-6} based on the logic that its happening requires PORV failure concurrent with the failure of at least two (out of three) HPIS pumps. Meanwhile the probability for the system-level top event, THP1 (HPIS fails) can be estimated to be approximately 1.4 x 10^{-4}. These are much smaller in comparison with the value 0.01 assigned to the primary event, UTHPIH, and thus constitute the reason why the cut sets containing the event UTHPIH are much more dominant. The failure of PORV to open for HPI cooling is further developed in the system-level fault tree TRC201 (Figure 16). It also suggests that any cut set which might contain the primary events related to PORV failures defined in the fault tree for TRC201 can only appear within the sum of all other cut sets in Table 5. The same conclusion was reached after examining all the dominant cut sets belonging to Types G through J of core melt bin III. By taking the total of the sum of all other cut sets for type A through type J in Table 5, one obtains a value of 8.7×10^{-7}/ry. It is clear that not all of this core melt frequency, which can be considered as an upper bound, is due to the failure of PORV or its block valve. In fact, a rough estimate based on an independent BNL calculation[4] indicates that no more than 10% of this value is actually PORV-related. The core melt frequency reduction attainable for the HPI cooling by improving the PORV reliability is, therefore, approximately $9. \times 10^{-8}$/ry.

Core Melt Frequencies for Bin IV

Another important core melt sequence that brings forth several of the dominant minimal cut sets for the transient initiating events is $TB\overline{U}\overline{Y}LX$, which involves failures of RCS heat removal by the SG and its recovery, and a failure to maintain long term core heat removal. This event tree sequence is categorized as core melt bin IV, the brief summary of which is presented in Table 6. Since this event tree sequence presupposes the success of HPI cooling, improving the PORV reliability essentially has no effect on reducing the core melt frequencies pertaining to this core melt bin.

Failure of PORV to Reclose After HPI Cooling

Under the circumstance that HPI cooling is actuated due to the failure of SG heat transfer, the operators are still expected to continue to attempt to recover feedwater flow to the SG so that the HPI cooling can be terminated. If they succeed in doing so, the PORV or the SRVs used in the HPI cooling still must face the challenge to reclose successfully in order that no loss of RCS integrity will occur. The failure of the PORV or the SRVs to reclose after they are opened for HPI cooling is represented by an event tree top event, W, failure to reestablish RCS integrity, in the Oconee PRA. The supporting logic for this top event is illustrated in Figure 17. As can be observed, the SRVs are assumed to be opened for HPI cooling only if the PORV fails to open. Based on the data presented in the Oconee PRA, the probability for the occurrence of W is estimated to be roughly 1.5×10^{-3}. Although improving the reliability of PORV can lower the probability of W, it is virtually ineffectual in reducing the total core melt frequencies because none of the dominant minimal cut sets belonging to bin I through bin IV comprises the failure of the top event W.

2.4 Low Temperature Overpressurization Protection Using PORVs

2.4.1 General Descriptions of Low Temperature Overpressurization Events

An overpressurization event at cold shutdown conditions can take place as a result of unanticipated addition of mass or energy to the RCS. Overpressurization of the RCS due to a "mass input event" can occur, for example, by a spurious actuation of the HPI (high pressure injection) system or by a failure in air supply system which causes the charging flow control valve to open, whereas that due to a "heat input event" can result from erroneous actuation of all pressurizer heaters or from thermal expansion of reactor coolant after starting an RC pump due to stored thermal energy in the steam generators.

Relatively speaking, the "mass input event" has a higher likelihood of occurrence, and is more difficult to alleviate because of the rapidity of its progression. However, it can be more easily characterized quantitatively without performing extensive system transient analysis. In this study, therefore, attention is focused mainly on the "mass input event." One of the most credible mass input events producing a net injection of mass into the reactor coolant system (RCS) involves failure in the air supply system, which causes the charging flow control valve to open and the letdown valve to fail closed. As a result, a net injection of mass by the centrifugal charging pump and a very high pressurization rate occurs. Failure of the charging flow control valve to operate as required can also lead to a mass input event. This could be caused by local valve failure or local failure in the air supply to the flow control valve. Because the reactor is at low pressure and the centrifugal charging pumps flow rate increases with decreasing RCS pressure, a large mass injection causing a rapid RCS pressurization rate would occur.

The PORVs can be utilized to protect the reactor vessel from low temperature overpressurization by lowering their set point from the regular high-pressure set point of 2335 psig (for Indian Point 3). Under cold shutdown conditions, the operator is instructed to reset PORV control selector switch to "Auto Low Temp Position" so that the low-pressure PORV set point of 500 psig can be chosen. To prevent overpressurization of the reactor vessel under low RCS temperature and pressure, Indian Point 3 is equipped with OPS (Overpressurization Protection System). The necessity to have such a protection system arises from the fact that the reactor steel and weld materials have less ductilities at low temperature so that when the reactor vessel is exposed to pressure stresses, it is more susceptible to crack or rupture. The fracture toughness of the vessel and weld materials is further reduced by irradiation during their lifetime. The increase in the nil-ductility transition reference temperature due to radiation damage is primarily a function of the fast-neutron fluence and the concentration of copper and nickel in the reactor vessel material. The Indian Point 3 OPS is a three-channel curve tracking scheme designed to automatically prevent a violation of the reactor vessel temperature/pressure limit curve defined in the Technical Specifications. It is activated when RCS temperature falls below 350°F by closing the states link inside the FCR panel. Its arming occurs when two out of three channels indicate 330°F. This will open the block valves, MOV 535 and MOV 536, if the control switch is in the auto position, thus rendering the PORVs available for pressure relief in case of an inadvertent overpressurization of the reactor vessel.

For plant cooldown from hot to cold shutdown conditions, the Indian Point 3 plant operation procedures further require that other precautionary measures be taken to prevent accidental overpressurization of the RCS. For example, the operator is instructed to transfer charging pump control to manual, and, if the RCS temperature is decreased below 350°F, to place the control switches for the safety injection pumps in the trip pullout position. The high-head safety injection pump discharge valves and all other valves that isolate the flow path into the RCS are also to be closed. To preclude inadvertent actuation of the safety injection system when RCS temperature is below 200°F, the operator is further advised to remove appropriate keys from the key locker to defeat the safety injection logics. All of these preventive actions can effectively reduce the chance of an inadvertent actuation of the safety injection system, the consequence of which is more difficult to mitigate due to its potentially rapid pressurization rate at low RCS pressure. Since the detrimental effect of low temperature overpressurization can be more severe if the reactor is in cold shutdown--water solid condition, the operator is urged to exercise extreme caution in such a case.

Besides the reactor vessel, the integrity of the RHR (Residual Heat Removal) loop is of primary concern under overpressurized condition since part of its piping is designed to operate at relatively low pressure. During plant shutdown and refueling operations, heat removal from the core and RCS is usually achieved by means of the RHR system. For Indian Point 3, the RHR system can be placed in service when RCS is depressurized to less than 450 psig and

RCS temperature is lowered to below 350°F. The system is designed to reduce the RCS temperature from 350°F to 140°F in approximately 24 hours. The RHR loop is partly protected from overpressurization by two RHR suction valves (motor-operated) located inside the containment. These RHR suction valves are pressure interlocked to prevent opening if RCS pressure is above 450 psig. This precludes inadvertent overpressurization of the RHR loop. Both valves are provided with separate pressure channels that sense RCS pressure. Each pressure channel provides two interlocks; first, it prevents the valve from being opened whenever RCS pressure is above 450 psig, and secondly, it automatically closes the valve when RCS pressure exceeds 550 psig. Moreover, to avoid any single failure from rendering both valve circuits inoperable, each valve operator and its pressure channel circuitry is powered from different instrument buses. When RHR is in operation, the RHR piping is further protected from overpressurization by a relief valve which is located near the two RHR suction valves. Since the RHR suction valves are designed to close automatically whenever RCS pressure exceeds 550 psig, this relief valve can serve little purpose of relieving the RCS pressure when the RCS is overpressurized to above 550 psig.

The Oconee 3 plant is also provided with a low temperature OPS (Overpressurization Protection System) similar to that described above (for Indian Point 3), although the detail of its mechanism and operation procedures may differ slightly. Its RHR suction line is also equipped with a relief valve and two normally closed and pressure-interlocked RHR suction valves for the protection of the RHR loop.

2.4.2 Calculation of Core-Melt Frequencies Resulting from Low Temperature Overpressurization Events

To estimate the core-melt frequency due to low temperature overpressurization events, a simple event tree was constructed as shown in Figure 18. The structure of this event tree, which is somewhat conservative, is applicable to both Indian Point 3 and Oconee 3. The conservatism is reflected in the fact that no credit is given to the RHR relief valve and the pressurizer safety valves and that a reactor vessel failure is always assumed to result in a core-melt with a probability of 1.0. As discussed earlier, the RHR relief valve is designed primarily to protect the integrity of the RHR lines from overpressurization events. It is also capable of mitigating overpressurization of the RCS if the RHR suction valves are open. For rapid pressurization rate (up to 100 psi/sec possible[12]), however, the RCS pressure may create a spurious high pressure interlock signal leading to automatic closure of the RHR suction valves (550 psig for Indian Point 3), thus isolating the RHR lines from the RCS. The initiating frequency of LTOP events is taken to be 0.13/ry based on the data shown in Reference 3. It was computed based on the number of events reported before 1979 including about 30 events for the violation of RCS pressure/temperature Technical Specifications. Since 1979, considerable effort has been exerted to enhance the protection of reactor vessel from LTOP events by, for example, implementing the OPS (Overpressurization Protection System) or amending the operation procedures, etc. Even with such enhancements, there have been two reported events[13] of overpressure excursions at low temperature. In addition, from 1979 to mid-1983, there were at least ten reported incidents in which the OPS prevented or mitigated a low temperature overpressurization event. It is noteworthy that some of these incidents were caused by inadvertent actuation of the safety injection system. This serves as a warning that, despite the efforts made to improve the operation procedures as discussed earlier, accidental actuation of the safety injection system at cold shutdown condition can still lead to low temperature over-pressure events due to human errors or failures of certain components. The first of the two LTOP events, which occurred at Turkey Point Unit 4 in November 1981,[13] was caused by one overpressure mitigation system (OMS) channel being out for maintenance, while the other (redundant) channel was disabled by undetected errors. The second event occurred as a result of undetected equipment malfunctions. Each event was initiated with a pressure spike resulting from the start of a reactor coolant pump which led to isolation of letdown by automatic closure of the RHR system isolation valve. In view of these data, it is not unreasonable to assume that the initiating frequency of LTOP events which can potentially challenge the PORVs remains roughly unchanged from the pre-1979 level (i.e., 0.13 events/reactor year). The probability for the failure of operator to reset the switch to the low-pressure PORV control mode is taken to be 0.01, based on NUREG/CR-1278[14] which gives the human error probability (HEP) for failure to follow this type of procedure. The probability of reactor vessel failure under low temperature and overpressurized conditions is conservatively assumed to be $2.\times10^{-3}$ (at an assumed mid-life of the vessel corresponding to a fluence of 8.2×10^{18} neutrons/cm^2) based on the results of probabilistic fracture-mechanics calculation performed at Oak Ridge National Laboratory for Oconee-1, by using Monte Carlo techniques.[2] In the Oak Ridge study, probabilistic fracture-mechanics calculations were carried out to determine the conditional probability of vessel failure for a number of postulated Oconee-1 transients involving overcooling and pressurization of the reactor vessel. This vessel failure probability is of the same order of magnitude as compared with those shown in Reference 3, which were obtained by using the Vessel Integrity Simulation Analysis (VISA) code. For the Oconee 3 vessel and a vessel with high copper (0.35%) and nickel (1.00%) contents, the probability of vessel failure at 2485 psia peak surge (with an estimated fluence of 8.5×10^{18} neutrons/cm^2 at the mid-life of the vessels) was found to be 1.5×10^{-3} and 2.2×10^{-3} respectively. If the peak surge pressure is lowered to 1200 psia, the vessel failure probability becomes much lower (7.0E-7 and 7.0E-6, respectively at the mid-life fluence). Note, however, that the predicted probabilities of vessel failure at 2485 psia peak surge and at the end of the life fluence (1.4×10^{19} neutrons/cm^2) are 2.6E-3 and 2.7E-3, respectively. At the present conditions (5.2 and 8 effective full power years for Indian Point 3 and Oconee, respectively) the conditional probability of vessel rupture is substantially smaller (about 3×10^{-5} for Indian Point and 8×10^{-5} for Oconee). The results for this study are summarized in Table 9. With the unavailability of one PORV taken to be 0.015 for Oconee 3, which has only one PORV, these conservative assumptions yield a core-melt frequency of about 4.8×10^{-6}/reactor year. For Indian Point 3, which has two PORVs, the probability for failure to provide RCS pressure relief with one PORV is roughly 3.6×10^{-3}. This yields a core-melt frequency due to low temperature overpressurization events for Indian Point 3 of approximately 2.6×10^{-6}/reactor year.

To quantify the benefit of improving the PORV reliability, it is necessary to remove the human error contribution from the unavailability of the PORV. In estimating the unavailabilities of the PORVs for both Oconee 3 and Indian Point 3, the block valves were assumed to be 60% of the time closed prior to demand. The probability for operator's failure to open the block valves when the PORVs are demanded for pressure relief due to LTOP was taken to be 0.01. The unavailabilities of the PORVs excluding the human error contribution are 9.1×10^{-3} and 3.5×10^{-5} respectively for Oconee 3 and Indian Point 3. Therefore, if the PORVs and the associated block valves can be improved to function perfectly, the core melt frequency can be reduced by (0.13) $(9.1 \times 10^{-3})(0.0015) = 1.8 \times 10^{-6}$/ry for Oconee 3 and (0.13) $(3.5 \times 10^{-5})(0.0015) = 6.8 \times 10^{-9}$/ry for Indian Point 3. Note that the larger core melt frequency reduction found for Oconee 3 is partly due to the larger unavailabilities of PORV and block valve for Oconee 3 as compared to those for Indian Point 3 (4.9E-3 vs. 5.0E-4 for PORV, and 6.0E-3 vs. 1.5E-3 for block valve). If the Indian Point 3 unavailability data were used in the Oconee 3 calculation, the unavailability of the PORV excluding the human error can be shown to be 2.4×10^{-3} so that the core melt frequency reduction would become 4.7×10^{-7}/ry rather than the 1.8×10^{-6}/ry shown above. The benefit of having two PORVs instead of one could further become less evident if credit were given to the RHR relief valve in the foregoing analysis.

It should be emphasized that, due to the complexity of the problem and the many uncertainties involved, these resu[illegible] only be considered as rough estimations based on several assumptions. [illegible] more precise analysis, further studies are required particularly i[illegible] areas of OPS control logic, RHR relief valve availability and capacity, [illegible]el failure probability, and vessel fracture size and location.

Finally, it should be added that some preliminary information developed from recent GI-94 studies suggests that the above BNL study is overly conservative in implicitly assuming that the probability of reaching a high pressure condition in the reactor vessel for every LTOP incidence is "one" (1). Generic Issue 94 is involved in performing an in-depth study of the LTOP issue. Data collected from operating plant experience shows that without protection against LTOP, there was about a 30% chance that the initiation of an LTOP event would result in vessel pressures sufficiently high to threaten the vessel. Furthermore, the most recent operating experience since LTOP protective measures have been in place, indicated that there is now only about a 10% chance of reaching high pressure on the initiation of an LTOP. On the other hand, recent preliminary GI-94 studies into the mechanism of vessel fracture and failure suggest that past estimates of the likelihood of vessel failure from the high pressures sometimes resulting from LTOP events, such as the estimates upon which the above BNL evaluation was based, may have been nonconservative. In summary, this most recent information indicates that there is probably less chance of achieving the high pressures required during LTOP events to cause vessel failure than was assumed in the BNL studies, but for that fraction of those events where the higher pressures are reached, there is probably a higher chance that the vessel will fail than was assumed by BNL. Since these two factors tend to offset each other it is not believed that the conclusions of this study should be changed at this time based on this preliminary GI-94 information.

2.4.3 Estimated Consequences for Low Temperature Overpressurization

The previous calculations provided an estimate of the core damage frequency on the assumption that a brittle fracture of the reactor vessel would lead to a core melt with a probability of 1.0. Although the ECC systems would be available, there is very little basis for an assumption that a reactor vessel rupture (if it occurs) would be small enough to prevent core uncovery even with ECC operation. Similarly, it is difficult to quantify the condition of the containment at cold shutdown conditions. It is assumed that the containment would be open about 50% of the time since refueling, tests and maintenance would be expected at cold shutdown conditions. Thus, we have attempted to determine the offsite radiological consequences for a low temperature overpressurization event assuming that the melt occurs into an open containment. The release estimates have been obtained[15] with the STCP for a late core melt with containment bypass. These estimated release fractions are summarized in Table 7 which forms the basis for the following consequence calculations.

The CRAC-2 code[16] was used to calculate offsite consequences for the "source term" given in Table 7. The input conditions are summarized in Table 8. The total health effects for this typical eastern site are found to be 9×10^6 person-rem over 30 years. Although there is some uncertainty in the source term given in Table 7 and the health effects may change depending on site specific weather patterns and population, the uncertainty in risk is dominated by the uncertainty in core melt frequency. Thus, no attempt has been made to provide an uncertainty range on the consequence estimates. It would also be inappropriate to obtain worst case consequence calculations and combine them with worst case frequency estimates.

The total risk is estimated using the range of core melt frequencies based on the vessel failure probabilities over the life of the plants along with the calculated consequence (9×10^6 person-rem) and is given in Table 10. Thus, the total present risk estimate for Oconee is about 1.2 person-rem/reactor year and approximately 0.4 person-rem/reactor year for Indian Point. However, the risk increases with the age of the vessel up to a maximum of 21 person-rem/reactor year for Indian Point and about 40 person-rem/reactor year for Oconee at the end of life.

2.5 Use of PORV as a High Point Vent to Supplement the Function of RVHVS

The primary purpose of the reactor vessel head vent system (RVHVS) is to remedy the undesirable situation in which inadequate core cooling or impaired natural circulation is created by the accumulation of noncondensable gases (such as hydrogen) in the RCS. This system is normally designed to remove noncondensable gases or steam from the upper part (upperhead) of the reactor vessel and discharge them through a reactor vessel head vent piping, which ultimately merges with the PORV discharge line, into the pressurizer relief tank. An additional letdown flow path is provided from the RVHV to the excess letdown heat exchanger in the chemical and volume control system. A typical schematic diagram of this system can be found, for example, in the Millstone Nuclear Power Station Unit 3 FSAR (Figure 5.4-17). The reactor vessel head vent piping consists of two parallel flow paths with redundant isolation valves provided for each flow path. The venting operation, however, employs only one of these flow paths at any one time. In other words, if for any reason, one flow path fails to provide venting, the redundant path is used as an alternative path.

Since the reactor vessel head vent piping eventually merges into the PORV discharge piping before reaching the pressurizer relief tank, a question can be raised as to whether the pressurizer PORVs can be effectively used as a high point vent to supplement the functions of RVHVS, in case the latter system fails. In particular, the NRC is interested in finding out how much reduction in core melt frequency can be achieved by improving the reliability of the PORVs if PORVs are used, in combination with the RVHVS, as a high point vent to remove the noncondensable gases.

To resolve this problem, it should first be pointed out that, for the noncondensable gases (such as hydrogen) to accumulate in the RCS in significant amounts so as to block natural circulation flow, certain degree of core uncovery must have already taken place. This post-core melt condition is beyond the scope of current PRA for computing the core melt frequency. Secondly, although the Reactor Vessel Head Vent System (RVHVS) or the PORVs may be effective in venting the hydrogen accumulated in the upper-head or hot-leg region, they may not be capable of venting hydrogen which could build up in the candy-cane region or at the top of the U-tube steam generator. Thirdly it should be remarked that, although the blocking of natural circulation flow can cause failure of RCS heat removal by the steam generators, core heat removal may still be achievable by using HPI cooling (feed and bleed) if both the HPI pumps and the PORVs are operational.

To estimate the unavailability of RVHVS, a fault tree was constructed based on the schematic diagram shown in Millstone 3 FSAR (Figure 5.4-17), to depict the logics for the failure of RVHVS path to open for venting on demand. Quantification of the fault tree, however, revealed that, due to high redundancy, the unavailability of the RVHVS was rather small (approximately 4.0×10^{-4}). The probability of failure to vent noncondensable gases using either the RVHVS or the PORV is even smaller (less than 10^{-5}).

To assess the impact of PORV unavailability on the core damage frequency when it is used in conjunction with the RVHVS for the venting, an attempt was made to merge the fault tree developed for the RVHVS with the existing fault trees (such as those shown in the Oconee PRA) delineating the failure of RCS heat removal due to loss of RCS recirculation. Such an attempt, however, was in vain, because the two kinds of fault trees were found to be incompatible due to discrepancy in the timing of relevant events. As pointed out earlier, before any significant amounts of noncondensable gases can accumulate in the upper part of the reactor vessel so as to hinder natural circulation, certain degree of core uncovery or damage must have already taken place. The fault trees or event trees developed in the Oconee PRA or the IPPSS, however, only deal with events which occur prior to onset of core damage. Unless these fault trees are drastically modified to include the events occurring after partial core melt, they cannot be logically combined with the fault trees involving post-core melt events, such as the formation or venting of noncondensable gases. This leads to a conclusion that, although the RVHVS, with or without the backup of PORVs, may be effective in mitigating the consequence of core damage at its early stage, we are unable to assess its possible impact on reducing the core damage frequency defined within the scope of the current PRA.

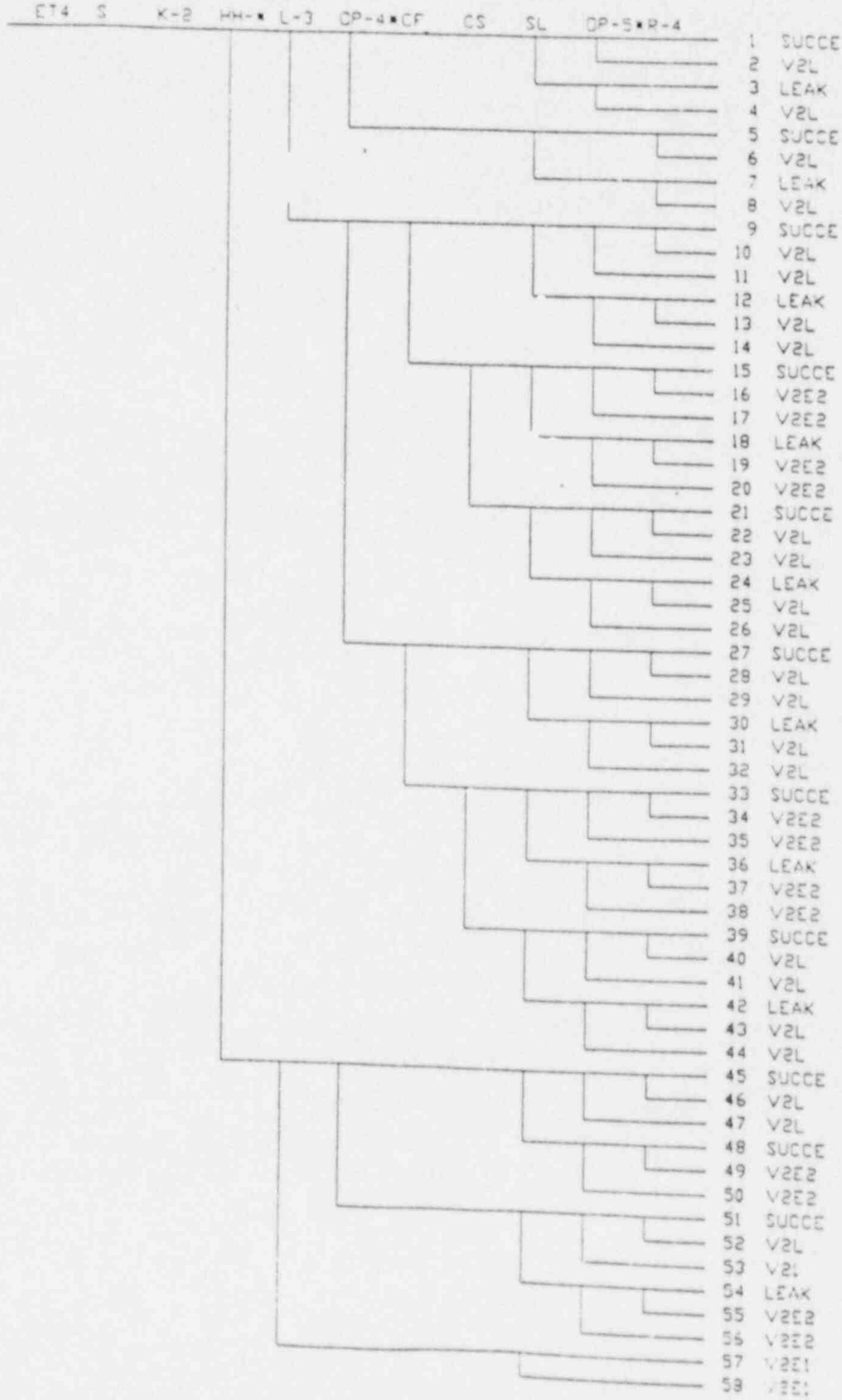


Figure 1. SGTR event tree (Indian Point 3).

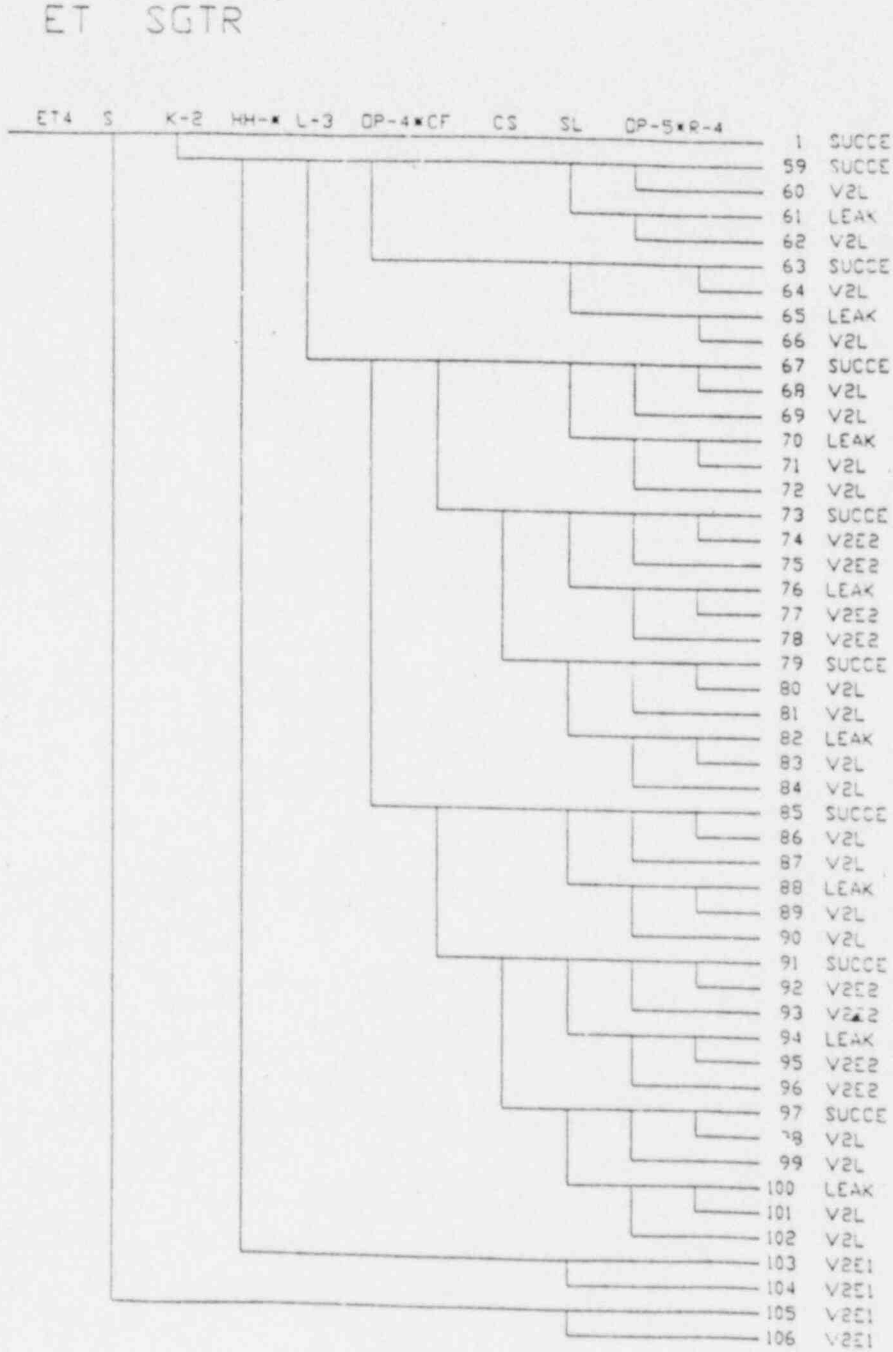


Figure 1. (Continued)

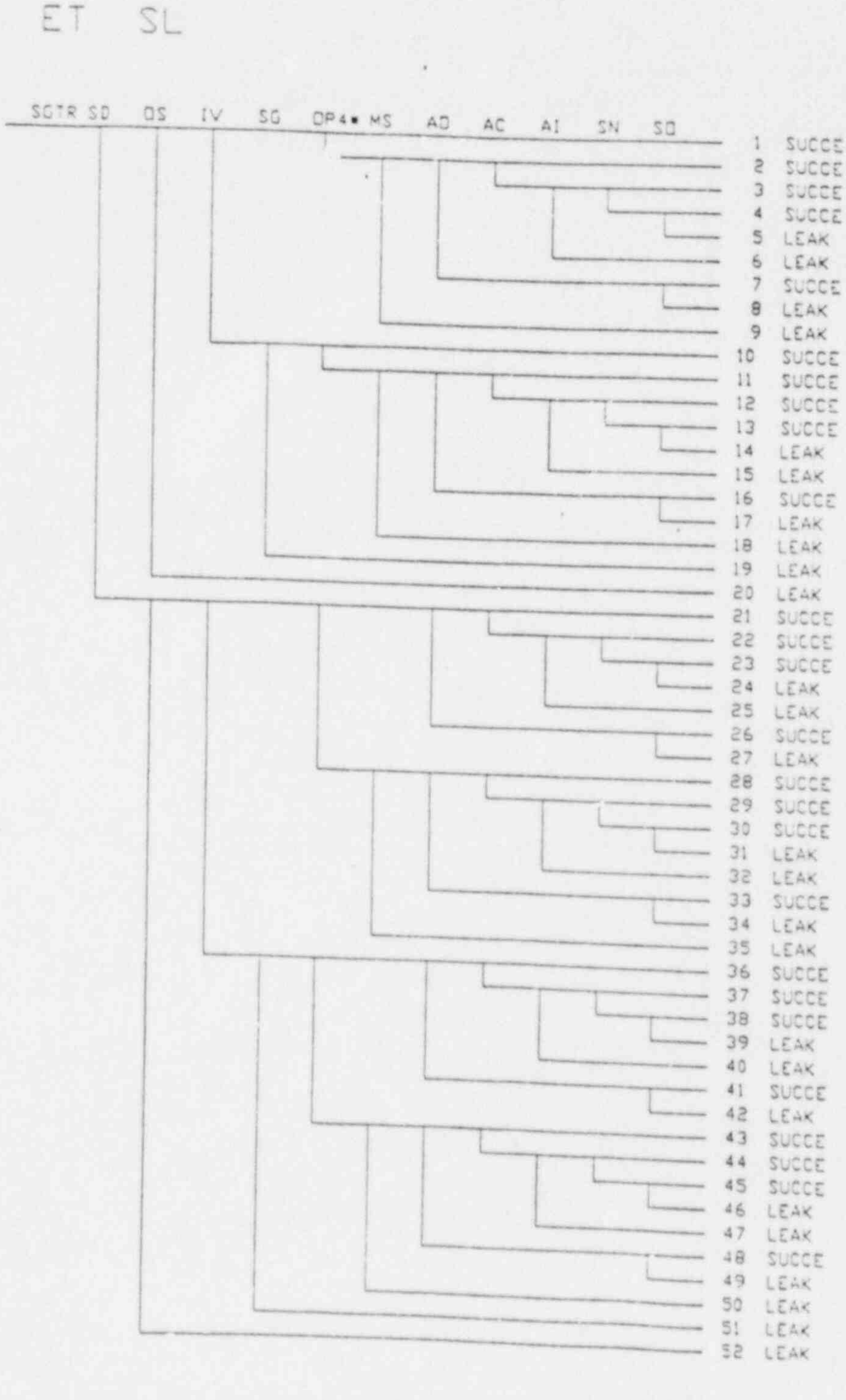


Figure 2. Event tree for SGTR event, SL (Indian Point 3).

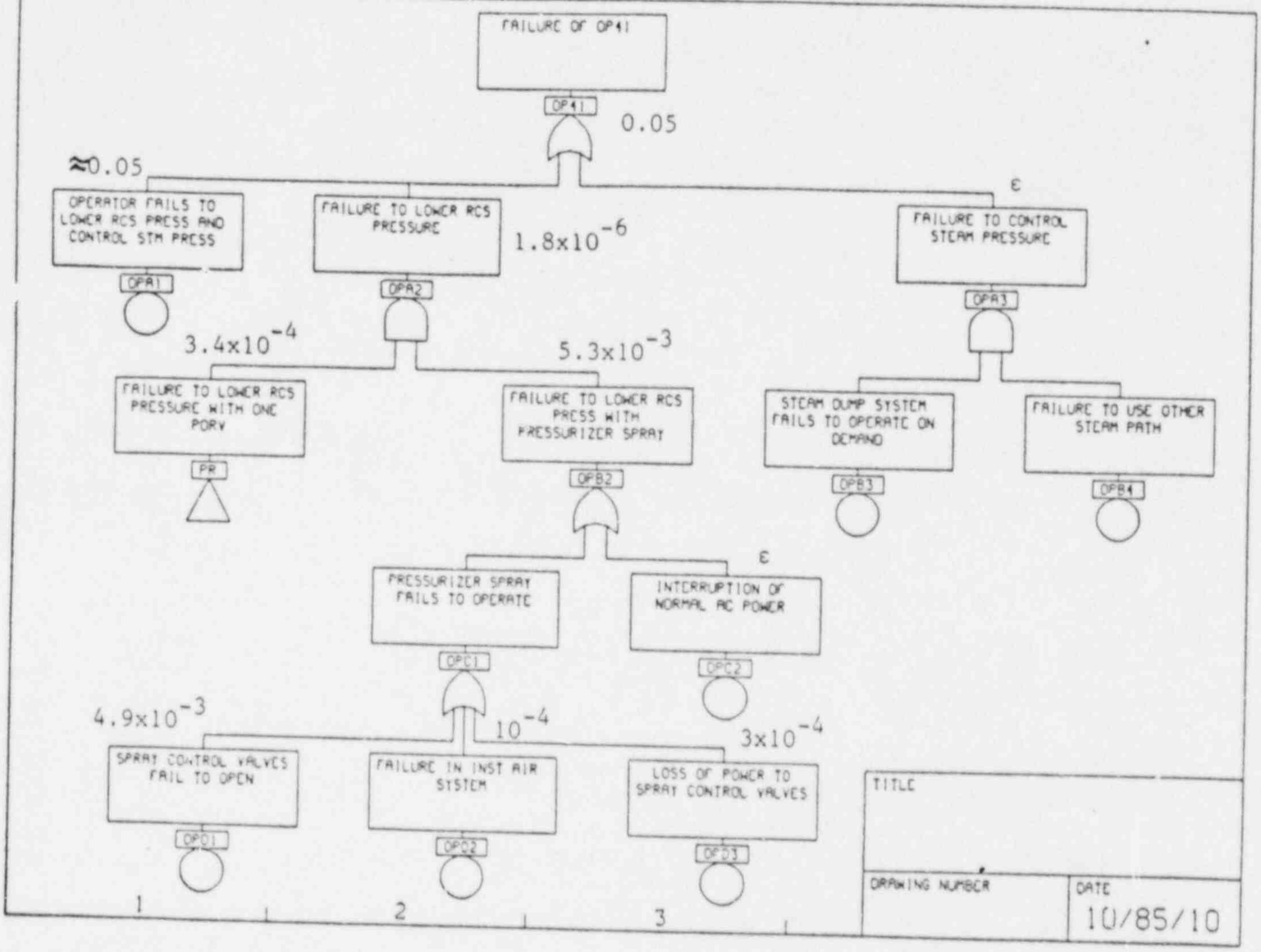


Figure 3. Fault tree for SGTR event, failure of OP-41.

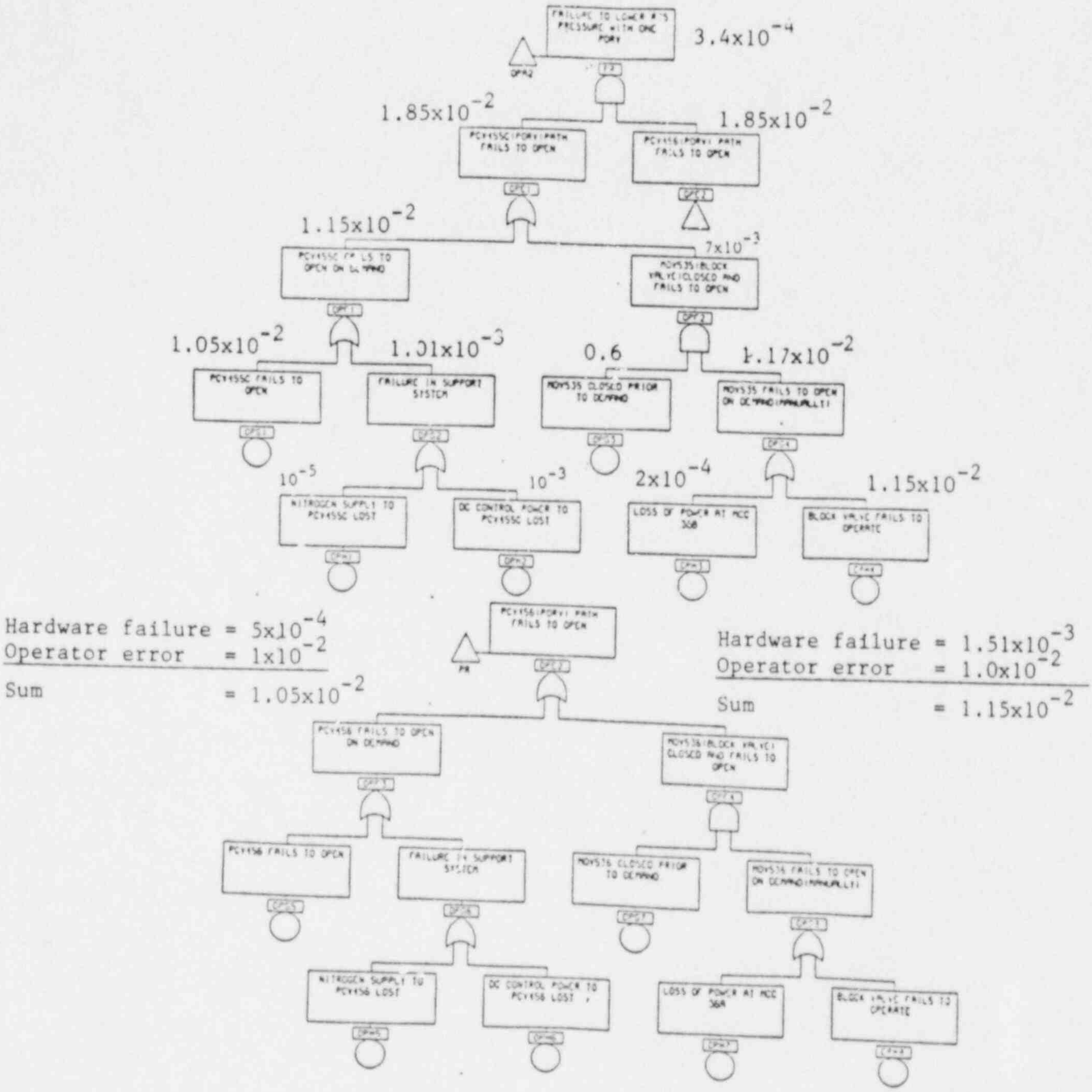
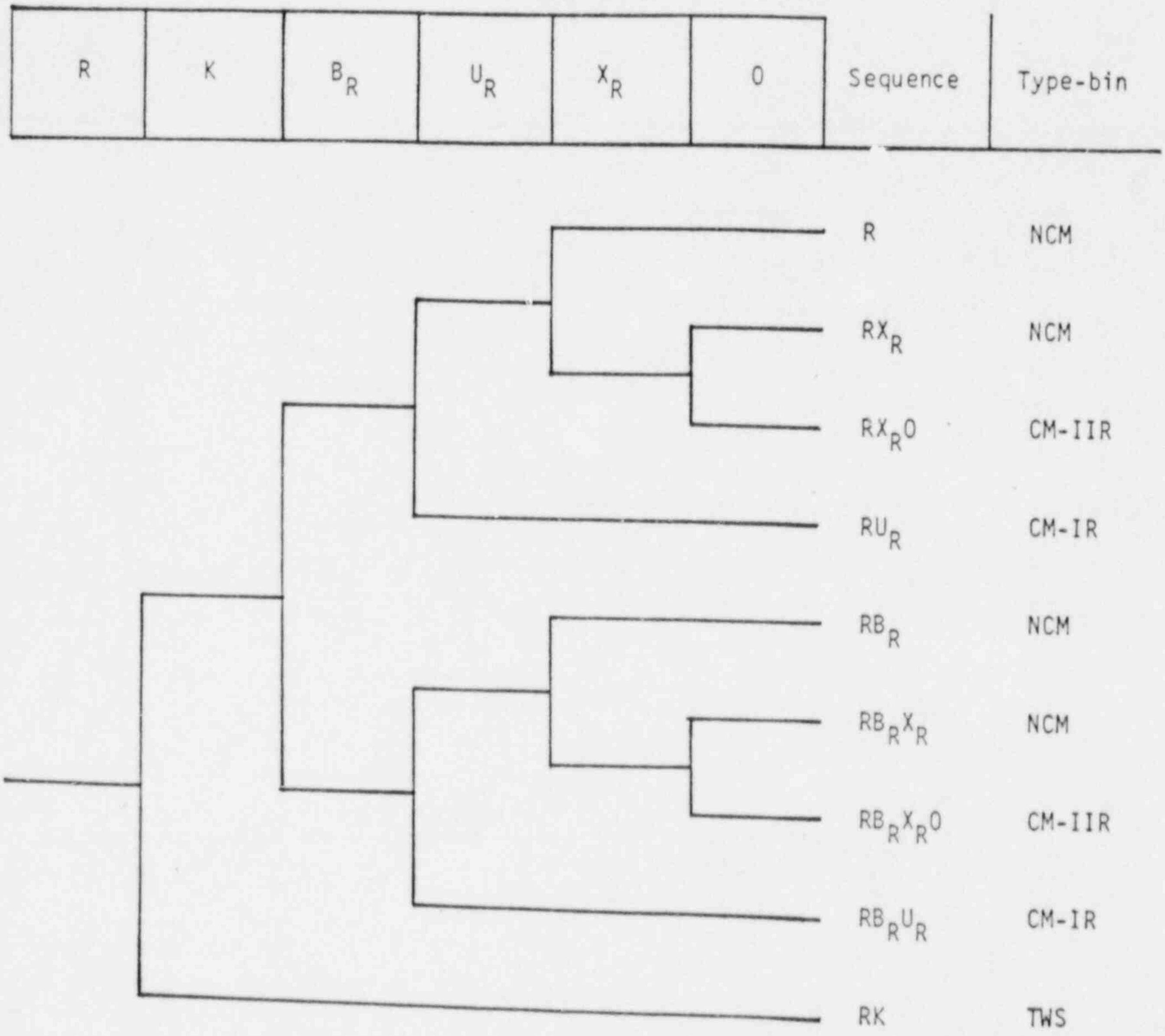


Figure 3. (Continued)



R : Steam Generator Tube Rupture, initiating event
K : Failure of the RPS (Reactor Protection System)
B_R : Failure of RCS heat removal
U_R : Failure of HPI
X_R : Failure to achieve long-term cooling at cold shutdown
O : Failure to maintain long-term cooling at hot conditions

Figure 4. Event tree for SGTR initiating events (Oconee 3).

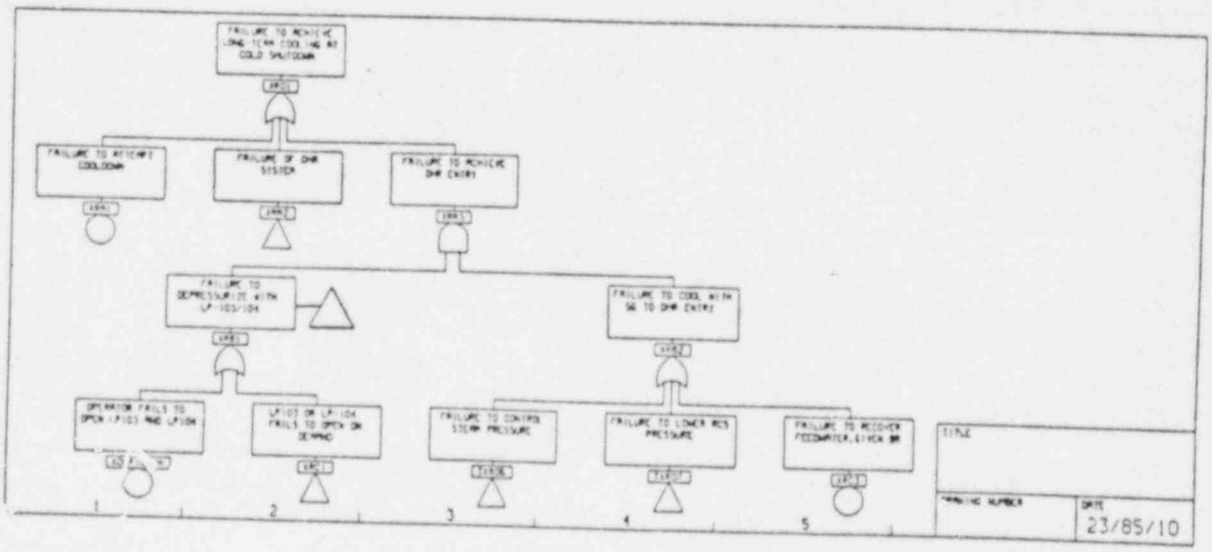
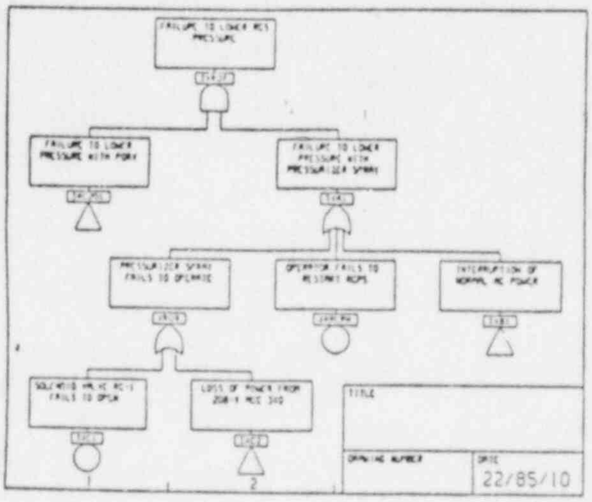


Figure 5. Fault tree for top event, X_R, failure to achieve lo..g term cooling at cold shutdown (Oconee PRA).



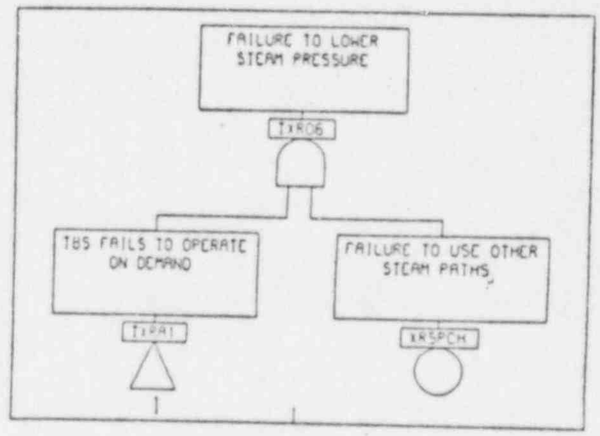


Figure 6. Fault tree for top event, X_R, continued.

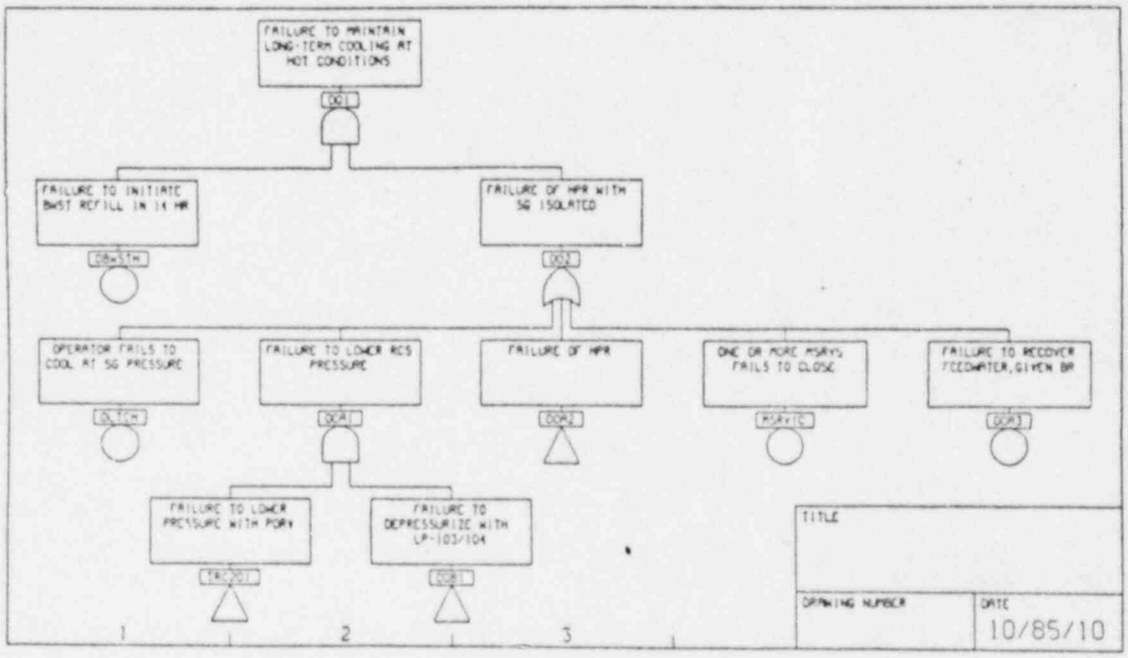


Figure 7. Fault tree for top event, X_e, failure to maintain long-term cooling at hot conditions (Oconee PRA).

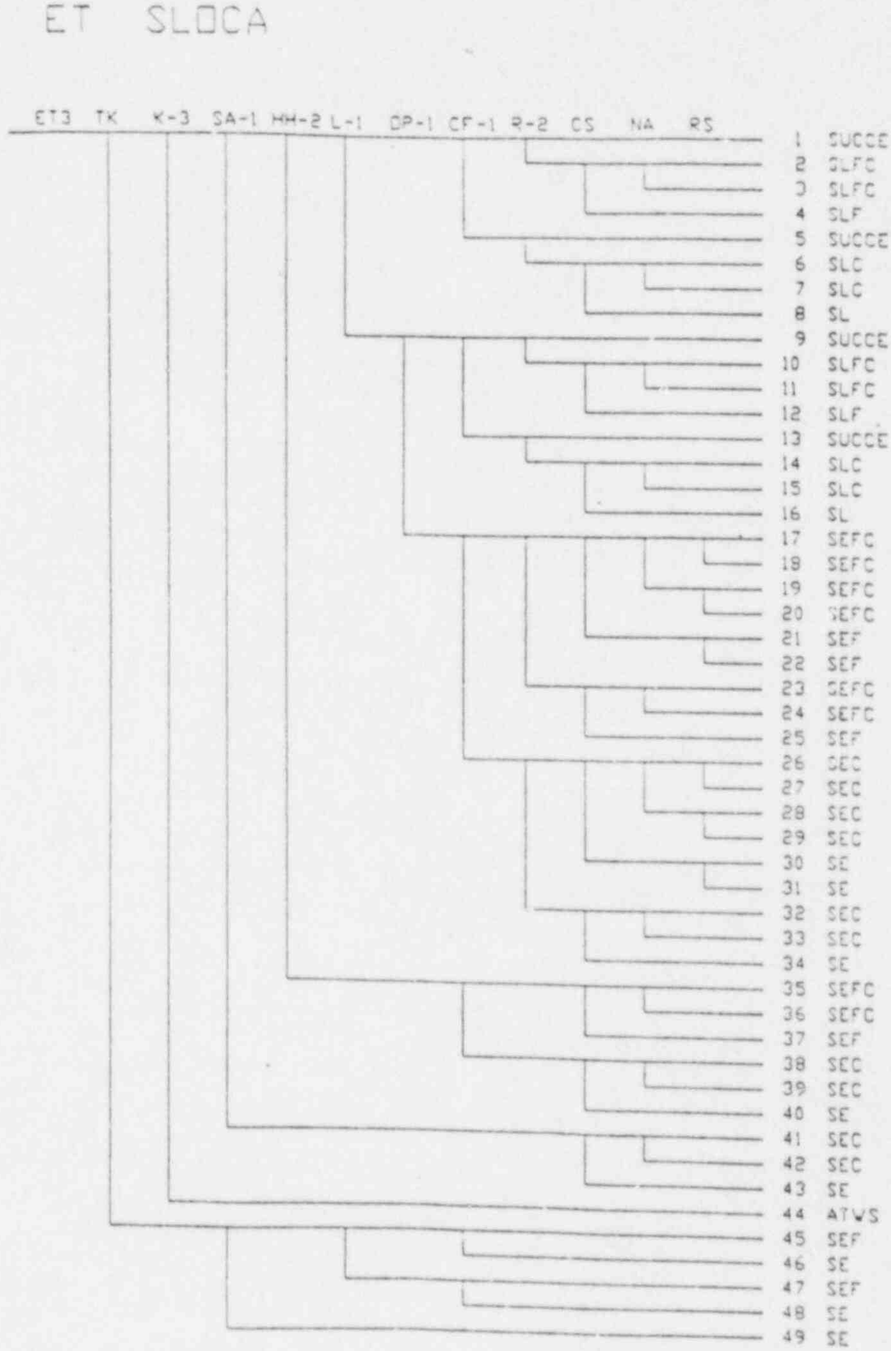


Figure 8. Small LOCA event tree (Indian Point 3).

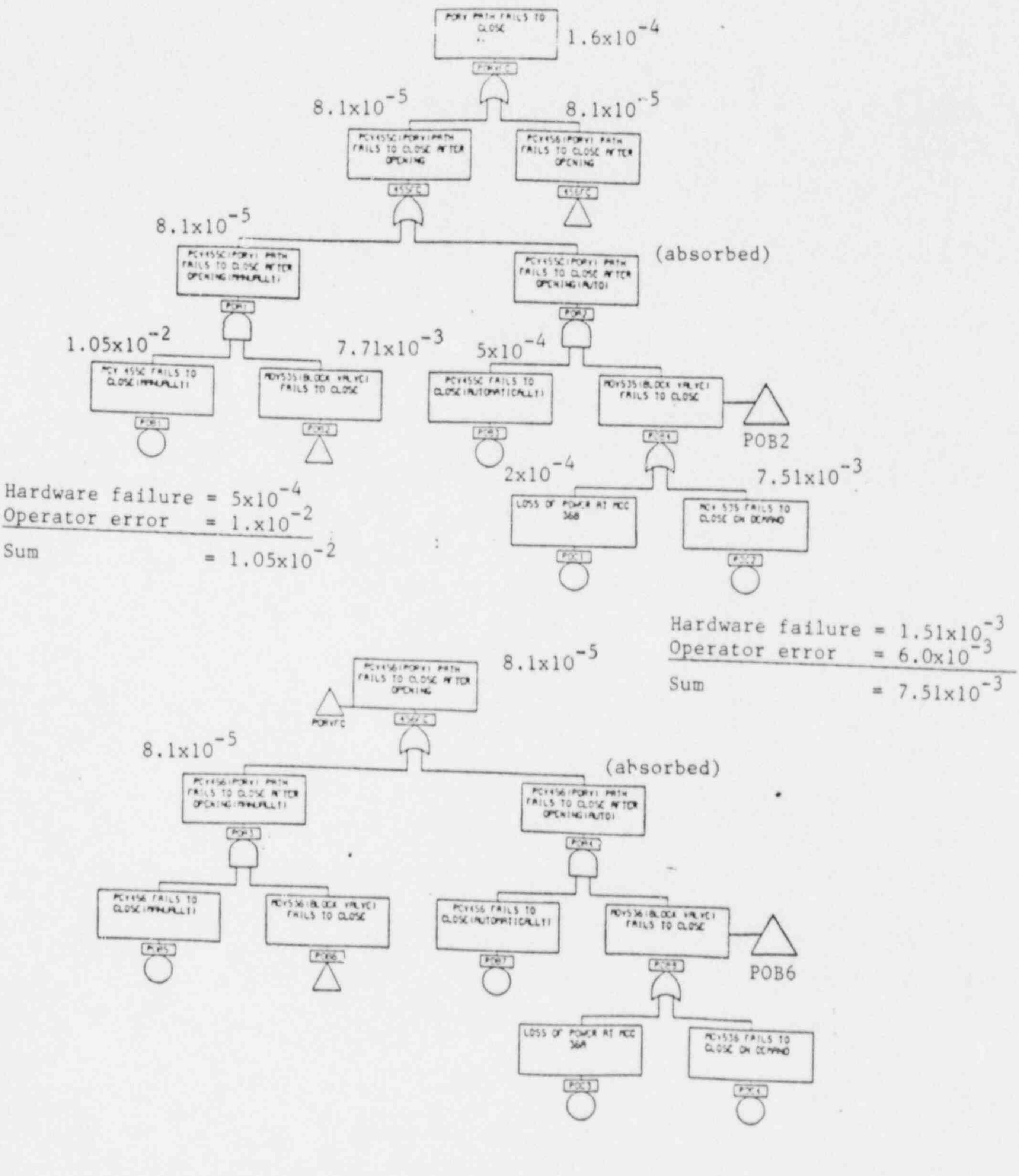
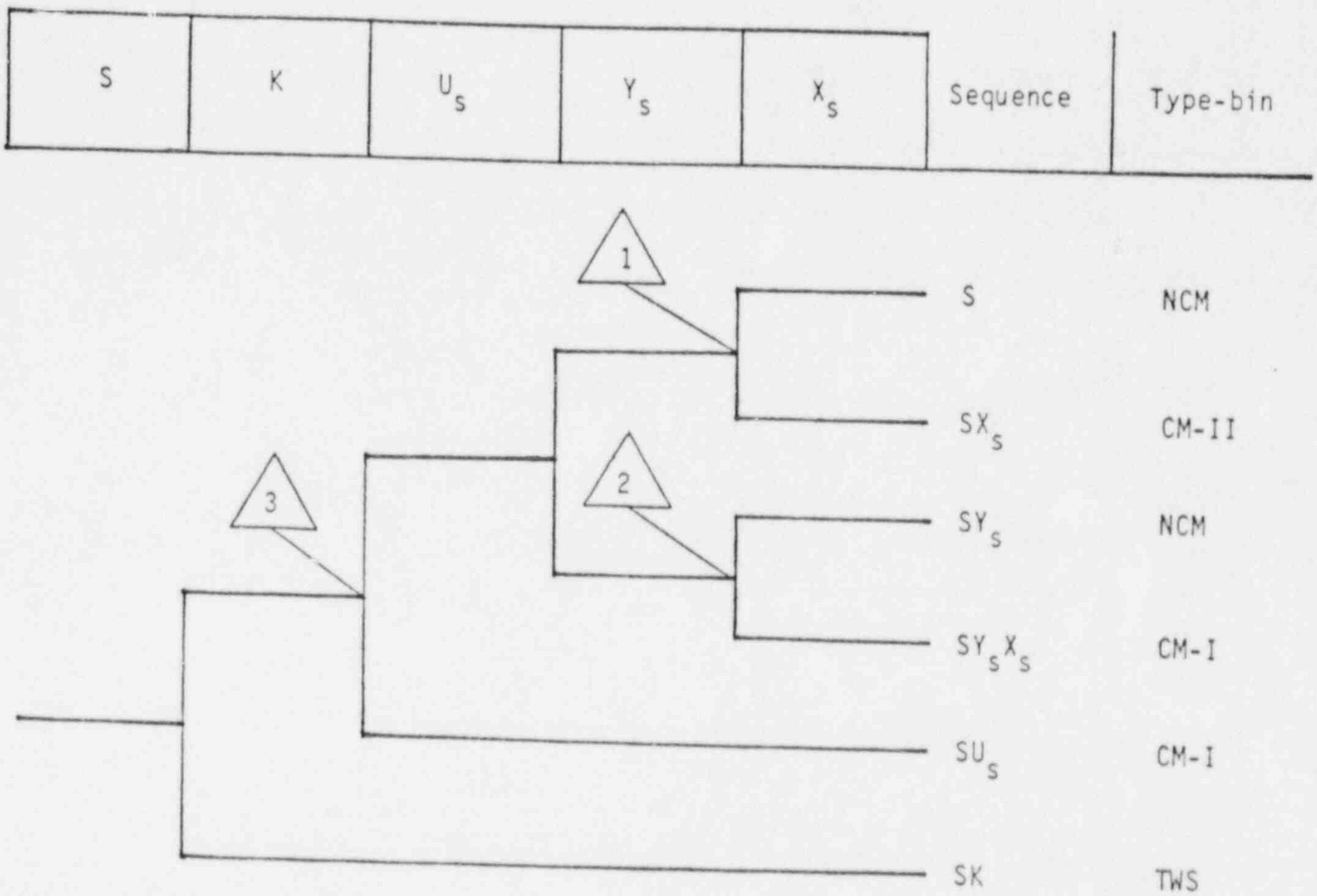


Figure 9. Fault tree for the event, PORV path fails to close (Indian Point 3).



S : Stuck-open PORV (small LOCA) initiating event

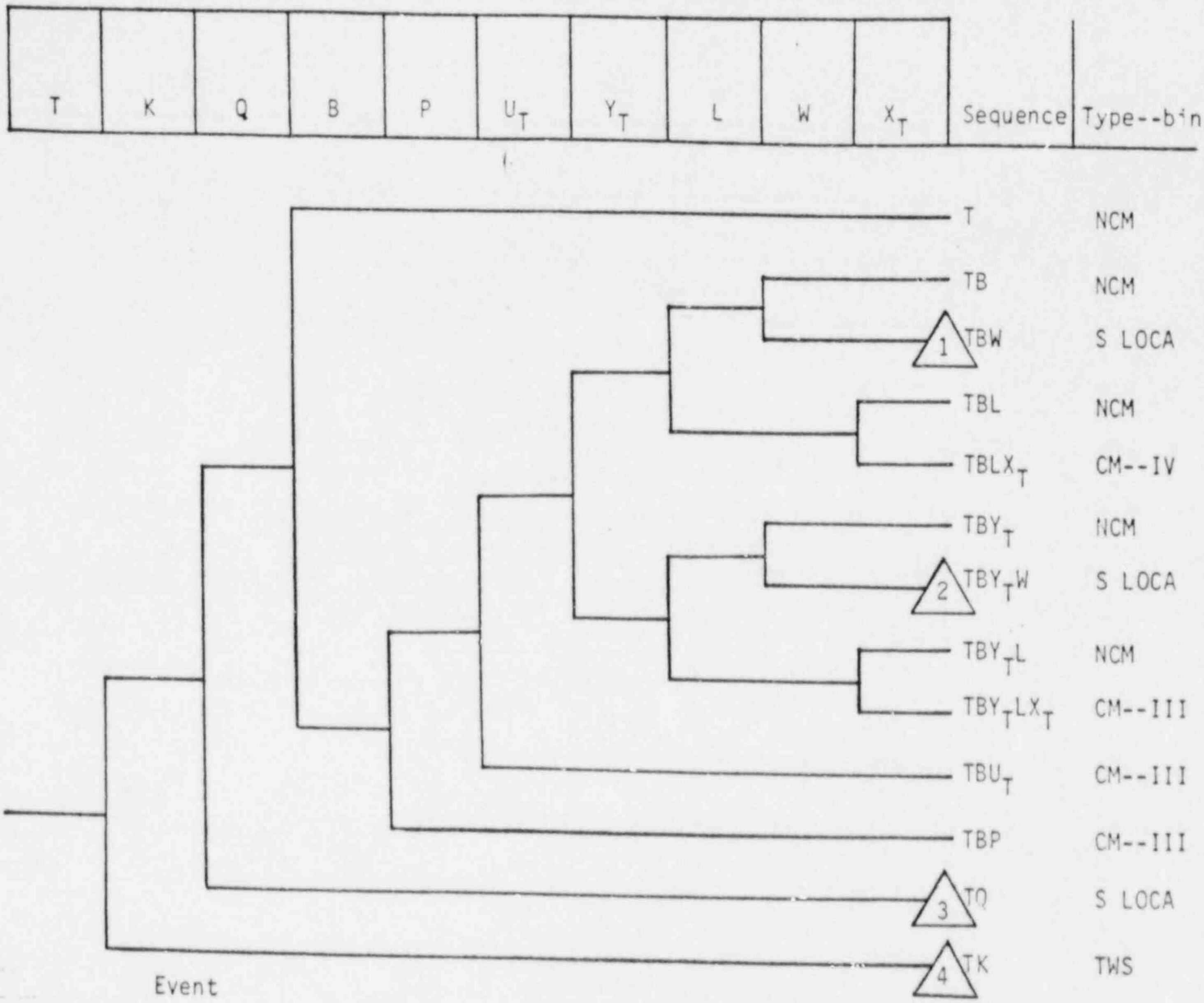
K : Failure of the RCS to trip the reactor

U_s : Failure of core-heat removal by HPI (1 of 3 pumps required)

Y_s : Failure to maintain RCS makeup supply

X_s : Failure to maintain long-term heat removal

Figure 10. Event tree for small-break LOCA events (Oconee 3).



Event

T	Occurrence of a transient initiating event
K	Failure of the RPS to trip the reactor
Q	Loss of RCS integrity
B	Failure of RCS heat removal via the steam generators
P	Failure to provide RCS pressure relief
U_T	Failure of core-heat removal by HPI cooling
Y_T	Failure to maintain RCS makeup supply
L	Failure to recover RCS heat removal
W	Failure to reestablish RCS integrity
X_T	Failure to maintain long-term core-heat removal

Figure 11. Event tree for transient initiating events (Oconee 3).

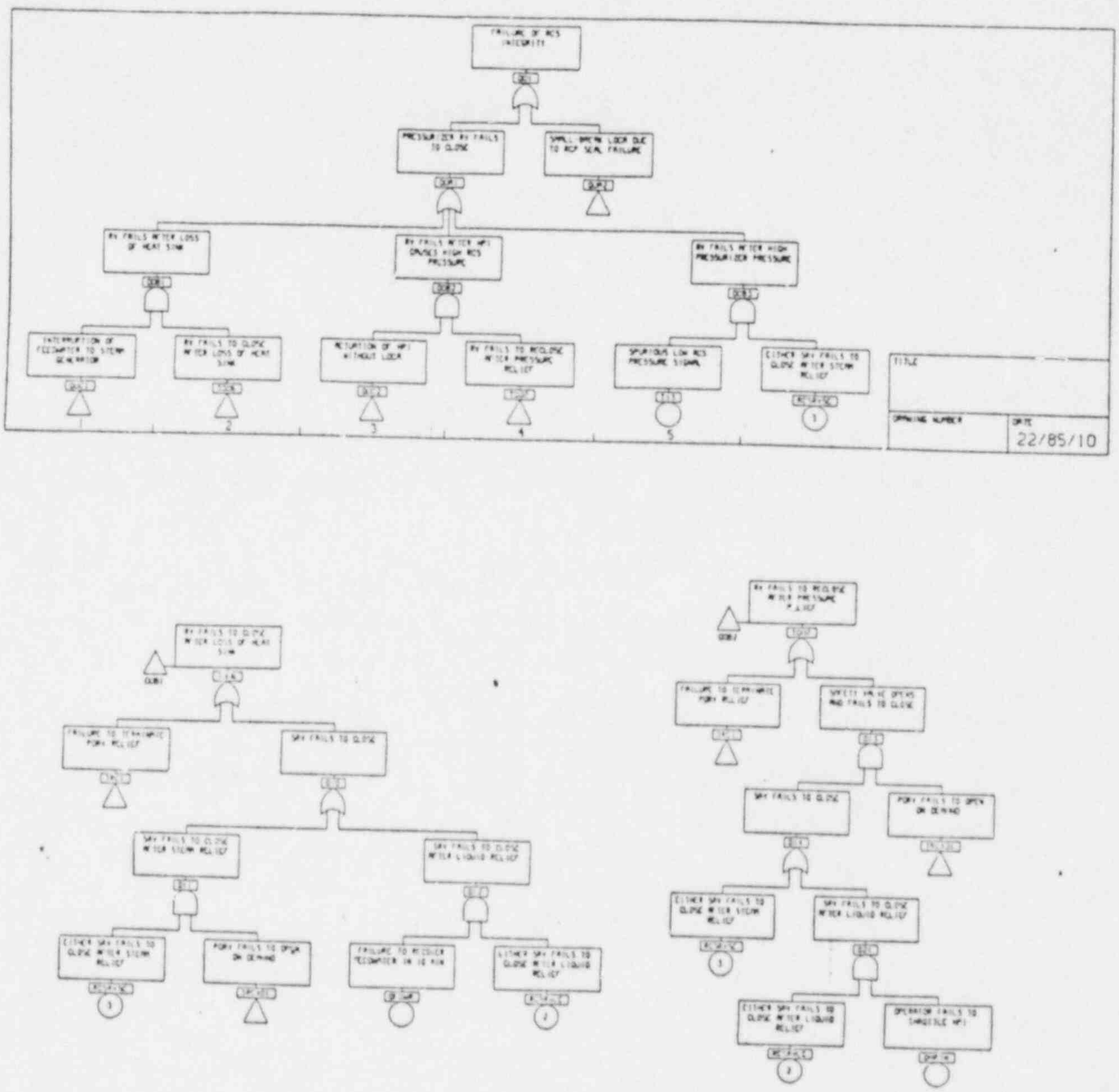


Figure 12. Fault tree for top event, Q, failure of RCS integrity (Oconee PRA).

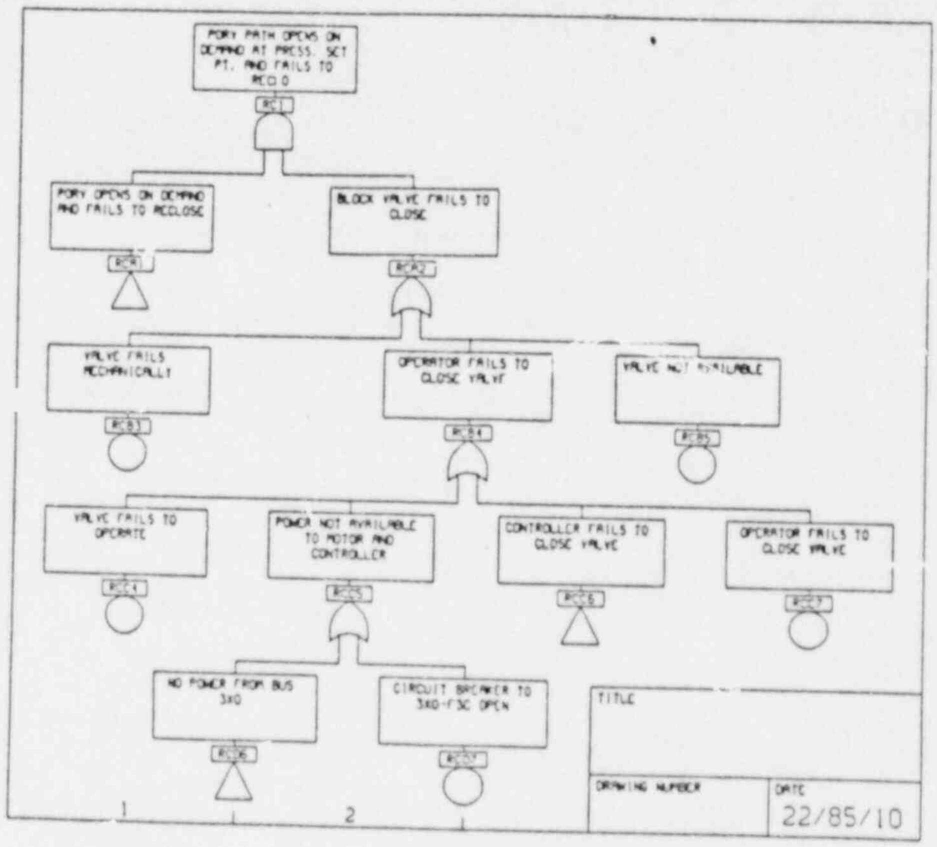
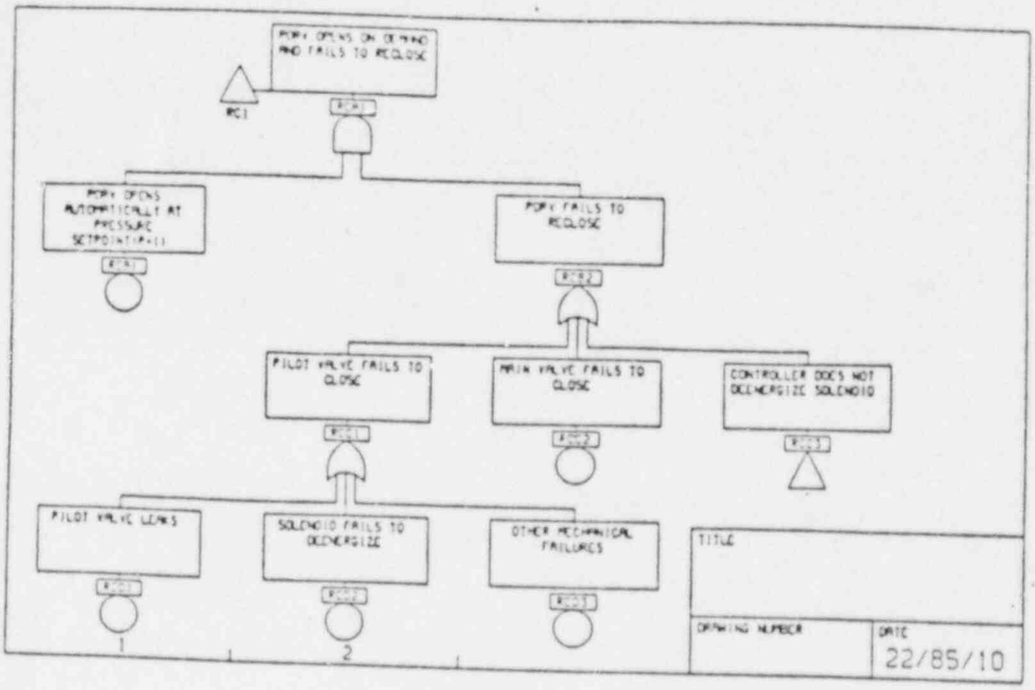


Figure 13. Fault tree for the primary pressure control system (Oconee PRA).



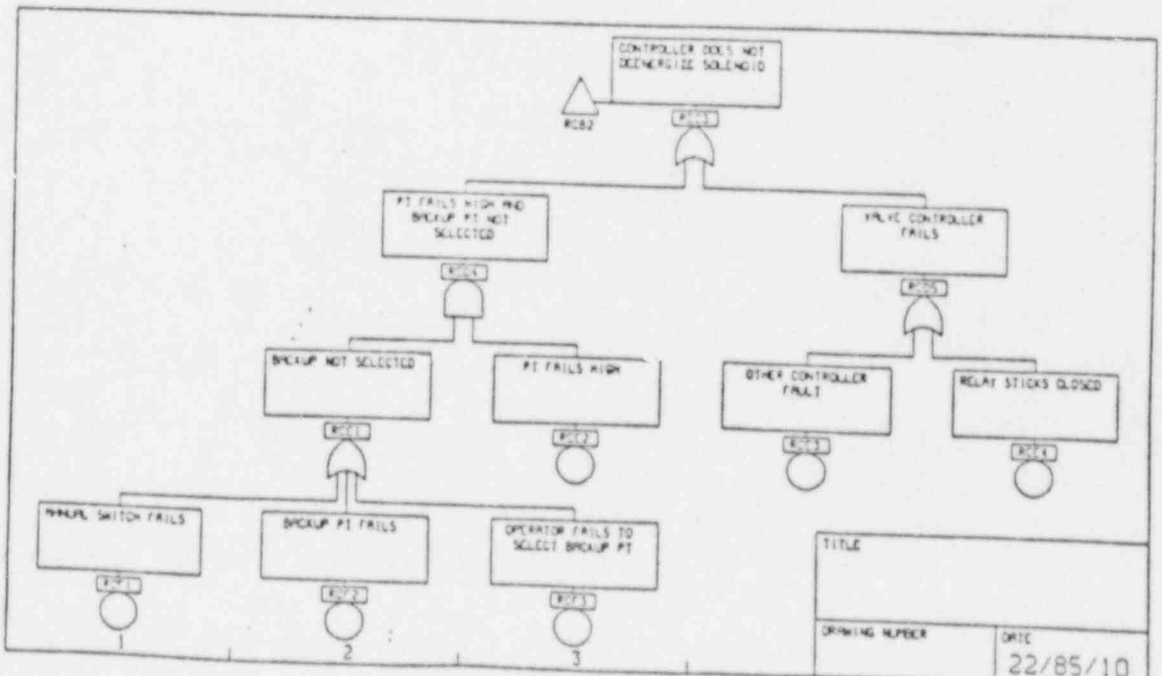
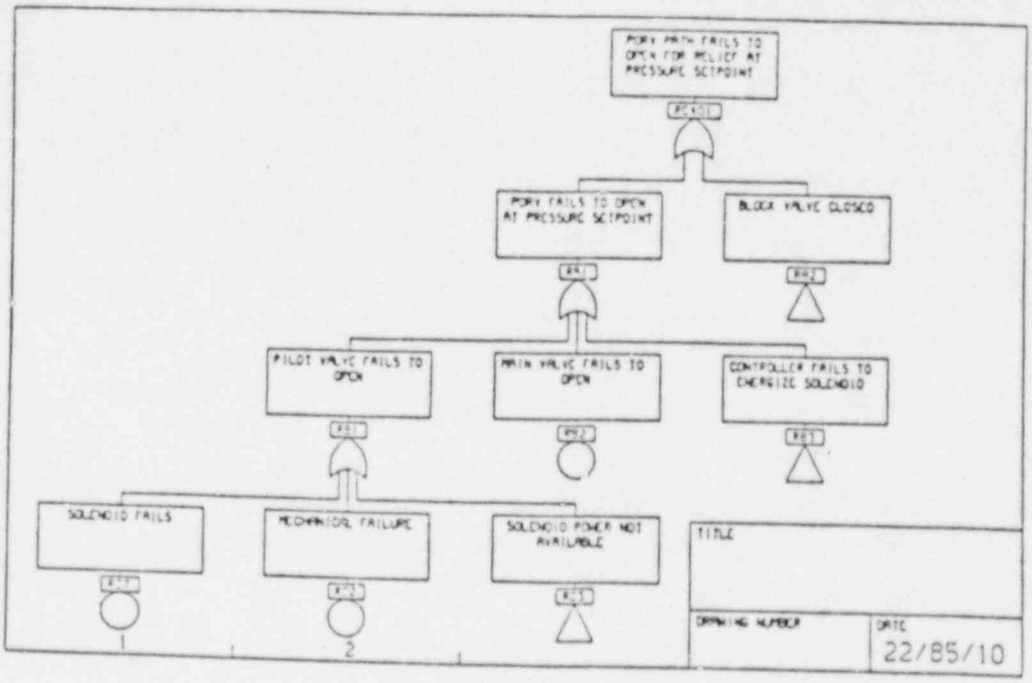


Figure 13. (Continued)



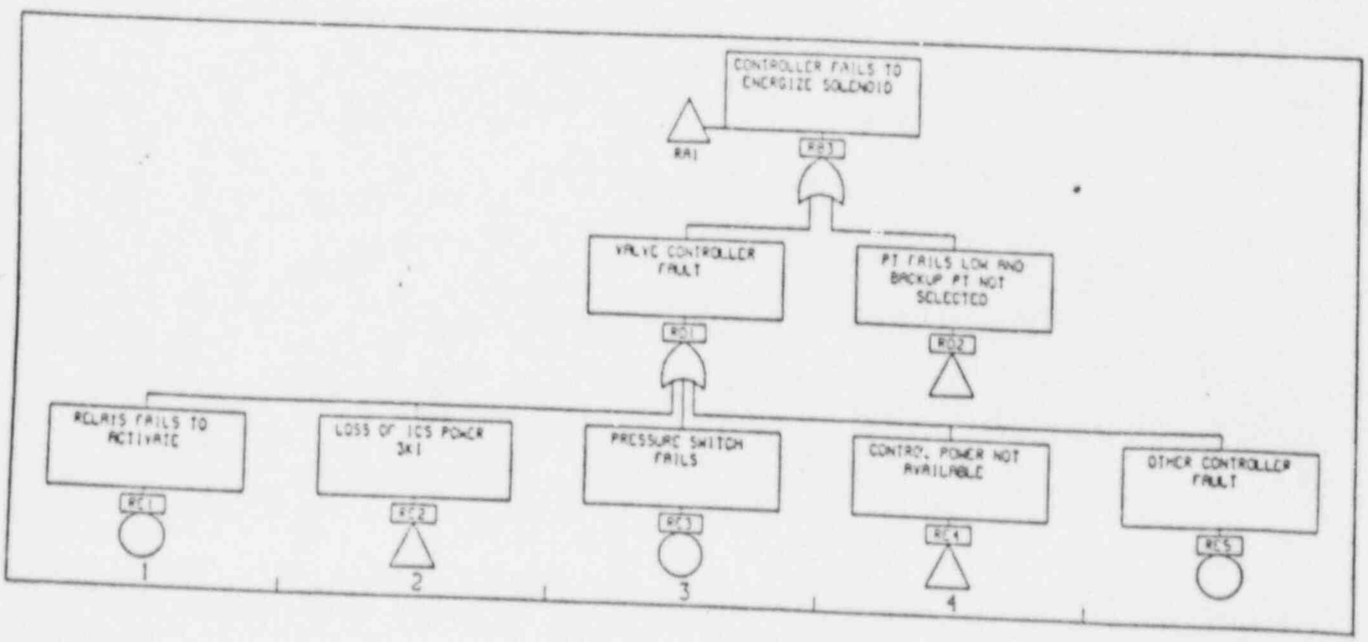
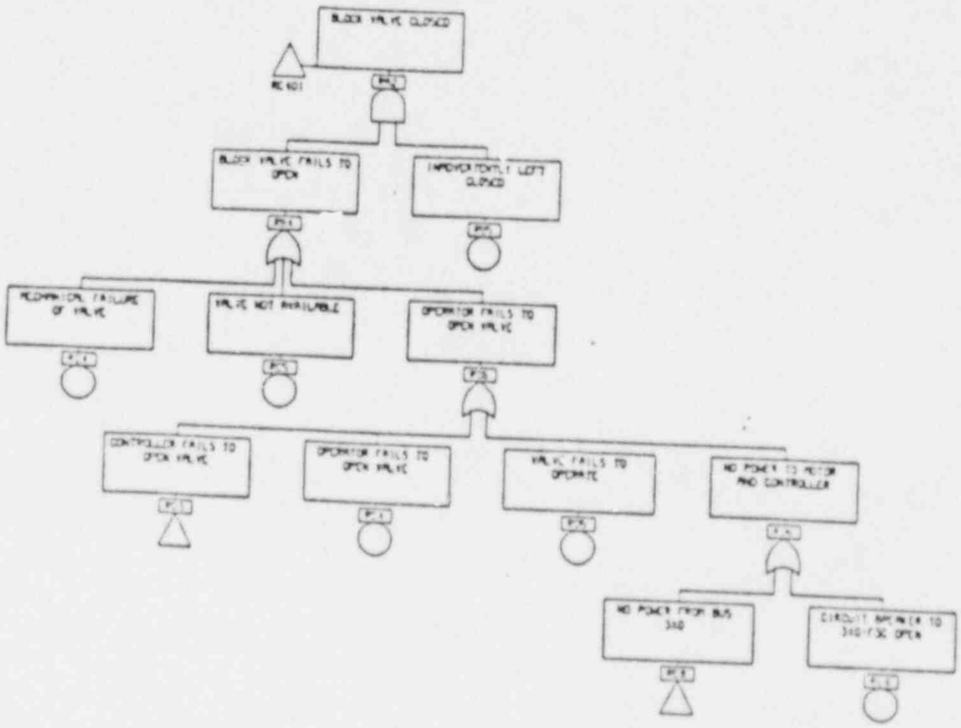
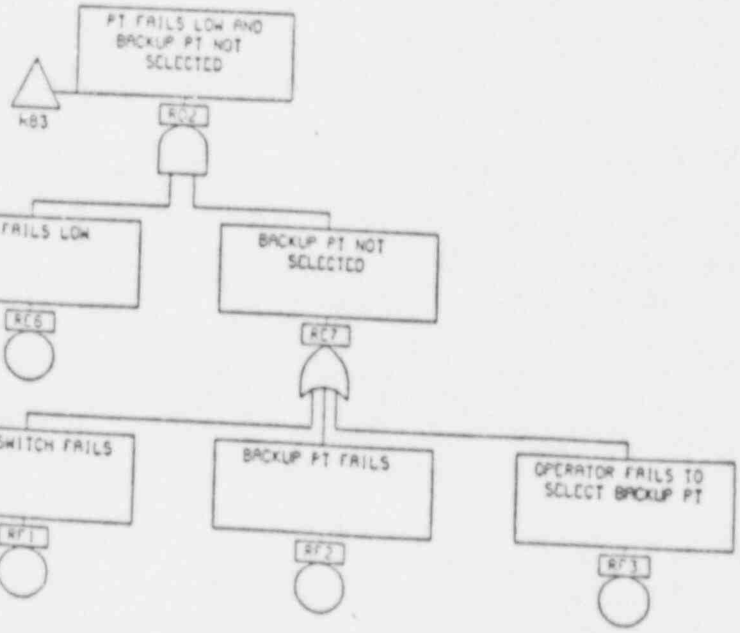


Figure 13. (Continued)





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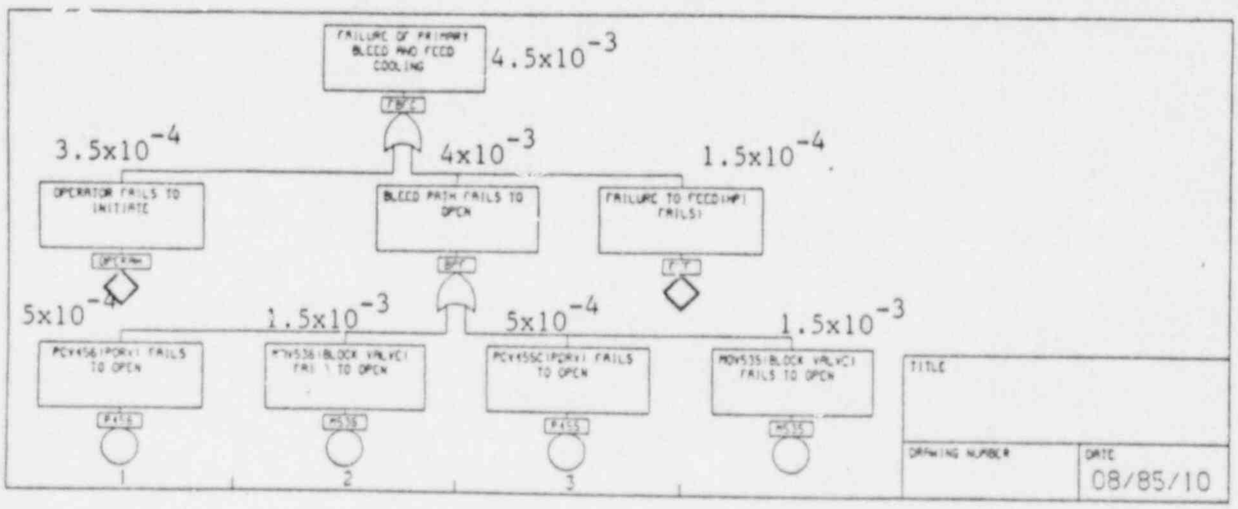

Figure 14. Fault tree for the event, failure of primary feed and bleed cooling (IPPSS).

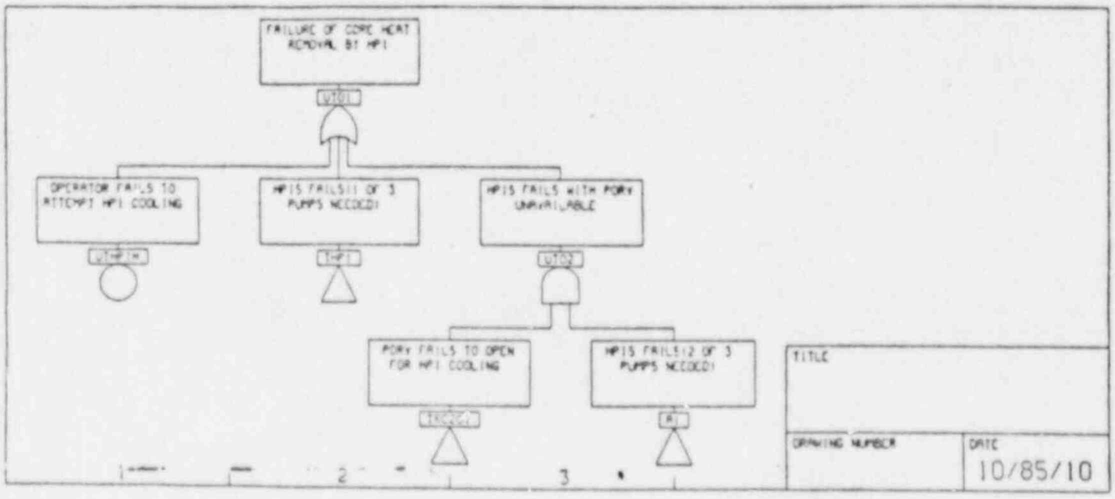

Figure 15. Fault tree for top events, U_T, failure of core heat removal by HPI (Oconee PRA).

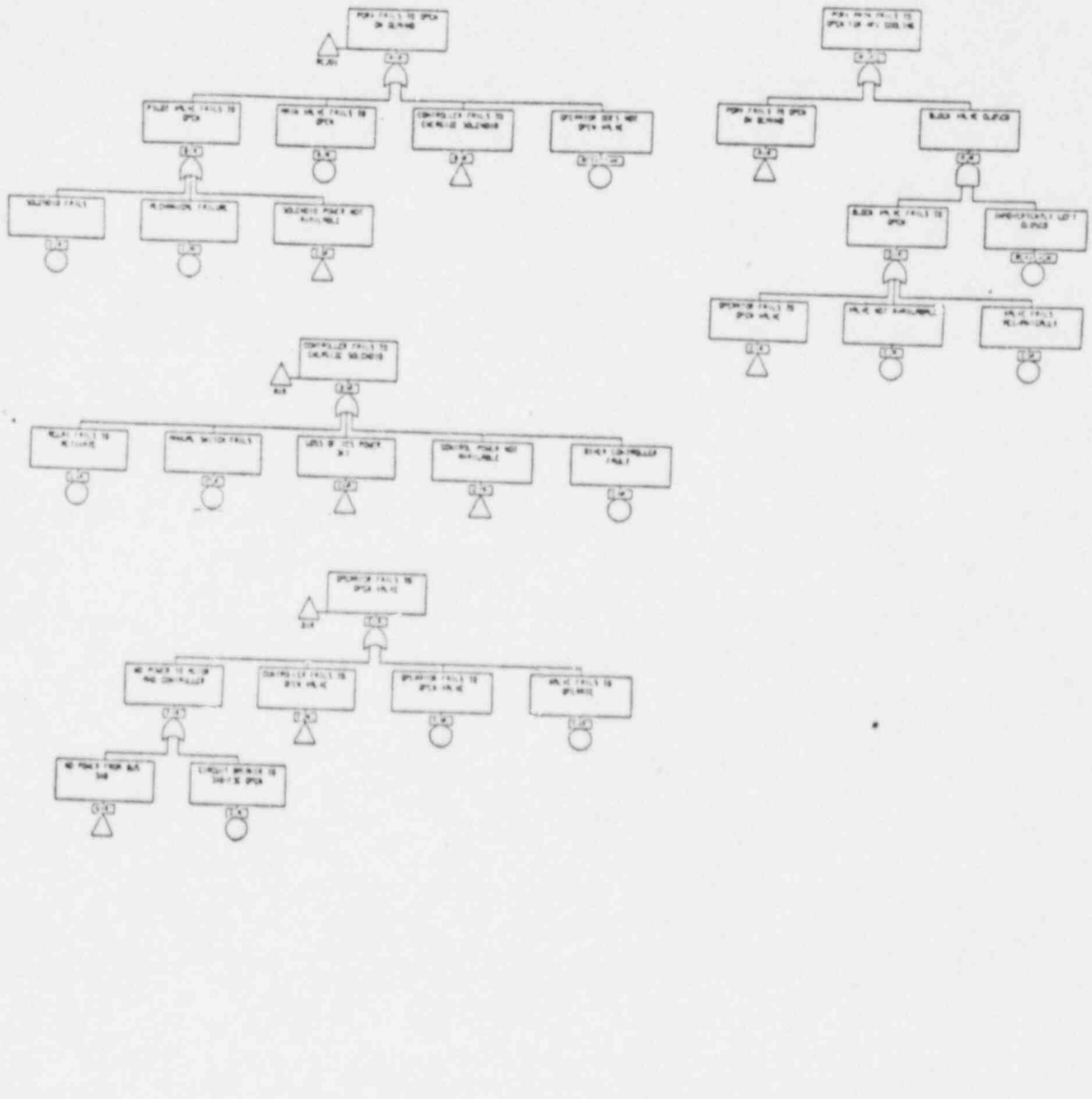

Figure 16. Fault tree for top event, U_T, failure of core heat removal by HPI (Oconee PRA , continued).

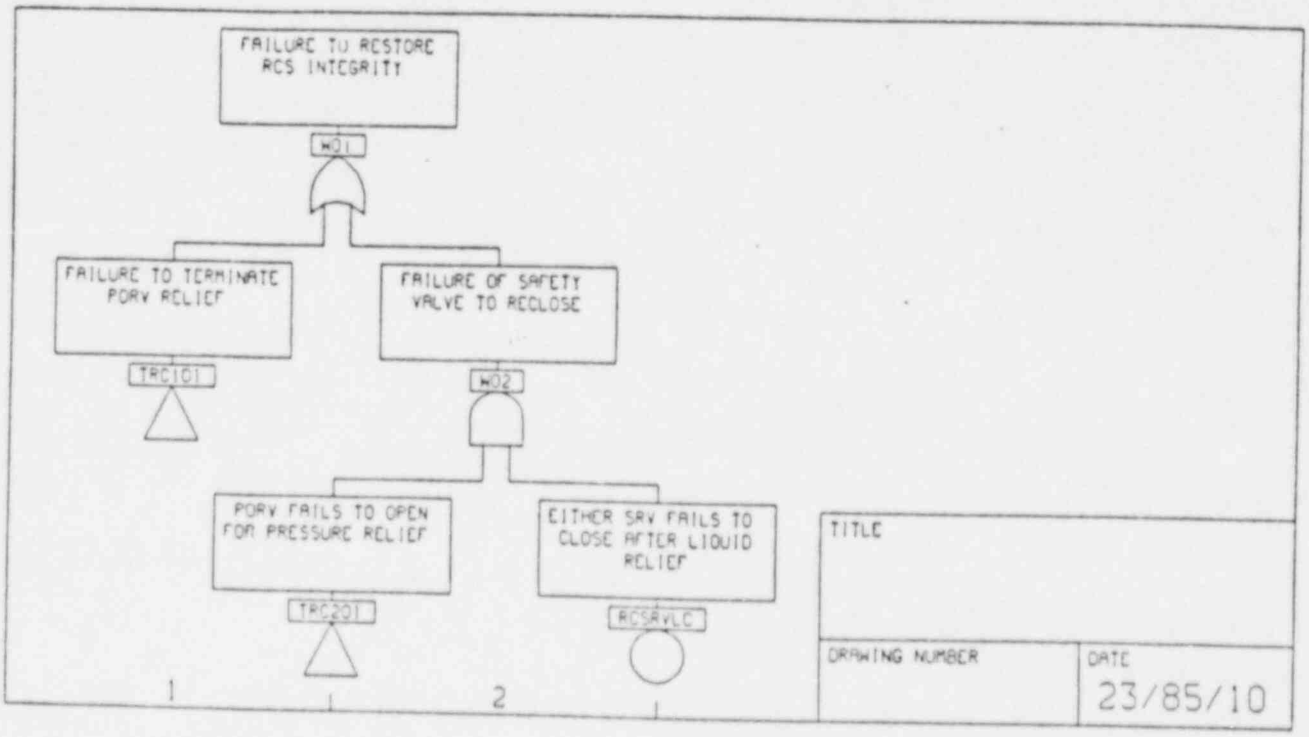

Figure 17. Fault tree for top event, W, failure to restore RCS integrity (Oconee PRA).

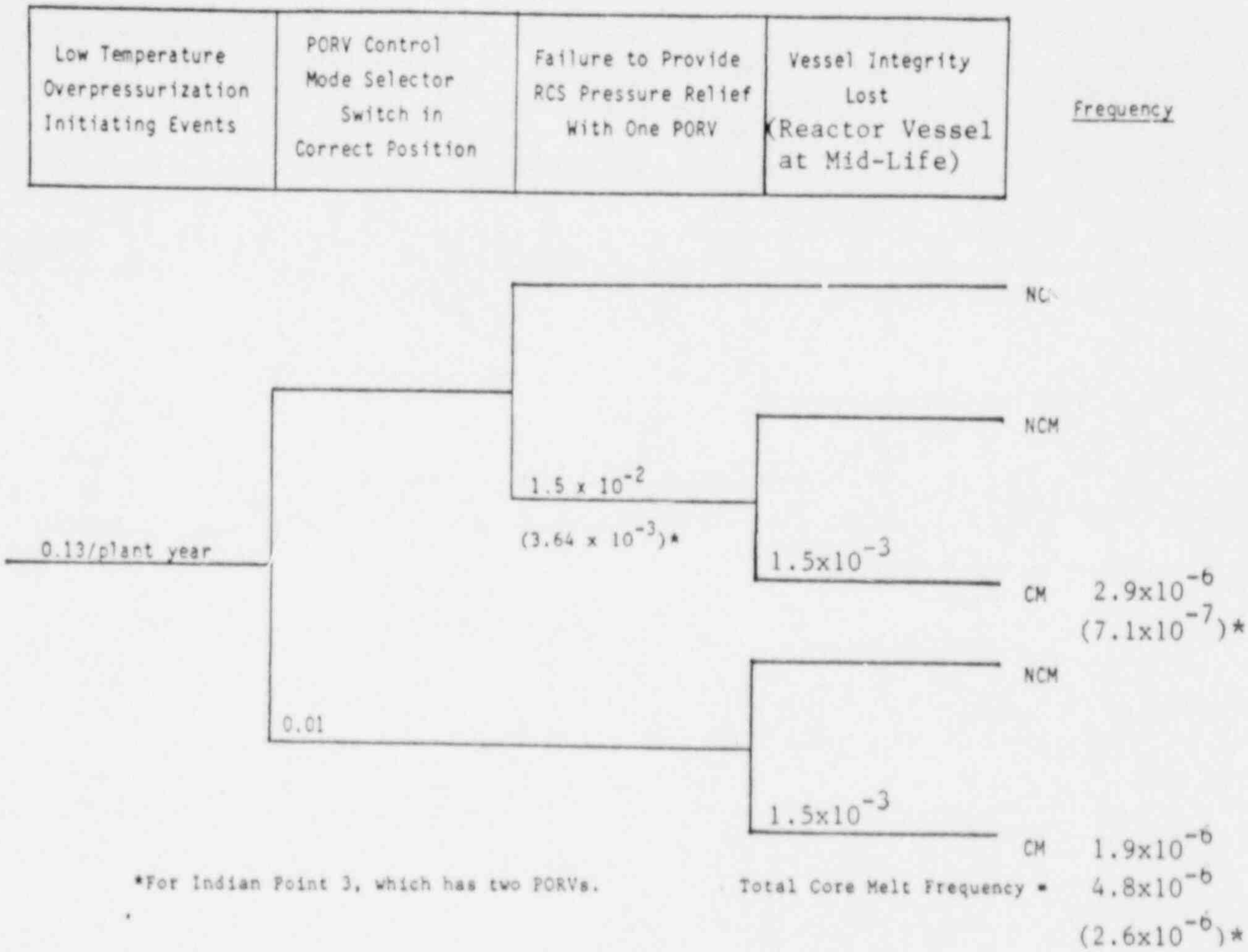

Figure 18. Event tree for low temperature overpressurization protection using PORVs.

Table 1
Summary of SGTR Core Melt Sequences and Frequencies
(Oconee 3)

Sequence Type	No. of Dominant Cut Sets	Core Melt Frequency/RY	Description of Sequence
		I. Bin IR: Event tree sequence $R \overline{B}_R U_R$	
A	2	$3.9x10^{-7}$	Failure of HPI due to BWST failure to provide suction to the HPI pumps
B	4	$8.5x10^{-7}$	HPI failure due to human errors and hardware faults in the HPI suction valve
C	1	$1.2x10^{-8}$	HPI failure due to loss of low pressure service water
		Sum = $1.3x10^{-6}$	
		Total Frequency for Bin IR = $1.3x10^{-6}$	
		II. Bin IIR: Event tree sequence $R \overline{B}_R \overline{U}_R X_R O$	
A	2	$4.0x10^{-7}$	A MSRV on the faulty SG stuck-open. Operator fails to refill the BWST and DHR fails.
B	7	$7.4x10^{-7}$	LPI system fails to operate for HPR or DHR. Operator fails to refill the BWST.
		Sum = $1.14x10^{-6}$	
		Sum of other Bin IIR cut sets = $2.7x10^{-7}$	
		Total frequency for Bin IIR seq. = $1.4x10^{-6}$	
		Total core melt frequency for SGTR initiating events = $2.7x10^{-6}$	

Table 2
A Brief Summary of Bin I Core-Melt Sequences (Oconee PRA)

Sequence Type	Event-Tree Sequence	Initiating Events	No. of Dominant Cut-Sets	Important Consequences Leading to Core Melt	Core-Melt Frequency from Dominant Cut Sets (yr^{-1})	$\sum$ (all other cut sets) (yr^{-1})	Total Core-Melt Frequency (yr^{-1})
A	SY_sX_s	Small break LOCA.	13	High pressure recirculation failure.	4.93×10^{-6}	3.4×10^{-8}	5.0×10^{-6}
B	TQU_s	Loss of LPSW, loss of power on bus 3TC, or reactor or turbine trip followed by LPSW pump failure.	4	RCP seal LOCA with no HPI available.	5.8×10^{-7}	9.3×10^{-9}	5.9×10^{-7}
C	SU_s	Small break LOCA.	4	HPI failure.	4.18×10^{-7}	2.5×10^{-8}	4.4×10^{-7}
D	T_6QU	Loss of instrument air.	5	RCP seal LOCA with no HPI available.	2.5×10^{-7}	----	2.5×10^{-7}
E	TQU_s	Failure of bus 3TC, or loss of LPSW followed by loss of MFW.	28	Loss of MFW, EFW and HPI. Stuck-open SRV (liquid discharge).	7.7×10^{-8}	2.4×10^{-8}	1.0×10^{-7}
F	TQU_s	Loss of instrument air, loss of offsite power or large feedwater line break.	40	Failure of EFW and HPI. Stuck-open SRV (liquid discharge).	6.3×10^{-8}	1.5×10^{-8}	7.8×10^{-8}
G	$T_{13}QU$	Failure of pressurizer control.	3	Stuck-open SRV (steam discharge) HPI failure.	6.3×10^{-8}	5.3×10^{-9}	6.83×10^{-8}
H	$T_{10}QY_sX_s$	Large feedwater line break inside reactor building.	2	Stuck-open SRV (liquid discharge). Failure to initiate HPR within 2 hrs.	1.4×10^{-8}	----	1.4×10^{-8}
I	TQU	Loss of all ac power.	2	Failures of TDEFWP and SSFASW. Stuck-open SRV (liquid discharge).	8.11×10^{-8}	----	8.11×10^{-8}
					6.48×10^{-6}	1.13×10^{-7}	6.6×10^{-6}

Table 3
A Brief Summary of Bin II Core-Melt Sequences (Oconee PRA)

Sequence Type	Event-Tree Sequence	Initiating Events	No. of Dominant Cut Sets	Important Consequences Leading to Core Melt	Core-Melt Frequency From Dominant Cut Sets(yr^{-1})	$\sum$ (all other cut sets) (yr^{-1})	Total Core-Melt Frequency (yr^{-1})
A	$T_5Q\overline{UY}X_s$	Loss of offsite power with failure to recover power to instrument air system for at least 12 hrs.	2	MFW and EFW failures. Stuck-open SRV (liquid discharge) HPR and LPR also fail.	1.0×10^{-8}	9×10^{-9}	1.9×10^{-8}
B	$TQ\overline{UY}X_s$	Loss of instrument air with failure to recover it for at least 12 hrs.	No individual cut sets above 1.0×10^{-8}	MFW and EFW failures. Stuck-open SRV (liquid discharge).	1.1×10^{-8}	----	1.1×10^{-8}
C	$\overline{SU}YXs$	Small LOCA followed by successful HPI.	5	Building-spray system terminated by the operator quickly. HPR and LPR fail to be initiated.	6.9×10^{-7}	----	6.9×10^{-7}
D	$TQ\overline{UY}X_s$	Large feedwater line break, total loss of main feedwater, and loss of condenser vacuum.	1	MFW not recovered in 10 min. Stuck-open SRV (liquid discharge).	2.8×10^{-8}	2.0×10^{-8}	4.8×10^{-8}
E	$TQ\overline{UY}X_s$	T, T_9, T_{11} that lead to overcooling events, or with additional failure such as stuck-open MSRV.	3	Stuck-open SRV (liquid discharge) HPR and LPR fail to be initiated.	4.9×10^{-8}	9.3×10^{-9}	5.83×10^{-8}
F	$TQ\overline{UY}X_s$	T_{13} (spurious low pressurizer pressure signal), or T_8 (inadvertent HPI actuation).	4	Stuck-open (steam discharge) for T_{13}, or liquid discharge for T_8. Recirculation failures.	1.88×10^{-7}	2.8×10^{-8}	2.16×10^{-7}
					9.8×10^{-7}	6.6×10^{-8}	1.05×10^{-6}

Table 4
Calculated Core-Melt Frequencies for Transient Initiating Events
(Indian Point 3) (All eight electric power states included.)

Event-Tree #	Description of Transient	Mean Annual Frequency (yr^{-1})	Core-Melt Frequency (yr^{-1})	Core-Melt Frequency with Improved PORVs and Block Valves Reliability (yr^{-1})
5	Steam-line break inside containment	2.16×10^{-3}	2.8×10^{-7} (2.8×10^{-7})*	1.5×10^{-7}
6	Steam-line break outside containment	2.16×10^{-3}	2.8×10^{-7} (2.8×10^{-7})	1.5×10^{-7}
7	Loss of main feedwater	3.8	5.7×10^{-7} (6.5×10^{-7})	3.4×10^{-7}
8	Closure of one main steam isolation valve	8.98×10^{-2}	1.4×10^{-8} (1.4×10^{-8})	8.0×10^{-9}
9	Loss or reactor coolant system flow	1.71×10^{-1}	2.6×10^{-8} (2.9×10^{-8})	1.5×10^{-8}
10	Core power excursion	2.57×10^{-2}	1.4×10^{-15} (0)	8.0×10^{-16}
11a	Turbine trip (general)	2.72	4.1×10^{-7} (4.6×10^{-7})	2.4×10^{-7}
11b	Turbine trip (loss of off-site power)	2.66×10^{-1}	5.8×10^{-6} ** (4.7×10^{-6})	5.8×10^{-6} **
11c	Turbine trip (loss of service water)	2.16×10^{-3}	8.5×10^{-8} (5.3×10^{-8})	8.5×10^{-8}
12a	Reactor trip	2.86	4.3×10^{-7} (3.9×10^{-7})	2.6×10^{-7}
12b	Reactor trip (loss of component cooling)	2.16×10^{-3}	3.3×10^{-8} (3.3×10^{-8})	3.3×10^{-8}
	Sum		7.93×10^{-6}	7.08×10^{-6}

*Values shown inside brackets denote the corresponding results presented in the IPPSS.

**Event-tree sequence 41 (a small LOCA due to RCP seal failure) and electric power recovery not considered

Table 5
A Brief Summary of Bin III Core-Melt Sequences (Oconee PRA)

Sequence Type	Event-Tree Sequence	Initiating Events	No. of Dominant Cut-Sets	Important Consequences Leading to Core Melt	Core-Melt Frequency From Dominant Cut Sets(yr^{-1})	$\sum$ (all other cut sets) (yr^{-1})	Total Core-Melt Frequency (yr^{-1})
A	T_2BU	Loss of main feed-water.	8	Failures of EFW and HPI cooling.	9.28×10^{-7}	2.6×10^{-7}	1.19×10^{-6}
B	T_4BU	Loss of condenser vacuum.	4	Loss of MFW and EFW HPI cooling also fails	2.9×10^{-7}	1.2×10^{-7}	4.1×10^{-7}
C	TBU	Turbine trip or other transient initiating event.	10	Loss of MFW and failure of EFW. Failure to initiate HPI cooling.	2.67×10^{-7}	1.6×10^{-7}	4.27×10^{-7}
D	$T_{11}BU$	Loss of ICS power.	1	Loss of MFW, followed by failures of HPI cooling and EFW.	4.0×10^{-8}	2.0×10^{-8}	6.0×10^{-8}
E	$T_{10}BU$	Large feedwater or condensate line break.	1	Loss of MFW and EFW. Operator fails to initiate HPI cooling.	4.7×10^{-6}	7.6×10^{-8}	4.78×10^{-6}
F	TBU	Loss of instrument air, or loss of offsite power.	15	Loss of MFW and failure of EFW. Failure to recover feedwater. Operator fail to initiate HPI cooling.	4.60×10^{-6}	6.6×10^{-8}	4.67×10^{-6}
G	TBU	Failure of low-pressure service water.	7	HPI pumps failure. Failure to initiate SSF seal injection within 30 min. leads to a small RCS leak with inability to makeup	1.46×10^{-5}	4.4×10^{-8}	1.47×10^{-5}
H	TBU	Loss of instrument air, or loss of offsite power which result in loss of instrument air.	4	Loss of letdown storage tank makeup. Flow to HPI pumps not available. Seal injection failure. Slow RCS leakage with no ability to makeup.	1.0×10^{-7}	1.2×10^{-7}	2.2×10^{-7}

Table 5 (continued)

Sequence Type	Event-Tree Sequence	Initiating Events	No. of Dominant Cut-Sets	Important Consequences Leading to Core Melt	Core-Melt Frequency From Dominant Cut Sets(yr^{-1})	$\sum$ (all other cut sets) (yr^{-1})	Total Core-Melt Frequency (yr^{-1})
I	TBU	Loss of ac power for more than 12 hrs.	1	Failure of standby shutdown facility RCVCS to provide seal injection in 30 min., followed by gradual loss of RCS inventory, loss of SG heat transfer, RCS boils off.	2.6×10^{-8}	----	2.6×10^{-8}
J	TBU	Loss of all ac power for more than 2 hrs.	2	Turbine-driven EFWP fails, SSF fails to provide SSF ASW within 30 min. RCS boils off.	2.9×10^{-8}	----	2.9×10^{-8}
					2.56×10^{-5}	8.7×10^{-7}	2.65×10^{-5}

Table 6
A Brief Summary of Bin IV Core-Melt Sequences (Oconee PRA)

Sequence Type	Event-Tree Sequence	Initiating Events	No. of Dominant Cut-Sets	Important Consequences Leading to Core Melt	Core-Melt Frequency From Dominant Cut Sets(yr^{-1})	$\sum$ (all other cut sets) (yr^{-1})	Total Core-Melt Frequency (yr^{-1})
A	$TB\overline{UYW}LX$	Loss of offsite power and failure to recover offsite power to MFW and instrument air system for at least 12 hrs.	1	MFW unavailable and EFW is lost due to various turbine-driven EFW pump failures, HPI cooling fails in about 6 hrs. due to HPI pump room flooding, SSF fails to provide seal injection and feed-water to the secondary side within 30 min. on loss of HPI.	2.0×10^{-8}	1.2×10^{-8}	3.2×10^{-8}
B	$TB\overline{UYW}LX$	Same as above.	3	MFW unavailable and EFW is lost due to various turbine-driven EFW pump failures, combined with loss of suction to motor-driven EFW pumps. HPI cooling is successful, HPR fails at 12 hrs. SSF fails to provide seal injection or feed-water to the secondary-side.	1.52×10^{-7}	1.0×10^{-8}	1.62×10^{-7}
					1.72×10^{-7}	2.2×10^{-8}	1.94×10^{-7}

Table 7
Estimated Environmental Release Fractions for a Core Melt Accident Resulting From a Low Temperature Overpressurization Event With an Open Containment

Species	Environmental Release Fraction
Kr	1.00
I	.12
Cs	.088
Te	.17
Sr	.05
Ru	.005
La	.014
Ce	.003
Ba	.005

Table 8
Summary of Input Assumptions for the CRAC-2 Consequence Calculations for a Core Melt Accident Resulting from a Low Temperature Overpressurization

Population Distribution	Uniform
Population Density	100/sq. mi.
Weather Conditions	Typical eastern wind rose
Accumulated Period	30 years
Evacuation Delay	2 hours
Warning Time	4 hours

Table 9
Estimated Reactor Vessel Failure Probability For a Low Temperature Overpressure Event (T<100C and P-2485 psia) Based on ORNL[2] Structural Analyses[2]

	Failure Probability (Per Reactor Year)	
	Indian Point 3	Oconee 3
Present Conditions	$3x10^{-5}$	$8x10^{-5}$
Mid-life of Reactor (22 full power years)	$1.5x10^{-3}$	$1.5x10^{-3}$
End of Reactor Vessel Life	$2.6x10^{-3}$	$2.6x10^{-3}$

Table 10
Total Estimated Risk for the Low-Temperature Overpressurization

	Risk (Person-Rem/Reactor Year)	
	Indian Pt. 3	Oconee 3
Present Conditions (Early in reactor vessel life)	0.2	1.2
Mid-life of Reactor (22 years full power)	12	23
End of Reactor Vessel Life	21	40

<u>REFERENCES</u>

1. T. J. Burns, et al., Pressurized Thermal Shock Evaluation of the Oconee-1 Nuclear Power Plant, NUREG/CR-3770, ORNL/TM-9176, (Draft), p. 4.36, Oak Ridge National Laboratory, Oak Ridge, Tennessee, April, 1984.

2. Ibid. P. 5.33.

3. Schedule for Resolving and Completing Generic Issue No. 94 - Additional Low-Temperature-Overpressure Protection for Light Water Reactors, A Memo from H. R. Denton to R. M. Bernero, July, 1985.

4. Nelson Hanan, Private Communications.

5. Indian Point Probabilistic Safety Study, Power Authority of the State of New York, Consolidated Edison Company of New York, Amendment 2, December, 1983.

6. Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3, NSAC-60, Vol. 1, p. 3-40, Co-sponsored by Electric Power Research Instititute, Palo Alto, California, and Duke Power Company, Charlotte, North Carolina, June, 1984.

7. Ibid. Vol. 1, p. 3-42.

8. Ibid. Vol. 4, Appendix D, D-17.

9. W. Tauche, "Loss of Feedwater Induced Loss of Coolant Accident Analysis Report," WCAP-9744, May, 1980.

10. Westinghouse Electric Corporation, "Report on Small Break Accidents for Westinghouse NSSS System," WCAP-9601, Vol. III, June, 1979.

11. R. E. Worrell and D. W. Stack, A SETS User's Manual for the Fault Tree Analyst, SAND77-2051, Sandia National Laboratories, Albuquerque, New Mexico, 1978.

12. A. Meliksetian and A. M. Sklencar, "Valve Inlet Fluid Conditions for Pressurized Safety and Relief Valves in Westinghouse - Designed Plants," EPRI Research Project V102-19, December 1982.

13. AEOD/C401, "Low Temperature - Overpressure Events at Turkey Point Unit 4," March 1984.

14. A. D. Swain and H. E. Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, October 1980.

15. M. Khatib-Rahbar et al., "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plant," Draft, NUREG/CR-4551, Volume 5, September 1986.

16. L. T. Ritchie et al., "CRC2 Model Description," Sandia National Laboratory, NUREG/CR-2552, March 1984.

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13. ABSTRACT (200 words or less)

An analysis was performed to explore the risk reduction potential of improving the pressurizer PORV and block valve reliability for two representative PWR plants, Indian Point 3, and Oconee 3. Attention was focused upon particular transient scenarios including steam generator tube rupture (SGTR), a stuck-open PORV, cooldown to cold shutdown, and the use of PORV in low-temperature overpressurization (LTOP) protection. The feasibility of using the PORV as a high point vent to supplement the function of Reactor Vessel Head Vent System (RVHVS) was also studied.

The pertinent event trees, fault trees, and the basic data presented in the Indian Point Probabilistic Safety Study (IPPSS) and the Oconee PRA were utilized to quantify the benefit of an improved PORV and block valve reliability in terms of potential reduction in core-melt frequencies. For Indian Point 3, independent core-melt frequency calculations were made based on the Boolean expressions derived for various plant damage states. For Oconee 3, the components of the dominant cut sets and the detailed fault trees shown in the Oconee PRA were given thorough scrutiny to determine their relevance to the hardware or operational failures of the PORV and its block valve.

With the exception of LTOP, the core-melt frequencies attributable to PORV or block valve failures were found to be relatively insignificant, only a very small fraction of the total core-melt frequency due to internal events.

For the case of LTOP, the core melt frequency and associated risk appear to be small for Indian Point 3 and Oconee 3 since the vessels have not had a substantial fraction of their estimated lifetime irradiation. The results of a conservative estimation of health effect, however, indicate that the public risk from LTOP events may become more significant late in plant life when the aging effects of the vessel contribute to increased vulnerability. These effects are being studied in more depth in the NRC's studies for Generic Issue Number 94.

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