

**Contribution of Anticipated Transients Without Scram
(ATWS) to Core Melt at United States Nuclear Power
Plants**

by

Richard T. Giachetti

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The research was performed under appointment to the Nuclear Engineering, Health Physics, and Radioactive Waste Management Fellowship program administered by Oak Ridge Associated Universities for the U. S. Department of Energy.

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Richard T. Giachetti
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TABLE OF CONTENTS

INTRODUCTION	0.1
CHAPTER 1 - AN ATWS HISTORY	1.1
CHAPTER 2 - THE <i>WASH-1400</i> PLANTS	2.1
CHAPTER 3 - BIG ROCK POINT	3.1
CHAPTER 4 - LIMERICK UNIT I	4.1
CHAPTER 5 - INDIAN POINT UNIT 2	5.1
CHAPTER 6 - INDIAN POINT UNIT 3	6.1
CHAPTER 7 - OCONEE UNIT 3	7.1
CHAPTER 8 - ZION UNIT 1	8.1
CHAPTER 9 - COMPARATIVE DISCUSSION	9.1
CHAPTER 10 - SUMMARY RESULTS AND CONCLUSIONS	10.1

LIST OF TABLES

- TABLE 2.1 - SURRY UNIT I DATA SHEET
- TABLE 2.2 - PEACH BOTTOM UNIT II DATA SHEET
- TABLE 3.1 - BIG ROCK POINT DATA SHEET
- TABLE 3.2 - STUCK OPEN SAFETY VALVE SEQUENCES AT BIG ROCK POINT
- TABLE 3.3 - ATWS SEQUENCES AT BIG ROCK POINT
- TABLE 4.1 - LIMERICK UNIT I DATA SHEET
- TABLE 5.1 - INDIAN POINT UNIT 2 DATA SHEET
- TABLE 5.2 - CONDITIONAL CORE MELT PROBABILITIES FOR INDIAN POINT 2
INTERNAL INITIATORS
- TABLE 5.3 - CONDITIONAL ATWS CORE MELT PROBABILITIES FOR INDIAN POINT 2
- TABLE 6.1 - INDIAN POINT UNIT 3 DATA SHEET
- TABLE 6.2 - CONDITIONAL CORE MELT PROBABILITIES FOR INDIAN POINT 3
INTERNAL INITIATORS
- TABLE 6.3 - CONDITIONAL ATWS CORE MELT PROBABILITIES FOR INDIAN POINT 3
- TABLE 7.1 - OCONEE UNIT 3 DATA SHEET
- TABLE 8.1 - ZION UNIT 1 DATA SHEET
- TABLE 8.2 - CONDITIONAL CORE MELT PROBABILITIES FOR ZION 1 INTERNAL
INITIATORS
- TABLE 8.3 - CONDITIONAL ATWS CORE MELT PROBABILITIES FOR ZION 1
- TABLE 9.1 - SUMMARY OF PLANT DESCRIPTIONS
- TABLE 9.2 - RPS UNAVAILABILITIES
- TABLE 9.3 - PLANT INITIATOR FREQUENCIES
- TABLE 9.4 - PLANT CORE MELT PROBABILITIES
- TABLE 9.5 - PERCENT CONTRIBUTIONS TO TOTAL CORE MELT PROBABILITY
- TABLE 9.6 - PLANT ATWS CORE MELT PROBABILITIES
- TABLE 9.7 - PERCENT CONTRIBUTIONS TO ATWS CORE MELT PROBABILITY

LIST OF TABLES, continued

TABLE 9.8 - PERCENT ATWS CONTRIBUTIONS TO TOTAL CORE MELT

TABLE 9.9 - COMPARISON OF PERCENT ATWS CONTRIBUTIONS FROM TWO STUDIES

TABLE 10.1 - SUMMARY OF PLANT DESCRIPTIONS

TABLE 10.2 - RPS UNAVAILABILITIES

TABLE 10.3 - PLANT INITIATOR FREQUENCIES

TABLE 10.4 - PLANT CORE MELT PROBABILITIES

TABLE 10.5 - PERCENT CONTRIBUTIONS TO TOTAL CORE MELT PROBABILITY

TABLE 10.6 - PLANT ATWS CORE MELT PROBABILITIES

TABLE 10.7 - PERCENT CONTRIBUTIONS TO ATWS CORE MELT PROBABILITY

TABLE 10.8 - PERCENT ATWS CONTRIBUTIONS TO TOTAL CORE MELT

TABLE 10.9 - COMPARISON OF PERCENT ATWS CONTRIBUTIONS FROM TWO
STUDIES

INTRODUCTION

0.0 Purpose

This report looks at WASH-1400 and several other Probabilistic Risk Assessments (PRAs) and Probabilistic Safety Studies (PSSs) to determine the contribution of Anticipated Transients Without Scram (ATWS) events to the total core melt probability at eight nuclear power plants in the United States. After considering each plant individually, the results are compared from plant to plant to see if any generic conclusions regarding ATWS, or core melt in general, can be made.

Chapter 1 presents a history of the ATWS issue - when it was born, what it is and what has been done about it over the past two decades. Chapters 2 through 8 present the information from the various probabilistic studies, describing in detail how I obtain the values presented. Chapter 2 deals with the WASH-1400 plants (Peach Bottom II and Surry I), while chapters 3 through 8 each deal with one of the other plants studied. Chapter 9 compares the results obtained in Chapters 2 through 8 while chapter 10 summarizes the results and presents the conclusions.

0.1 Plants Studied

Three Boiling Water Reactors (BWRs) and five Pressurized Water Reactors (PWRs) were studied. All three BWRs are of General Electric design. All but one PWR are by Westinghouse. The Oconee PWR is by Babcock and Wilcox. The following table lists each plant that was studied along with the reference study used.

Boiling Water Reactors

Peach Bottom Unit II
Big Rock Point
Limerick Unit I

Reference Study

WASH-1400 (1975)
Big Rock Point PRA (1981)
Limerick Generating Station PRA (1982) and Limerick
Generating Station Severe Accident Risk Assessment
(1983)

Pressurized Water Reactors

Surry Unit 1
Indian Point Unit 2
Indian Point Unit 3
Oconee Unit 3
Zion Unit 1

Reference Study

WASH-1400 (1975)
Indian Point PSS (1982)
Indian Point PSS (1982)
Oconee Unit 3 PRA (1984)
Zion PSS (1981)

0.3 Limitations of this Study

The conclusions in this study are subject to several limitations. First and foremost is the fact that in the United States each nuclear power plant has its own individual design. This means that problems, advantages and disadvantages can vary widely from plant to plant even if designed and built by the same firms and using the same nuclear steam supply systems. Thus it is impossible to make generalizations about all plants based on the limited study of a few plants. One could estimate the contribution of a particular initiator at a given plant based on analyses of plants with similar designs, but that approach is no substitute for a specific analysis of the plant in question.

Second, this study looks at core melt probabilities rather than radioactive release and public consequences. Consequently one must assume that there is a direct relationship between core melt and public consequences, which is not necessarily true. The lack of such a relationship is significant in that ATWS events typically result in high containment pressure, increasing the probability of breaching containment leading to increased consequences. Thus it is possible that an ATWS event could result in greater public consequences than another core melt initiator with the same core melt probability but a lower containment failure probability.

CHAPTER 1 - AN ATWS HISTORY

1.0 Introduction

The history of ATWS dates back to January of 1969, to a letter from a consultant to the Advisory Committee for Reactor Safety (ACRS). [1] Since then, the regulatory staff has taken various positions regarding ATWS while receiving various degrees of resistance from the reactor vendors and utilities. The crucial aspect of ATWS, and the reason it has taken so long to deal with, is the difficulty of determining the reliability of the trip system. This difficulty arises because a major contributor to unavailability is deemed to be the effect of common cause failure on the system availability and there is considerable uncertainty in the calculation of common cause failure rates. [2] The vendors have consistently calculated unavailability values at least as low as 10^{-6} to 10^{-7} per demand, while the regulatory staff has estimated unavailability values 1000 times greater than these. [1] To date, the operational experience data is insufficient to support the vendors' claims. The following history outlines the major developments of the ATWS issue.

1.1 The Birth of ATWS

Prior to the letter of January 1969, no ATWS events had occurred at any operating commercial power reactor, but there had been one precursor at the Kahl reactor in Germany in 1963. This incident involved common cause failure of the trip relays. Due to a change in the manufacturing process, the protective coating on the relays became sticky during operation and caused the relays to stay closed when they were required to open. Fortunately, these failures were discovered during routine testing and the situation was rectified. [3]

As mentioned above, the ATWS issue was officially born in January 1969 in a letter

from ACRS consultant E.P. Epler to ACRS Executive Secretary R.F. Fraley. The letter discussed scram reliabilities for Boiling Water Reactors (BWRs), and predicted that unavailabilities less than 10^{-4} could not be expected due to systematic (common cause) failure. In the following months of 1969, the ACRS required several BWR applicants (Hatch and Brunswick) and one PWR applicant (Three Mile Island) to study ways to prevent common mode scram failures as well as ways to make ATWS consequences tolerable. [1]

By mid-1970, three of the four reactor vendors had provided summaries of predicted ATWS consequences to the Atomic Energy Commission (AEC) regulatory staff. Westinghouse appeared to be in the best position due to the large pressure relief capacity of their reactor pressure vessels. They felt that slightly increased relief capacity or a boron injection system could solve the problem. Babcock and Wilcox feared that they would have to do more than Westinghouse due to a smaller relief capacity and a lower water inventory on the secondary side of the steam generators. General Electric's primary concern was that the operators would only have a few seconds to respond to successfully mitigate an ATWS. They postulated that tripping the recirculation pumps could solve the problem. Combustion Engineering had provided no information. [1]

In September 1970, the AEC regulatory staff reported a position on ATWS to the ACRS. This position indicated that the current reliability of scram systems combined with the occurrence rate of anticipated transients would result in an unacceptable ATWS frequency when many plants (i.e. >1000) were in operation. Thus the staff recommended that applicants be required to: 1) demonstrate that with their present designs, ATWS consequences would be acceptable, or 2) make design changes to significantly improve the reliability of scram systems. The ACRS was not ready to endorse this position and decided that further investigation was needed. [1]

1.2 ATWS: A Generic Safety Issue

By 1971, ATWS was declared a generic safety issue. An ad hoc ACRS subcommittee

was set up in January of 1971 to look at this issue. In meetings with the committee, the vendors claimed that their scram system unavailabilities were less than 10^{-6} or 10^{-7} per demand. General Electric even claimed their unavailability was as low as 10^{-15} per demand. The committee's meeting minutes showed dissatisfaction with the slow pace of the vendors and the AEC staff. [1]

In February of 1971, General Electric proposed recirculation pump trip as a back-up for the scram system at the Newbold Island plant. However, in March, during the operating license review for the Quad Cities BWRs, GE and Commonwealth Edison said that recirculation pump trip was not needed at these plants. [1] This dichotomous treatment of proposed and existing plants exemplified the vendors' and utilities' aversion to any type of backfitting at existing plants. As will be seen, this desire to avoid backfitting played a major role in prolonging the ATWS issue. On July 30, 1971 the AEC staff forwarded a report to the ACRS which concluded that recirculation pump trip substantially increased the probability that a plant would be able to handle an ATWS event. By August 1971, both Newbold Island and Limerick were committed to using recirculation pump trip. [1]

On April 28, 1972, the Director of Regulation, L. Manning Muntzing, sent a letter to the ACRS outlining a new staff position. Specifically, this position would require applicants to: 1) demonstrate that with their present designs the consequences of ATWS would be acceptable, or 2) make design changes which would make ATWS consequences acceptable, or 3) make design changes to significantly improve the reliability of scram systems. This was essentially the same staff position as that of September 1970. The ACRS quickly agreed with the staff position and sent a letter of approval on May 10, 1972, apparently resolving the ATWS issue. [1]

1.3 WASH-1270 and a "Change in Position"

However, the regulatory staff changed its position in a draft report dated November

30, 1972. This new position would require "new" plants (i.e. those filing for construction permits in the near future) to have two independent scram systems, but they would not have to be able to withstand the consequences of an ATWS. This position was forwarded to the ACRS in January of 1973, and the ACRS gave its approval in April 1973, but more to get action on the issue, not because it was a better position. This position was publicly adopted in September 1973 with the publication of WASH-1270. [1]

WASH-1270 also gave an overall safety goal for ATWS as well as a quantitative safety goal. Based on some rather arbitrary numbers for such things as the recurrence interval for accidents not within the design basis (ATWS apparently fitting this category), the AEC staff concluded that ATWS events should not lead to a serious accident (i.e. core melt or consequences greater than those permitted in 10 CFR 100) with a frequency greater than 10^{-7} per R-yr. [1,3]

The staff felt that with the publication of WASH-1270, ATWS was resolved except for implementation and in February 1974 the ACRS moved ATWS to the resolved column on its list of generic items. Despite these views, the vendors did not submit any firm proposals for the soon to be required diverse scram systems. Several vendors had submitted analyses of protection systems showing unavailabilities less than 10^{-7} , but the staff considered this approach unacceptable since it did not satisfy the intent of WASH-1270 or the staff criteria. The vendors complained that the staff criteria did not give enough guidance for the initiation of new designs. [1]

In October 1975, the AEC staff met with the ACRS to discuss a possible major revision to the WASH-1270 position with respect to Class A plants (i.e. those plants filing for construction permits after October 1, 1976 for which a diverse scram system was required). Four alternatives were discussed, ranging from no change at all to a re-evaluation of WASH-1270. The staff recommended alternative number four, i.e. that ATWS consequences at Class A plants be made tolerable by other design changes if

necessary. The ACRS agreed and sent a letter of concurrence. This position was effectively the "pre-WASH-1270" position, which was the position of April 1972 and September 1970, and was the position that the ACRS preferred. [1] Thus, almost seven years after the ATWS issue was born, over five years were wasted as the staff and the ACRS arrived at the same position three different times.

1.4 Resistance from Industry

During the next several years, and with the support of the ACRS, the regulatory staff attempted to implement this policy but received strong resistance from the vendors and utilities. It is during this period that the vendors' and utilities' aversion to backfitting was perhaps most prominent in delaying the resolution of the ATWS issue. They felt that the costs to meet the staff requirements were significant but not justified. [4] They claimed that no design changes were necessary because WASH-1400 (published in October 1975) found that ATWS was not a major contributor to risk. In the fall of 1976, the Electric Power Research Institute (EPRI) began publishing a series of reports entitled *ATWS: A Reappraisal*. They concluded that scram unavailabilities were much lower than 10^{-4} and that ATWS was not a significant threat to the health and safety of the public. Additionally, in early 1976 the ACRS discovered that even though new BWRs since Newbold Island (1971) had incorporated recirculation pump trip, operating BWRs had not. The ACRS sent a memorandum to the staff recommending that all BWRs employ recirculation pump trip, and the staff in turn sent letters to all BWR operators requesting their specific plans to incorporate this feature. [1] (Despite these letters, many BWR operators did not install automatic recirculation pump trip until a show cause order requiring such a system was issued on February 21, 1980: nine years after Newbold Island. [2])

Finally, in March 1977, the AEC's successor, the Nuclear Regulatory Commission (NRC), established a task force to resolve the ATWS issue. In July 1977 the NRC staff reported the task force's findings to the ACRS. These findings reiterated the position of

October 1975, which was essentially the same position of April 1972 and of September 1970. [1]

1.5 NUREG-0460 to Brown's Ferry

However, in April 1978, the NRC staff took a new position on ATWS in a report entitled *Anticipated Transients without Scram for Light Water Reactors* (NUREG-0460). First, the staff changed the safety objective from 10^{-7} unacceptable ATWS events per R-yr to 10^{-6} unacceptable ATWS events per R-yr. This change was apparently based on the core melt probability of 10^{-5} per R-yr calculated in WASH-1400, as well as on the idea that the intent of WASH-1270 was to exclude ATWS from the design basis envelope, whereas now the idea was to include ATWS in the design basis. [1,3] Secondly, the staff estimated that the current frequency of ATWS events with unacceptable consequences was 2×10^{-4} per R-yr. Lastly, the staff used deterministic as well as probabilistic methods to determine the design approaches needed for each vendor to meet the objective. [1] Additionally, this new position would require diversity of the rod absorber component of the diverse scram system, whereas previous positions did not. [4] There was strong opposition from the industry, and even the ACRS was not ready to endorse this position. [1]

In early 1979, after the Lewis Committee report (NUREG/CR-1400) and the Commissioners' endorsement of the report, the staff proposed a greatly revised position on ATWS; one that reflected the difficulty of backfitting. For operating plants or plants under construction, the emphasis was on circuitry changes which might improve scram reliability. For plants to be constructed, the emphasis was still on hardware changes to mitigate ATWS consequences. The staff used engineering judgment to arrive at this new staff position since the commissioners stated that probabilistic methods alone could not be used to provide a basis for licensing. [1]

In early 1980, in the aftermath of the Three Mile Island accident, the regulatory staff

proposed a more stringent position on ATWS and industry objected. Even the ACRS took a position requiring less backfitting than the staff position required. On June 28, 1980 a second ATWS precursor occurred, this time at the Browns Ferry Unit 3 BWR when 76 of 185 control rods failed to fully insert when a scram was called for. [1] This event underscored the urgency of the ATWS issue and the need to deal with it quickly and effectively.

1.6 Back on Track

On September 4, 1980, the regulatory staff finally seemed to get the ATWS issue back on track. In a report numbered SECY-80-409, the staff recommended that the commissioners publish a proposed rule to require improvements in reactor designs to reduce the likelihood of ATWS and to mitigate the consequences of ATWS. [5] This position is almost identical to the positions of July 1977, October 1975, April 1972 and the original staff position of September 1970. Thus, almost twelve years after Epler's letter, the staff was recommending a position very similar to the original position it had arrived at ten years earlier.

On November 4, 1980 a utility group on ATWS published a proposed rule in a petition for rulemaking (PRM-50-29). The modifications required by this rule were keyed to the type of plant and the manufacturer. Over a year later, on November 24, 1981, the NRC published two alternative proposed rules for public comment. One was the "staff" rule which would require plant specific analyses to identify needed improvements. The other was the "Hendrie" rule which would require a reliability assurance program along with some hardware changes. Additionally, the public comment period for the utility group's petition for rulemaking was reopened, and all three proposed rules were under consideration. By June 28, 1982, 39 comments had been received, 31 of which were from utilities. Most of the comments said that no rule was needed, but that if one was adopted, the utility group's proposal was preferred. [5]

On July 13, 1982 an NRC task force and steering group was formed to consider four alternatives regarding the ATWS rule: [2]

1. Make no ATWS rule at all, or include it in the Severe Accident Program,
2. Adopt the proposed utility group rule, or a modified version of it,
3. Adopt the staff rule, or a modified version of it, or
4. Adopt the sections of the Hendrie rule for which there are technical bases.

Then, on February 22 and 25, 1983, two ATWS events occurred at Salem I, a Westinghouse PWR. Fortunately, in both cases the reactor was scrammed manually within 30 seconds after automatic trip failed and there was no core damage and no radioactive release. [2,5] Surprisingly, this did not cause a major re-evaluation of the proposed ATWS rules and in July 1983 the NRC staff recommended publication of a final ATWS rule and a related proposed rule for Westinghouse reactors. The final rule would be a modified version of the utility group's proposal, while the proposed rule for Westinghouse plants was a result of the Salem I ATWS events and would require diverse scram systems for Westinghouse reactors. [5]

On June 26, 1984, the NRC published the final rule on ATWS in the *Federal Register*. The rule adopted the utility group's approach with some exceptions, notably increased (86 gpm) Stand-by Liquid Control System (SLCS) flow for all BWRs (not just new ones) and including the reactor trip breakers as part of the scram system requiring diversity. Other requirements of the rule included diverse scram systems for Babcock and Wilcox, Combustion Engineering, and GE reactors; automatic recirculation pump trip for BWRs; and automatic turbine trip and Auxiliary Feedwater System (AFWS) initiation for PWRs. [2,5]

The effective date of the ATWS rule was July 26, 1984, and implementation was to be completed by the second refueling outage after the effective date, or by the date of issuance of a license authorizing operation greater than 5% of full power. If implementation was scheduled to occur after these deadlines, a justification was to be

submitted and a final schedule was to be worked out between the licensee and the commission. Older plants could apply for exemption on the basis of power level, unique design features, remaining plant lifetime or remote siting. [2]

References:

- [1] *Nuclear Reactor Safety: on the History of the Regulatory Process.* David Okrent. Madison, Wisconsin: University of Wisconsin Press, 1981. p. 243-260.
- [2] "Amendments r.e. Design and Ops to Reduce Risk From ATWS." *Federal Register of the United States.* Vol. 49, June 26, 1984. p. 26036-26045.
- [3] *Anticipated Transients Without Scram for Water-Cooled Power Reactors.* WASH-1270. U.S. Atomic Energy Commission. Washington, D.C.: U.S. Atomic Energy Commission, 1973.
- [4] *Anticipated Transients Without Scram for Light Water Reactors.* NUREG-0460. U.S. Nuclear Regulatory Commission. Washington, D.C.: U.S. Nuclear Regulatory Commission, 1978.
- [5] "ATWS - An Update from NRC", Roger J. Matson. *Anticipated and Abnormal Plant Transients in Light Water Reactors. Vol. I.* Pamela L. Lassahn, Debu Majumdar and George F. Brockett, eds. New York: Plenum Press, 1984. p. 305-309.

CHAPTER 2 - THE WASH-1400 PLANTS

2.0 Introduction

WASH-1400 [1] was the first study that tried to quantify the public risk due to nuclear power plants by means of a systematic investigation of plant performance. The probabilistic approach introduced in WASH-1400 has been the basis for all of the probabilistic studies and assessments that have followed. This is not to say that WASH-1400 is flawless. On the contrary, the validity of WASH-1400 was hotly debated, and many of the specific methods it used were deemed invalid by the NRC Commissioners and others. But the general approach was considered a step in the right direction, and today this approach is seen as a useful tool in assessing plant designs.

I did not blindly accept the information presented in WASH-1400. Instead I tried to trace how results were arrived at and in many cases I made adjustments to correct for faulty methods or to maintain consistency with the PRA methods used in the other reports referenced in this study. The following sections of this chapter discuss how the values for initiator frequencies, RPS unavailability, core melt probability and ATWS core melt probability were obtained from the information in WASH-1400.

2.1 Plant Descriptions

The PWR studied in WASH-1400 was the Surry Unit I plant located in Surry, Virginia. The steam supply system is a Westinghouse three loop design using upright U-tube steam generators. The power rating is 2441 MWt (788 MWe). The large dry containment building is made of reinforced concrete with a steel lining. The WASH-1400 authors chose this reactor because they felt that it would be representative of PWRs to be built in the future. For the most part, today's PWRs are very similar to Surry, with perhaps the biggest exception being the use of four cooling loops instead of three.

The BWR studied in WASH-1400 was the Peach Bottom Unit II plant. This reactor is a

General Electric BWR/4. The power rating is 3293 MWt (1065 MWe). The containment structure is a Mark I design using an inert argon atmosphere. Again, the WASH-1400 authors chose this plant because they felt it was representative of the BWRs to follow. Although many of today's BWRs are BWR/4 reactors, many of the containment structures are of the Mark II design.

2.2 The Initiator Frequencies

WASH-1400 identified four basic classes of initiators: loss of coolant accidents (LOCAs), transients, catastrophic vessel rupture and external events such as earthquakes and tornados. The frequencies of the internal initiators (i.e. LOCAs, transients and vessel rupture) were determined in Appendix II of WASH-1400 from equipment failure data and plant operating experience data. Specific transient frequencies (i.e. frequencies for turbine trips, loss of feedwater flow, MSIV closure, loss of off-site power, etc.) were not determined, but rather a total transient frequency of ten events per year was assumed, and the analysis of transients was carried out on this generic basis. The external initiator frequencies were determined from published data on seismic activity, tornado strike probabilities, aircraft crash frequencies and turbine missile frequencies. No other external initiators were studied.

The methodology used in WASH-1400 to arrive at the initiator frequencies appears sound, even though some of the information was rather inadequate for my purposes (i.e. the lumping together of the transient frequencies). Despite this drawback, I accepted the initiator frequencies presented in WASH-1400 as valid, and have included these values in tables 2.1 and 2.2, the data sheets for Surry Unit I and Peach Bottom Unit II, respectively.

2.3 The Reactor Protection System

WASH-1400 used a fault tree analysis to determine the unavailability of the reactor protection system (RPS). For the PWR RPS, contributions from single and multiple channel failures as well as test and maintenance failures were considered. A common cause failure

analysis was performed which assumed that common cause failure would occur due to miscalibration of redundant instrument channels. The WASH-1400 authors viewed the calibrations as a loosely coupled system which was analyzed as a combination of a completely dependent system and a completely independent system. Specifically, the common cause failure probability was calculated as the log-normal mean of the miscalibration probabilities found for the completely dependent and the completely independent systems. This calculation received much criticism and is considered today to be a poor approximation. However, for the PWR RPS, the common cause failure analysis was not included in the RPS unavailability since the WASH-1400 authors felt that there are many diverse sensors and they are not all calibrated at the same time. Using this approach, the WASH-1400 authors obtained a PWR RPS unavailability value of 3.6×10^{-5} per demand.

Evidently the WASH-1400 authors did not think that common cause failures would contribute significantly to the PWR RPS unavailability since they did not include the common cause failure contribution in their result. The reasoning they used was that the PWR RPS has many diverse sensors which are not all calibrated at the same time. A favorable argument can be made for this reasoning since part of the RPS may fail due to miscalibration, while the many other diverse channels will still function properly. However, I did not accept the WASH-1400 authors' approach since there must be some contribution due to common cause failures. From WASH-1400 the probability that two completely independent systems would be miscalibrated is 1×10^{-6} . I assumed that this would be the common cause failure contribution to the PWR RPS unavailability. As will be seen, this value is consistent with the common cause failure probabilities calculated for the other PWRs in this study.

Thus, by following the approach outlined above, the RPS unavailability that I used for the WASH-1400 PWR was 3.7×10^{-5} per demand which is the value shown in table 2.1, the Surry Unit I data sheet. Since the RPS unavailability is a direct multiplier in determining

the probabilities of the various ATWS sequences, the WASH-1400 PWR ATWS results had to be adjusted by the ratio of 3.7/3.6 to account for this revised PWR RPS unavailability. The specific application of this adjustment is explained at the appropriate points in the sections below.

For the BWR system, contributions from single and multiple channel failures, test and maintenance failures, and common cause failures were considered. Again, the common cause failure analysis assumed that the common cause failure was miscalibration of several instrument channels, and the failure probability was assumed to be the log-normal mean between complete dependence and complete independence of the calibration events. Since the BWR RPS does not have as many diverse sensors as the PWR RPS, the WASH-1400 authors included their common cause failure calculation in their calculation of the BWR RPS unavailability. I chose to accept the WASH-1400 BWR RPS unavailability value of 1.3×10^{-5} per demand, which is the value shown in table 2.2, the Peach Bottom Unit II data sheet.

2.4 Core Melt Probabilities

One of the criticisms WASH-1400 received was for its "probability smoothing" technique. In this technique, each dominant accident sequence is assigned to a specific "release category" based on the expected radioactive release (if any). However, the authors recognized that a given accident could be better or worse than they estimated, so that an accident sequence assigned to a specific release category could actually be in an adjacent release category. To account for this possibility when analyzing a given release category, the authors decided to include ten percent of the adjacent category probabilities in the category being analyzed. This approach is conservative when determining the *risk* associated with a given release category, but will artificially increase the total probabilities since the contribution from each sequence is increased from 100% to 110% or 120%, depending upon the number of adjacent release categories (one or two).

To avoid the effects of probability smoothing for internal initiators, I began with the

point estimate probabilities for the dominant accident sequences listed in table 5.2 (PWRs) and table 5.3 (BWRs) in WASH-1400. These probabilities were found through event tree analysis. I added these probabilities together across the core melt release categories to obtain a core melt probability for each internal initiator for each type of plant (PWR and BWR). The only adjustments made to the point estimates were to the ATWS sequences for the PWR (the TK sequences), which were multiplied by the ratio 3.7/3.6 to account for the adjustment made to the PWR RPS unavailability which was explained above in Section 2.3. The internal core melt values obtained using this method are listed here in tables 2.1 and 2.2, the Surry Unit I and Peach Bottom Unit II data sheets. As expected, the core melt probabilities were less than WASH-1400 found when using the probability smoothing technique.

The analysis of external initiators is another area where WASH-1400 received substantial criticism. Many critics felt that the treatment of external events was inadequate and superficial, and a quick look at WASH-1400 reveals that external events certainly were not given the same extensive analyses that the internal events were. The use of fault and event trees is conspicuously absent in the external event analyses. In contrast, today's PRAs exhibit much more in-depth analyses of external events, including the extensive use of fault and event trees. However, the objective of this paper is not to correct the shortcomings of the various reference reports, but rather to obtain and compare the information reported, and to take account of the various shortcomings in the comparative analysis in Chapter 9. In light of this objective, and in light of the fact that probability smoothing was not applied to the external event core melt probabilities, I decided to accept the external core melt probabilities reported on pages 67 to 70 in WASH-1400, and to account for possible inadequacies in the comparative analysis chapter. These external core melt probabilities are shown in tables 2.1 and 2.2, the Surry Unit I and Peach Bottom Unit II data sheets.

2.5 ATWS Core Melt Probabilities

WASH-1400 did not explicitly calculate an ATWS core melt probability, but such a value can be easily determined from the dominant accident sequence probabilities listed in table 5.2 (PWRs) and table 5.3 (BWRs) in WASH-1400. For PWRs, an ATWS sequence is indicated by any sequence beginning with the letters TK, since T represents a transient, and K represents failure of the scram system. By summing all of the point estimate probabilities for all of the core melt TK sequences found in table 5.2, I obtained a point estimate probability of 4.04×10^{-6} per R-yr for the PWR ATWS core melt probability. However, this probability needed to be revised due to the adjustment I made to the PWR RPS unavailability value. As mentioned in Section 2.3, the RPS unavailability is a direct multiplier in determining the ATWS sequence probabilities. So to account for the revised RPS unavailability, I needed to multiply the ATWS core melt probability of 4.04×10^{-6} per R-yr by the ratio 3.7/3.6, i.e. the ratio of the revised RPS unavailability to the unavailability used by the WASH-1400 authors. The ATWS core melt probability obtained by using this method is 4.15×10^{-6} per R-yr, and is the value listed here in table 2.1, the Surry Unit I data sheet.

For BWRs, an ATWS sequence is represented by any TC sequence since T represents a transient, and C represents failure of the scram system. By summing all of the core melt TC sequences listed in table 5.3 in WASH-1400, I obtained a point estimate probability of 1.01×10^{-5} per R-yr for the BWR ATWS core melt probability, which is the value listed here in table 2.2, the Peach Bottom Unit II data sheet.

2.6 Contribution of ATWS to Core Melt

From table 2.1, the Surry Unit I data sheet, we see that the ATWS contribution to the total core melt probability is 9.54 %. Also, the ATWS core melt probability of 4.15×10^{-6} per R-yr is the same order of magnitude as the NUREG-0460 safety objective of 1×10^{-6}

per R-yr. Both of these points indicate that ATWS is not a major contributor to core melt at Surry Unit I. Instead, a very strong argument could be made that LOCA events at Surry pose a larger threat than ATWS since LOCAs account for 69.43 % of the total core melt probability. In fact, small LOCAs are the largest single contributor to core melt at Surry, contributing 39.08% to the total core melt.

From table 2.2, the Peach Bottom Unit II data sheet, we see that the ATWS contribution to the total core melt is 40.32%. Also, the ATWS core melt probability of 1.01×10^{-5} per R-yr is an order of magnitude greater than the NUREG-0460 safety objective of 1×10^{-6} per R-yr. Both of these points indicate that ATWS events at Peach Bottom pose a serious risk.

Since WASH-1400 did not separate the various transient initiators, I could not determine which transient initiators, if any, dominated the ATWS core melt probability. As will be seen in the following chapters, identification of such initiators could show a pattern from plant to plant. It may also be useful to know if one or two initiators dominate the ATWS core melt probability, or if the probability is evenly distributed across the many transient initiators. The comparative analysis chapter will identify and discuss any such trends or patterns.

References:

- [1] *Reactor Safety Study*. WASH-1400. U.S. Nuclear Regulatory Commission. Washington, D.C.: U.S. Nuclear Regulatory Commission, 1975.

TABLE 2.1 - SURRY UNIT I DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per YR)	% OF TOTAL Pcm	Pcm/ATWS (per YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	1.00E-04	3.10E-06	7.13%	0E+00	0.00%	0.00%
MEDIUM LOCA	3.00E-04	6.10E-06	14.02%	0E+00	0.00%	0.00%
SMALL LOCA	1.00E-03	1.70E-05	39.08%	0E+00	0.00%	0.00%
INTERFACING SYST. LOCA	4.00E-06	4.00E-06	9.20%	0E+00	0.00%	0.00%
LOCA TOTALS	1.40E-03	3.02E-05	69.43%	0E+00	0.00%	0.00%
TRANSIENTS	10.00	1.31E-05	30.11%	4.15E-06	100.00%	9.54%
VESSEL RUPTURE	1.00E-07	1.00E-07	0.23%	0E+00	0.00%	0.00%
INTERNAL TOTALS	10.00	4.34E-05	99.77%	4.15E-06	100.00%	9.54%
SEISMIC EVENTS	4.90E-04	1.00E-07	0.23%	0E+00	0.00%	0.00%
TORNADOS	5.00E-06	0E+00	0.00%	0E+00	0.00%	0.00%
AIRCRAFT CRASHES	1.00E-07	0E+00	0.00%	0E+00	0.00%	0.00%
TURBINE MISSILES	1.00E-04	0E+00	0.00%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	5.95E-04	1.00E-07	0.23%	0E+00	0.00%	0.00%
GRAND TOTALS	10.00	4.35E-05	100.00%	4.15E-06	100.00%	9.54%
MEAN RPS UNRELIABILITY 3.70E-05/d						

CHAPTER 3 - BIG ROCK POINT

3.0 Introduction

The *Big Rock Point Probabilistic Risk Assessment* (PRA) was completed in 1981, six years after WASH-1400 was published. The Big Rock Point PRA uses the same basic approach as WASH-1400, but it does not employ the specific methods of WASH-1400 that were deemed inadequate. Specifically, one notices a more extensive treatment of external events as well as the absence of the probability smoothing technique.

The purpose of the Big Rock Point PRA was to demonstrate that the plant had a sufficiently low risk to the health and safety of the public so as to be exempt from many of the backfitting requirements that were being imposed on nuclear plants during the late nineteen seventies and early nineteen eighties - many of these as a result of the Three Mile Island II accident. Despite this objective, several modifications were identified that would help reduce the risk of an accident at the Big Rock Point plant. These modifications included use of an emergency condenser make-up system, increased post-incident system reliability, increased reactor depressurization system/core spray reliability, and environmental qualification of components. The PRA reported that these modifications would reduce the probability of core melt at Big Rock Point from 9.75×10^{-4} per R-yr to 4.22×10^{-4} per R-yr, a decrease of 56.7%. However, since the modifications had yet to be implemented when the PRA was published, I decided to base my results on the unmodified plant.

3.1 Plant Description

The Big Rock Point plant is located in Charlevoix, Michigan. It is a General Electric first generation BWR with a power rating of 240 MWt (72 MWe). It has a spherical steel containment structure. It should be noted that, being an older plant, the power rating of Big Rock Point is a full order of magnitude lower than the power rating of a "typical" BWR such

as the WASH-1400 BWR. The importance of this fact will be explained at the appropriate points in the sections below.

3.2 The Initiator Frequencies

The initiator frequencies developed by the PRA authors appear in the PRA on pages 25-27 in table 3.2. However, for the sake of consistency between the various PRAs studied as well as for the sake of consistency within the Big Rock Point PRA itself, I determined that several modifications to the list of initiators and their frequencies were required. The remainder of this section discusses these modifications.

Table 3.2 in the PRA lists ATWS and a stuck open safety valve as initiators in and of themselves. However, when I traced the results through the PRA event trees, I discovered that these special cases were in effect treated as branches of the event trees for several of the more common initiators. Specifically, the loss of off-site power, loss of main condenser, loss of instrument air, and MSIV (main steam isolation valve) closure event trees all branch to the stuck open safety valve event tree, while the turbine trip, loss of feedwater, loss of off-site power, spurious turbine by-pass, loss of main condenser, loss of instrument air, and MSIV closure event trees all branch to the ATWS event tree. The Big Rock Point PRA obtained the stuck open safety valve and ATWS initiating frequencies by combining the appropriate initiating frequencies listed above with the respective probabilities of a safety valve sticking open and the reactor protection system failing. The other PRAs studied also treated ATWS and stuck open safety valves as branches of other initiating event event trees, but, quite properly, did not list these events as separate initiators. Thus for the sake of consistency, and to accurately reflect the total initiating event frequency at Big Rock Point, I felt it was appropriate to delete the ATWS and stuck open safety valve initiators from the list of initiators at Big Rock Point and to treat those events as branches in the event trees of the remaining initiators.

The initiating frequency for tornados and aircraft crashes were not listed in table 3.2

of the PRA but were found in appendix VI of the PRA. The tornado frequency was found on page VI-149 of the PRA and was equal to 1.11×10^{-3} per year. The frequency of aircraft crashes at the Big Rock Point site was found on page VI-158 and was found to be the sum of two components: commercial flights and army B-52 flights that train in the area. The total frequency of aircraft crashes at Big Rock Point was calculated to be 8.53×10^{-7} per year.

The final list of initiators along with the frequencies used appears in table 3.1, the Big Rock Point data sheet.

3.3 The Reactor Protection System

The Big Rock Point PRA authors used the NUREG-0460 model with updated plant experience data to determine the Reactor Protection System (RPS) unavailability. This model assumes a constant failure rate for the RPS with monthly testing. The updated plant experience data assumed 996 plant-years of experience with only one failure, the Kahl reactor. The PRA did not include the Brown's Ferry incident in the data base due to "substantial design differences at BRP."

Using this approach, the total RPS unavailability was calculated to be 7.0×10^{-5} per demand (50% confidence level). As in NUREG-0460, this failure probability was assumed to be equally divided between the electrical and mechanical portions of the RPS. The PRA authors maintained that electrical failures could be easily remedied by a manual scram since the electrical failure would almost certainly be in the trip relays or logic circuits, and the manual scram button in the control room by-passes these circuits. However, the PRA authors felt that mechanical failures could not be easily remedied, so the value used in the event tree(s) for the RPS unavailability was the 3.5×10^{-5} per demand due to mechanical failures.

Although the PRA's treatment of the electrical failures is somewhat dubious, I decided to accept the value of 3.5×10^{-5} per demand as the RPS unavailability for Big Rock Point, which is the value shown on the Big Rock Point data sheet. The implications of effectively

ignoring electrical failures will be discussed in section 3.6 below.

3.4 Core Melt Probabilities

The PRA values for core melt probability by initiator are listed on pages 4 and 64 of the PRA in tables 1.1 and 4.1 respectively (table 1.1 is identical to table 4.1). As with the initiator frequencies, both ATWS and a stuck open safety valve are listed as initiators in and of themselves. Consequently, the core melt probabilities assigned to these "initiators" should actually be assigned to the initiators that branch to the ATWS and stuck open safety valve event trees. Thus I had to trace through the event trees to determine the appropriate reallocation of these core melt probabilities. The rest of this section discusses this process.

Table 4.3 on page 76 of the PRA lists all of the dominant accident sequences along with the core melt probability for each sequence. From this table, it is seen that there are eight dominant sequences involving a stuck open safety valve (represented by the symbol K). There are four different initiators involved in these eight sequences: loss of off-site power, loss of main condenser, loss of instrument air, and spurious MSIV closure. Each initiator is responsible for two stuck open safety valve sequences. Table 3.2 lists each of these eight sequences by initiator along with their corresponding core melt probabilities. The right hand column of table 3.2 totals the core melt probabilities by initiator.

Also from table 4.3 in the PRA, we find that there are 13 ATWS sequences listed. The initiators used in the ATWS analysis were slightly different than those used in the overall analysis. Specifically, the turbine trip initiator was divided up into four separate initiators: turbine trip IPR/PR failure, loss of one feed pump, load rejection, and miscellaneous scrams (represented by T1, T4, T5 and T8). Also, three of the "overall" initiators (loss of condenser, loss of instrument air, and spurious MSIV closure) were combined and analyzed as a single ATWS initiator: loss of condenser (represented by T6). The three other initiators used in the ATWS analysis were spurious turbine by-pass (T2),

loss of feedwater (T3), and loss of off-site power (T7). The 13 ATWS sequences listed in table 4.3 of the PRA are listed here in table 3.3 by initiator along with their corresponding core melt probabilities. The right hand column of table 3.3 totals the core melt probabilities by initiator.

In order to obtain the core melt probabilities listed in table 3.1, the Big Rock Point data sheet, I began with the values listed in table 1.1 in the PRA. This table in the PRA list a single core melt value for LOCAs rather than listing separate values for small, medium and large LOCAs. To determine these separate values I turned to table I.2 in Appendix I of the PRA (p. I-160 and I-161 in the PRA). There were two sequences for small LOCAs (S1), two sequences for medium LOCAs (S2) and one sequence for large LOCAs (S7). The small LOCA core melt probabilities were 3.7×10^{-5} per R-yr and 4×10^{-6} per R-yr for a total small LOCA core melt probability of 4.1×10^{-5} per R-yr, which is the value listed in table 3.1 the Big Rock Point data sheet. Similarly for medium LOCAs, the sequence probabilities are 3.7×10^{-6} per R-yr and 4×10^{-7} per R-yr for a total probability of 4.1×10^{-6} per R-yr, which is the value listed in table 3.1. The probability for the single large LOCA sequence is 3.7×10^{-7} per R-yr which is the value listed in table 3.1.

The core melt probability for interfacing system LOCAs was taken directly from table 1.1 in the PRA.

The core melt probabilities for steam line breaks inside and outside containment were taken directly from table 1.1 in the PRA.

The core melt probability for loss of main feedwater is the sum of the probability listed in table 1.1 of the PRA (4.3×10^{-7} per R-yr) and the portion of the ATWS core melt probability that is actually due to loss of main feedwater events. From table 3.3 we see that this second probability is 2.4×10^{-6} per R-yr, which when added to the first probability yields a probability of 2.8×10^{-6} per R-yr which is the core melt probability

listed in table 3.1 for loss of main feedwater at Big Rock Point.

The core melt probability due to spurious MSIV closure is the sum of the value listed in table 1.1 of the PRA (3.2×10^{-6} per R-yr), the portion of the stuck open safety valve core melt probability that is actually due to MSIV closure and the portion of the ATWS core melt probability that is due to MSIV closure. From table 3.2 we see that the portion of the stuck open safety valve probability that is due to spurious MSIV closure is 1.6×10^{-5} per R-yr. Determining the portion of the ATWS core melt probability that should be assigned to MSIV closure is a little more involved. From table 3.3 we see that three initiators (loss of main condenser, loss of instrument air and MSIV closure) were treated as a single initiator (loss of condenser) in the ATWS analysis. Since each of these three initiators has an initiating frequency of 6.0×10^{-2} per year, and the ATWS initiator (loss of main condenser) has an initiating frequency of 1.8×10^{-1} per year, I assumed that the ATWS core melt probability for each of the three initiators was one-third of the value listed in table 3.3 for the loss of condenser sequence (i.e. 2.1×10^{-6} per R-yr). Thus the core melt probability for spurious MSIV closure was calculated to be the sum of 3.2×10^{-6} per R-yr, 1.6×10^{-5} per R-yr and 2.1×10^{-6} per R-yr which is 2.1×10^{-5} per R-yr and is the value shown in table 3.1, the Big Rock Point data sheet.

The core melt probability due to loss of the main condenser was found in the same fashion as that for MSIV closure. The value from table 1.1 of the PRA was 4.0×10^{-6} per R-yr, the value from table 3.2 (stuck open safety valve) was 1.6×10^{-5} per R-yr and the value from table 3.3 (ATWS) was one-third of the loss of condenser value or 2.1×10^{-6} per R-yr. Thus the core melt probability due to loss of condenser is 2.2×10^{-5} per R-yr, which is the value shown in table 3.1, the Big Rock Point data sheet.

The core melt probability for loss of instrument air was determined in the same fashion as that for MSIV closure and loss of main condenser. The result was 1.7×10^{-4} per

per R-yr, which is the value shown in table 3.1, the Big Rock Point data sheet.

The core melt probability for turbine trips at Big Rock Point was found by summing the value presented in table 1.1 of the PRA (9.0×10^{-7} per R-yr) with the value listed in table 3.3 here (1.02×10^{-5} per R-yr) which represents the core melt probability from ATWS sequences initiated by turbine trips. The result is 1.11×10^{-5} per R-yr which is the value listed for turbine trips in table 3.1, the Big Rock Point data sheet.

The core melt probability for spurious turbine by-pass at Big Rock Point was found by summing the value presented in table 1.1 of the PRA (7.1×10^{-5} per R-yr) with the value listed in table 3.3 here (3.2×10^{-6} per R-yr) which represents the core melt probability from ATWS sequences initiated by spurious turbine by-pass. The result is 7.4×10^{-5} per R-yr which is the value listed for spurious turbine by-pass in table 3.1.

The core melt probability for loss of off-site power at Big Rock Point was found by summing the value presented in table 1.1 of the PRA (4.7×10^{-5} per R-yr) with the values listed in table 3.2 (1.23×10^{-4} per R-yr) and table 3.3 (4.6×10^{-6} per R-yr) which represent, respectfully, the core melt probabilities from stuck open safety valve and ATWS sequences initiated by loss of off-site power. The result is 1.75×10^{-4} per R-yr which is the value listed for loss of off-site power in table 3.1.

The core melt probability for spurious opening of the RDS isolation valve was taken directly from table 1.1 of the PRA. This value is 1.7×10^{-5} per R-yr, which is the value listed in table 3.1, the Big Rock Point data sheet.

The core melt probability due to seismic events at Big Rock Point was found on page VI-118 of the PRA. This value is 1.2×10^{-7} per R-yr and is listed in table 3.1, the Big Rock Point data sheet.

The core melt probability due to fires can be determined from table VI.10 on page VI-53 of the PRA. This table shows the sequence probabilities for the fire event tree. By

summing all of the probabilities, I obtained a total core melt probability due to fires of 2.3×10^{-4} per R-yr, which is the value shown in table 3.1.

The probability of core melt due to tornados was deemed all but impossible in section VI.9 of the PRA, and was therefore assigned a probability of 0.0 in table 3.1, the Big Rock Point data sheet.

The probability of core melt due to aircraft crashes was calculated on page VI-159 of the PRA. The value calculated was 8.82×10^{-8} per R-yr and assumed a 0.1 conditional probability of core melt given a direct hit. This value is listed in table 3.1.

The core melt probability due to high energy line breaks was taken directly from table 1.1 of the PRA. This value is 1.5×10^{-6} per R-yr and is the value listed in table 3.1, the Big Rock Point data sheet.

3.5 ATWS Core Melt Probabilities

The Big Rock Point PRA was one of the few PRAs that explicitly calculated ATWS core melt probabilities. As mentioned above, the PRA authors even went as far as to list ATWS as an initiator in and of itself. Also as mentioned in section 3.4 above, the dominant ATWS sequences from table 4.3 in the PRA are listed here in table 3.3. The ATWS core melt frequencies for turbine trips, spurious turbine by-pass, loss of feedwater and loss of off-site power were taken directly from table 3.3. The remaining initiator listed in table 3.3 (loss of condenser) is actually three initiators in one: loss of main condenser, loss of instrument air and spurious MSIV closure. Since each of these three initiators has a frequency of 0.06 per year, and the loss of condenser initiator used in the ATWS analysis had a frequency of 0.18 per year, I assumed that the core melt probability assigned to loss of condenser in table 3.3 could be divided evenly among these three initiators. Therefore, the ATWS core melt probabilities listed in table 3.1 (the Big Rock Point data sheet) for loss of main condenser, loss of instrument air and spurious MSIV closure are all equal to 2.1×10^{-6} per R-yr, which is one-third of the value from table 3.3 for loss of condenser. No

other initiators were found to contribute to ATWS events by the Big Rock Point PRA authors.

3.6 Contribution of ATWS to Core Melt

From table 3.1, the Big Rock Point data sheet, we see that the ATWS contribution to the total core melt probability is 2.76%. This suggests that ATWS events at Big Rock Point do not pose an undue risk to the health and safety of the public. However, the ATWS core melt probability of 2.68×10^{-5} per R-yr is an order of magnitude larger than the NUREG-0460 safety objective of 1×10^{-6} per R-yr, which suggests that ATWS events at Big Rock Point may pose a health and safety risk to the public. Also, the total core melt probability of 9.71×10^{-4} per R-yr at Big Rock Point is a full order of magnitude larger than the WASH-1400 BWR total core melt probability, suggesting that core melt in general at Big Rock Point poses an excessive risk to the public. To resolve these differences, we turn to the fact that the power rating of Big Rock Point is a full order of magnitude lower than that for the WASH-1400 BWR, or any other "typical" BWR. Consequently, the radioactive inventory would be approximately one order of magnitude lower than a "typical" plant, and the consequences of any given accident would also be an order of magnitude lower. It is therefore reasonable to assume that even though the core melt probability is an order of magnitude greater than the WASH-1400 BWR, and even though the ATWS core melt probability is an order of magnitude greater than the NUREG-0460 safety objective, the expected consequences at Big Rock Point would not be substantially different due to the much lower radioactive inventory. Thus, if we assume that the total risk from the WASH-1400 BWR is acceptable, then we can say that the total risk from Big Rock Point is acceptable, and since ATWS contributes only 2.76% of the total core melt probability, we can make a very strong argument that ATWS does not represent an undue risk to the health and safety of the public at Big Rock Point.

We can also see from table 3.1 that the major contributors to the ATWS core melt

probability are turbine trip(38.06%), loss of off-site power(17.16%) and spurious turbine by-pass(12.31%). However, only loss of off-site power and spurious turbine by-pass contribute significantly to the total core melt probability (18.02% and 7.62% respectively), suggesting that modifications to decrease the ATWS core melt probability would have a limited effect on the total core melt probability.

One final note, as mentioned in section 3.3, the PRA authors effectively ignored electrical failures of the RPS. If we assume that these failures are not easily remedied, the RPS unavailability would be double that used in the PRA. Since the RPS unavailability is a direct multiplier in determining the ATWS core melt probability, the ATWS core melt probability would double to 5.36×10^{-5} per R-yr. The total core melt probability would increase by 2.68×10^{-5} per R-yr to a total of 9.98×10^{-4} per R-yr. The new contribution of ATWS events to total core melt would then be 5.37%. Although this is almost double the previous value of 2.76%, the percent contributions of each initiator to the ATWS core melt would not change at all, and the percent contributions to the total core melt would change minimally, having no significant effect on the discussion above.

TABLE 3.1 - BIG ROCK POINT DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per R-YR)	% OF TOTAL Pcm	Pcm/ATWS (per R-YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	1.00E-05	3.70E-07	0.04%	0E+00	0.00%	0.00%
MEDIUM LOCA	1.00E-04	4.10E-06	0.42%	0E+00	0.00%	0.00%
SMALL LOCA	1.00E-03	4.10E-05	4.22%	0E+00	0.00%	0.00%
INTERFACING SYST. LOCA	1.98E-03	9.10E-05	9.37%	0E+00	0.00%	0.00%
LOCA TOTALS	3.09E-03	1.36E-04	14.05%	0E+00	0.00%	0.00%
STM. LINE BREAK INSIDE CONTAINMENT	1.11E-03	1.10E-04	11.33%	0E+00	0.00%	0.00%
STM. LINE BREAK OUTSIDE CONTAINMENT	1.11E-04	0E+00	0.00%	0E+00	0.00%	0.00%
LOSS OF MAIN FEEDWATER	1.60E-01	2.80E-06	0.29%	2.40E-06	8.96%	0.25%
SPURIOUS MSIV CLOSURE	6.00E-02	2.10E-05	2.16%	2.10E-06	7.84%	0.22%
LOSS OF MAIN CONDENSER	6.00E-02	2.20E-05	2.27%	2.10E-06	7.84%	0.22%
LOSS OF INSTRUMENT AIR	6.00E-02	1.70E-04	17.51%	2.10E-06	7.84%	0.22%
TURBINE TRIP	1.39E+00	1.11E-05	1.14%	1.02E-05	38.06%	1.05%
SPURIOUS TURB. BY-PASS	1.00E-01	7.40E-05	7.62%	3.30E-06	12.31%	0.34%
LOSS OF OFF-SITE POWER	1.30E-01	1.75E-04	18.02%	4.60E-06	17.16%	0.47%
SPURIOUS OPENING OF THE RDS ISOLATION VALVE	1.20E-03	1.70E-05	1.75%	0E+00	0.00%	0.00%
TRANSIENT TOTALS	1.96	6.03E-04	62.09%	2.68E-05	100.00%	2.76%
INTERNAL TOTALS	1.97	7.39E-04	76.14%	2.68E-05	100.00%	2.76%
SEISMIC EVENTS	1.11E-03	1.20E-07	0.01%	0E+00	0.00%	0.00%
FIRES	6.10E-03	2.30E-04	23.69%	0E+00	0.00%	0.00%
TORNADOS	1.11E-03	0E+00	0.00%	0E+00	0.00%	0.00%
AIRCRAFT CRASHES	8.53E-07	8.82E-08	0.01%	0E+00	0.00%	0.00%
HIGH ENERGY LINE BREAKS	4.19E-06	1.50E-06	0.15%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	8.33E-03	2.32E-04	23.86%	0E+00	0.00%	0.00%
GRAND TOTALS	1.97	9.71E-04	100.00%	2.68E-05	100.00%	2.76%
MEAN RPS UNRELIABILITY 3.50E-05/d						

TABLE 3.2 - STUCK OPEN SAFETY VALVE SEQUENCES AT BIG ROCK POINT

SEQUENCE	INITIATOR	CORE MELT PROBABILITY (per R-Yr)	TOTAL C.M. PROBABILITY BY INITIATOR (per R-Yr)
PEvFsKC	Loss of off-site power	2.6 E-5	1.23 E-4
PEmFsKC	Loss of off-site power	9.7 E-5	
MEvNKC	Loss of main condenser	1.2 E-5	1.6 E-5
MEmNKC	Loss of main condenser	3.7 E-6	
UEvUKC	Loss of instrument air	1.2 E -5	1.3 E-4
UEmUKC	Loss of instrument air	1.2 E-4	
WEvKC	Spurious MSIV closure	1.2 E-5	1.6 E-5
WEmKC	Spurious MSIV closure	3.7 E-6	

TABLE 3.3 - ATWS SEQUENCES AT BIG ROCK POINT

SEQUENCE	INITIATOR	CORE MELT PROBABILITY (per R-Yr)	TOTAL C.M. PROBABILITY BY INITIATOR (per R-Yr)
T1AYfOLr	Turbine trip	1.4 E-6	
T1ABoOLr	Turbine trip	4.2 E-7	
T1ABoLr	Turbine trip	8.8 E-7	
T4ABoLr	Turbine trip	2.9 E-7	
T5AYfLr	Turbine trip	4.1 E-6	
T5ABoLr	Turbine trip	2.9 E-6	
T8ALr	Turbine trip	1.7 E-7	1.02 E-5
T2ABoLr	Spurious turbine by-pass	3.2 E-6	3.2 E-6
T3AYfLr	Loss of main feedwater	1.2 E-6	
T3AYfOLr	Loss of main feedwater	3.7 E-7	
T3ABoLr	Loss of main feedwater	7.8 E-7	2.4 E-6
T6ABoLr	Loss of condenser*	6.3 E-6	6.3 E-6
T7ABoLr	Loss of off-site power	4.6 E-6	4.6 E-6

*Three initiators in one: Loss of main cond., Loss of instr. air and Spur. MSIV closure

CHAPTER 4 - LIMERICK UNIT I

4.0 Introduction

The *Limerick Unit I Probabilistic Risk Assessment* (PRA) was published in 1982 in the aftermath of the Three Mile Island II accident. The purpose of the PRA was to establish the risk to the health and safety of the public to help aid in the licensing process of the Limerick Generating Station. The Limerick PRA uses the same basic approach as WASH-1400 but does not employ the methods of WASH-1400 that were deemed inadequate. However, the treatment of external events in the Limerick PRA is conspicuously absent. To correct this major oversight, the *Limerick Generating Station Severe Accident Risk Assessment* (SARA) was published in 1983 which included updated results for the PRA as well as in-depth analyses of external initiators. The information presented here was taken from both of these studies.

4.1 Plant Description

The Limerick Unit I Generating Station is located in Limerick, Pennsylvania. It is a General Electric BWR/4 with a power rating of 3293 MWt (1055 MWe). It has a Mark II type containment structure with an inert nitrogen atmosphere.

4.2 The Initiator Frequencies

Table 3.2.1 in the Limerick Generating Station PRA lists the transient initiator frequencies used in the PRA, while table 3.2.2 lists the initiating frequencies for LOCAs. These frequencies are based on generic plant data. However, table 3.2.1 of the PRA notes that the initiating frequency used for loss of off-site power was not the value listed, but a value determined from site specific data for Limerick. From the event trees on pages 3-35 and 3-8 of the PRA, the initiator frequency used for loss of off-site power appears to be 5.3×10^{-2} per R-yr.

Also, on page A-10 of Appendix A in the PRA, it is noted that the loss of feedwater initiator was later combined and analyzed with the MSIV events. Although the PRA authors felt this was advantageous, I decided that for my purposes it was best to keep these initiators separate.

Thus the internal initiator frequencies shown in the Limerick data sheet, table 4.1, are taken directly from tables 3.2.1 and 3.2.2 of the PRA with the exception of the loss of off-site power initiator which was obtained from the event trees for the reasons mentioned above.

The initiating frequencies for external events had to be determined from the *Limerick Generating Station Severe Accident Risk Assessment* (SARA). Unfortunately, the SARA authors did not explicitly calculate initiating frequencies for each external initiator. In the paragraphs below I explain how I obtained initiating frequencies for external events from the information presented by the SARA authors.

For seismic events I turned to Appendix A of the SARA. Table 5 in Appendix A lists annual frequencies of exceedence for specific peak ground accelerations for several seismic zones considered to affect the Limerick site. Since these seismic zones came from several different models, the SARA authors assigned each zone a subjective weight. The lowest acceleration listed is 0.1g which is equivalent to a small earth tremor. Thus I felt it a reasonable approach to take the seismic initiating frequency to be the sum of the weighted probabilities of exceeding 0.1g peak ground acceleration for the six weighted seismic zones listed in table 5 of Appendix A of the SARA. I calculated this value to be 1.67×10^{-3} per year, which is the value listed in table 4.1, the Limerick data sheet.

The frequency of fires at Limerick can be determined from table 4.2 on pages 4-33 to 4-36 of the SARA. The fire initiation frequencies used were based on data related to the type of fire (i.e. a cable fire, a cabinet fire, or a fire with transient fuels) as opposed to data on the area of the plant where the fire was, as is common in most other PRAs. By

adding the frequencies listed in table 4.2 of the SARA, I obtained an initiating frequency for fires of 5.28×10^{-2} per year.

The frequency of tornados was determined from Section 6 of the SARA. Page 6-6 lists the frequency of a tornado of any size striking a point as 1.13×10^{-4} per year, while page 6-9 gives the frequency of a tornado at class F1 or above as 2.3×10^{-4} per year. Even though both of these numbers could be considered acceptable, I chose to use the latter value since it was slightly higher and therefore slightly more conservative.

In Section 7 of the SARA, the authors calculated the probability of an aircraft crashing at the Limerick site. They obtained a value of 9.4×10^{-8} per year which I found to be reasonable and is the value I used in this report.

The frequency of floods was based on industry experience and was reported in Appendix H of the SARA. Table H-2 lists these flood frequencies from industry experience. Unfortunately, the authors noted that some flood events contributed to more than one plant location since the location of some floods could not be determined exactly from the event description. Since I had no way of knowing how many flood events had contributed to flood frequencies in more than one plant location, I was forced to accept the overall (all plant locations) BWR flood frequency listed in table H-2 as the flood frequency at Limerick. This value, 7.0×10^{-2} per year, is the value shown in table 4.1, the Limerick data sheet.

On page 8-1 of the SARA, the SARA authors estimated the frequency of turbine missiles as 1×10^{-5} per year. Since there has been very little research in the area of turbine missiles, I assumed this value to be acceptable even though it is an order of magnitude lower than the values estimated in the other PRAs used in this study.

4.3 The Reactor Protection System

The Limerick PRA authors chose to use the NUREG-0460 RPS unavailability value of 3.0×10^{-5} per demand as the RPS unavailability value for Limerick since they felt that the

RPS at Limerick was not significantly different in design from other RPS designs. This approach seemed reasonable to me, so I accepted the RPS unavailability value of 3.0×10^{-5} per demand, which is the value listed in table 4.1, the Limerick data sheet.

However, the authors decided to assign two-thirds of this value to electrical failures and one-third of the value to mechanical failures, unlike NUREG-0460 which assigned 50% of the value to electrical failures and the other 50% to mechanical failures. The justification for this different approach was that two-thirds of the experienced precursors (Kahl and Monticello) had been electrical failures while one-third of the precursors (Brown's Ferry 3) had been mechanical failures.

Although it seems questionable for the authors to make such an assumption based upon the evidence of only three precursors, many experts are beginning to feel that electrical failures will dominate the RPS unavailability. We must also remember that the 50-50 split assigned in NUREG-0460 was an arbitrary assumption. So taking these points into account, I felt that the PRA authors' approach was acceptable and did not modify the event trees which displayed separate headings for electrical and mechanical RPS failures.

4.4 Core Melt Probabilities

The core melt probabilities presented on page 3-108 of the Limerick PRA are grouped into six types of accidents: loss of off-site power, loss of main feedwater/MSIV closures followed by loss of coolant make-up, other loss of make-up events, ATWS events, loss of containment cooling events and LOCA events. On page 12-27 of the PRA, table 12-5 presents core melt probabilities by class, which is determined by the core melt timing and containment failure timing, i.e. fast or slow core melt with containment remaining intact or failing before core melt.

Unfortunately, neither of these presentations were adequate to assign specific core melt probabilities for all of the initiators that I had identified. I therefore turned to the event trees presented in the PRA which provided enough information to calculate the core

melt probabilities for each initiator. This process was complicated by the fact that several initiators (MSIV closure and loss of main feedwater) were analyzed as a single initiator: MSIV closure. Also, several of the turbine trip ATWS sequences were transferred to the MSIV ATWS event tree, and both the small and medium LOCA ATWS sequences were analyzed in the Inadvertent Open Relief Valve (IORV) ATWS event tree. The adjustments made to ensure that the core melt values for these particular sequences were assigned to the appropriate initiators are explained at the appropriate points below.

The large LOCA event tree is on page 3-53 of the PRA. There are four core melt sequences listed which give a total core melt probability of 7.1×10^{-8} per R-yr. This is the value shown here in table 4.1, the Limerick data sheet.

The medium LOCA event tree is on page 3-57 of the PRA and shows only three sequences. The value of 1.7×10^{-8} per R-yr shown in table 4.1, the Limerick Data Sheet, is the sum of the core melt probabilities listed for these three sequences as well as the ATWS core melt frequency for medium LOCAs, which was analyzed in the IORV ATWS event tree (page 3-83 in the PRA). The calculation of the medium LOCA ATWS core melt frequency is explained in Section 4.5 below.

The core melt probability of 2.8×10^{-8} per R-yr due to small LOCAs was calculated from the small LOCA event tree (which is shown on page 3-60 of the PRA) along with the core melt probability for small LOCA ATWS events which was obtained from the IORV ATWS event tree (on page 3-83 of the PRA). The calculation of the small LOCA ATWS core melt probability is discussed in Section 4.5 below.

The core melt probability due to an Inadvertent Open Relief Valve (IORV) was calculated from the IORV event tree on page 3-48 of the PRA and the IORV ATWS event tree on page 3-83 of the PRA. However, if we add the probabilities for the sequences listed, we find that the calculated value is 1.06×10^{-6} per R-yr, not 9.1×10^{-7} per R-yr as shown on the Limerick data sheet. This difference is due to the fact that the PRA authors treated

the small and medium LOCA ATWS sequences as IORV sequences.

As I reported earlier, the IORV initiator frequency used in the PRA is 0.06 per year. However, the initiator frequency used in the IORV event tree is 0.07 per year, which is the sum of the IORV initiator frequency and both the small and medium LOCA initiating frequencies (0.06, 1.0×10^{-2} and 2.0×10^{-3} per R-yr, respectively) to two significant figures.

Thus, to find the "true" IORV core melt frequency, I needed to adjust the initiator frequency used in the event tree from 0.07 per year to 0.06 per year. Fortunately, the initiator frequency is a direct multiplier so that we can simply multiply the core melt probability calculated with the 0.07 per year frequency (1.06×10^{-6} per R-yr) by the ratio 0.06/0.07 to obtain the corrected IORV core melt frequency. The result is the 9.1×10^{-7} per R-yr value shown in table 4.1, the Limerick data sheet.

The core melt probability for MSIV closure events was calculated from the MSIV event tree on page 3-33 of the PRA and the MSIV ATWS event tree on pages 3-75 and 3-76 of the PRA. However, the initiator frequency used in the regular event tree includes loss of main feedwater events (0.70 per year) as well as MSIV closure events (1.08 per year), while the initiator frequency for the ATWS event tree includes loss of feedwater (0.70 per year), MSIV closure (1.08 per year) and some turbine trip ATWS events (0.42 per year). Thus, to obtain the "true" MSIV core melt probability, the values calculated from these event trees need to be pro-rated from the initiator frequencies used to the 1.08 per year frequency for MSIV. For the MSIV event tree on page 3-33, this ratio is 1.08/1.78, while for the MSIV ATWS event tree this value is 1.08/2.2. Thus, the MSIV closure core melt probability that I calculated in this fashion is 2.79×10^{-6} per R-yr which is the value shown in table 4.1, the Limerick data sheet.

As mentioned above, the core melt probability for loss of main feedwater events was calculated in the MSIV event trees. We therefore must pro-rate the values calculated in

the MSIV event trees to determine the "true" loss of main feedwater core melt probability. Since the loss of main feedwater initiating frequency was determined to be 0.70 per year, the ratio used for the regular MSIV event tree on page 3-33 of the PRA was 0.70/1.78 while the ratio for the MSIV ATWS event tree on pages 3-75 and 3-76 of the PRA was 0.70/2.2. Thus by using this approach, the value for the loss of main feedwater core melt probability was found to be 1.81×10^{-6} per R-yr which is the value shown in table 4.1, the Limerick data sheet.

The core melt probability for turbine trip sequences at Limerick was determined from the turbine trip event tree (page 3-17 in the PRA), the turbine trip ATWS event tree (pages 3-67 and 3-69 in the PRA) and the MSIV ATWS event tree (pages 3-75 and 3-76 in the PRA). The total core melt probability for the turbine trip sequences is the sum of the probabilities calculated in the turbine trip event trees (regular and ATWS) as well as the appropriate portion of the total value calculated in the MSIV ATWS event tree. As mentioned above, the turbine trip ATWS sequences that were analyzed in the MSIV ATWS event tree made up 0.42 per year of the initiator frequency of 2.2 per year used in that event tree. Thus $0.42/2.2$ (i.e. 19%) of the core melt probability calculated in the MSIV ATWS event tree should actually be assigned to the turbine trip ATWS core melt probability. By following this procedure, I determined the total turbine trip core melt probability to be 1.76×10^{-6} per R-yr, which is the value shown in table 4.1, the Limerick data sheet.

The loss of off-site power core melt probability was determined directly from the loss of off-site power event trees on pages 3-35 and 3-81 of the PRA. The value calculated was 6.70×10^{-6} per R-yr which is the value shown in table 4.1, the Limerick data sheet.

The core melt probability for a manual reactor trip was also determined directly from that sequence's event tree which is on page 3-31 of the PRA. The value obtained was 2.28×10^{-7} per R-yr as is shown in table 4.1, the Limerick data sheet.

The core melt probability for seismic events at Limerick was calculated from the seismic event trees (Figures 3-7, 3-8 and 3-9) in the *Limerick Severe Accident Risk Assessment* (SARA). The total core melt probability for seismic events was calculated to be 5.86×10^{-6} per R-yr, which is the value shown in table 4.1, the Limerick data sheet.

The core melt probability due to fires was determined in the Limerick SARA. Table 4-6 in the SARA lists the results of this analysis. By summing the values in this table, I obtained a core melt probability of 2.24×10^{-5} per R-yr which is the value shown in table 4.1 in this report.

The SARA analysis of the core melt probability due to tornados was not quite as straightforward. A specific value was not determined, rather some estimates were made and tornados were judged to have a negligible impact on the total core melt probability at Limerick. So in order to obtain a value, I began with the SARA estimates and included some estimates of my own. Figure 6.5 in the SARA lists the probability of a core melt due to a tornado-generated missile as 8.48×10^{-9} per R-yr. Page 6-11 in the SARA gives the core melt probability due to tornado caused loss of off-site power as 2.16×10^{-8} per R-yr. Page 6-12 of the SARA estimates the frequency of a tornado large enough to cause structural damage and core melt directly (i.e. by wind force alone, not by causing loss of off-site power or by generating missiles) as 10^{-7} per year, and estimates that the frequency of core melt due to such a scenario would therefore be much less than 10^{-7} per R-yr. To obtain a value, I assumed that the conditional probability of core melt given such a tornado would be 0.10, so that the probability of core melt due to a large tornado alone would be 10^{-8} per R-yr. Therefore, the total core melt probability due to tornados would be the sum of the core melt frequencies due to tornado generated missiles (8.48×10^{-9} per R-yr), tornado caused loss of off-site power (2.16×10^{-8} per R-yr) and large tornados (10^{-8} per R-yr), or 4.0×10^{-8} per R-yr which is the value shown in table 4.1, the

Limerick data sheet.

The Limerick SARA estimated that the core melt probability due to aircraft crashes would be negligible. Again, I felt that it was desirable to calculate a value for comparison purposes. The PRA found the initiating frequency for aircraft crashes at Limerick to be 9.40×10^{-8} per year. I assumed that the conditional probability of a core melt given an aircraft crash at Limerick was 0.10. Using this assumption I calculated the probability of a core melt from an aircraft crash at Limerick to be 9.40×10^{-9} per R-yr, which is the value shown in table 4.1, the Limerick data sheet.

The results of the SARA flood analysis found that the core melt probability due to flooding was 5.7×10^{-7} per R-yr. Page 5-35 of the SARA states that table 5.5 in the SARA summarizes the flooding core melt probability data, however I did not find this to be the case. I therefore attempted to trace their results using the data in tables 5.1 through 5.4 in the SARA. However, the results I got with this data were more than an order of magnitude greater than the SARA authors reported. Upon further investigation I discovered that the reason for this difference was that the data in tables 5.1 through 5.4 were a "level two analysis" which the PRA authors used to eliminate lesser contributing flooding sequences. After eliminating these sequences, a more in-depth "level three" analysis was performed on the remaining sequences which were found to contribute significantly less than they appeared to contribute in the level two analysis. Since the SARA authors apparently did not include the data for the level three analysis, I could not trace the results and therefore decided to accept their value of 5.7×10^{-7} per R-yr as the core melt probability due to flooding. This is the value shown in table 4.1, the Limerick data sheet.

Finally, the core melt probability due to turbine missiles was considered negligible by the SARA authors. Again they did not explicitly calculate a value, but on pages 8-2 and 8-3 of the SARA they estimated the probability of a turbine missile hitting a piece of safety

equipment to be 10^{-8} per year. Once again I assumed that the conditional probability of core melt given this event was 0.10, which yielded a core melt probability for turbine missiles of 10^{-9} per R-yr. This value is shown in table 4.1, the Limerick data sheet.

4.5 ATWS Core Melt Probabilities

As mentioned in Section 4.4 above, the medium LOCA ATWS core melt sequence was transferred from the medium LOCA event tree (page 3-57 in the PRA) to the Inadvertent Open Relief Valve (IORV) ATWS event tree (pages 3-82 and 3-83 of the PRA). As also mentioned above, the initiator frequency used in the IORV ATWS event tree is 0.07 per year, which is the sum of the initiator frequencies for an IORV (0.06 per year), a small LOCA (0.01 per year) and a medium LOCA (0.002 per year), to two significant figures. Thus to determine the medium LOCA ATWS contribution to core melt, we need to substitute the medium LOCA initiating frequency for the total initiating frequency used in the IORV ATWS event tree, i.e. we should multiply the probabilities calculated in the IORV ATWS event tree by the ratio $0.002/0.07$. By following this procedure, I obtained a medium LOCA ATWS core melt probability of 2.4×10^{-9} per R-yr, which is the value shown in table 4.1, the Limerick data sheet.

The small LOCA ATWS sequence was also analyzed in the IORV ATWS event tree, and therefore a procedure analogous to calculating the medium LOCA ATWS core melt probability was used to determine the small LOCA ATWS core melt probability. In this case the multiplying ratio was $0.01/0.07$, and the small LOCA ATWS core melt probability was found to be 1.2×10^{-8} per R-yr, as shown in table 4.1.

Since the IORV ATWS event tree also included the analysis of the above LOCA ATWS sequences, a multiplying factor was again needed to determine the "true" IORV ATWS core melt probability. In this case the factor was $0.06/0.07$, and the value calculated for the IORV ATWS core melt probability was 7.3×10^{-8} per R-yr, which is the value shown in table 4.1, the Limerick data sheet.

MSIV closure, loss of main feedwater and several turbine trip ATWS events were all analyzed in the MSIV ATWS event tree. The initiating frequencies for each of these three components comprise the 2.2 per year frequency shown on the MSIV ATWS event tree on pages 3-75 and 3-76 of the PRA. In order to separate each component out, I had to pro-rate the results calculated in the event tree. For the MSIV ATWS events, the initiating frequency was 1.08 per year, so that $1.08/2.2$ or 49.1% of the sequences analyzed in the MSIV ATWS event tree are attributable to the MSIV initiator. Similarly, the initiating frequency for loss of main feedwater is 0.70 per year, so that $0.70/2.2$ or 31.8% of the sequences are attributable to the loss of main feedwater initiator. By following this procedure, I obtained an ATWS core melt probability of 3.14×10^{-7} per R-yr for MSIV closure ATWS events and a loss of main feedwater ATWS core melt probability of 2.03×10^{-7} per R-yr. Both of these values are listed here in table 4.1.

The majority of the turbine trip ATWS events were analyzed in the turbine trip ATWS event tree (pages 3-67 and 3-69 in the PRA), however, as mentioned above, several sequences were transferred to the MSIV ATWS event tree. Thus, as in the case of MSIV closure and loss of main feedwater, the core melt probabilities calculated in the MSIV ATWS event tree should be pro-rated (in this case by a factor of $0.42/2.2$) to determine the contribution of the turbine trip sequences. This process yields a value of 1.22×10^{-7} per R-yr. When added to the value of 4.33×10^{-7} per R-yr obtained in the turbine trip ATWS event tree, I obtained a final turbine trip ATWS core melt probability of 5.55×10^{-7} per R-yr, which is the value listed in table 4.1, the Limerick data sheet.

The ATWS core melt probability for loss of off-site power sequences was calculated directly from the LOOP ATWS event tree which is on page 3-81 of the PRA. The value obtained was 5.26×10^{-8} per R-yr, which is the value shown in table 4.1, the Limerick data sheet.

The ATWS core melt probability for seismic events was determined from the seismic

event tree, figures 3-7 through 3-9 in the Limerick SARA. ATWS events are those containing the symbol for failure of the reactor protection system, C_m. The total probability of these eight sequences is 8.09×10^{-7} per R-yr, which is the value shown for the seismic ATWS core melt probability in table 4.1, the Limerick data sheet.

Finally, tornados were analyzed in Section 6 of the SARA. Again the SARA authors estimated that the contribution to ATWS core melt sequences due to tornados was negligible, but they did not calculate a specific value. Their analysis estimated that the probability of a tornado causing an ATWS event that would lead to core melt was less than 10^{-10} per R-yr. To be conservative, I assumed that the tornado ATWS core melt probability was equal to 10^{-10} , which is the value shown in table 4.1, the Limerick data sheet.

4.6 Contribution of ATWS to Core Melt

From table 4.1, the Limerick data sheet, we see that the contribution of ATWS events to the total core melt probability is 4.68%. This value is an order of magnitude lower than the value calculated for the WASH-1400 BWR (40.32%). The Limerick ATWS core melt probability of 2.02×10^{-6} per R-yr is also the same order of magnitude as the NUREG -0460 safety objective of 1×10^{-6} per R-yr. Both of these points indicate that ATWS events at Limerick are not major contributors to core melt and that they do not pose an undue risk to the health and safety of the public.

We also see from table 4.1 that the major contributors to the ATWS core melt probability are seismic events (40.03%), turbine trips (27.46%), MSIV closure (15.54%) and loss of main feedwater (10.04%). These four initiators account for 28.29% of the total core melt probability (with seismic events contributing 13.57%), which suggests that modifications to reduce the ATWS core melt probability at Limerick would have a limited effect on the total core melt probability.

TABLE 4.1 - LIMERICK UNIT I DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per R-YR)	% OF TOTAL Pcm	Pcm/ATWS (per R-YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	4.00E-04	7.10E-08	0.16%	0E+00	0.00%	0.00%
MEDIUM LOCA	2.00E-03	1.70E-08	0.04%	2.40E-09	0.12%	0.01%
SMALL LOCA	1.00E-02	2.80E-08	0.06%	1.20E-08	0.59%	0.03%
LOCA TOTALS	1.24E-02	1.16E-07	0.27%	1.44E-08	0.71%	0.03%
INADVERTANT OPEN RELIEF VALVE	6.00E-02	9.10E-07	2.11%	7.30E-08	3.61%	0.17%
MSIV CLOSURE	1.08E+00	2.79E-06	6.46%	3.14E-07	15.54%	0.73%
LOSS OF MAIN FEEDWATER	7.00E-01	1.81E-06	4.19%	2.03E-07	10.04%	0.47%
TURBINE TRIP	3.98E+00	1.76E-06	4.07%	5.55E-07	27.46%	1.28%
LOSS OF OFF-SITE POWER	5.30E-02	6.70E-06	15.51%	5.26E-08	2.60%	0.12%
MANUAL REACTOR TRIP	3.20E+00	2.28E-07	0.53%	0E+00	0.00%	0.00%
TRANSIENT TOTALS	9.07	1.42E-05	32.87%	1.20E-06	59.25%	2.77%
INTERNAL TOTALS	9.09	1.43E-05	33.14%	1.21E-06	59.97%	2.81%
SEISMIC EVENTS	1.67E-03	5.86E-06	13.57%	8.09E-07	40.03%	1.87%
FIRES	5.28E-02	2.24E-05	51.86%	0E+00	0.00%	0.00%
TORNADOS	2.30E-04	4.00E-08	0.09%	1.00E-10	0.00%	0.00%
AIRCRAFT CRASHES	9.40E-08	9.40E-09	0.02%	0E+00	0.00%	0.00%
FLOODS	7.00E-02	5.70E-07	1.32%	0E+00	0.00%	0.00%
TURBINE MISSILES	1.00E-05	1.00E-09	0.00%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	1.25E-01	2.89E-05	66.86%	8.09E-07	40.03%	1.87%
GRAND TOTALS	9.21	4.32E-05	100.00%	2.02E-06	100.00%	4.68%

MEAN RPS UNRELIABILITY 3.00E-05/d

CHAPTER 5 - INDIAN POINT UNIT 2

5.0 Introduction

The *Indian Point Probabilistic Safety Study* (PSS) was published in 1982 in the aftermath of the Three Mile Island II accident. The study estimated the core melt and public risk associated with accidents at Indian Point Units 2 and 3. The methodology used was similar to the WASH-1400 methodology, but many of the questionable techniques used in WASH-1400 were replaced with more acceptable approaches. The information presented in this chapter was derived from the Indian Point PSS.

5.1 Plant Description

Indian Point Unit 2 is located at Indian Point, New York. It is a Westinghouse pressurized water reactor (PWR) with four coolant loops. Its power rating is 2758 MWt (873 MWe). The large dry containment structure is made of reinforced concrete with a steel lining.

5.2 The Initiator Frequencies

Table 1.0-1 on page 1.0-5 of the Indian Point PSS gives a list of 16 internal initiators which were determined specifically for Indian Point Unit 2 from plant operating data. Since these values appeared consistent with generic plant data and with the values used in the other PRAs I studied, I chose to accept these values and have listed them in table 5.1, the Indian Point 2 data sheet.

For external events, specific initiating frequencies were not calculated. I therefore developed initiating frequencies for external initiators as accurately as I could from the data presented in the PSS. The following paragraphs describe how I arrived at each external initiating frequency that is listed in table 5.1.

Section 7.2 of the PSS discusses seismic events, and notes that two seismicity studies were commissioned by the PSS: one from Dames and Moore, the other from Woodward-

Clyde. These studies are presented in Section 7.9 of the PSS and both list probabilities of exceedence for various ground accelerations. In order to calculate an initiating frequency for earthquakes at Indian Point, I assumed that this frequency was equal to the probability of exceeding a ground acceleration of 0.1g. Table 3 in the Dames and Moore study lists ground accelerations for two tectonic zones which I assumed were given equal weight. The probability of exceeding 0.1g ground acceleration for the Northeast zones is listed as 1.5×10^{-3} per year, while this same probability for the Piedmont/Piedmont-Cape Ann zones is given as 2.8×10^{-3} per year. Since I assumed that each of these values should be given equal weight, I calculated a seismic initiating frequency of 2.2×10^{-3} per year for the Dames and Moore study. Tables 2 and 4 in the Woodward-Clyde study list the probability of exceeding 0.1g ground acceleration as 1.01×10^{-2} per year. Since page 7.2-5 of the PSS states that each seismicity study was given equal weight, I averaged the two values obtained from the studies to obtain a seismic initiating frequency of 6.15×10^{-3} per year, which is the value shown in table 5.1, the Indian Point 2 data sheet.

The fire analysis for Indian Point 2 is presented in Section 7.3 of the PSS. The fire initiation frequencies were based on industry experience. There were four basic areas of concern: the cable spreading room, the switchgear room, the electrical tunnel and the diesel generating room. The frequencies for fires in these areas were given on pages 7.3-6, 7.3-18, 7.3-18 and 7.3-38 of the PSS, respectively. The sum of these frequencies (7.2×10^{-3} , 9.4×10^{-3} , 2.2×10^{-3} and 7.0×10^{-4} per year, respectively) is 1.95×10^{-2} per year which is the value shown in table 5.1, the Indian Point 2 data sheet.

The effects of wind events at Indian Point 2 were studied in Section 7.5 of the PSS. A Research Triangle Institute report addressing wind speed probabilities (for tornados and hurricanes) is presented in Section 7.9.5 of the PSS. However, no specific initiating frequencies were determined for tornados or hurricanes. Section 7.5 of the PSS gives lower bound wind capacities for various structures and equipment at Indian Point 2. I

assumed that an event would be initiated when the upper bound wind speed from the Research Triangle Institute report exceeded the smallest lower bound capacity listed in Section 7.5 of the PSS. For tornados, the lowest lower bound capacity is 125 mph for the transmission lines. We can find the mean probability that the upper bound wind speed (fastest 1/4 mile) exceeds 125 mph by interpolating in table III-14 (p. III-38 in the PSS) and in Figure III-7 (p. III-40 in the PSS) of the Research Triangle Institute report. I calculated this probability to be 8.32×10^{-5} per year. For hurricanes the lowest lower bound capacity is 94 mph for the auxiliary feedwater pump building. In this case we interpolate in table IV-10 (p. IV-38 in the PSS) and Figure IV-8 (p. IV-39 in the PSS) of the Research Triangle Institute report to find that the mean probability of the upper bound wind speed (fastest mile) exceeding 94 mph is 6.85×10^{-3} per year. (NOTE: the plots in Figures III-7 and IV-8 are log-log plots, so that it is a logarithmic interpolation.) So based on my assumption above, the total frequency of wind induced events at Indian Point 2 is the sum of these two probabilities, i.e. 6.93×10^{-3} per year which is the value shown here in table 5.1, the Indian Point 2 data sheet.

Section 7.6 of the PSS analyzed the probability of an aircraft crash at the Indian Point site. Page 7.6-6 of the PSS lists the probability of a light aircraft hitting the site (2.4×10^{-7} per year) as well as the probability of a large aircraft hitting the site (4.6×10^{-8} per year). I assumed that the initiating frequency for aircraft crash events was the sum of these two probabilities, i.e. 2.9×10^{-7} per year which is the value listed in table 5.1, the Indian Point 2 data sheet.

Section 7.4 of the PSS analyzed the effects of flooding. Page 7.4-2 of the PSS conservatively lists the probability of flooding from the circulating water system as 1×10^{-3} per year while all other flood initiator probabilities are less than or equal to 1.0×10^{-6} per year. I therefore took the flood initiator frequency to be 1.0×10^{-3} per year, which

is the value shown in table 5.1, the Indian Point 2 data sheet.

Section 7.8 of the PSS analyzed the effects of turbine missiles. The authors estimated the probability of a turbine generated missile as 1×10^{-4} per year. Since there has been very little research in this area, I accepted this estimate and have displayed this value in table 5.1, the Indian Point 2 data sheet.

5.3 The Reactor Protection System

The PSS authors obtained a reactor protection system unavailability value of 2.01×10^{-5} per demand by using fault tree analysis. This analysis included failure of multiple control rods to insert, trip breaker failures, logic channel failures (relay and matrix failures), wiring faults, instrument channel failures, unavailability due to testing and a common cause failure analysis based on instrument miscalibration. The authors did not explicitly calculate separate probabilities for electrical and mechanical failures.

As with all common cause failure analyses, the PSS common cause failure analysis is somewhat questionable in its assumptions and yields a common cause failure rate of 4.61×10^{-7} per demand. However, since the total calculated unavailability value of 2.01×10^{-5} per demand is the same order of magnitude as the NUREG-0460 value of 3.0×10^{-5} per demand, I decided to accept the PSS authors' RPS unavailability value of 2.01×10^{-5} per demand, which is the value shown here in table 5.1, the Indian Point 2 data sheet.

5.4 Core Melt Probabilities

Table 1.0-2 of the PSS shows a split fraction matrix for the internal initiators. This matrix is basically a listing of conditional core melt probabilities given the particular initiating events. These split fractions, or conditional core melt probabilities, were derived through event tree analysis in the PSS. I reviewed each event tree and traced many of the results listed in table 1.0-2 of the PSS. I found the event trees to be reasonable and the results listed in table 1.0-2 to be consistent with the event trees. I

therefore accepted the values in table 1.0-2 of the PSS and used them as I describe below to calculate the core melt probabilities for the internal initiators.

Each matrix entry is the conditional probability of getting a particular *class* of core melt accident (i.e. a particular "plant state" as the PSS refers to it) given a specific initiating event. Thus if we sum these split fractions horizontally in table 1.0-2 of the PSS (i.e. across the various plant states), we obtain the *total* conditional core melt probabilities for each of the internal initiators. I calculated these values and they are shown here in table 5.2. Once I calculated these conditional probabilities, I simply had to multiply each value by the corresponding initiator frequency to obtain the core melt probability for each initiator. These core melt probabilities are shown in table 5.1, the Indian Point 2 data sheet.

The effects of external initiators at Indian Point 2 were discussed in Section 7 of the PSS. The seismic analysis was derived from two consultant studies as mentioned in Section 5.3 above. I had to assume that the studies were accurate since it was difficult to trace their results. The seismic core melt frequency presented in the PSS was found in an amendment 2 modification on page 7.2-14A-3. This value is 7.7×10^{-6} per R-yr which is the value shown here in table 5.1, the Indian Point 2 data sheet.

The effects of fires on core melt probability were addressed in Section 7.3 of the PSS. Page 7.3-3 of the PSS states that the fire analysis was not detailed, but instead used conservative assumptions. The analysis used a variety of functions and probabilities including fire initiation frequencies, fire propagation times, mean extinguishing times, and critical zone factors to estimate the probability of a core melt due to a fire in a particular area of the plant. The analysis seemed fairly straightforward and the assumptions appropriate. I chose to accept the PSS value of 2.0×10^{-4} per R-yr as the mean probability of a core melt due to fires. This value is shown in table 5.1, the Indian Point 2 data sheet.

The effects of wind and wind induced missiles (i.e. hurricanes and tornados) at Indian Point 2 were analyzed in Section 7.5 of the PSS. The analysis, as in the case of fires, is rather general and uses conservative assumptions. The PSS authors obtained a value of 4.5×10^{-5} per R-yr as the mean probability of a core melt due to tornados and hurricanes. I accepted this value which is the value shown here in table 5.1, the Indian Point 2 data sheet.

The PSS authors addressed the effects of aircraft crashes in Section 7.6 of the PSS. They determined that the probability of an aircraft hitting a critical structure was 2.9×10^{-7} per year and that the probability of core melt is "much less" and therefore insignificant. However, I felt that it was desirable to obtain such a core melt value, for the sake of completeness and for the sake of comparison to the other PRAs studied in this project. To obtain this value I assumed that the conditional probability of a core melt given an aircraft impact on a critical structure is 0.10. Thus the core melt probability at Indian Point 2 due to aircraft crashes is 2.9×10^{-8} per R-yr which is the value shown in table 5.1, the Indian Point 2 data sheet.

The flooding analysis presented in Section 7.4 of the PSS concludes that no transients would result from flooding and that flooding is therefore a negligible contributor to core melt. However, as in the case of aircraft crashes, I felt it desirable to estimate a core melt probability due to flooding for completeness and comparison purposes. From table 5.1, the Indian Point 2 data sheet, negligible contributors appear to be those with a core melt probability on the order of 10^{-8} per R-yr or less. The initiating frequency for floods is 1×10^{-3} per year which is the probability of flooding from the circulating water system, while all other flooding occurs with a frequency of 1×10^{-6} per year. If we assume that the core melt probability due to floods is 1×10^{-8} per R-yr, this would require the conditional probability of core melt given a flood from the circulating water

system to be 1×10^{-5} , while the conditional probability of core melt given any other flood would have to be 1×10^{-2} . Considering the fact that the circulating water system is primarily in the turbine building and the fact that the PSS authors determined that *no* transients would result from flooding, these conditional probabilities seem very reasonable. I therefore assumed that the core melt probability due to flooding at Indian Point 2 is 1×10^{-8} per R-yr, which is the value shown in table 5.1, the Indian Point 2 data sheet.

The PSS authors determined in Section 7.8 of the PSS that the probability of a turbine missile hitting a piece of safety equipment is 1.0×10^{-7} per year, and that the probability of core melt from a turbine missile would therefore be much less than this value. Again, I felt it desirable to obtain a value for the core melt probability due to turbine missiles. To obtain this value I assumed that the conditional probability of core melt given a turbine missile hitting a piece of safety equipment was 0.10. Thus I calculated the core melt probability due to turbine missiles at Indian Point 2 to be 1.0×10^{-8} per R-yr, as shown in table 5.1, the Indian Point 2 data sheet.

5.5 ATWS Core Melt Probabilities

The ATWS core melt split fractions are presented in table 1.3.5.14-2 on page 1.3-444 of the PSS. As with the total core melt probability split fractions, these values were derived through event tree analysis and represent the probability of getting a particular class of ATWS core melt (i.e. plant state) for the specific internal initiators. By summing horizontally in table 1.3.5.14-2 (i.e. across the various plant states), I obtained the total conditional ATWS core melt probability for each internal initiator. These values are listed here in table 5.3. To obtain the ATWS core melt probabilities, I multiplied the initiator frequencies by the corresponding conditional core melt probabilities. These ATWS core melt probabilities are listed in table 5.1, the Indian Point 2 data sheet.

It should be noted that the PSS authors did not feel that external events would initiate

an ATWS sequence.

5.6 Contribution of ATWS to Core Melt

From table 5.1, the Indian Point 2 data sheet, we see that the contribution of ATWS events at Indian Point 2 is 0.65%. This value is more than an order of magnitude lower than the corresponding contribution of ATWS events at Surry I, the WASH-1400 PWR, and it is so low as to suggest that ATWS events at Indian Point 2 do not pose an undue risk to the health and safety of the public. Also, the core melt probability due to ATWS events at Indian Point 2 is 2.21×10^{-6} per R-yr, which is the same order of magnitude as the Surry I ATWS core melt probability of 4.15×10^{-6} per R-yr and the NUREG-0460 safety objective of 1×10^{-6} per R-yr, which again supports the idea that ATWS events do not pose a major risk at Indian Point 2.

We also see from table 5.1 that the major contributors to the ATWS core melt probability are turbine trip (42.36%), loss of main feedwater (38.79%) and turbine trip due to loss of off-site power (17.13%). However, of these three, only the turbine trip due to loss of off-site power contributes significantly to the total core melt probability (11.14%), suggesting that modifications to lower the core melt probability due to ATWS events would not significantly reduce the overall core melt probability.

TABLE 5.1 - INDIAN POINT UNIT 2 DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per R-YR)	% OF TOTAL Pcm	Pcm/ATWS (per R-YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	1.95E-03	1.60E-05	4.70%	0E+00	0.00%	0.00%
MEDIUM LOCA	1.95E-03	1.30E-05	3.82%	0E+00	0.00%	0.00%
SMALL LOCA	1.85E-02	1.66E-05	4.88%	2.37E-09	0.11%	0.00%
INTERFACING SYST. LOCA	4.58E-07	4.58E-07	0.13%	0E+00	0.00%	0.00%
LOCA TOTALS	2.24E-02	4.61E-05	13.54%	2.37E-09	0.11%	0.00%
STM. GEN. TUBE RUPTURE	2.74E-02	1.42E-07	0.04%	9.97E-13	0.00%	0.00%
STM. LINE BREAK INSIDE CONTAINMENT	1.95E-03	1.99E-07	0.06%	2.50E-10	0.01%	0.00%
STM. LINE BREAK OUTSIDE CONTAINMENT	1.95E-03	1.99E-07	0.06%	2.50E-10	0.01%	0.00%
LOSS OF MAIN FEEDWATER TRIP OF ONE MSIV	6.70E+00	1.80E-06	0.53%	8.58E-07	38.79%	0.25%
LOSS OF RCS FLOW	1.25E+00	1.75E-07	0.05%	0E+00	0.00%	0.00%
CORE POWER EXCURSION	1.36E-01	3.28E-08	0.01%	1.58E-08	0.71%	0.00%
TURBINE TRIP	2.21E-02	2.08E-15	0.00%	9.99E-16	0.00%	0.00%
TURBINE TRIP: LOOP	7.32E+00	1.96E-06	0.58%	9.37E-07	42.36%	0.28%
TURBINE TRIP: LOSS OF SERVICE WATER	1.82E-01	3.79E-05	11.14%	3.79E-07	17.13%	0.11%
TURBINE TRIP: LOSS OF COMPONENT COOLING	1.95E-03	6.03E-08	0.02%	1.95E-08	0.88%	0.01%
REACTOR TRIP	6.84E+00	9.58E-07	0.28%	0E+00	0.00%	0.00%
REACTOR TRIP: LOSS OF COMPONENT COOLING	1.95E-03	3.71E-08	0.01%	0E+00	0.00%	0.00%
TRANSIENT TOTALS	22.49	4.35E-05	12.77%	2.21E-06	99.89%	0.65%
INTERNAL TOTALS	22.51	8.95E-05	26.31%	2.21E-06	100.00%	0.65%
SEISMIC EVENTS	6.15E-03	7.70E-06	2.26%	0E+00	0.00%	0.00%
FIRES	1.95E-02	2.00E-04	58.78%	0E+00	0.00%	0.00%
TORNADOS	6.93E-03	4.30E-05	12.64%	0E+00	0.00%	0.00%
AIRCRAFT CRASHES	2.90E-07	2.90E-08	0.01%	0E+00	0.00%	0.00%
FLOODS	1.00E-03	1.00E-08	0.00%	0E+00	0.00%	0.00%
TURBINE MISSILES	1.00E-04	1.00E-08	0.00%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	3.37E-02	2.51E-04	73.69%	0E+00	0.00%	0.00%
GRAND TOTALS	22.54	3.40E-04	100.00%	2.21E-06	100.00%	0.65%
MEAN RPS UNRELIABILITY 2.01E-05/d						

TABLE 5.2 - CONDITIONAL CORE MELT PROBABILITIES FOR I. P. 2 INTERNAL INITIATORS

INITIATOR	CONDITIONAL CORE MELT PROBABILITY
LARGE LOCA	8.20E-03
MEDIUM LOCA	6.69E-03
SMALL LOCA	8.95E-04
INTERFACING SYST. LOCA	1.00E+00
STM. GEN. TUBE RUPTURE	5.18E-06
STM. LINE BREAK INSIDE CONTAINMENT	1.02E-04
STM. LINE BREAK OUTSIDE CONTAINMENT	1.02E-04
LOSS OF MAIN FEEDWATER	2.68E-07
TRIP OF ONE MSIV	1.40E-07
LOSS OF RCS FLOW	2.41E-07
CORE POWER EXCURSION	9.41E-14
TURBINE TRIP	2.68E-07
TURBINE TRIP: LOOP	2.08E-04
TURBINE TRIP:LOSS OF SERVICE WATER	3.09E-05
REACTOR TRIP	1.40E-07
REACTOR TRIP: LOSS OF COMPONENT COOLING	1.90E-05

TABLE 5.3 - CONDITIONAL ATWS CORE MELT PROBABILITIES FOR INDIAN POINT 2

INITIATOR	COND. ATWS CORE MELT PROBABILITY
LARGE LOCA	0E+00
MEDIUM LOCA	0E+00
SMALL LOCA	1.28E-07
INTERFACING SYST. LOCA	0E+00
STM. GEN. TUBE RUPTURE	3.64E-11
STM. LINE BREAK INSIDE	1.28E-07
CONTAINMENT	
STM. LINE BREAK OUTSIDE	1.28E-07
CONTAINMENT	
LOSS OF MAIN FEEDWATER	1.28E-07
TRIP OF ONE MSIV	0E+00
LOSS OF RCS FLOW	1.16E-07
CORE POWER EXCURSION	4.52E-14
TURBINE TRIP	1.28E-07
TURBINE TRIP: LOOP	2.08E-06
TURBINE TRIP:LOSS OF	1.00E-05
SERVICE WATER	
REACTOR TRIP	0E+00
REACTOR TRIP: LOSS OF	0E+00
COMPONENT COOLING	

CHAPTER 6 - INDIAN POINT UNIT 3

6.0 Introduction

The *Indian Point Probabilistic Safety Study* (PSS) was published in 1982 in the aftermath of the Three Mile Island II accident. The study estimated the core melt and public risk associated with accidents at Indian Point Units 2 and 3. The methodology used was similar to the WASH-1400 methodology, but many of the questionable techniques used in WASH-1400 were replaced with more acceptable approaches. The information presented in this chapter is derived from the Indian Point PSS.

6.1 Plant Description

Indian Point Unit 3 is located at Indian Point, New York. It is a Westinghouse pressurized water reactor (PWR) with four coolant loops. Its power rating is 3025 MWt (965 MWe). The large dry containment structure is made of reinforced concrete with a steel lining.

6.2 The Initiator Frequencies

Table 1.0-3 on page 1.0-7 of the Indian Point PSS gives a list of 16 internal initiators which were determined for Indian Point Unit 3 by combining generic plant data with plant specific data. Since these values appeared consistent with the values used in the other PRAs I studied, I chose to accept these values and have listed them in table 6.1, the Indian Point 3 data sheet.

For external events, specific initiating frequencies were not calculated. I therefore developed initiating frequencies for external initiators as accurately as I could from the data presented in the PSS. The following paragraphs describe how I arrived at each external initiating frequency that is listed in table 6.1, the Indian Point 3 data sheet.

With the exception of fires, the PSS authors determined that the frequency of external initiators at Indian Point 3 would be equal to or, for some initiators, slightly less than the

frequency of these initiators at Indian Point 2. In light of this fact, the frequencies listed in table 6.1 for seismic events, tornados, aircraft crashes, floods and turbine missiles are the same as those listed in table 5.1 for Indian Point 2. Explanations of how I arrived at each of these values can be found in Section 5.2 of Chapter 5.

The fire analysis for Indian Point 3 is presented in Section 7.3 of the PSS. The fire initiation frequencies were based on industry experience. There were four basic areas of concern: the cable spreading room, the cable spreading room tunnel entrance, the upper electrical tunnel and the diesel generating room. The frequencies for fires in these areas were given on pages 7.3-79, 7.3-79, 7.3-97 and 7.3-96 of the PSS, respectively. The sum of these frequencies (7.2×10^{-3} , 7.2×10^{-3} , 2.2×10^{-3} and 9.8×10^{-3} per year, respectively) is 2.64×10^{-2} per year which is the value shown in table 6.1, the Indian Point 3 data sheet.

6.3 The Reactor Protection System

The PSS authors obtained a reactor protection system unavailability value of 3.93×10^{-5} per demand by using a fault tree analysis. This analysis included failure of multiple control rods to insert, trip breaker failures, logic channel failures (relay and matrix failures), wiring faults, instrument channel failures, unavailability due to testing and a common cause failure analysis based on instrument miscalibration. The authors did not explicitly calculate separate probabilities for electrical and mechanical failures.

As with all common cause failure analyses, the PSS common cause failure analysis is somewhat questionable in its assumptions and yields a common cause failure rate of 4.61×10^{-7} per demand. However, since the total calculated unavailability value of 3.93×10^{-5} per demand is greater than the NUREG-0460 value of 3.0×10^{-5} per demand, I decided to accept the PSS authors' RPS unavailability value of 3.93×10^{-5} per demand, which is the value shown in table 6.1, the Indian Point 3 data sheet.

6.4 Core Melt Probabilities

Table 1.0-4 on page 1.0-8 of the PSS shows a split fraction matrix for the internal initiators. This matrix is basically a listing of conditional core melt probabilities given the particular initiating events. These split fractions, or conditional core melt probabilities, were derived through event tree analysis in the PSS. I reviewed each event tree and traced many of the results listed in table 1.0-4. I found the event trees to be reasonable and the results listed in table 1.0-4 to be consistent with the event trees. I therefore accepted the values in table 1.0-4 and used them as I describe below to calculate the core melt probabilities for the internal initiators.

Each matrix entry is the conditional probability of getting a particular *class* of core melt accident (i.e. a particular "plant state" as the PSS refers to it) given a specific initiating event. Thus if we sum these split fractions horizontally in table 1.0-4 in the PSS (i.e. across the various plant states), we obtain the *total* conditional core melt probabilities for each of the internal initiators. I calculated these values and they are shown here in table 6.2. Once I calculated these conditional probabilities, I simply had to multiply each value by the corresponding initiator frequency to obtain the core melt probability for each initiator. These core melt probabilities are shown in table 6.1, the Indian Point 3 data sheet.

The effects of external initiators at Indian Point 3 were discussed in Section 7 of the PSS. The seismic analysis was derived from two consultant studies as mentioned in Section 5.3 above. I had to assume that the studies were accurate since it was difficult to trace their results. The seismic core melt frequency presented in the PSS was found in an amendment 2 modification on page 7.2-21A-1. This value is 3.6×10^{-6} per R-yr which is the value shown in table 6.1, the Indian Point 3 data sheet.

The effects of fires on core melt probability were addressed in Section 7.3 of the PSS. Page 7.3-3 of the PSS states that the fire analysis was not detailed, but instead used

conservative assumptions. The analysis used a variety of functions and probabilities including fire initiation frequencies, fire propagation times, mean extinguishing times, and critical zone factors to estimate the probability of a core melt due to a fire in a particular area of the plant. The analysis seemed fairly straightforward and the assumptions appropriate. I chose to accept the PSS value of 6.3×10^{-5} per R-yr as the mean probability of a core melt due to fires. This value is shown in table 6.1, the Indian Point 3 data sheet.

The effects of wind and wind induced missiles (i.e. hurricanes and tornados) at Indian Point 3 were analyzed in Section 7.5 of the PSS. The analysis, as in the case of fires, is rather general and uses conservative assumptions. The PSS authors obtained a value of 1.3×10^{-6} per R-yr as the mean probability of a core melt due to tornados and hurricanes. I accepted this value which is the value shown in table 6.1, the Indian Point 3 data sheet.

The PSS authors addressed the effects of aircraft crashes in Section 7.6 of the PSS. They determined that the probability of an aircraft hitting a critical structure was less than 2.9×10^{-7} per year and that the probability of core melt is "much less" and therefore insignificant. However, I felt that it was desirable to obtain such a core melt value, for the sake of completeness and for the sake of comparison to the other PRAs studied in this project. To obtain this value I assumed that the conditional probability of a core melt given an aircraft impact on a critical structure is 0.10. Thus the core melt probability at Indian Point 3 due to aircraft crashes is 2.9×10^{-8} per R-yr which is the value shown in table 6.1, the Indian Point 3 data sheet.

The flooding analysis presented in Section 7.4 of the PSS concludes that no transients would result from flooding and that flooding is therefore a negligible contributor to core melt. However, as in the case of Indian Point 2, I felt it desirable to estimate a core melt probability due to flooding for completeness and comparison purposes. Using the same arguments as those presented for flooding at Indian Point 2 in Section 5.4 above, I

estimated that the core melt probability due to flooding at Indian Point 3 is 1×10^{-8} per R-yr, which is the value shown in table 6.1, the Indian Point 3 data sheet.

The PSS authors determined in Section 7.8 of the PSS that the probability of a turbine missile hitting a piece of safety equipment is 1.0×10^{-7} per year, and that the probability of core melt from a turbine missile would therefore be much less than this value. Again, I felt it desirable to obtain a value for the core melt probability due to turbine missiles. To obtain this value I assumed that the conditional probability of core melt given a turbine missile hitting a piece of safety equipment was 0.10. Thus I calculated the core melt probability due to turbine missiles at Indian Point 3 to be 1.0×10^{-8} per R-yr, as shown in table 6.1, the Indian Point 3 data sheet.

6.5 ATWS Core Melt Probabilities

The ATWS core melt split fractions are presented in table 1.3.6.14-2 on page 1.3-650 of the PSS. As with the total core melt probability split fractions, these values were derived through event tree analysis and represent the probability of getting a particular class of ATWS core melt (plant state) for the specific internal initiators. By summing horizontally in table 1.3.6.14-2 (i.e. across the various plant states), I obtained the total conditional ATWS core melt probability for each internal initiator. These values are listed in table 6.3. To obtain the ATWS core melt probabilities, I multiplied the initiator frequencies by the corresponding conditional core melt probabilities. These ATWS core melt probabilities are listed in table 6.1, the Indian Point 3 data sheet.

It should be noted that the PSS authors did not feel that external events would initiate any ATWS sequences.

6.6 Contribution of ATWS to Core Melt

From table 6.1, the Indian Point 3 data sheet, we see that the contribution of ATWS events at Indian Point 3 is 1.55%. This value is only one-sixth of the corresponding value for Surry I, the WASH-1400 PWR, and it is so low as to suggest that ATWS events at

Indian Point 3 do not pose an undue risk to the health and safety of the public. Also from table 6.1, the core melt probability due to ATWS events at Indian Point 3 is 3.03×10^{-6} per R-yr, which is the same order of magnitude as the Surry I ATWS core melt probability of 4.15×10^{-6} per R-yr and the NUREG-0460 safety objective of 1×10^{-6} per R-yr, which again supports the idea that ATWS events do not pose a major risk at Indian Point 3.

We also see from table 6.1 that the major contributors to the ATWS core melt probability at Indian Point 3 are loss of main feedwater (46.22%), turbine trip (33.02%), and turbine trip due to loss of off-site power (17.20%). However, these three initiators account for only 3.82% of the total core melt probability, suggesting that modifications to lower the core melt probability due to ATWS events would have a negligible effect on the overall core melt probability at Indian Point 3.

TABLE 6.1 - INDIAN POINT UNIT 3 DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per R-YR)	% OF TOTAL Pcm	Pcm/ATWS (per R-YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	2.16E-03	1.81E-05	9.27%	0E+00	0.00%	0.00%
MEDIUM LOCA	2.16E-03	1.37E-05	7.02%	0E+00	0.00%	0.00%
SMALL LOCA	2.01E-02	8.56E-05	43.86%	7.40E-09	0.24%	0.00%
INTERFACING SYST. LOCA	4.64E-07	4.64E-07	0.24%	0E+00	0.00%	0.00%
LOCA TOTALS	2.44E-02	1.18E-04	60.39%	7.40E-09	0.24%	0.00%
STM. GEN. TUBE RUPTURE	3.37E-02	6.98E-07	0.36%	2.91E-12	0.00%	0.00%
STM. LINE BREAK INSIDE CONTAINMENT	2.16E-03	2.83E-07	0.14%	7.95E-10	0.03%	0.00%
STM. LINE BREAK OUTSIDE CONTAINMENT	2.16E-03	2.83E-07	0.14%	7.95E-10	0.03%	0.00%
LOSS OF MAIN FEEDWATER TRIP OF ONE MSIV	3.80E+00	1.91E-06	0.98%	1.40E-06	46.22%	0.72%
LOSS OF RCS FLOW	8.98E-02	1.35E-08	0.01%	0E+00	0.00%	0.00%
CORE POWER EXCURSION	1.71E-01	7.99E-08	0.04%	5.66E-08	1.87%	0.03%
TURBINE TRIP	2.57E-02	4.68E-15	0.00%	3.32E-15	0.00%	0.00%
TURBINE TRIP: LOOP	2.72E+00	1.37E-06	0.70%	1.00E-06	33.02%	0.51%
TURBINE TRIP: LOSS OF SERVICE WATER	2.66E-01	4.18E-06	2.14%	5.21E-07	17.20%	0.27%
TURBINE TRIP: LOSS OF COMPONENT COOLING	2.16E-03	1.27E-07	0.07%	4.23E-08	1.40%	0.02%
REACTOR TRIP	2.86E+00	3.95E-07	0.20%	0E+00	0.00%	0.00%
REACTOR TRIP: LOSS OF COMPONENT COOLING	2.16E-03	3.28E-08	0.02%	0E+00	0.00%	0.00%
TRANSIENT TOTALS	9.97	9.37E-06	4.80%	3.02E-06	99.76%	1.55%
INTERNAL TOTALS	10.00	1.27E-04	65.19%	3.03E-06	100.00%	1.55%
SEISMIC EVENTS	6.15E-03	3.60E-06	1.84%	0E+00	0.00%	0.00%
FIRES	2.64E-02	6.30E-05	32.28%	0E+00	0.00%	0.00%
TORNADOS	6.93E-03	1.30E-06	0.67%	0E+00	0.00%	0.00%
AIRCRAFT CRASHES	2.90E-07	2.90E-08	0.01%	0E+00	0.00%	0.00%
FLOODS	1.00E-03	1.00E-08	0.01%	0E+00	0.00%	0.00%
TURBINE MISSILES	1.00E-04	1.00E-08	0.01%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	4.06E-02	6.79E-05	34.81%	0E+00	0.00%	0.00%
GRAND TOTALS	10.04	1.95E-04	100.00%	3.03E-06	100.00%	1.55%
MEAN RPS UNRELIABILITY 3.93E-05/d						

TABLE 6.2 - CONDITIONAL CORE MELT PROBABILITIES FOR I. P. 3 INTERNAL INITIATORS

INITIATOR	CONDITIONAL CORE MELT PROBABILITY
LARGE LOCA	8.36E-03
MEDIUM LOCA	6.36E-03
SMALL LOCA	4.26E-03
INTERFACING SYST. LOCA	1.00E+00
STM. GEN. TUBE RUPTURE	2.07E-05
STM. LINE BREAK INSIDE	1.31E-04
CONTAINMENT	
STM. LINE BREAK OUTSIDE	1.31E-04
CONTAINMENT	
LOSS OF MAIN FEEDWATER	5.02E-07
TRIP OF ONE MSIV	1.50E-07
LOSS OF RCS FLOW	4.67E-07
CORE POWER EXCURSION	1.82E-13
TURBINE TRIP	5.02E-07
TURBINE TRIP: LOOP	1.57E-05
TURBINE TRIP:LOSS OF	5.86E-05
SERVICE WATER	
REACTOR TRIP	1.38E-07
REACTOR TRIP: LOSS OF	1.52E-05
COMPONENT COOLING	

TABLE 6.3 - CONDITIONAL ATWS CORE MELT PROBABILITIES FOR INDIAN POINT 3

INITIATOR	COND. ATWS CORE MELT PROBABILITY
LARGE LOCA	0E+00
MEDIUM LOCA	0E+00
SMALL LOCA	3.68E-07
INTERFACING SYST. LOCA	0E+00
STM. GEN. TUBE RUPTURE	8.63E-11
STM. LINE BREAK INSIDE CONTAINMENT	3.68E-07
STM. LINE BREAK OUTSIDE CONTAINMENT	3.68E-07
LOSS OF MAIN FEEDWATER	3.68E-07
TRIP OF ONE MSIV	0E+00
LOSS OF RCS FLOW	3.31E-07
CORE POWER EXCURSION	1.29E-13
TURBINE TRIP	3.68E-07
TURBINE TRIP: LOOP	1.96E-06
TURBINE TRIP:LOSS OF SERVICE WATER	1.96E-05
REACTOR TRIP	0E+00
REACTOR TRIP: LOSS OF COMPONENT COOLING	0E+00

CHAPTER 7 - OCONEE UNIT 3

7.0 Introduction

The *Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3* was published in 1984. The study estimated the core melt and public risk associated with accidents at Oconee Unit 3. The methodology used was similar to the WASH-1400 methodology, but many of the questionable techniques used in WASH-1400 were replaced with more acceptable approaches. The information presented in this chapter is derived from the Oconee PRA.

7.1 Plant Description

Oconee Unit 3 is located at Oconee, South Carolina. It is a Babcock and Wilcox pressurized water reactor (PWR) with two coolant loops. Its power rating is 2568 MWt (887 MWe). The large dry containment structure is made of reinforced concrete with a steel lining. A stand-by shut down facility with independent cooling systems and a diesel generator to deal with fire and flood situations was under construction at the time this PRA was published. The PRA assumed that this facility was operational.

7.2 The Initiator Frequencies

Table 5-9 on page 5-53 of the Oconee PRA gives a list of internal initiators which were determined for Oconee Unit 3 by combining generic plant data with plant specific data. Since these values appeared consistent with the values used in the other PRAs I studied, I chose to accept these values with two minor modifications. First, the separate loss of off-site power values for loss of the grid and loss of the substation were combined into a single loss of off-site power frequency. Second, the three values listed for loss of low pressure service water are actually two component values (for loss with and without LPWS 108) as well as the aggregate value of 4.0×10^{-3} per year. To avoid duplication, I have only listed the aggregate value among the internal initiator frequencies which are

shown in table 7.1, the Oconee 3 data sheet.

For external events, specific initiating frequencies were not always calculated in the PRA. I therefore developed initiating frequencies for external initiators as accurately as I could from the data presented in the PRA. The following paragraphs describe how I arrived at each external initiating frequency that is listed in table 7.1.

The initiating frequency for seismic events was calculated from table 9-9 of the PRA. This table tabulates 18 frequency of exceedence curves for the Oconee site. Each curve was assigned a certain weight by the PRA authors. For my purposes, I decided that an earthquake would be initiated if the ground acceleration exceeded 0.1g. Then, from each curve I obtained the frequency of exceeding 0.1g ground acceleration and multiplied this frequency by the weighting factor for the given curve. I then took the sum of these weighted frequencies to be the initiating frequency for seismic events. The value I obtained was 4.79×10^{-4} per year which is the value shown in table 7.1, the Oconee 3 data sheet.

The fire analysis for Oconee 3 is given in Section 9 of the PRA. This analysis used industry data to get a prior distribution then combined this distribution with Oconee 3 data to get a posterior distribution. The two main areas of concern were the equipment room and the cable shaft. Each area had a fire initiation frequency of 2.3×10^{-2} per year. I therefore took the total fire initiator frequency to be 4.6×10^{-2} per year which is the value listed in table 7.1, the Oconee 3 data sheet.

The initiating frequency for tornados used in the PRA was based on historical data for tornados in the Oconee area. A severe tornado was considered to have winds exceeding 150 mph and occurred with a frequency of 3.5×10^{-5} per year. I accepted this frequency as the initiating frequency for tornados at Oconee 3 and have listed this value in table 7.1.

The effect of aircraft crashes at Oconee is addressed on page 9-299 of the PRA. The PRA authors developed a formula to calculate the crash frequency at Oconee 3. This formula took into account the area of the plant structures (target area), the number of

flights in the Oconee area, the accident rate per mile for the aircraft types operating in the Oconee area, and the probability of getting a ground crash as a function of perpendicular distance from an airway. The authors calculated a median crash frequency of 2.5×10^{-9} per year which I accepted as the initiating frequency for aircraft crashes at Oconee 3. This value is listed as such in table 7.1, the Oconee 3 data sheet.

The flood analysis at Oconee 3 is presented in Section 9 of the PRA. The authors analyzed internal floods (i.e. those from pipe breaks, pump seal failures, vessel leaks, etc.) separately from external floods (i.e. those from rain and river swelling). The main areas of concern for internal flooding were the turbine and auxiliary buildings. Fault tree analyses were done to determine the flooding frequencies due to the circulating cooling water, the high pressure service water and the low pressure service water systems. The flooding frequency in the turbine building was calculated to be 2.93×10^{-2} per year while in the auxiliary building flood frequency was 1.14×10^{-3} per year. A second analysis based on industry data yielded similar results. I therefore took the internal flood frequency to be the sum of the turbine and auxiliary building flood frequencies, i.e. 3.04×10^{-2} per year, which is the value listed in table 7.1, the Oconee 3 data sheet.

The PRA authors found that the failure of the Jocassee dam dominated the external flood frequency. They used failure data from similar dams to calculate a failure frequency of 2.5×10^{-5} per year. I accepted this value as the initiating frequency for external floods and have listed it in table 7.1, the Oconee 3 data sheet.

7.3 The Reactor Protection System

The PRA authors performed an overview analysis of the RPS to determine if the Oconee 3 system was significantly different from reactor protection systems at other plants. This analysis showed that the Oconee 3 system was not significantly different from other systems. As a result, rather than calculating a specific value for Oconee 3, the

authors chose to use the NUREG-0460 value of 3.0×10^{-5} per demand as the RPS unavailability at Oconee 3. This approach seemed reasonable, so I accepted this value as the unavailability of the Oconee 3 RPS and have listed it in table 7.1, the Oconee 3 data sheet. However, it should be noted that unlike the NUREG-0460 authors, the Oconee 3 authors did not differentiate between electrical and mechanical failures of the RPS.

7.4 Core Melt Probabilities

Table 2 on page 16 of the PRA lists the mean core melt probabilities for the internal initiators. The PRA did not include the event trees used to arrive at these values, so I was forced to accept the values without being able to trace the results. Several modifications had to be made to the results listed in table 2 since the list of initiators did not correspond exactly to the list that I developed for this report. Specifically, ATWS was listed as a separate initiator, so it was necessary to determine which of the other initiators were actually responsible for, as well as the extent to which each initiator contributed to, the ATWS value listed. Also, many of the initiators that I had identified were lumped together as "Other Transients" and assigned a single aggregate core melt probability. The following paragraphs explain the modifications that I made to the table 2 results.

In order to reassign the ATWS core melt probability to the appropriate initiators, I turned to the scoping analysis for ATWS in Appendix E of the PRA. This analysis looked at what the authors felt were the four dominant initiators: loss of condenser vacuum, turbine trip, loss of main feedwater, and loss of off-site power. As mentioned in Section 7.3 above, the authors used the NUREG-0460 value of 3.0×10^{-5} per demand as the RPS unavailability. From the event trees on pages E-15 through E-18 I obtained the ATWS core melt probability values shown here in table 7.1 for the four initiators involved. I then added these ATWS core melt probabilities to the "non-ATWS" core melt probabilities listed in table 2 of the PRA to obtain a "total" core melt probability for each of the four initiators. (As discussed below, loss of condenser vacuum was apparently included in the

"Other Transients" category, so the ATWS core melt probability for loss of condenser vacuum was added to the "Other Transients" core melt probability listed in table 2 of the PRA.) Thus, for these four initiators, it is this "total" core melt probability which is listed as the core melt probability in table 7.1, the Oconee 3 data sheet.

It should be noted that the ATWS core melt probabilities calculated above add up to 6.11×10^{-6} per R-yr which is slightly larger than, but very close to, the value of 6.0×10^{-6} per R-yr listed in table 2 of the PRA. This difference is most likely due to round off differences between myself and the PRA authors and should not be a cause for concern.

The second problem with table 2 of the PRA is that it apparently lumped together eight initiators into the category "Other Transients." By comparing the list of initiators established in Section 7.2 above with table 2 of the PRA, these eight initiators appear to be: steam line break, partial loss of feedwater, excessive feedwater flow, loss of condenser vacuum, spurious safety actuation, loss of ICS power bus, loss of switchgear 3TC, and spurious low pressurizer pressure signal. Since the PRA did not include the event trees, it was impossible for me to determine the extent to which each of these initiators contributed to the core melt probability listed in table 2 of the PRA. I was therefore forced to list these initiators here under the subcategory "Other Transients" in table 7.1 and display a subtotal for the core melt probability. As mentioned above, the value shown in table 7.1 is the value from table 2 of the PRA plus the ATWS core melt probability due to loss of condenser vacuum since this initiator is apparently one of the "Other Transients."

The remaining internal initiator core melt probabilities listed in table 7.1 were taken directly from table 2 of the PRA.

The core melt probabilities due to external initiators are listed in table 3 on page 16 of the PRA. Again, the event trees used to obtain these values were not included in the copy of the PRA which was available to me, so there was no way for me to trace these results. The only value missing from table 3 was a core melt probability due to aircraft crashes.

This omission was apparently the belief of the PRA authors that their value calculated for aircraft induced core melt was negligible. However, for the sake of completeness and consistency with the other PRAs that I studied, I estimated that the conditional probability of core melt given an aircraft crash at the Oconee site would be 0.10. Since the initiating frequency for aircraft crashes at Oconee 3 was estimated to be 2.5×10^{-9} per year, I estimated that the core melt probability at Oconee 3 due to aircraft crashes was 2.5×10^{-10} per R-yr, which is the value listed in table 7.1. The core melt probabilities listed in table 7.1 for the remaining external initiators were taken directly from table 3 of the PRA.

7.5 ATWS Core Melt Probabilities

As mentioned above, the ATWS analysis presented in Appendix E of the PRA was a scoping analysis. The PRA authors developed relatively simplistic event trees with conservative assumptions to obtain a conservative estimate of the ATWS core melt probability. They only considered the four initiators which are considered to be the dominant initiators in ATWS sequences: loss of condenser vacuum, turbine trip, loss of main feedwater, and loss of off-site power. The event trees listed on pages E-15 through E-18 of the PRA calculate the ATWS core melt probabilities for each of these initiators. I traced these calculations and found the event trees to be reasonable and the values to be appropriate, and have therefore accepted the PRA values and have listed them in table 7.1, the Oconee 3 data sheet.

7.6 Contribution of ATWS to Core Melt

From table 7.1, the Oconee 3 data sheet, we see that the contribution of ATWS events at Oconee 3 is 2.41%. This value is approximately one-fourth of the corresponding value for Surry 1, the WASH-1400 PWR, and it is so low as to suggest that ATWS events at Oconee 3 do not pose an undue risk to the health and safety of the public. The core melt probability due to ATWS events at Oconee 3 is 6.11×10^{-6} per R-yr, which is the same

order of magnitude as the Surry I ATWS core melt probability of 4.15×10^{-6} per R-yr and the NUREG-0460 safety objective of 1×10^{-6} per R-yr. If we take into account that the ATWS analysis at Oconee 3 was a scoping analysis and that the actual probability of an ATWS event leading to core melt at Oconee 3 is undoubtedly lower, we can make a very good argument that ATWS events at Oconee 3 are acceptable.

We also see from table 7.1 that the major contributors to the ATWS core melt probability at Oconee 3 are turbine trip (81.83%) and loss of main feedwater (10.97%). However, these two initiators account for only 3.43% of the total core melt probability, suggesting that modifications to lower the core melt probability due to ATWS events would have a negligible effect on the overall core melt probability at Oconee 3.

TABLE 7.1 - OCONEE UNIT 3 DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per R-YR)	% OF TOTAL Pcm	Pcm/ATWS (per R-YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	9.30E-04	9.00E-06	3.56%	0E+00	0.00%	0.00%
SMALL LOCA	3.00E-03	6.10E-06	2.41%	0E+00	0.00%	0.00%
INTERFACING SYST. LOCA	1.40E-07	1.40E-07	0.06%	0E+00	0.00%	0.00%
REACTOR VESSEL RUPTURE	1.10E-06	1.10E-06	0.43%	0E+00	0.00%	0.00%
LOCA TOTALS	3.93E-03	1.63E-05	6.45%	0E+00	0.00%	0.00%
STM. GEN. TUBE RUPTURE	8.60E-03	2.70E-06	1.07%	0E+00	0.00%	0.00%
FW LINE BREAK (LARGE)	9.30E-04	4.80E-06	1.90%	0E+00	0.00%	0.00%
LOSS OF MAIN FEEDWATER	6.40E-01	1.87E-06	0.74%	6.70E-07	10.97%	0.26%
REACTOR/TURBINE TRIP	4.90E+00	6.80E-06	2.69%	5.00E-06	81.83%	1.98%
LOSS OF OFF-SITE POWER	1.70E-01	2.65E-06	1.05%	2.50E-07	4.09%	0.10%
LOSS OF LP SERV. WATER	4.00E-03	1.30E-05	5.14%	0E+00	0.00%	0.00%
LOSS OF INSTR. AIR	1.70E-01	3.20E-06	1.26%	0E+00	0.00%	0.00%
OTHER TRANSIENTS:						
STEAM LINE BREAK	3.00E-03	N/A	N/A	N/A	N/A	0.00%
PARTIAL LOSS OF FW	6.90E-01	N/A	N/A	N/A	N/A	0.00%
EXCESSIVE FW FLOW	9.20E-02	N/A	N/A	N/A	N/A	0.00%
LOSS OF CONDENSER VAC.	2.10E-01	N/A	N/A	1.90E-07	3.11%	0.08%
SPUR. SAFE. ACTUATION	1.00E-02	N/A	N/A	N/A	N/A	0.00%
LOSS OF ICS POWER BUS	2.00E-02	N/A	N/A	N/A	N/A	0.00%
LOSS OF SWITHGEAR 3TC	5.40E-03	N/A	N/A	N/A	N/A	0.00%
SPUR. LOW PRESSURIZER PRESSURE SIGNAL	4.40E-02	N/A	N/A	N/A	N/A	0.00%
SUBTOTAL: OTHER TRANS.	1.07E+00	2.79E-06	1.10%	1.90E-07	3.11%	0.08%
TRANSIENT TOTALS	6.97	3.78E-05	14.94%	6.11E-06	100.00%	2.41%
INTERNAL TOTALS	6.97	5.42E-05	21.39%	6.11E-06	100.00%	2.41%
SEISMIC EVENTS	4.79E-04	6.30E-05	24.89%	0E+00	0.00%	0.00%
FIRES	4.60E-02	1.00E-05	3.95%	0E+00	0.00%	0.00%
TORNADOS	3.50E-05	1.30E-05	5.14%	0E+00	0.00%	0.00%
AIRCRAFT CRASHES	2.50E-09	2.50E-10	0.00%	0E+00	0.00%	0.00%
INTERNAL FLOODS	3.04E-02	8.80E-05	34.76%	0E+00	0.00%	0.00%
EXTERNAL FLOODS	2.50E-05	2.50E-05	9.88%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	7.69E-02	1.99E-04	78.61%	0E+00	0.00%	0.00%
GRAND TOTALS	7.05	2.53E-04	100.00%	6.11E-06	100.00%	2.41%

MEAN RPS UNRELIABILITY 3.00E-05/d

CHAPTER 8 - ZION UNIT 1

8.0 Introduction

The *Zion Probabilistic Safety Study (PSS)* was published in 1981 in the aftermath of the Three Mile Island II accident. The study estimated the core melt and public risk associated with accidents at Zion Unit 1. The methodology used was similar to the WASH-1400 methodology, but many of the questionable techniques used in WASH-1400 were replaced with more acceptable approaches. The information presented in this chapter is derived from the Zion PSS.

8.1 Plant Description

Zion Unit 1 is located in Illinois on the western shore of Lake Michigan, about 40 miles north of Chicago, Illinois and about 42 miles south of Milwaukee, Wisconsin. It is a Westinghouse pressurized water reactor (PWR) with four coolant loops. Its power rating is 3250 MWt (1040 MWe). The large dry containment structure is made of reinforced concrete with a steel lining.

8.2 The Initiator Frequencies

Table 1.4-1 on page 1.4-3 of the Zion PSS gives a list of internal initiators which were determined for Zion Unit 1 by using generic plant data to obtain a prior distribution with plant specific data used to get a posterior distribution. Since these values appeared consistent with the values used in the other PRAs I studied, I chose to accept these values, which are the frequencies listed for the internal initiators in table 8.1, the Zion Unit 1 data sheet.

For external events, specific initiating frequencies were not always calculated in the PSS. I therefore developed initiating frequencies for external initiators as accurately as I could from the data presented in the PSS. The following paragraphs describe how I arrived at each external initiating frequency that is listed in table 8.1.

Section 7.9.1 of the PSS is a Dames and Moore seismicity study for the Zion site. This

study did not explicitly calculate an initiating frequency for seismic events. However, table 3 in the study listed several probabilities of exceeding certain peak ground accelerations. In an effort to be consistent with the other PRAs studied, I assumed that the initiating frequency of seismic events was equal to the composite Bayesian estimate of the probability of exceeding 0.10g peak ground acceleration. This value was listed as 3.9×10^{-4} per year in table 3 of the Dames and Moore study and is also listed here in table 8.1 as the frequency of seismic events.

The fire analysis for Zion 1 is given in Section 7.3 of the PSS. This analysis used American Nuclear Insurers data to obtain a gamma distribution for fire frequencies. The two main areas of concern were the cable spreading room and the auxiliary electrical equipment room. These areas had fire initiation frequencies of 7.2×10^{-3} per year and 4.1×10^{-3} per year, respectively. I therefore took the total fire initiator frequency to be the sum of these two frequencies, i.e. 1.13×10^{-2} per year, which is the value listed in table 8.1, the Zion Unit 1 data sheet.

The initiating frequency for tornados used in the PSS was based on tornado risk evaluation reports and tornado missile simulation reports. The PSS authors determined the frequency of tornados, the frequency of tornado generated missiles hitting critical structures, and the frequency of loss of off-site power due to tornados. These values were calculated to be 2.8×10^{-5} , 7.0×10^{-5} , and 1.0×10^{-3} per year, respectively. I accepted the sum of these frequencies (1.1×10^{-3} per year) as the initiating frequency for tornado events at Zion 1 and have listed this value in table 8.1, the Zion Unit 1 data sheet.

The effect of aircraft crashes at Zion is addressed in Section 7.6 of the PSS. The PSS authors used an NRC algorithm which takes into account the area of the plant structures (target area), the number of takeoffs and landings in the plant area, and the crash density function for the aircraft and airports involved. The authors calculated a crash frequency

of 2.0×10^{-7} per year which I accepted as the initiating frequency for aircraft crashes at Zion 1. This value is listed as such in table 8.1, the Zion Unit 1 data sheet.

The flood analysis at Zion 1 is presented in Section 7.4 of the PSS. The authors determined that floods from external sources (i.e. those from rain and lake swelling) would be negligible. The main sources of internal flooding were the component cooling system, the service water system and the fire fighting system. The PSS authors used failure data in a formula which assumed that 99 percent of the failures were leaks and that there was a probability of detecting the leak before flooding occurred. The flood initiation frequency calculated in the PSS was 4.7×10^{-6} per year, which is the value listed in table 8.1, the Zion Unit 1 data sheet.

On page 7.8-1 of the PSS, the authors estimated the frequency of turbine missiles to be 1.0×10^{-4} per year. Since there has been very little research in the area of turbine missiles, I assumed that this value was acceptable and have listed it in table 8.1, the Zion Unit 1 data sheet.

8.3 The Reactor Protection System

The PSS authors obtained a reactor protection system unavailability value of 1.8×10^{-4} per demand by using a fault tree analysis. This analysis included failure of multiple control rods to insert, trip breaker failures, logic channel failures (relay failures), wiring faults, unavailability due to testing and a common cause failure value arbitrarily taken to be equal to the failure rate of a single instrument channel. The authors did not explicitly state separate probabilities for electrical and mechanical failures.

As with all common cause failure analyses, the PSS common cause failure analysis is rather arbitrary and yields a common cause failure rate of 4.61×10^{-7} per demand. However, since the total calculated unavailability value is an order of magnitude greater than the NUREG-0460 value of 3.0×10^{-5} per demand, I decided to accept the PSS authors'

RPS unavailability value of 1.8×10^{-4} per demand, which is the value shown in table 8.1, the Zion Unit 1 data sheet.

8.4 Core Melt Probabilities

Table 1.4-2 of the PSS shows a split fraction matrix for the internal initiators. This matrix is basically a listing of conditional core melt probabilities given the particular initiating events. These split fractions, or conditional core melt probabilities, were derived through event tree analysis in the PSS. I reviewed each event tree and traced many of the results listed in table 1.4-2. I found the event trees to be reasonable and the results listed in table 1.4-2 to be consistent with the event trees. I therefore accepted the values in table 1.4-2 and used them as I describe below to calculate the core melt probabilities for the internal initiators.

Each matrix entry is the conditional probability of getting a particular "plant state" involving core melt given a specific initiating event. Thus if we sum these split fractions horizontally in table 1.4-2 of the PSS (i.e. across the various plant states), we obtain the *total* conditional core melt probabilities for each of the internal initiators. I calculated these values and they are shown here in table 8.2. Once I calculated these conditional probabilities, I simply had to multiply each value by the corresponding initiator frequency to obtain the core melt probability for each initiator. These core melt probabilities are shown in table 8.1, the Zion Unit 1 data sheet.

The effects of seismic events at Zion are analyzed in Section 7.2 of the PSS. The basic approach used a fault tree which is shown in figure 7.2-3 of the PSS. I crudely traced the PSS results by combining the seismic initiator frequencies listed in table 8.8-1 of the PSS to obtain a single weighted initiator frequency for each ground acceleration listed. I then combined tables 7.2-4a through 7.2-4e with equal weight to obtain a conditional probability of plant state SE (seismic initiated core melt) for each ground acceleration. By multiplying the initiator frequencies for each acceleration by the corresponding conditional probability

for plant state SE, I obtained core melt probabilities for various sized earthquakes as represented by the various ground accelerations. The sum of these probabilities was 5.73×10^{-6} per R-yr which agreed very well with the PSS seismic core melt probability of 5.6×10^{-6} per R-yr. Since my method was much cruder than that used in the PSS, I chose to accept the PSS value of 5.6×10^{-6} per R-yr as the seismic core melt probability which is the value listed in table 8.1, the Zion Unit 1 data sheet.

The effects of fires on core melt were presented in Section 7.3 of the PSS. The two main areas of concern were the cable spreading room and the auxiliary electrical equipment room. The respective core melt probabilities due to fires in these areas are given on page 7.3-1 of the PSS as 1.8×10^{-6} per R-yr and 2.8×10^{-6} per R-yr for a total core melt probability due to fires of 4.6×10^{-6} per R-yr. The analysis seemed somewhat cursory, but the PSS authors maintained that their intention was to obtain a conservative estimate of the core melt probability due to fires. Since the results were difficult to trace, I had little choice but to accept their value of 4.6×10^{-6} per R-yr as the core melt probability due to fires at Zion 1 which is the value shown here in table 8.1, the Zion Unit 1 data sheet.

The effects of tornados and tornado generated missiles at Zion 1s presented in Section 7.5 of the PSS. The authors estimated that the additional probability of off-site power loss would be 1.0×10^{-3} per year, which they determined would have a negligible effect on the total core melt probability. From table 8.2 in here we see that the conditional probability of core melt due to loss of off-site power is 3.76×10^{-6} per reactor. Thus, the increased frequency of loss of off-site power due to tornados would lead to an increase in core melt probability of 3.76×10^{-9} per R-yr, which is indeed negligible when compared to the total core melt probability of 5.25×10^{-5} per R-yr listed here in table 8.1. However, as will be seen below, it is not negligible when compared to the core melt probability due to tornados. Thus, to be consistent with the other PRAs studied, I decided to include this value as part

of the core melt probability due to tornados.

The PSS authors obtained the probability of a tornado generated missile hitting a safety structure from a generic study done in 1978, which determined this value to be 2.0×10^{-6} per year. The authors then estimated that the probability of a core melt given such a hit was between 10^{-2} and 10^{-4} per reactor. Thus the PSS authors concluded that the core melt probability due to tornado generated missiles is between 2.0×10^{-8} and 2.0×10^{-10} per R-yr. To be conservative, I assumed that the core melt probability due to tornado generated missiles was 2.0×10^{-8} per R-yr. After adding the value calculated above for the core melt due to loss of off-site power, I obtained a total core melt probability due to tornados of 2.38×10^{-8} per R-yr which is the value listed in table 8.1, the Zion Unit 1 data sheet.

Section 7.6 of the PSS addresses the effects of aircraft crashes at Zion. After calculating the probability of an aircraft crash at the site, the authors apparently assumed that the conditional core melt probability was much less than 10^{-1} since they state that the core melt probability due to aircraft crashes is much less than 10^{-8} per R-yr. Again, to be conservative, I assumed that the conditional probability equal to 10^{-1} which yielded a core melt due probability due to aircraft crashes of 2.0×10^{-8} per R-yr, which is the value shown in table 8.1, the Zion Unit 1 data sheet.

The flood analysis is presented in Section 7.4 of the PSS. The PSS authors felt that external flooding would be negligible and that flooding of the turbine building would not even threaten any safety related systems. They also determined that flooding of the auxiliary building alone would not lead to core melt since a turbine driven auxiliary feedwater pump and feed and bleed capability would still be available. Flooding of the containment building was found to be unlikely since at least two reactor water storage tank volumes of water would need to accumulate to cause problems, and the PSS authors felt that any flooding

would be detected long before this much water could accumulate. Thus, the PSS authors did not calculate a core melt probability due to flooding. However, for comparison purposes in this report, I felt it desirable to estimate a value. I therefore assumed that the conditional probability of core melt given a flood was 10^{-2} per reactor, which when multiplied by the initiating frequency for floods yielded a total probability of core melt due to flooding of 4.7×10^{-8} per R-yr, which is the value listed in table 8.1, the Zion Unit 1 data sheet.

The analysis of turbine missiles is given in Section 7.8 of the PSS. The PSS authors estimated that the probability of a turbine missile hitting a "vital" target is 1×10^{-7} per R-yr and that the probability of core melt from such an event is even less. In order to get a core melt probability for turbine missiles, I assumed the conditional probability of core melt given such a hit was 10^{-1} , so that the probability of core melt due to turbine missiles at Zion is 1×10^{-8} per R-yr, which is the value shown in table 8.1, the Zion Unit 1 data sheet.

8.5 ATWS Core Melt Probabilities

The ATWS core melt split fractions are presented in table 1.3.4.15-2 on page 1.3-364 of the PSS. As with the total core melt probability split fractions, these values were derived through event tree analysis and represent the conditional probability of getting a particular "plant state" involving an ATWS core melt given a specific initiating event. By summing horizontally in table 1.3.4.15-2 (i.e. across the various plant states), I obtained the total conditional ATWS core melt probability for each internal initiator. These values are listed here in table 8.3. To obtain the ATWS core melt probabilities, I multiplied the initiator frequencies by the corresponding conditional core melt probabilities. These ATWS core melt probabilities are listed in table 8.1, the Zion Unit 1 data sheet.

It should be noted that the PSS authors did not feel that external events would initiate an ATWS sequence.

I should also note that there was a revised ATWS analysis presented in revision

number one to the PSS which yielded an ATWS core melt probability of 9.48×10^{-8} per R-yr, but I rejected this analysis for several reasons. First, this result is almost two orders of magnitude lower than the ATWS core melt probabilities calculated in the other PSSs, yet the Zion RPS unavailability of 1.8×10^{-4} per demand that was used in the revised analysis is almost a full order of magnitude greater than the unavailability values used in the other studies. Such large discrepancies in the results suggest that there were major differences in the assumptions and/or methodologies used in the revised Zion analysis and the other PSSs since the Zion 1 plant is not significantly different in design from the other PWRs involved.

Second, the revised analysis included several additional event headings in the revised ATWS event tree, including the heading "OP-6: Rods in by One Minute" which accounts for the operators being able to manually scram the reactor with a manual trip button and/or being able to insert the rods manually. The revision authors assumed a failure probability of 8.0×10^{-3} for this branch point. Since success at this branch point by-passes many core melt sequences, this single assumption more than any other accounts for the huge reduction in the revised ATWS core melt probability. However, the revision authors did not specifically explain how the operators would achieve manual shutdown in the event of an ATWS. I therefore felt that their assumption of 8.0×10^{-3} for the branch point failure probability was one to two orders of magnitude too low. If one were to increase this value by one to two orders of magnitude, the revised event tree would yield an ATWS core melt probability on the order of 10^{-6} per R-yr, which is consistent with the other PSSs and the original Zion analysis.

Thus, in light of these two points, I rejected the revised ATWS analysis and determined the ATWS core melt probabilities from the original ATWS analysis as described above.

8.6 Contribution of ATWS to Core Melt

From table 8.1, the Zion Unit 1 data sheet, we see that the contribution of ATWS events at Zion 1 is 14.81%. This value is 55% greater than the corresponding value for Surry I, the WASH-1400 PWR. The core melt probability due to ATWS events at Zion 1 is 7.77×10^{-6} per R-yr, which is nearly double the Surry I ATWS core melt probability of 4.15×10^{-6} per R-yr and almost eight times greater than the NUREG-0460 safety objective of 1×10^{-6} per R-yr. These numbers suggest that ATWS is a significant contributor at Zion 1.

However, we should not totally discount the fact that the PSS authors performed a revised ATWS analysis since they obviously felt that the original analysis was inadequate. Although I rejected the specific revised analysis presented, a revised analysis with acceptable assumptions could show that the actual ATWS core melt probability is less than 7.77×10^{-6} per R-yr, and that ATWS events are not major contributors to core melt at Zion 1. I feel that the ATWS core melt probability presented here is an upper bound limit.

We also see from table 8.1 that the major contributors to the ATWS core melt probability at Zion 1 are loss of main feedwater (52.50%) and turbine trip (37.31%). However, these two initiators account for only 15.09% of the total core melt probability, suggesting that modifications to lower the core melt probability due to ATWS events would have a limited effect on the overall core melt probability at Zion 1.

TABLE 8.1 - ZION UNIT 1 DATA SHEET

INITIATOR	FREQUENCY (per YR)	Pcm (per R-YR)	% OF TOTAL Pcm	Pcm/ATWS (per R-YR)	% OF TOTAL Pcm/ATWS	% OF TOTAL Pcm
LARGE LOCA	9.40E-04	6.20E-06	11.81%	0E+00	0.00%	0.00%
MEDIUM LOCA	9.40E-04	5.33E-06	10.16%	0E+00	0.00%	0.00%
SMALL LOCA	3.54E-02	1.64E-05	31.25%	2.75E-08	0.35%	0.05%
INTERFACING SYST. LOCA	1.05E-07	1.05E-07	0.20%	0E+00	0.00%	0.00%
LOCA TOTALS	3.73E-02	2.80E-05	53.42%	2.75E-08	0.35%	0.05%
STM. GEN. TUBE RUPTURE	2.44E-02	2.24E-07	0.43%	1.09E-14	0.00%	0.00%
STM. LINE BREAK INSIDE CONTAINMENT	9.40E-04	5.99E-09	0.01%	7.35E-10	0.01%	0.00%
STM. LINE BREAK OUTSIDE CONTAINMENT	9.40E-04	5.02E-09	0.01%	7.35E-10	0.01%	0.00%
LOSS OF MAIN FEEDWATER	5.17E+00	4.62E-06	8.80%	4.08E-06	52.50%	7.77%
TRIP OF ONE MSIV	2.52E-01	4.03E-08	0.08%	0E+00	0.00%	0.00%
LOSS OF RCS FLOW	3.58E-01	1.61E-06	3.07%	2.55E-07	3.28%	0.49%
CORE POWER EXCURSION	2.28E-02	2.83E-15	0.00%	2.35E-15	0.00%	0.00%
TURBINE TRIP	3.69E+00	3.30E-06	6.29%	2.90E-06	37.31%	5.53%
TURBINE TRIP: LOOP	5.76E-02	2.17E-07	0.41%	4.21E-09	0.05%	0.01%
TURBINE TRIP:LOSS OF SERVICE WATER	9.40E-04	1.12E-09	0.00%	7.64E-10	0.01%	0.00%
SPURIOUS SAFETY INJECT.	6.36E-01	3.70E-06	7.05%	5.03E-07	6.47%	0.96%
REACTOR TRIP	3.77E+00	4.18E-07	0.80%	0E+00	0.00%	0.00%
TRANSIENT TOTALS	13.98	1.41E-05	26.95%	7.74E-06	99.65%	14.76%
INTERNAL TOTALS	14.02	4.22E-05	80.37%	7.77E-06	100.00%	14.81%
SEISMIC EVENTS	3.90E-04	5.60E-06	10.67%	0E+00	0.00%	0.00%
FIRES	1.13E-02	4.60E-06	8.77%	0E+00	0.00%	0.00%
TORNADOS	1.10E-03	2.38E-08	0.05%	0E+00	0.00%	0.00%
AIRCRAFT CRASHES	2.00E-07	2.00E-08	0.04%	0E+00	0.00%	0.00%
FLOODS	4.70E-06	4.70E-08	0.09%	0E+00	0.00%	0.00%
TURBINE MISSILES	1.00E-04	1.00E-08	0.02%	0E+00	0.00%	0.00%
EXTERNAL TOTALS	1.29E-02	1.03E-05	19.63%	0E+00	0.00%	0.00%
GRAND TOTALS	14.03	5.25E-05	100.00%	7.77E-06	100.00%	14.81%

MEAN RPS UNRELIABILITY 1.80E-04/d

TABLE 8.2 - CONDITIONAL CORE MELT PROBABILITIES FOR ZION 1 INTERNAL INITIATORS

INITIATOR	CONDITIONAL CORE MELT PROBABILITY
LARGE LOCA	6.60E-03
MEDIUM LOCA	5.67E-03
SMALL LOCA	4.62E-04
INTERFACING SYST. LOCA	1.00E+00
STM. GEN. TUBE RUPTURE	9.19E-06
STM. LINE BREAK INSIDE CONTAINMENT	6.37E-06
STM. LINE BREAK OUTSIDE CONTAINMENT	5.34E-06
LOSS OF MAIN FEEDWATER	8.94E-07
TRIP OF ONE MSIV	1.60E-07
LOSS OF RCS FLOW	4.50E-06
CORE POWER EXCURSION	1.24E-13
TURBINE TRIP	8.94E-07
TURBINE TRIP: LOOP	3.76E-06
TURBINE TRIP: LOSS OF SERVICE WATER	1.19E-06
SPURIOUS SAFETY INJECT.	5.82E-06
REACTOR TRIP	1.11E-07

TABLE 8.3 - CONDITIONAL ATWS CORE MELT PROBABILITIES FOR ZION 1

INITIATOR	COND. ATWS CORE MELT PROBABILITY
LARGE LOCA	0E+00
MEDIUM LOCA	0E+00
SMALL LOCA	7.78E-07
INTERFACING SYST. LOCA	0E+00
STM. GEN. TUBE RUPTURE	4.46E-13
STM. LINE BREAK INSIDE	7.82E-07
CONTAINMENT	
STM. LINE BREAK OUTSIDE	7.82E-07
CONTAINMENT	
LOSS OF MAIN FEEDWATER	7.90E-07
TRIP OF ONE MSIV	0E+00
LOSS OF RCS FLOW	7.12E-07
CORE POWER EXCURSION	1.03E-13
TURBINE TRIP	7.86E-07
TURBINE TRIP: LOOP	7.31E-08
TURBINE TRIP: LOSS OF	8.13E-07
SERVICE WATER	
SPURIOUS SAFETY INJECT.	7.91E-07
REACTOR TRIP	0E+00

CHAPTER 9 - COMPARATIVE DISCUSSION

9.0 Introduction

The preceding seven chapters discussed how I obtained the initiator frequencies, the core melt probabilities and the ATWS core melt probabilities for the various initiators at the eight plants involved in this study. In this chapter I will take a closer look at these numbers, in particular comparing the results from plant to plant to identify similarities as well as differences. Through such an approach I hope to determine if there are any apparent generic conclusions regarding the contribution of ATWS events at nuclear power plants in the United States.

9.1 Plant Description

Table 9.1 presents a summary of the various plant descriptions. The biggest similarity is between the WASH-1400 BWR (Peach Bottom) and Limerick Unit I. Both are General Electric BWR/4 design, both have a licensed thermal power of 3293 MWt and both employ inert containment atmospheres. Their designed net electric output differs by only 10 MWe (less than one percent of either's total designed net electric power). The only significant difference is that Peach Bottom has a Mark I containment structure while Limerick has a Mark II type structure.

In stark contrast to these two BWRs is the Big Rock Point BWR. This plant is significantly older than Peach Bottom and Limerick and uses the original General Electric BWR design. The power ratings at Big Rock Point are more than an order of magnitude lower than those at Peach Bottom and Limerick and the containment is a spherical steel structure rather than the more common Mark I and II type designs. The difference in the power ratings is significant since the radioactive inventory of the core is proportional to the power generated.

Although the Big Rock Point plant is significantly different than the other two BWRs studied, the separate PRAs for each plant take these different designs into account.

However, since I am only using the PRAs to obtain core melts and not consequences, the differences in power ratings (and hence radioactive inventories) must be taken into account. The implications of the different power ratings will be discussed in the sections below on core melt probability and ATWS core melt probability.

The five PWRs studied all have a large dry containment structure. The thermal power ratings range from 2441 MWt to 3250 MWt, while the designed net electric power ranges from 788 MWe to 1040 MWe. Only one unit, Oconee 3, is a Babcock and Wilcox design while the other four units are by Westinghouse. The smallest unit is the WASH-1400 PWR (Surry) and is a Westinghouse three loop design. The other three Westinghouse units are of the four loop design. The largest unit is Zion 1, a Westinghouse four loop system.

The differences among the PWR power ratings was significant in that the largest PWR power rating was 33 percent larger than the smallest PWR power rating. However, I felt that these differences were not significant enough to warrant special treatment in the discussions below, as was the case with the BWRs, since the individual reference studies took the specific plant designs into account.

9.2 The Reactor Protection System

Table 9.2 lists the RPS unavailability values used for each of the plants in this study as well as the values proposed in WASH-1270 and NUREG-0460. The NUREG-0460 value was chosen as a baseline value, and table 9.2 compares each value to the NUREG-0460 value. Two plants (Peach Bottom and Indian Point 2) had values lower than NUREG-0460, two plants (Limerick I and Oconee 3) had values equal to the NUREG-0460 value and the remaining four plants (Surry I, Big Rock Point, Indian Point 3 and Zion 1) had values greater than the NUREG-0460 value. Those plants with RPS unavailabilities equal to or greater than the NUREG-0460 value of 3.0×10^{-5} per demand were deemed acceptable since the results achieved would be at least as conservative as if the NUREG-0460 value had been used. However, the two plants with values lower than the NUREG-0460 value

need closer attention.

Peach Bottom Unit II was the WASH-1400 BWR. The technique used in WASH-1400 to determine the RPS unavailability received much criticism following the report's publication. Specifically the use of the log-normal mean between complete independence and complete dependence of instrument miscalibrations as the common cause failure contribution to RPS unavailability was viewed as completely wrong and an under estimation. Consequently, the actual RPS unavailability value at Peach Bottom may be higher than the value which was used (1.3×10^{-5} per demand). In light of this situation, in the discussions below I will consider the possibility that the RPS unavailability at Peach Bottom is greater than 1.3×10^{-5} per demand. Specifically I will assume that this greater value is equal to the NUREG-0460 unavailability of 3.0×10^{-5} per demand.

The Indian Point Unit 2 RPS unavailability was calculated using fault tree analysis. Although the value of 2.01×10^{-5} per demand is lower than the NUREG-0460 value of 3.0×10^{-5} per demand, it is the same order of magnitude and very close to the NUREG-0460 value. Also, the fault tree methodology used in the Indian Point PRA is very similar to the methodology used in the other PRAs studied and has found wide acceptance today. Consequently, I felt that there was no need to revise the Indian Point 2 data. However, for the sake of consistency I decided that in the discussions below I would consider the possibility that the Indian Point 2 RPS unavailability was equal to the NUREG-0460 value. As will be seen this will have very little impact on the Indian Point 2 results.

9.3 The Initiator Frequencies

Table 9.3 summarizes the initiator frequencies for the plants studied in this study. The majority of the studies combined generic plant data with plant specific data to obtain the internal initiator frequencies listed. The exceptions were the Big Rock Point PRA which used plant specific data exclusively, the Limerick PRA which used generic BWR data for all but the loss of off-site power frequency which was plant specific, and the WASH-1400

plants (Peach Bottom and Surry) which used generic data. The only study to explicitly calculate external event frequencies was the Big Rock Point PRA. The frequencies listed for the external initiators for the other studies were my best estimates based on the data presented in each study. From table 9.3 we see that there is generally very good agreement between the initiator frequencies used in the various studies. There are, however, several exceptions which warrant discussion.

The first exception is the Big Rock Point study which used many initiator frequencies that were one to two orders of magnitude below those used in the other studies. The specific initiators involved include the small, medium and large LOCA initiators, steam line break outside containment, loss of main feedwater, MSIV closure, loss of condenser and fires. Additionally, no frequencies were reported for floods and turbine missiles.

There are several explanations for these lower frequencies at Big Rock Point. First, four of the six plants listed in table 9.3 are PWRs and Big Rock Point is a BWR. Since there are significant design differences between PWRs and BWRs, one might expect significant differences in initiator frequencies. Second, the Big Rock Point data is plant specific whereas the other studies used generic plant data or a combination of generic data and plant specific data. Third, Big Rock Point is an older plant with a simpler and therefore perhaps more reliable design.

The omission of floods and turbine missiles as initiators at Big Rock Point reflects the PRA authors' belief that these initiators are insignificant at Big Rock Point. Although this may be the case, these omissions create problems when comparing Big Rock Point to the other plants studied since the specific core melt contributions of floods and turbine missiles were not calculated at Big Rock Point. As will be seen later, at all of the other plants, floods and turbine missiles did not initiate any ATWS core melt sequences. Therefore in the discussions below I will assume that floods and turbine missiles contribute insignificantly to core melt and ATWS core melt at Big Rock Point.

The second exception is the Indian Point Unit 2 study which calculated a total transient

frequency that was more than double the average total transient frequency listed in table 9.3. The specific transients which contribute most to this difference are loss of main feedwater, MSIV closure, turbine trip and reactor trip. As seen from table 9.3, these are in general the most frequent transients for each of the plants listed. Since the Indian Point 2 study combined plant specific data with generic plant data, it is entirely possible that these "frequent" transients occurred with even greater frequency at Indian Point 2 and hence yielded a higher total transient frequency. For this reason (and also because higher initiator frequencies yield more conservative results) I decided that there was no need for special treatment of the Indian Point 2 frequencies and results to this point.

The third exception is the Oconee 3 study where the small LOCA (and consequently the LOCA total frequency) was an order of magnitude lower than in the the other studies and where the total transient frequency was slightly lower than the average listed in table 9.3. There can be several explanations for these differences. First, the Oconee 3 PRA classified LOCAs as either large or small whereas the other PRAs used three classifications: small, medium and large. This difference in methodology could have resulted in the lower small LOCA frequency at Oconee 3. Second, Oconee 3 is a Babcock and Wilcox reactor while the other PWRs studied are Westinghouse reactors. Since there are significant design differences between these two manufacturers (i.e. number of coolant loops, steam generator design), it is entirely possible that the Oconee 3 frequencies are lower than those for the other plants listed in table 9.3. Additionally, the LOCA frequencies listed for Oconee 3 are the same order of magnitude as those for Surry I, the WASH-1400 PWR. In light of these latter two points, I felt that special treatment of the Oconee 3 results was not necessary.

9.4 Core Melt Probabilities

Table 9.4 compares the core melt probabilities calculated for the eight plants involved in this study. In general there is very good agreement with the core melt probabilities for

any given initiator being the same order of magnitude from plant to plant. However, there are some significant differences which require further discussion. The most notable of these differences involve the extremely low LOCA values at Peach Bottom and Limerick and the extremely low external values for Peach Bottom and Surry, the WASH-1400 plants. It is also notable that the Big Rock Point values are on the high side for almost every initiator category, even though the initiator frequencies at Big Rock Point were found to be on the low side. These discrepancies are discussed in more detail below.

As just mentioned, the core melt probability values for external events at Peach Bottom Unit II and Surry Unit I (the WASH-1400 plants) were significantly lower than the external event core melt probabilities of the other plants involved in this study. This result should come as no surprise since WASH-1400 was heavily criticized for its inadequate treatment of external events and their effect on core melt. The evidence of this inadequate treatment is seen in table 9.4 where fires and floods were not even considered by the WASH-1400 authors and where the contributions from tornados, aircraft crashes and turbine missiles were deemed all but impossible. The effects of these lower external probabilities on this study are unknown. However, in all of the other studies except for Limerick, external events did not contribute to the ATWS core melt probability. Therefore, a larger external core melt probability at Surry or Peach Bottom would most likely increase the total core melt probability without increasing the ATWS core melt probability. Consequently, the percent contribution of ATWS to the total core melt probability would be lower than that calculated in this study. Thus it would be reasonable to assume that the contribution of ATWS calculated with the lower external core melt numbers presented here would be an upper limit. I assumed that this was the case and calculated the WASH-1400 ATWS contributions using the core melt frequencies presented in table 9.4.

The second set of discrepancies in table 9.4 are the low LOCA values for Peach Bottom and Limerick. These values are two to three orders of magnitude lower than the values

listed for the other plants. The most probable explanation is that these two plants are BWRs which differ significantly in design from PWRs and as such could have much lower LOCA core melt probabilities. However, the other BWR in this study, Big Rock Point, has LOCA core melt probabilities that are very similar to the PWRs studied here. This difference could be explained by the fact that Big Rock Point is an older plant and as such could be significantly different in design from Limerick and Peach Bottom to yield higher LOCA core melt probabilities. This theory is supported by the fact that Big Rock Point shows a significant core melt contribution from the interfacing systems LOCA - an initiator that is typically considered a PWR initiator only. In any case, the question to be answered at this point is what effect do these lower values have on this study? As will be seen in the next section, LOCA initiators have a negligible contribution to the ATWS core melt probability. Therefore, a larger LOCA core melt probability at Peach Bottom or Limerick would most likely increase the total core melt probability without increasing the ATWS core melt probability. Consequently, the percent contribution of ATWS to the total core melt probability would be lower than that calculated in this study. Thus it would be reasonable to assume that the contribution of ATWS calculated with the lower LOCA core melt numbers presented here would be an upper limit. I assumed that this was the case and calculated the Peach Bottom and Limerick contributions using the core melt frequencies presented in table 9.4.

The third discrepancy in table 9.4 is that almost all of the Big Rock Point core melt values are higher than those for the other plants, even though Big Rock Point had consistently low initiator frequencies. This difference seems ironic at first until we remember that Big Rock Point is an older plant that lacks many of the safety features of newer plants. Also, the Big Rock Point PRA was one of the earliest PRAs, and as such may have used more conservative assumptions and methodology than some of the more recent studies. Both of these facts lead to higher calculated core melt frequencies at Big Rock Point than at some of the other plants of newer design and more realistic assumptions and

methodologies. In light of these points, I accepted the core melt probabilities presented in table 9.4 for Big Rock Point.

Perhaps a better way to look at the core melt probabilities is to consider the percent contribution of each initiator to the total core melt at each plant. Table 9.5 does just that. At first glance there are no apparent trends. But if we take a closer look and identify those initiators at each plant which contribute more than five percent to the total core melt at that plant we discover some interesting trends. If we exclude Peach Bottom and Surry (due to WASH-1400's inadequate treatment of external events), five of the six remaining plants studied show fires as significant (i.e. greater than five percent) contributors to core melt. Four of these plants (Big Rock Point, Limerick, Indian Point 2 and Indian Point 3) show contributions greater than 20 percent while two plants (Limerick and Indian Point 2) top the 50 percent mark. Undeniably, fires can pose a serious risk at nuclear power generation plants in the United States.

Continuing our examination of table 9.5, we see that three of the five PWRs studied (Surry, Indian Point 3 and Zion) show LOCAs as significant contributors to core melt. All three of these plants show contributions greater than 50 percent.

Three of the eight plants studied (Big Rock Point, Limerick and Indian Point 2) show loss of off-site power as a significant contributor with this initiator contributing more than ten percent to the total core melt at each plant.

And finally, three of the eight plants (Limerick, Oconee and Zion) list seismic events as a significant contributor to core melt, again with this initiator contributing more than ten percent of the total core melt at each plant.

In light of these facts, a very good argument could be made that regulatory efforts and modifications to plants should be directed at reducing the risks of core melt due to fires, LOCAs, loss of off-site power and seismic events since these events seem to be the largest contributors to core melt.

9.5 ATWS Core Melt Probabilities

Table 9.6 shows the contributions of the various initiators at each plant to the ATWS core melt probability for each plant (i.e. the core melt probability due to ATWS sequences only). In general there is very good agreement between the plants. Most of the plants calculated an ATWS core melt probability on the order of 10^{-6} per R-yr. Peach Bottom and Big Rock Point were slightly higher, on the order of 10^{-5} per R-yr.

The right hand column in table 9.6 shows the average ATWS core melt probability for all of the plants that calculated a value for a particular initiator. Big Rock Point had several initiators whose contributions were an order of magnitude greater than this average, specifically MSIV trip, turbine trip and loss of off-site power. Limerick had several initiators whose contributions were an order of magnitude less than the average (loss of main feedwater, turbine trip and loss of off-site power). Zion had two initiators whose contributions were two orders of magnitude below the average (loss of off-site power and steam generator tube rupture). The results for Big Rock Point are likely due to the fact that Big Rock Point is an older plant and lacks some safety features of the later plants. Limerick's results are probably due to the fact that Limerick is a BWR while four of the six plants involved are PWRs. Zion's results are due to the greater redundancy of off-site power supplies and apparently different assumptions about steam generator tube rupture.

Again, a better way to compare the data would be to look at the percent contributions of each initiator to the total ATWS core melt probability. Table 9.7 does just that. As with the overall core melt probability discussed in Section 9.4, if we identify those initiators in table 9.7 which contribute more than five percent to the ATWS core melt probability, we see some interesting trends. At six out of six plants, we see that turbine trips and loss of main feedwater are significant contributors. In fact, turbine trips account for anywhere from 27.46% (Limerick) to 81.83% (Oconee 3) of the ATWS core melt, while loss of main

feedwater accounts for anywhere from 8.96% (Big Rock Point) to 52.50% (Zion). MSIV trip was significant at both BWRs (Big Rock Point and Limerick) while loss of off-site power was significant at three of the six plants (Big Rock Point, Indian Point 2 and Indian Point 3). Only one plant, Limerick, showed any contributions to ATWS from external events, but the 40.03% contribution from seismic events is the largest single contributor to ATWS core melt at Limerick.

In light of these facts, if one wishes to reduce the probability of an ATWS event leading to core melt, one should start by looking more closely at turbine trip and loss of main feedwater events.

However, even though turbine trips and loss of main feedwater are the major contributors to core melt, if we return to table 9.5 we see that, in general, these initiators do not contribute significantly to the total core melt probability. Typically these initiators contribute a few percent to the total core melt probability, with only Zion showing significant contributions (8.80% for loss of main feedwater and 6.29% for turbine trip). These facts suggest that efforts and modifications to reduce the core melt probability due to ATWS events would have little if any effect on the overall core melt probability.

As mentioned in Section 9.2 above, I said I would consider the possibility that the RPS unavailabilities at Peach Bottom and Indian Point 2 were equal to the NUREG-0460 value of 3.0×10^{-5} . In these cases, the ATWS core melt probability for Peach Bottom would change from 1.01×10^{-5} per R-yr to 2.33×10^{-5} per R-yr, while the Indian Point 2 value would change from 2.21×10^{-6} per R-yr to 3.30×10^{-6} per R-yr. These differences are relatively small and would not significantly change the discussion presented above.

9.6 Contribution of ATWS to Core Melt

Table 9.8 shows the percent ATWS contribution of each initiator at each plant to the total core melt probability at each plant. If we identify contributions greater than five percent, we see problems at only three of the eight plants (Peach Bottom, Surry and Zion).

The five remaining plants show total ATWS contributions of less than five percent, with four of these plants (Big Rock Point, Indian Point 2, Indian Point 3 and Oconee) showing contributions less than three percent. Such results suggest that ATWS is not a major contributor to core melt at many commercial power plants in the United States, and that ATWS does not present an undue risk to the health and safety of the public.

The plants where ATWS is a significant contributor to core melt are Peach Bottom and Surry (the WASH-1400 plants) and Zion. The WASH-1400 contributions are in all likelihood artificially inflated due to the poor treatment of external events. If we return to table 9.4, we see that proper treatment of external events could easily double the total core melt probabilities calculated for the WASH-1400 plants, assuming that contributions from external events at these plants are similar to those calculated at the other plants listed in table 9.4. If we assume that external events would have a negligible effect on the ATWS core melt probabilities at the WASH-1400 plants, we would see the percent contribution of ATWS at Surry drop to 4.77% while Peach Bottom would drop to 20.16%. Although the contribution at Peach Bottom would remain unacceptably high, we see that the contribution at Surry would drop below the five percent value.

Zion is a different story. We see that loss of main feedwater and turbine trip make up essentially all of the ATWS contribution. Thus it appears that modifications to Zion may be justified to reduce the ATWS contribution to core melt. However, as mentioned in Section 8.5, the Zion study presented a revised ATWS analysis in revision number one to the PSS. Although I rejected this analysis as presenting an artificially low ATWS core melt probability, it is conceivable that the probability of the operators inserting the control rods within one minute (which was not accounted for in the analysis I used) could reduce the ATWS contribution significantly, even as much as the two-thirds reduction that would be needed to show an ATWS contribution to total core melt of less than five percent. Obviously, a closer look at the Zion plant is necessary before any firm conclusions could be drawn.

As mentioned above in Section 9.2, I said I would consider the possibility that the RPS unavailabilities at Peach Bottom and Indian Point 2 were equal to the NUREG-0460 value of 3.0×10^{-5} . If this were the case, the Indian Point 2 ATWS core melt probability value would change from 2.21×10^{-6} per R-yr to 3.30×10^{-6} per R-yr. The percent contribution value in table 9.8 would change from 0.65% to 0.97 %, making a negligible difference for my purposes. However, the Peach Bottom ATWS core melt probability value would change from 1.01×10^{-5} per R-yr to 2.33×10^{-5} per R-yr. This would change the percent contribution value in table 9.8 from 40.32% to 92.83%. If we adjust for the inadequate external event treatment, the percent ATWS contribution at Peach Bottom would still be 46.42%. Obviously, and regardless of the RPS value used, ATWS contributions at Peach Bottom appear to be unacceptable.

9.7 Comparison Study

During my research I came across a study entitled "Dominant Accident Sequences Derived from Review of Five PRA Studies" [1]. Table 2 in that study lists percent contributions to total core melt of various sequences at eleven different plants, including the eight plants involved in this study. Table 9.9 in here lists the percent contribution to core melt due to ATWS events reported in that study and this study for the eight plants involved. As can be seen in the table, the data agree fairly well with only one or two major discrepancies. Values for four of the plants (Surry, Indian point 2, Indian Point 3 and Zion) are within one percent and two of the plants (Big Rock Point and Limerick) are within three percent. The most notable difference is at Oconee 3 where they calculated an 11% ATWS contribution and I calculated a 2.41% ATWS contribution. The only explanation for this discrepancy is that the Oconee 3 PRA that I used assumed that a stand by shut down facility that was under construction was operational. The authors of the reference study [1] may not have incorporated this same assumption. The other major difference involves the Peach Bottom plant where I calculated a 40.32 percent ATWS contribution and

they calculated a 47 percent contribution. Although this difference is rather significant, if we consider the sketchy analysis in WASH-1400 and the fact that both contributions are very large and would not significantly change the results of this study, it would be pointless to concern ourselves with this difference.

However, the most significant result we can obtain from table 9.9 is the fact that the correlation coefficient between the two sets of data is 0.97. Such good agreement between these two studies suggest that we can put a certain level of confidence in both studies.

References:

- [1] "Dominant Accident Sequences Derived from Review of Five PRA Studies", V. Joksimovich, M.V. Frank and D.R. Worledge. *Anticipated and Abnormal Plant Transients in Light Water Reactors, Vol. 2*, Pamela L. Lassahn, Debu Majumdar and George F. Brockett, eds. New York: Plenum Press, 1984. p. 926.

TABLE 9.1 - SUMMARY OF PLANT DESCRIPTIONS

POWER PLANT	PLANT TYPE AND MANUFACTURER	LICENSED THERMAL PWR (MWt)	DESIGNED NET ELECTRIC PWR (MWe)	TYPE OF CONTAINMENT
WASH-1400 BWR	GENERAL ELECTRIC BWR/4	3293	1065	MARK I - ARGON ATM.
WASH-1400 PWR	WESTINGHOUSE PWR (3 LOOPS)	2441	788	LARGE DRY
BIG ROCK POINT	GENERAL ELECTRIC BWR	240	72	SPHERICAL STEEL
LIMERICK UNIT I	GENERAL ELECTRIC BWR/4	3293	1055	MARK II - NITROGEN ATM.
INDIAN POINT 2	WESTINGHOUSE PWR (4 LOOPS)	2758	873	LARGE DRY
INDIAN POINT 3	WESTINGHOUSE PWR (4 LOOPS)	3025	965	LARGE DRY
OCONEE UNIT 3	BABCOCK & WILCOX PWR (2 LOOPS)	2568	887	LARGE DRY
ZION UNIT I	WESTINGHOUSE PWR (4 LOOPS)	3250	1040	LARGE DRY
AVERAGES:		2608.50	843.13	

TABLE 9.2 - RPS UNAVAILABILITIES

POWER PLANT/STUDY	ELECTRICAL UNAVAIL. (per demand)	MECHANICAL UNAVAIL. (per demand)	TOTAL UNAVAIL. (per demand)	% OF NUREG-0460 UNAVAIL.
NUREG-0460	1.50E-05	1.50E-05	3.00E-05	100.00%
WASH-1270	N/A	N/A	5.00E-05	166.67%
PEACH BOTTOM II	N/A	N/A	1.30E-05	43.33%
SURRY UNIT I	N/A	N/A	3.70E-05	123.33%
BIG ROCK POINT	0E+00	3.50E-05	3.50E-05	116.67%
LIMERICK UNIT I	2.00E-05	1.00E-05	3.00E-05	100.00%
INDIAN POINT 2	1.63E-05	3.77E-06	2.01E-05	66.90%
INDIAN POINT 3	3.01E-05	9.19E-06	3.93E-05	130.97%
OCONEE UNIT 3	N/A	N/A	3.00E-05	100.00%
ZION UNIT I	1.77E-04	2.98E-06	1.80E-04	599.93%
AVERAGES:	4.31E-05	1.27E-05	4.64E-05	154.78%

TABLE 9.3 - PLANT INITIATOR FREQUENCIES

INITIATOR	PEACH BOTTOM II (per YR)	SURRY UNIT I (per YR)	BIG ROCK POINT (per YR)	LIMERICK UNIT I (per YR)	IND. PT. UNIT 2 (per YR)	IND. PT. UNIT 3 (per YR)	OCONEE UNIT 3 (per YR)	ZION UNIT I (per YR)	AVERAGE FREQUENCY (per YR)
LARGE LOCA	1.00E-04	1.00E-04	1.00E-05	4.00E-04	1.95E-03	2.16E-03	9.30E-04	9.40E-04	8.24E-04
MEDIUM LOCA	3.00E-04	3.00E-04	1.00E-04	2.00E-03	1.95E-03	2.16E-03	N/A	9.40E-04	1.11E-03
SMALL LOCA	1.00E-03	1.00E-03	1.00E-03	1.00E-02	1.85E-02	2.01E-02	3.00E-03	3.54E-02	1.13E-02
INTER. SYST. LOCA	4.00E-06	4.00E-06	1.98E-03	N/A	4.58E-07	4.64E-07	1.40E-07	1.05E-07	2.84E-04
REACTOR VESSEL RUPTURE	1.10E-07	1.00E-07	N/A	N/A	N/A	N/A	1.10E-06	N/A	4.37E-07
LOCA TOTALS	1.40E-03	1.40E-03	3.09E-03	1.24E-02	2.24E-02	2.44E-02	3.93E-03	3.73E-02	1.33E-02
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	1.11E-03	N/A	1.95E-03	2.16E-03	1.50E-03	9.40E-04	1.53E-03
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	1.11E-04	N/A	1.95E-03	2.16E-03	1.50E-03	9.40E-04	1.33E-03
LOSS OF MAIN F.W.	N/A	N/A	1.60E-01	7.00E-01	6.70E+00	3.80E+00	6.40E-01	5.17E+00	2.86E+00
MSIV TRIP	N/A	N/A	6.00E-02	1.08E+00	1.25E+00	8.98E-02	N/A	2.52E-01	5.46E-01
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	6.00E-02	N/A	N/A	N/A	2.10E-01	N/A	1.35E-01
TURBINE TRIP	N/A	N/A	1.39E+00	3.98E+00	7.32E+00	2.72E+00	4.90E+00	3.69E+00	4.00E+00
TURBINE TRIP: LOOP	N/A	N/A	1.30E-01	5.30E-02	1.82E-01	2.66E-01	1.70E-01	5.76E-02	1.43E-01
REACTOR TRIP	N/A	N/A	N/A	3.20E+00	6.84E+00	2.86E+00	INCL. IN T.T.	3.77E+00	4.17E+00
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	1.36E-01	1.71E-01	N/A	3.58E-01	2.22E-01
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	2.74E-02	3.37E-02	8.60E-03	2.44E-02	2.35E-02
OTHER TRANSIENTS	N/A	N/A	1.61E-01	6.00E-02	2.60E-02	3.00E-02	1.04E+00	6.60E-01	3.29E-01
TRANSIENT TOTALS	10.00	10.00	1.96	9.07	22.49	9.97	6.97	13.98	10.56
INTERNAL TOTALS	10.00	10.00	1.97	9.09	22.51	10.00	6.97	14.02	10.57
SEISMIC EVENTS	4.90E-04	4.90E-04	1.11E-03	1.67E-03	6.15E-03	6.15E-03	4.79E-04	3.90E-04	2.12E-03
FIRES	N/A	N/A	6.10E-03	5.28E-02	1.95E-02	2.64E-02	4.60E-02	1.13E-02	2.70E-02
TORNADOS	5.00E-06	5.00E-06	1.11E-03	2.30E-04	6.93E-03	6.93E-03	3.50E-05	1.10E-03	2.04E-03
AIRCRAFT CRASHES	1.00E-07	1.00E-07	8.53E-07	9.40E-08	2.90E-07	2.90E-07	2.50E-09	2.00E-07	2.41E-07
FLOODS	N/A	N/A	N/A	7.00E-02	1.00E-03	1.00E-03	3.04E-02	4.70E-06	2.05E-02
TURBINE MISSILES	1.00E-04	1.00E-04	N/A	1.00E-05	1.00E-04	1.00E-04	N/A	1.00E-04	8.50E-05
HIGH ENERGY LINE BREAKS	N/A	N/A	4.19E-06	N/A	N/A	N/A	N/A	N/A	4.19E-06
EXTERNAL TOTALS	5.95E-04	5.95E-04	8.33E-03	1.25E-01	3.37E-02	4.06E-02	7.69E-02	1.29E-02	3.73E-02
GRAND TOTALS	10.00	10.00	1.97	9.21	22.54	10.04	7.05	14.03	10.61
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	4.81E-05

TABLE 9.4 - PLANT CORE MELT PROBABILITIES

INITIATOR	PEACH BOTTOM II (per YR)	SURRY UNIT I (per YR)	BIG ROCK POINT (per YR)	LIMERICK UNIT I (per YR)	IND. PT. UNIT 2 (per YR)	IND. PT. UNIT 3 (per YR)	OCONEE UNIT 3 (per YR)	ZION UNIT I (per YR)	AVERAGE PROB. (per YR)
LARGE LOCA	1.80E-07	3.10E-06	3.70E-07	7.10E-08	1.60E-05	1.81E-05	9.00E-06	6.20E-06	6.63E-06
MEDIUM LOCA	2.70E-07	6.10E-06	4.10E-06	1.70E-08	1.30E-05	1.37E-05	N/A	5.33E-06	6.07E-06
SMALL LOCA	3.90E-07	1.70E-05	4.10E-05	2.80E-08	1.66E-05	8.56E-05	6.10E-06	1.64E-05	2.29E-05
INTER. SYST. LOCA	0E+00	4.00E-06	9.10E-05	N/A	4.58E-07	4.64E-07	1.40E-07	1.05E-07	1.37E-05
REACTOR VESSEL RUPTURE	1.10E-07	1.00E-07	N/A	N/A	N/A	N/A	1.10E-06	N/A	4.37E-07
LOCA TOTALS	9.50E-07	3.03E-05	1.36E-04	1.16E-07	4.61E-05	1.18E-04	1.63E-05	2.80E-05	4.70E-05
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	1.10E-04	N/A	1.99E-07	2.83E-07	INCL. IN O.T.	5.99E-09	2.76E-05
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0E+00	N/A	1.99E-07	2.83E-07	INCL. IN O.T.	5.02E-09	1.22E-07
LOSS OF MAIN F.W. MSIV TRIP	N/A	N/A	2.80E-06	1.81E-06	1.80E-06	1.91E-06	1.87E-06	4.62E-06	2.47E-06
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.10E-05	2.79E-06	1.75E-07	1.35E-08	N/A	4.03E-08	4.80E-06
TURBINE TRIP: LOOP	N/A	N/A	2.20E-05	N/A	N/A	N/A	INCL. IN O.T.	N/A	2.20E-05
REACTOR TRIP	N/A	N/A	1.11E-05	1.76E-06	1.96E-06	1.37E-06	6.80E-06	3.30E-06	4.38E-06
LOSS OF RCS FLOW	N/A	N/A	1.75E-04	6.70E-06	3.79E-05	4.18E-06	2.65E-06	2.17E-07	3.78E-05
S.G. TUBE RUPTURE	N/A	N/A	N/A	2.28E-07	9.58E-07	3.95E-07	INCL. IN T.T.	4.18E-07	5.00E-07
OTHER TRANSIENTS	N/A	N/A	N/A	N/A	3.28E-08	7.99E-08	N/A	1.61E-06	5.74E-07
TRANSIENT TOTALS	2.40E-05	1.31E-05	6.03E-04	1.42E-05	4.35E-05	9.37E-06	3.78E-05	1.41E-05	9.49E-05
INTERNAL TOTALS	2.50E-05	4.34E-05	7.39E-04	1.43E-05	8.95E-05	1.27E-04	5.42E-05	4.22E-05	1.42E-04
SEISMIC EVENTS	1.00E-07	1.00E-07	1.20E-07	5.86E-06	7.70E-06	3.60E-06	6.30E-05	5.60E-06	1.08E-05
FIRES	N/A	N/A	2.30E-04	2.24E-05	2.00E-04	6.30E-05	1.00E-05	4.60E-06	8.83E-05
TORNADOS	0E+00	0E+00	0E+00	4.00E-08	4.30E-05	1.30E-06	1.30E-05	2.38E-08	7.17E-06
AIRCRAFT CRASHES	0E+00	0E+00	8.82E-08	9.40E-09	2.90E-08	2.90E-08	2.50E-10	2.00E-08	2.20E-08
FLOODS	N/A	N/A	N/A	5.70E-07	1.00E-08	1.00E-08	1.13E-04	4.70E-08	2.27E-05
TURBINE MISSILES	0E+00	0E+00	N/A	1.00E-09	1.00E-08	1.00E-08	N/A	1.00E-08	5.17E-09
HIGH ENERGY LINE BREAKS	N/A	N/A	1.50E-06	N/A	N/A	N/A	N/A	N/A	1.50E-06
EXTERNAL TOTALS	1.00E-07	1.00E-07	2.32E-04	2.89E-05	2.51E-04	6.79E-05	1.99E-04	1.03E-05	9.86E-05
GRAND TOTALS	2.51E-05	4.35E-05	9.71E-04	4.32E-05	3.40E-04	1.95E-04	2.53E-04	5.25E-05	2.40E-04
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	4.81E-05

TABLE 9.5 - PERCENT CONTRIBUTIONS TO TOTAL CORE MELT PROBABILITY

INITIATOR	PEACH BOTTOM II	SURRY UNIT I	BIG ROCK POINT	LIMERICK UNIT I	IND. PT. UNIT 2	IND. PT. UNIT 3	OCONEE UNIT 3	ZION UNIT I	AVERAGE CONTR.
LARGE LOCA	0.72%	7.13%	0.04%	0.16%	4.70%	9.27%	3.56%	11.81%	4.67%
MEDIUM LOCA	1.08%	14.02%	0.42%	0.04%	3.82%	7.02%	N/A	10.16%	5.22%
SMALL LOCA	1.56%	39.08%	4.22%	0.06%	4.88%	43.86%	2.41%	31.25%	15.91%
INTER. SYST. LOCA	0.00%	9.20%	9.37%	N/A	0.13%	0.24%	0.06%	0.20%	2.74%
REACTOR VESSEL RUPTURE	0.44%	0.23%	N/A	N/A	N/A	N/A	0.43%	N/A	0.37%
LOCA TOTALS	3.79%	69.66%	14.05%	0.27%	13.54%	60.39%	6.45%	53.42%	27.70%
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	11.33%	N/A	0.06%	0.14%	INCL. IN O.T.	0.01%	2.89%
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.06%	0.14%	INCL. IN O.T.	0.01%	0.05%
LOSS OF MAIN F.W.	N/A	N/A	0.29%	4.19%	0.53%	0.98%	0.74%	8.80%	2.59%
MSIV TRIP	N/A	N/A	2.16%	6.46%	0.05%	0.01%	N/A	0.08%	1.75%
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.27%	N/A	N/A	N/A	INCL. IN O.T.	N/A	2.27%
TURBINE TRIP	N/A	N/A	1.14%	4.07%	0.58%	0.70%	2.69%	6.29%	2.58%
TURBINE TRIP: LOOP	N/A	N/A	18.02%	15.51%	11.14%	2.14%	1.05%	0.41%	8.05%
REACTOR TRIP	N/A	N/A	N/A	0.53%	0.28%	0.20%	INCL. IN T.T.	0.80%	0.45%
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	0.01%	0.04%	N/A	3.07%	1.04%
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	0.04%	0.36%	1.07%	0.43%	0.47%
OTHER TRANSIENTS	N/A	N/A	26.88%	2.11%	0.03%	0.08%	9.40%	7.05%	7.59%
TRANSIENT TOTALS	95.81%	30.11%	62.09%	32.87%	12.77%	4.80%	14.94%	26.94%	35.04%
INTERNAL TOTALS	99.60%	99.77%	76.14%	33.14%	26.31%	65.19%	21.39%	80.37%	62.74%
SEISMIC EVENTS	0.40%	0.23%	0.01%	13.57%	2.26%	1.84%	24.89%	10.67%	6.73%
FIRES	N/A	N/A	23.69%	51.86%	58.78%	32.28%	3.95%	8.77%	29.89%
TORNADOS	0.00%	0.00%	0.00%	0.09%	12.64%	0.67%	5.14%	0.05%	2.32%
AIRCRAFT CRASHES	0.00%	0.00%	0.01%	0.02%	0.01%	0.01%	0.00%	0.04%	0.01%
FLOODS	N/A	N/A	N/A	1.32%	0.00%	0.01%	44.64%	0.09%	9.21%
TURBINE MISSILES	0.00%	0.00%	N/A	0.00%	0.00%	0.01%	N/A	0.02%	0.00%
HIGH ENERGY LINE BREAKS	N/A	N/A	0.15%	N/A	N/A	N/A	N/A	N/A	0.15%
EXTERNAL TOTALS	0.40%	0.23%	23.86%	66.86%	73.69%	34.81%	78.61%	19.63%	37.26%
GRAND TOTALS	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	4.81E-05

TABLE 9.6 - PLANT ATWS CORE MELT PROBABILITIES

INITIATOR	PEACH BOTTOM II (per YR)	SURRY UNIT I (per YR)	BIG ROCK POINT (per YR)	LIMERICK UNIT I (per YR)	IND. PT. UNIT 2 (per YR)	IND. PT. UNIT 3 (per YR)	OCONEE UNIT 3 (per YR)	ZION UNIT I (per YR)	AVERAGE PROB. (per YR)
LARGE LOCA	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
MEDIUM LOCA	0E+00	0E+00	0E+00	2.40E-09	0E+00	0E+00	N/A	0E+00	4.80E-10
SMALL LOCA	0E+00	0E+00	0E+00	1.20E-08	2.37E-09	7.40E-09	0E+00	2.75E-08	8.21E-09
INTER. SYST. LOCA	0E+00	0E+00	0E+00	N/A	0E+00	0E+00	0E+00	0E+00	0E+00
REACTOR VESSEL RUPTURE	0E+00	0E+00	N/A	N/A	N/A	N/A	0E+00	N/A	0E+00
LOCA TOTALS	0E+00	0E+00	0E+00	1.44E-08	2.37E-09	7.40E-09	0E+00	2.75E-08	8.61E-09
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	0E+00	N/A	2.50E-10	7.95E-10	N/A	7.35E-10	4.45E-10
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0E+00	N/A	2.50E-10	7.95E-10	N/A	7.35E-10	4.45E-10
LOSS OF MAIN F.W.	N/A	N/A	2.40E-06	2.03E-07	8.58E-07	1.40E-06	6.70E-07	4.08E-06	1.60E-06
MSIV TRIP	N/A	N/A	2.10E-06	3.14E-07	0E+00	0E+00	N/A	0E+00	4.83E-07
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.10E-06	N/A	N/A	N/A	1.90E-07	N/A	1.15E-06
TURBINE TRIP	N/A	N/A	1.02E-05	5.55E-07	9.37E-07	1.00E-06	5.00E-06	2.90E-06	3.43E-06
TURBINE TRIP: LOOP	N/A	N/A	4.60E-06	5.26E-08	3.79E-07	5.21E-07	2.50E-07	4.21E-09	9.68E-07
REACTOR TRIP	N/A	N/A	N/A	0E+00	0E+00	0E+00	INCL. IN T.T.	0E+00	0E+00
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	1.58E-08	5.66E-08	N/A	2.55E-07	1.09E-07
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	9.97E-13	2.91E-12	0E+00	1.09E-14	9.79E-13
OTHER TRANSIENTS	N/A	N/A	5.40E-06	7.30E-08	1.95E-08	4.23E-08	0E+00	5.04E-07	1.01E-06
TRANSIENT TOTALS	1.01E-05	4.15E-06	2.68E-05	1.20E-06	2.21E-06	3.02E-06	6.11E-06	7.74E-06	7.85E-06
INTERNAL TOTALS	1.01E-05	4.15E-06	2.68E-05	1.21E-06	2.21E-06	3.03E-06	6.11E-06	7.77E-06	7.86E-06
SEISMIC EVENTS	0E+00	0E+00	0E+00	8.09E-07	0E+00	0E+00	0E+00	0E+00	1.35E-07
FIRES	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
TORNADOS	0E+00	0E+00	0E+00	1.00E-10	0E+00	0E+00	0E+00	0E+00	1.67E-11
AIRCRAFT CRASHES	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
FLOODS	0E+00	0E+00	N/A	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
TURBINE MISSILES	0E+00	0E+00	N/A	0E+00	0E+00	0E+00	N/A	0E+00	0E+00
HIGH ENERGY LINE BREAKS	0E+00	0E+00	0E+00	N/A	N/A	N/A	N/A	N/A	0E+00
EXTERNAL TOTALS	0E+00	0E+00	0E+00	8.09E-07	0E+00	0E+00	0E+00	0E+00	1.35E-07
GRAND TOTALS	1.01E-05	4.15E-06	2.68E-05	2.02E-06	2.21E-06	3.03E-06	6.11E-06	7.77E-06	7.99E-06
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	5.57E-05

TABLE 9.7 - PERCENT CONTRIBUTIONS TO ATWS CORE MELT PROBABILITY

INITIATOR	PEACH BOTTOM II	SURRY UNIT I	BIG ROCK POINT	LIMERICK UNIT I	IND. PT. UNIT 2	IND. PT. UNIT 3	OCONEE UNIT 3	ZION UNIT I	AVERAGE CONTR.
LARGE LOCA	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
MEDIUM LOCA	0.00%	0.00%	0.00%	0.12%	0.00%	0.00%	N/A	0.00%	0.02%
SMALL LOCA	0.00%	0.00%	0.00%	0.59%	0.11%	0.24%	0.00%	0.35%	0.22%
INTER. SYST. LOCA	0.00%	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
REACTOR VESSEL RUPTURE	0.00%	0.00%	N/A	N/A	N/A	N/A	0.00%	N/A	0.00%
LOCA TOTALS	0.00%	0.00%	0.00%	0.71%	0.11%	0.24%	0.00%	0.35%	0.24%
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.01%	0.03%	N/A	0.01%	0.01%
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.01%	0.03%	N/A	0.01%	0.01%
LOSS OF MAIN F.W.	N/A	N/A	8.96%	10.04%	38.79%	46.22%	10.97%	52.50%	27.91%
MSIV TRIP	N/A	N/A	7.84%	15.54%	0.00%	0.00%	N/A	0.00%	4.67%
LOSS OF CONDENSER	N/A	N/A	7.84%	N/A	N/A	N/A	3.11%	N/A	5.47%
TURBINE TRIP	N/A	N/A	38.06%	27.46%	42.36%	33.02%	81.83%	37.31%	43.34%
TURBINE TRIP: LOOP	N/A	N/A	17.16%	2.60%	17.13%	17.20%	4.09%	0.05%	9.71%
REACTOR TRIP	N/A	N/A	N/A	0.00%	0.00%	0.00%	INCL. IN T.T.	0.00%	0.00%
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	0.71%	1.87%	N/A	3.28%	1.95%
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
OTHER TRANSIENTS	N/A	N/A	20.15%	3.61%	0.88%	1.40%	0.00%	6.48%	5.42%
TRANSIENT TOTALS	100.00%	100.00%	100.00%	59.25%	99.89%	99.76%	100.00%	99.64%	93.09%
INTERNAL TOTALS	100.00%	100.00%	100.00%	59.97%	100.00%	100.00%	100.00%	100.00%	93.33%
SEISMIC EVENTS	0.00%	0.00%	0.00%	40.03%	0.00%	0.00%	0.00%	0.00%	6.67%
FIRES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TORNADOS	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
AIRCRAFT CRASHES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
FLOODS	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TURBINE MISSILES	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	N/A	0.00%	0.00%
HIGH ENERGY LINE BREAKS	0.00%	0.00%	0.00%	N/A	N/A	N/A	N/A	N/A	0.00%
EXTERNAL TOTALS	0.00%	0.00%	0.00%	40.03%	0.00%	0.00%	0.00%	0.00%	6.67%
GRAND TOTALS	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	5.57E-05

TABLE 9.8 - PERCENT ATWS CONTRIBUTIONS TO TOTAL CORE MELT

INITIATOR	PEACH BOTTOM II	SURRY UNIT I	BIG ROCK POINT	LIMERICK UNIT I	IND. PT. UNIT 2	IND. PT. UNIT 3	OCONEE UNIT 3	ZION UNIT I	AVERAGE CONTR.
LARGE LOCA	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
MEDIUM LOCA	0.00%	0.00%	0.00%	0.01%	0.00%	0.00%	N/A	0.00%	0.00%
SMALL LOCA	0.00%	0.00%	0.00%	0.03%	0.00%	0.00%	0.00%	0.05%	0.01%
INTER. SYST. LOCA	0.00%	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
REACTOR VESSEL RUPTURE	0.00%	0.00%	N/A	N/A	N/A	N/A	0.00%	N/A	0.00%
LOCA TOTALS	0.00%	0.00%	0.00%	0.03%	0.00%	0.00%	0.00%	0.05%	0.02%
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.00%	0.00%	N/A	0.00%	0.00%
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.00%	0.00%	N/A	0.00%	0.00%
LOSS OF MAIN F.W.	N/A	N/A	0.25%	0.47%	0.25%	0.72%	0.26%	7.77%	1.62%
MSIV TRIP	N/A	N/A	0.22%	0.73%	0.00%	0.00%	N/A	0.00%	0.19%
LOSS OF CONDENSER	N/A	N/A	0.22%	N/A	N/A	N/A	0.08%	N/A	0.15%
TURBINE TRIP	N/A	N/A	1.05%	1.28%	0.28%	0.51%	1.98%	5.53%	1.77%
TURBINE TRIP: LOOP	N/A	N/A	0.47%	0.12%	0.11%	0.27%	0.10%	0.01%	0.18%
REACTOR TRIP	N/A	N/A	N/A	0.00%	0.00%	0.00%	INCL. IN T.T.	0.00%	0.00%
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	0.00%	0.03%	N/A	0.49%	0.17%
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
OTHER TRANSIENTS	N/A	N/A	0.56%	0.17%	0.01%	0.02%	0.00%	0.96%	0.29%
TRANSIENT TOTALS	40.32%	9.54%	2.76%	2.77%	0.65%	1.55%	2.41%	14.76%	4.15%
INTERNAL TOTALS	40.32%	9.54%	2.76%	2.81%	0.65%	1.55%	2.41%	14.81%	4.17%
SEISMIC EVENTS	0.00%	0.00%	0.00%	1.87%	0.00%	0.00%	0.00%	0.00%	0.31%
FIRES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TORNADOS	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
AIRCRAFT CRASHES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
FLOODS	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TURBINE MISSILES	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	N/A	0.00%	0.00%
HIGH ENERGY LINE BREAKS	0.00%	0.00%	0.00%	N/A	N/A	N/A	N/A	N/A	0.00%
EXTERNAL TOTALS	0.00%	0.00%	0.00%	1.87%	0.00%	0.00%	0.00%	0.00%	0.31%
GRAND TOTALS	40.32%	9.54%	2.76%	4.68%	0.65%	1.55%	2.41%	14.81%	4.48%
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	5.57E-05

TABLE 9.9 - COMPARISON OF PERCENT ATWS CONTRIBUTIONS FROM TWO STUDIES

PLANT	% ATWS CONTRIBUTION FROM REFERENCE STUDY [1]	% ATWS CONTRIBUTION FROM THIS STUDY
PEACH BOTTOM	47	40.32
BIG ROCK POINT	0	2.76
LIMERICK	2	4.68
SURRY	9	9.54
INDIAN POINT 2	0	0.65
INDIAN POINT 3	1	1.55
OCONEE 3	11	2.41
ZION	15	14.81

CHAPTER 10 - SUMMARY RESULTS AND CONCLUSIONS

10.0 Introduction

This study examined probabilistic risk studies for eight light water nuclear power plants located in the United States. The purpose of the study was to estimate the contribution of Anticipated Transients without Scram (ATWS) to the total core melt probability and to identify any trends from plant to plant. This chapter summarizes the results and draws conclusions. Limitations of this study are also discussed.

10.1 Plant Descriptions

Table 9.1 summarized the plant descriptions for the eight plants studied. Table 9.1 is reproduced here as table 10.1. Three boiling water reactors (BWRs) were studied: Peach Bottom Unit II, Limerick Unit I and Big Rock Point. Peach Bottom and Limerick were found to be fairly representative of most BWRs in operation today. Big Rock Point is a much older plant, with a much lower power rating and a simpler containment structure. Specifically, Big Rock Point's power rating is a full order of magnitude lower than the other BWR power ratings and the containment is a spherical steel structure rather than the more sophisticated Mark I and Mark II designs.

Five pressurized water reactors (PWRs) were studied: Surry Unit I, Indian Point Unit 2, Indian Point Unit 3, Oconee Unit 3 and Zion Unit 1. All five plants were typical of operating PWRs. Oconee 3 is a Babcock and Wilcox two loop unit, while the other four designs are by Westinghouse. Surry is a three loop design while the Indian Point plants and Zion employ four coolant loops.

10.2 The Reactor Protection Systems

Table 9.2 summarized the RPS unavailabilities used for each plant as well as the values proposed in WASH-1270 and NUREG-0460. Table 9.2 is reproduced here as table 10.2. The reactor protection systems at all eight reactors are typical of those in use today. They employ diverse sensors and multiple channels to trip the reactor. The

system unavailabilities ranged from 1.3×10^{-5} per demand (Peach Bottom) to 1.8×10^{-4} per demand (Zion). The average unavailability for the eight plants is 4.81×10^{-5} per demand.

Most of the plants used individual analyses to calculate their RPS unavailability values. Indian Point 2, Indian Point 3 and Zion used event tree analyses to develop their RPS unavailability values while WASH-1400 used similar, yet looser, analyses. Big Rock Point used the NUREG-0460 methodology with revised plant experience data while Limerick and Oconee 3 simply used the NUREG-0460 value since they determined that their RPS designs were not significantly different than a typical RPS.

A baseline unavailability equal to the NUREG-0460 value of 3.0×10^{-5} per demand was used for comparison purposes. Only two plants (Peach Bottom and Indian Point 2) had unavailability values below this baseline. The possibility that the RPS unavailability at these two plants was equal to the NUREG-0460 value was considered and found to have no impact on the findings of this study.

10.3 The Initiator Frequencies

The initiator frequencies determined from the probability studies are listed in table 9.3, which is reproduced here as table 10.3. Half of the plants (Indian Point 2, Indian Point 3, Oconee 3 and Zion 1) used generic data combined with plant specific data to obtain the initiating frequencies. Big Rock Point used plant specific data exclusively. Limerick used generic data with the exception of the loss of off-site power frequency which was plant specific. The WASH-1400 plants (Peach Bottom and Surry) used generic data exclusively. Many of the studies did not calculate explicit frequencies for external events. The frequencies listed were my best estimates based on the information presented in each study.

Transients were seen to dominate the initiating frequencies. The average total initiating frequency was found to be 10.61 per year, with transients contributing an

average of 10.56 per year. Overall the frequencies were consistent from plant to plant with the exceptions being Big Rock Point which showed unusually low frequencies for transients (1.96 per year) and Indian Point 2 which showed unusually high frequencies for transients (22.49 per year).

10.4 Core Melt Probabilities

The core melt probabilities determined for each plant are shown in table 9.4 which is reproduced here as table 10.4. In general there was very good agreement from plant to plant with only a few exceptions. Big Rock Point showed core melt frequencies approximately one order of magnitude greater than the other plants. This difference was attributed to the fact that Big Rock Point is an older plant and lacks many of the safety features of newer designs. Peach Bottom and Limerick showed LOCA values almost two orders of magnitude lower than the other plants. This difference was attributed to design differences between BWRs and PWRs. The WASH-1400 plants showed external values that were two orders of magnitude lower than the other plants. This difference is due to inadequate treatment of external events in the WASH-1400 study. The average core melt probability for these eight plants was 2.4×10^{-4} per R-yr.

To gain greater insight into which initiators contribute significantly (i.e. greater than five percent) to core melt, table 9.5 was prepared which showed the percent contribution to core melt for each initiator for each plant. Table 9.5 is reproduced here as table 10.5. Some interesting trends were noticed in this table. First and foremost, fires were seen to be significant contributors to core melt at five of the six plants where fires were considered as initiators. In fact at four of these plants (Big Rock Point, Limerick, Indian Point 2 and Indian Point 3) fires contributed over 20 percent to core melt with fires at Limerick and Indian Point 2 contributing over 50 percent to core melt. Undeniably, fires are major contributors to core melt and can present a significant risk at nuclear power plants in the United States.

Second, loss of coolant accidents (LOCAs) contribute significantly at three of the five PWRs (Surry, Indian Point 3 and Zion). At all three of these plants, total LOCA contributions are greater than 50 percent, suggesting that LOCA events at PWRs pose a significant risk.

Third, loss of off-site power was identified as a significant contributor at three of the six plants that identified loss of off-site power as a specific initiator. In fact the loss of off-site power contributions at Big Rock Point, Limerick and Indian Point 2 were all greater than ten percent.

Finally, three of the eight plants (Limerick, Oconee and Zion) showed seismic events as a significant contributor to core melt. Again the contribution at each plant was greater than ten percent.

10.5 ATWS Core Melt Probabilities

The ATWS core melt probabilities determined for each plant are shown in table 9.6 which is reproduced here as table 10.6. In general there was very good agreement from plant to plant. The only notable exceptions are Peach Bottom and Big Rock Point which showed ATWS core melt probabilities approximately one order of magnitude greater than the other plants. The average ATWS core melt probability for these eight plants was 7.99×10^{-6} per R-yr.

To gain greater insight into which initiators contribute significantly (i.e. greater than five percent) to ATWS core melt, table 9.7 was prepared which showed the percent contribution to ATWS core melt for each initiator for each plant. Table 9.7 is reproduced here as table 10.7. Some interesting trends were noticed in this table. First and foremost, turbine trips and loss of main feedwater were seen to be significant contributors to core melt at all six of the six plants where individual transients were considered as initiators. In fact turbine trips accounted for anywhere from 27.46% (Limerick) to 81.83% (Oconee) of the ATWS core melt probability while loss of main

feedwater contributed anywhere from 8.96% (Big Rock Point) to 52.5% (Zion).

MSIV trip contributed significantly at both BWRs that calculated a value (Big Rock Point and Limerick) and loss of off-site power contributed significantly to three of the six plants (Big Rock Point, Indian Point 2 and Indian Point 3). Only one plant, Limerick, showed a contribution to ATWS core melt probability from external initiators, but the 40.03% contribution from seismic events is indeed significant.

An important observation made at this point is that although turbine trips and loss of main feedwater are the major contributors to the ATWS core melt probability, they contribute only a few percent to the overall core melt probability as seen in table 10.5. This suggests that efforts to reduce the ATWS core melt probability due to turbine trips and loss of main feedwater will have little or no effect on the overall core melt probability.

10.6 Contribution of ATWS to Core Melt

The percent contributions of ATWS events to the total core melt probability for each plant were presented in table 9.8 which has been reproduced as table 10.8. The total percent contributions ranged from 0.65% (Indian Point 2) to 40.32% (Peach Bottom) and the average contribution was found to be 4.48%. Five of the plants showed total ATWS contributions of less than five percent, while four of these plants (Big Rock Point, Indian Point 2, Indian Point 3 and Oconee) showed contributions less than three percent. Again, this suggests that efforts to reduce the ATWS core melt probability at many plants will have a very small, if any, effect on the total core melt probability.

The WASH-1400 plants (Peach Bottom and Surry) were considered to have underestimated the contributions to core melt due to external events. If proper treatment of external events were considered, the core melt probabilities at Peach Bottom and Surry could conceivably double without increasing the ATWS core melt probabilities at these plants. This would decrease the Surry ATWS contribution to less

than five percent, although the Peach Bottom contribution would remain at approximately 20 percent.

The 14.81% ATWS contribution to core melt at Zion was also seen as an upper bound limit since the Zion PSS authors felt that they had not adequately allowed for manual trip of the reactor following the start of an ATWS. Although it is difficult to determine how much lower the ATWS core melt probability actually is, it is conceivable that a revised analysis with realistic assumptions about manual trip could show a two-thirds reduction in the ATWS core melt probability which would show an ATWS contribution of less than five percent to the total core melt at Zion.

10.7 Comparison Study

During my research I came across a study entitled "Dominant Accident Sequences Derived from Review of Five PRA Studies" [1]. Table 9.9 compared the data presented in that study for ATWS contributions to core melt with the contributions calculated in this study. Table 9.9 is reproduced here as table 10.9. The only major discrepancy noted was at Oconee 3 where they calculated an 11 % ATWS contribution while my data shows a 2.41% contribution. The most probable cause for this difference is that I assumed that a stand-by shut down facility that was under construction at Oconee was operational, where as the authors of the comparison study may not have made this same assumption.

A second discrepancy between the two studies is at Peach Bottom where I calculated an ATWS contribution of 40.32% and they calculated a 47% contribution. Although this difference could be considered significant, substitution of their value in this study would have no effect on the conclusions drawn here since the contribution is so large in either case.

The most significant relationship between the two sets of data in table 10.9 is that the correlation coefficient is 0.97, suggesting that a great deal of confidence can be put in both studies.

10.8 Conclusions

As discussed in the next section, it is not possible to make broad generalizations regarding all nuclear power plants in the United States based on the limited study of a few plants since plant designs are very individual. However, if we assume that the eight plants studied here are representative of the plants in operation today, it can be useful to draw general conclusions based on the information derived in this study as long as we remember that these conclusions may not be applicable to each and every nuclear plant and that each plant must still be evaluated on an individual basis. With this caveat in mind, I offer the following conclusions.

First, reactor protection system unavailability values can be expected to be on the order of 10^{-5} per demand. This value is consistent with the NUREG-0460 value of 3.0×10^{-5} per demand which was used as the baseline for this study.

Second, The initiator frequencies are dominated by transient events and can be expected to be close to 10 events per year.

Third, the core melt probability at a nuclear power plant can be expected to be on the order of 10^{-4} per R-yr. This is an order of magnitude larger than the findings of WASH-1400, the original study designed to estimate this probability. The difference is due to the inadequate treatment of external events in the WASH-1400 study. The dominant contributor to the core melt probability at most plants is fire. PWRs show a dominant contribution from LOCAs as well. Other significant contributors are loss of off-site power and seismic events.

Fourth, the core melt probability due to ATWS events is on the order of 10^{-6} per R-yr. The dominant contributors to ATWS sequences leading to core melt are turbine trips and loss of main feedwater. A significant contributor at BWRs appears to be MSIV trip.

Fifth, and perhaps most important to this study, is that ATWS sequences at the majority of the plants studied contributed less than five percent to the total core melt

frequency, suggesting that ATWS does not pose an undue risk to the health and safety of the public. At most plants, fires, LOCAs and/or loss of off-site power contribute significantly more to the core melt probability than ATWS. Additionally, at most plants, those contributors that make up the majority of the ATWS contribution contribute insignificantly to the total core melt, which means that efforts to reduce the probability of core melt due to ATWS will have little if any effect on the total core melt probability.

Thus if one is interested in decreasing the total core melt probability, ATWS should be considered a low priority in relation to fires, LOCAs and loss of off-site power.

10.9 Limitations of this Study

The conclusions in this study are subject to several limitations. First and foremost is the fact that in the United States each nuclear power plant has its own individual design. This means that problems, advantages and disadvantages can vary widely from plant to plant even if designed and built by the same firms and using the same nuclear steam supply systems. Thus it is impossible to make generalizations about all plants based on the limited study of a few plants. One could estimate the contribution of a particular initiator at a given plant based on analyses of plants with similar designs, but that approach is no substitute for a specific analysis of the plant in question.

Second, this study looks at core melt probabilities rather than radioactive release and public consequences. This is significant in that ATWS events typically result in high containment pressure, increasing the probability of breaching containment leading to increased consequences. Thus it is possible that ATWS events could result in greater public consequences than one of the more dominant core melt initiators, such as fires, which may have a lower containment failure probability.

Despite these limitations, I feel that some stock should be placed in the conclusions presented here since many of the plants studied are typical of today's operating plants and since the first line of defense against public consequences is preventing core melt.

References:

- [1] "Dominant Accident Sequences Derived from Review of Five PRA Studies", V. Joksimovich, M.V. Frank and D.R. Worledge. *Anticipated and Abnormal Plant Transients in Light Water Reactors, Vol. 2*, Pamela L. Lassahn, Debu Majumdar and George F. Brockett, eds. New York: Plenum Press, 1984. p. 926.

TABLE 10.1 - SUMMARY OF PLANT DESCRIPTIONS

POWER PLANT	PLANT TYPE AND MANUFACTURER	LICENSED THERMAL PWR (MWt)	DESIGNED NET ELECTRIC PWR (MWe)	TYPE OF CONTAINMENT
WASH-1400 BWR	GENERAL ELECTRIC BWR/4	3293	1065	MARK I - ARGON ATM.
WASH-1400 PWR	WESTINGHOUSE PWR (3 LOOPS)	2441	788	LARGE DRY
BIG ROCK POINT	GENERAL ELECTRIC BWR	240	72	SPHERICAL STEEL
LIMERICK UNIT I	GENERAL ELECTRIC BWR/4	3293	1055	MARK II - NITROGEN ATM.
INDIAN POINT 2	WESTINGHOUSE PWR (4 LOOPS)	2758	873	LARGE DRY
INDIAN POINT 3	WESTINGHOUSE PWR (4 LOOPS)	3025	965	LARGE DRY
OCONEE UNIT 3	BABCOCK & WILCOX PWR (2 LOOPS)	2568	887	LARGE DRY
ZION UNIT I	WESTINGHOUSE PWR (4 LOOPS)	3250	1040	LARGE DRY
AVERAGES:		2608.50	843.13	

TABLE 10.2 - RPS UNAVAILABILITIES

POWER PLANT/STUDY	ELECTRICAL UNAVAIL. (per demand)	MECHANICAL UNAVAIL. (per demand)	TOTAL UNAVAIL. (per demand)	% OF NUREG-0460 UNAVAIL.
NUREG-0460	1.50E-05	1.50E-05	3.00E-05	100.00%
WASH-1270	N/A	N/A	5.00E-05	166.67%
PEACH BOTTOM II	N/A	N/A	1.30E-05	43.33%
SURRY UNIT I	N/A	N/A	3.70E-05	123.33%
BIG ROCK POINT	0E+00	3.50E-05	3.50E-05	116.67%
LIMERICK UNIT I	2.00E-05	1.00E-05	3.00E-05	100.00%
INDIAN POINT 2	1.63E-05	3.77E-06	2.01E-05	66.90%
INDIAN POINT 3	3.01E-05	9.19E-06	3.93E-05	130.97%
OCONEE UNIT 3	N/A	N/A	3.00E-05	100.00%
ZION UNIT I	1.77E-04	2.98E-06	1.80E-04	599.93%
AVERAGES:	4.31E-05	1.27E-05	4.64E-05	154.78%

TABLE 10.3 - PLANT INITIATOR FREQUENCIES

INITIATOR	PEACH BOTTOM II (per YR)	SURRY UNIT I (per YR)	BIG ROCK POINT (per YR)	LIMERICK UNIT I (per YR)	IND. PT. UNIT 2 (per YR)	IND. PT. UNIT 3 (per YR)	OCONEE UNIT 3 (per YR)	ZION UNIT I (per YR)	AVERAGE FREQUENCY (per YR)
LARGE LOCA	1.00E-04	1.00E-04	1.00E-05	4.00E-04	1.95E-03	2.16E-03	9.30E-04	9.40E-04	8.24E-04
MEDIUM LOCA	3.00E-04	3.00E-04	1.00E-04	2.00E-03	1.95E-03	2.16E-03	N/A	9.40E-04	1.11E-03
SMALL LOCA	1.00E-03	1.00E-03	1.00E-03	1.00E-02	1.85E-02	2.01E-02	3.00E-03	3.54E-02	1.13E-02
INTER. SYST. LOCA	4.00E-06	4.00E-06	1.98E-03	N/A	4.58E-07	4.64E-07	1.40E-07	1.05E-07	2.84E-04
REACTOR VESSEL RUPTURE	1.10E-07	1.00E-07	N/A	N/A	N/A	N/A	1.10E-06	N/A	4.37E-07
LOCA TOTALS	1.40E-03	1.40E-03	3.09E-03	1.24E-02	2.24E-02	2.44E-02	3.93E-03	3.73E-02	1.33E-02
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	1.11E-03	N/A	1.95E-03	2.16E-03	1.50E-03	9.40E-04	1.53E-03
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	1.11E-04	N/A	1.95E-03	2.16E-03	1.50E-03	9.40E-04	1.33E-03
LOSS OF MAIN F.W.	N/A	N/A	1.60E-01	7.00E-01	6.70E+00	3.80E+00	6.40E-01	5.17E+00	2.86E+00
MSIV TRIP	N/A	N/A	6.00E-02	1.08E+00	1.25E+00	8.98E-02	N/A	2.52E-01	5.46E-01
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	6.00E-02	N/A	N/A	N/A	2.10E-01	N/A	1.35E-01
TURBINE TRIP: LOOP	N/A	N/A	1.39E+00	3.98E+00	7.32E+00	2.72E+00	4.90E+00	3.69E+00	4.00E+00
REACTOR TRIP	N/A	N/A	1.30E-01	5.30E-02	1.82E-01	2.66E-01	1.70E-01	5.76E-02	1.43E-01
LOSS OF RCS FLOW	N/A	N/A	N/A	3.20E+00	6.84E+00	2.86E+00	INCL. IN T.T.	3.77E+00	4.17E+00
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	1.36E-01	1.71E-01	N/A	3.58E-01	2.22E-01
OTHER TRANSIENTS	N/A	N/A	1.61E-01	6.00E-02	2.60E-02	3.00E-02	1.04E+00	6.60E-01	3.29E-01
TRANSIENT TOTALS	10.00	10.00	1.96	9.07	22.49	9.97	6.97	13.98	10.56
INTERNAL TOTALS	10.00	10.00	1.97	9.09	22.51	10.00	6.97	14.02	10.57
SEISMIC EVENTS	4.90E-04	4.90E-04	1.11E-03	1.67E-03	6.15E-03	6.15E-03	4.79E-04	3.90E-04	2.12E-03
FIRES	N/A	N/A	6.10E-03	5.28E-02	1.95E-02	2.64E-02	4.60E-02	1.13E-02	2.70E-02
TORNADOS	5.00E-06	5.00E-06	1.11E-03	2.30E-04	6.93E-03	6.93E-03	3.50E-05	1.10E-03	2.04E-03
AIRCRAFT CRASHES	1.00E-07	1.00E-07	8.53E-07	9.40E-08	2.90E-07	2.90E-07	2.50E-09	2.00E-07	2.41E-07
FLOODS	N/A	N/A	N/A	7.00E-02	1.00E-03	1.00E-03	3.04E-02	4.70E-06	2.05E-02
TURBINE MISSILES	1.00E-04	1.00E-04	N/A	1.00E-05	1.00E-04	1.00E-04	N/A	1.00E-04	8.50E-05
HIGH ENERGY LINE BREAKS	N/A	N/A	4.19E-06	N/A	N/A	N/A	N/A	N/A	4.19E-06
EXTERNAL TOTALS	5.95E-04	5.95E-04	8.33E-03	1.25E-01	3.37E-02	4.06E-02	7.69E-02	1.29E-02	3.73E-02
GRAND TOTALS	10.00	10.00	1.97	9.21	22.54	10.04	7.05	14.03	10.61
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	4.81E-05

TABLE 10.4 - PLANT CORE MELT PROBABILITIES

INITIATOR	PEACH BOTTOM II (per YR)	SURRY UNIT I (per YR)	BIG ROCK POINT (per YR)	LIMERICK UNIT I (per YR)	IND. PT. UNIT 2 (per YR)	IND. PT. UNIT 3 (per YR)	OCONEE UNIT 3 (per YR)	ZION UNIT I (per YR)	AVERAGE PROB. (per YR)
LARGE LOCA	1.80E-07	3.10E-06	3.70E-07	7.10E-08	1.60E-05	1.81E-05	9.00E-06	6.20E-06	6.63E-06
MEDIUM LOCA	2.70E-07	6.10E-06	4.10E-06	1.70E-08	1.30E-05	1.37E-05	N/A	5.33E-06	6.07E-06
SMALL LOCA	3.90E-07	1.70E-05	4.10E-05	2.80E-08	1.66E-05	8.56E-05	6.10E-06	1.64E-05	2.29E-05
INTER. SYST. LOCA	0E+00	4.00E-06	9.10E-05	N/A	4.58E-07	4.64E-07	1.40E-07	1.05E-07	1.37E-05
REACTOR VESSEL RUPTURE	1.10E-07	1.00E-07	N/A	N/A	N/A	N/A	1.10E-06	N/A	4.37E-07
LOCA TOTALS	9.50E-07	3.03E-05	1.36E-04	1.16E-07	4.61E-05	1.18E-04	1.63E-05	2.80E-05	4.70E-05
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	1.10E-04	N/A	1.99E-07	2.83E-07	INCL. IN O.T.	5.99E-09	2.76E-05
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0E+00	N/A	1.99E-07	2.83E-07	INCL. IN O.T.	5.02E-09	1.22E-07
LOSS OF MAIN F.W. MSIV TRIP	N/A	N/A	2.80E-06	1.81E-06	1.80E-06	1.91E-06	1.87E-06	4.62E-06	2.47E-06
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.10E-05	2.79E-06	1.75E-07	1.35E-08	N/A	4.03E-08	4.80E-06
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.20E-05	N/A	N/A	N/A	INCL. IN O.T.	N/A	2.20E-05
TURBINE TRIP	N/A	N/A	1.11E-05	1.76E-06	1.96E-06	1.37E-06	6.80E-06	3.30E-06	4.38E-06
TURBINE TRIP: LOOP	N/A	N/A	1.75E-04	6.70E-06	3.79E-05	4.18E-06	2.65E-06	2.17E-07	3.78E-05
REACTOR TRIP	N/A	N/A	N/A	2.28E-07	9.58E-07	3.95E-07	INCL. IN T.T.	4.18E-07	5.00E-07
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	3.28E-08	7.99E-08	N/A	1.61E-06	5.74E-07
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	1.42E-07	6.98E-07	2.70E-06	2.24E-07	9.41E-07
OTHER TRANSIENTS	N/A	N/A	2.61E-04	9.10E-07	9.74E-08	1.60E-07	2.38E-05	3.70E-06	4.83E-05
TRANSIENT TOTALS	2.40E-05	1.31E-05	6.03E-04	1.42E-05	4.35E-05	9.37E-06	3.78E-05	1.41E-05	9.49E-05
INTERNAL TOTALS	2.50E-05	4.34E-05	7.39E-04	1.43E-05	8.95E-05	1.27E-04	5.42E-05	4.22E-05	1.42E-04
SEISMIC EVENTS	1.00E-07	1.00E-07	1.20E-07	5.86E-06	7.70E-06	3.60E-06	6.30E-05	5.60E-06	1.08E-05
FIRES	N/A	N/A	2.30E-04	2.24E-05	2.00E-04	6.30E-05	1.00E-05	4.60E-06	8.83E-05
TORNADOS	0E+00	0E+00	0E+00	4.00E-08	4.30E-05	1.30E-06	1.30E-05	2.38E-08	7.17E-06
AIRCRAFT CRASHES	0E+00	0E+00	8.82E-08	9.40E-09	2.90E-08	2.90E-08	2.50E-10	2.00E-08	2.20E-08
FLOODS	N/A	N/A	N/A	5.70E-07	1.00E-08	1.00E-08	1.13E-04	4.70E-08	2.27E-05
TURBINE MISSILES	0E+00	0E+00	N/A	1.00E-09	1.00E-08	1.00E-08	N/A	1.00E-08	5.17E-09
HIGH ENERGY LINE BREAKS	N/A	N/A	1.50E-06	N/A	N/A	N/A	N/A	N/A	1.50E-06
EXTERNAL TOTALS	1.00E-07	1.00E-07	2.32E-04	2.89E-05	2.51E-04	6.79E-05	1.99E-04	1.03E-05	9.86E-05
GRAND TOTALS	2.51E-05	4.35E-05	9.71E-04	4.32E-05	3.40E-04	1.95E-04	2.53E-04	5.25E-05	2.40E-04
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	4.81E-05

TABLE 10.5 - PERCENT CONTRIBUTIONS TO TOTAL CORE MELT PROBABILITY

INITIATOR	PEACH BOTTOM II	SURRY UNIT I	BIG ROCK POINT	LIMERICK UNIT I	IND. PT. UNIT 2	IND. PT. UNIT 3	OCONEE UNIT 3	ZION UNIT I	AVERAGE CONTR.
LARGE LOCA	0.72%	7.13%	0.04%	0.16%	4.70%	9.27%	3.56%	11.81%	4.67%
MEDIUM LOCA	1.08%	14.02%	0.42%	0.04%	3.82%	7.02%	N/A	10.16%	5.22%
SMALL LOCA	1.56%	39.08%	4.22%	0.06%	4.88%	43.86%	2.41%	31.25%	15.91%
INTER. SYST. LOCA	0.00%	9.20%	9.37%	N/A	0.13%	0.24%	0.06%	0.20%	2.74%
REACTOR VESSEL RUPTURE	0.44%	0.23%	N/A	N/A	N/A	N/A	0.43%	N/A	0.37%
LOCA TOTALS	3.79%	69.66%	14.05%	0.27%	13.54%	60.39%	6.45%	53.42%	27.70%
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	11.33%	N/A	0.06%	0.14%	INCL. IN O.T.	0.01%	2.89%
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.06%	0.14%	INCL. IN O.T.	0.01%	0.05%
LOSS OF MAIN F.W.	N/A	N/A	0.29%	4.19%	0.53%	0.98%	0.74%	8.80%	2.59%
MSIV TRIP	N/A	N/A	2.16%	6.46%	0.05%	0.01%	N/A	0.08%	1.75%
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.27%	N/A	N/A	N/A	INCL. IN O.T.	N/A	2.27%
TURBINE TRIP	N/A	N/A	1.14%	4.07%	0.58%	0.70%	2.69%	6.29%	2.58%
TURBINE TRIP: LOOP	N/A	N/A	18.02%	15.51%	11.14%	2.14%	1.05%	0.41%	8.05%
REACTOR TRIP	N/A	N/A	N/A	0.53%	0.28%	0.20%	INCL. IN T.T.	0.80%	0.45%
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	0.01%	0.04%	N/A	3.07%	1.04%
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	0.04%	0.36%	1.07%	0.43%	0.47%
OTHER TRANSIENTS	N/A	N/A	26.88%	2.11%	0.03%	0.08%	9.40%	7.05%	7.59%
TRANSIENT TOTALS	95.81%	30.11%	62.09%	32.87%	12.77%	4.80%	14.94%	26.94%	35.04%
INTERNAL TOTALS	99.60%	99.77%	76.14%	33.14%	26.31%	65.19%	21.39%	80.37%	62.74%
SEISMIC EVENTS	0.40%	0.23%	0.01%	13.57%	2.26%	1.84%	24.89%	10.67%	6.73%
FIRES	N/A	N/A	23.69%	51.86%	58.78%	32.28%	3.95%	8.77%	29.89%
TORNADOS	0.00%	0.00%	0.00%	0.09%	12.64%	0.67%	5.14%	0.05%	2.32%
AIRCRAFT CRASHES	0.00%	0.00%	0.01%	0.02%	0.01%	0.01%	0.00%	0.04%	0.01%
FLOODS	N/A	N/A	N/A	1.32%	0.00%	0.01%	44.64%	0.09%	9.21%
TURBINE MISSILES	0.00%	0.00%	N/A	0.00%	0.00%	0.01%	N/A	0.02%	0.00%
HIGH ENERGY LINE BREAKS	N/A	N/A	0.15%	N/A	N/A	N/A	N/A	N/A	0.15%
EXTERNAL TOTALS	0.40%	0.23%	23.86%	66.86%	73.69%	34.81%	78.61%	19.63%	37.26%
GRAND TOTALS	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	4.81E-05

TABLE 10.6 - PLANT ATWS CORE MELT PROBABILITIES

INITIATOR	PEACH BOTTOM II (per YR)	SURRY UNIT I (per YR)	BIG ROCK POINT (per YR)	LIMERICK UNIT I (per YR)	IND. PT. UNIT 2 (per YR)	IND. PT. UNIT 3 (per YR)	OCONEE UNIT 3 (per YR)	ZION UNIT I (per YR)	AVERAGE PROB. (per YR)
LARGE LOCA	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
MEDIUM LOCA	0E+00	0E+00	0E+00	2.40E-09	0E+00	0E+00	N/A	0E+00	4.80E-10
SMALL LOCA	0E+00	0E+00	0E+00	1.20E-08	2.37E-09	7.40E-09	0E+00	2.75E-08	8.21E-09
INTER. SYST. LOCA	0E+00	0E+00	0E+00	N/A	0E+00	0E+00	0E+00	0E+00	0E+00
REACT. VESSEL RUPTURE	0E+00	0E+00	N/A	N/A	N/A	N/A	0E+00	N/A	0E+00
LOCA TOTALS	0E+00	0E+00	0E+00	1.44E-08	2.37E-09	7.40E-09	0E+00	2.75E-08	8.61E-09
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	0E+00	N/A	2.50E-10	7.95E-10	N/A	7.35E-10	4.45E-10
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0E+00	N/A	2.50E-10	7.95E-10	N/A	7.35E-10	4.45E-10
LOSS OF MAIN F.W.	N/A	N/A	2.40E-06	2.03E-07	8.58E-07	1.40E-06	6.70E-07	4.08E-06	1.60E-06
MSIV TRIP	N/A	N/A	2.10E-06	3.14E-07	0E+00	0E+00	N/A	0E+00	4.83E-07
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	2.10E-06	N/A	N/A	N/A	1.90E-07	N/A	1.15E-06
TURBINE TRIP: LOOP	N/A	N/A	1.02E-05	5.55E-07	9.37E-07	1.00E-06	5.00E-06	2.90E-06	3.43E-06
REACTOR TRIP	N/A	N/A	4.60E-06	5.26E-08	3.79E-07	5.21E-07	2.50E-07	4.21E-09	9.68E-07
LOSS OF RCS FLOW	N/A	N/A	N/A	0E+00	0E+00	0E+00	INCL. IN T.T.	0E+00	0E+00
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	1.58E-08	5.66E-08	N/A	2.55E-07	1.09E-07
OTHER TRANSIENTS	N/A	N/A	N/A	N/A	9.97E-13	2.91E-12	0E+00	1.09E-14	9.79E-13
TRANSIENT TOTALS	1.01E-05	4.15E-06	2.68E-05	1.20E-06	2.21E-06	3.02E-06	6.11E-06	7.74E-06	7.85E-06
INTERNAL TOTALS	1.01E-05	4.15E-06	2.68E-05	1.21E-06	2.21E-06	3.03E-06	6.11E-06	7.77E-06	7.86E-06
SEISMIC EVENTS	0E+00	0E+00	0E+00	8.09E-07	0E+00	0E+00	0E+00	0E+00	1.35E-07
FIRES	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
TORNADOS	0E+00	0E+00	0E+00	1.00E-10	0E+00	0E+00	0E+00	0E+00	1.67E-11
AIRCRAFT CRASHES	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
FLOODS	0E+00	0E+00	N/A	0E+00	0E+00	0E+00	0E+00	0E+00	0E+00
TURBINE MISSILES	0E+00	0E+00	N/A	0E+00	0E+00	0E+00	N/A	0E+00	0E+00
HIGH ENERGY LINE BREAKS	0E+00	0E+00	0E+00	N/A	N/A	N/A	N/A	N/A	0E+00
EXTERNAL TOTALS	0E+00	0E+00	0E+00	8.09E-07	0E+00	0E+00	0E+00	0E+00	1.35E-07
GRAND TOTALS	1.01E-05	4.15E-06	2.68E-05	2.02E-06	2.21E-06	3.03E-06	6.11E-06	7.77E-06	7.99E-06
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	5.57E-05

TABLE 10.7 - PERCENT CONTRIBUTIONS TO ATWS CORE MELT PROBABILITY

INITIATOR	PEACH BOTTOM II	SURRY UNIT I	BIG ROCK POINT	LIMERICK UNIT I	IND. PT. UNIT 2	IND. PT. UNIT 3	OCONEE UNIT 3	ZION UNIT I	AVERAGE CONTR.
LARGE LOCA	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
MEDIUM LOCA	0.00%	0.00%	0.00%	0.12%	0.00%	0.00%	N/A	0.00%	0.02%
SMALL LOCA	0.00%	0.00%	0.00%	0.59%	0.11%	0.24%	0.00%	0.35%	0.22%
INTER. SYST. LOCA	0.00%	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
REACTOR VESSEL RUPTURE	0.00%	0.00%	N/A	N/A	N/A	N/A	0.00%	N/A	0.00%
LOCA TOTALS	0.00%	0.00%	0.00%	0.71%	0.11%	0.24%	0.00%	0.35%	0.24%
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.01%	0.03%	N/A	0.01%	0.01%
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.01%	0.03%	N/A	0.01%	0.01%
LOSS OF MAIN F.W.	N/A	N/A	8.96%	10.04%	38.79%	46.22%	10.97%	52.50%	27.91%
MSIV TRIP	N/A	N/A	7.84%	15.54%	0.00%	0.00%	N/A	0.00%	4.67%
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	7.84%	N/A	N/A	N/A	3.11%	N/A	5.47%
TURBINE TRIP: LOOP	N/A	N/A	38.06%	27.46%	42.36%	33.02%	81.83%	37.31%	43.34%
REACTOR TRIP	N/A	N/A	17.16%	2.60%	17.13%	17.20%	4.09%	0.05%	9.71%
LOSS OF RCS FLOW	N/A	N/A	N/A	0.00%	0.00%	0.00%	INCL. IN T.T.	0.00%	0.00%
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	0.71%	1.87%	N/A	3.28%	1.95%
OTHER TRANSIENTS	N/A	N/A	20.15%	3.61%	0.88%	1.40%	0.00%	6.48%	5.42%
TRANSIENT TOTALS	100.00%	100.00%	100.00%	59.25%	99.89%	99.76%	100.00%	99.64%	93.09%
INTERNAL TOTALS	100.00%	100.00%	100.00%	59.97%	100.00%	100.00%	100.00%	100.00%	93.33%
SEISMIC EVENTS	0.00%	0.00%	0.00%	40.03%	0.00%	0.00%	0.00%	0.00%	6.67%
FIRES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TORNADOS	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
AIRCRAFT CRASHES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
FLOODS	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TURBINE MISSILES	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	N/A	0.00%	0.00%
HIGH ENERGY LINE BREAKS	0.00%	0.00%	0.00%	N/A	N/A	N/A	N/A	N/A	0.00%
EXTERNAL TOTALS	0.00%	0.00%	0.00%	40.03%	0.00%	0.00%	0.00%	0.00%	6.67%
GRAND TOTALS	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	5.57E-05

TABLE 10.8 - PERCENT ATWS CONTRIBUTIONS TO TOTAL CORE MELT

INITIATOR	PEACH BOTTOM II	SURRY UNIT I	BIG ROCK POINT	LIMERICK UNIT I	IND. PT. UNIT 2	IND. PT. UNIT 3	OCONEE UNIT 3	ZION UNIT I	AVERAGE CONTR.
LARGE LOCA	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
MEDIUM LOCA	0.00%	0.00%	0.00%	0.01%	0.00%	0.00%	N/A	0.00%	0.00%
SMALL LOCA	0.00%	0.00%	0.00%	0.03%	0.00%	0.00%	0.00%	0.05%	0.01%
INTER. SYST. LOCA	0.00%	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
REACTOR VESSEL RUPTURE	0.00%	0.00%	N/A	N/A	N/A	N/A	0.00%	N/A	0.00%
LOCA TOTALS	0.00%	0.00%	0.00%	0.03%	0.00%	0.00%	0.00%	0.05%	0.02%
STM. LINE BREAK INSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.00%	0.00%	N/A	0.00%	0.00%
STM. LINE BREAK OUTSIDE CONTAIN.	N/A	N/A	0.00%	N/A	0.00%	0.00%	N/A	0.00%	0.00%
LOSS OF MAIN F.W.	N/A	N/A	0.25%	0.47%	0.25%	0.72%	0.26%	7.77%	1.62%
MSIV TRIP	N/A	N/A	0.22%	0.73%	0.00%	0.00%	N/A	0.00%	0.19%
LOSS OF CONDENSER TURBINE TRIP	N/A	N/A	0.22%	N/A	N/A	N/A	0.08%	N/A	0.15%
TURBINE TRIP: LOOP	N/A	N/A	1.05%	1.28%	0.28%	0.51%	1.98%	5.53%	1.77%
REACTOR TRIP	N/A	N/A	N/A	0.00%	0.00%	0.00%	INCL. IN T.T.	0.00%	0.00%
LOSS OF RCS FLOW	N/A	N/A	N/A	N/A	0.00%	0.03%	N/A	0.49%	0.17%
S.G. TUBE RUPTURE	N/A	N/A	N/A	N/A	0.00%	0.00%	0.00%	0.00%	0.00%
OTHER TRANSIENTS	N/A	N/A	0.56%	0.17%	0.01%	0.02%	0.00%	0.96%	0.29%
TRANSIENT TOTALS	40.32%	9.54%	2.76%	2.77%	0.65%	1.55%	2.41%	14.76%	4.15%
INTERNAL TOTALS	40.32%	9.54%	2.76%	2.81%	0.65%	1.55%	2.41%	14.81%	4.17%
SEISMIC EVENTS	0.00%	0.00%	0.00%	1.87%	0.00%	0.00%	0.00%	0.00%	0.31%
FIRES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TORNADOS	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
AIRCRAFT CRASHES	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
FLOODS	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
TURBINE MISSILES	0.00%	0.00%	N/A	0.00%	0.00%	0.00%	N/A	0.00%	0.00%
HIGH ENERGY LINE BREAKS	0.00%	0.00%	0.00%	N/A	N/A	N/A	N/A	N/A	0.00%
EXTERNAL TOTALS	0.00%	0.00%	0.00%	1.87%	0.00%	0.00%	0.00%	0.00%	0.31%
GRAND TOTALS	40.32%	9.54%	2.76%	4.68%	0.65%	1.55%	2.41%	14.81%	4.48%
RPS UNAVAIL. (per d)	1.30E-05	3.70E-05	3.50E-05	3.00E-05	2.01E-05	3.93E-05	3.00E-05	1.80E-04	5.57E-05

TABLE 10.9 - COMPARISON OF PERCENT ATWS CONTRIBUTIONS FROM TWO STUDIES

PLANT	% ATWS CONTRIBUTION FROM REFERENCE STUDY [1]	% ATWS CONTRIBUTION FROM THIS STUDY
PEACH BOTTOM	47	40.32
BIG ROCK POINT	0	2.76
LIMERICK	2	4.68
SURRY	9	9.54
INDIAN POINT 2	0	0.65
INDIAN POINT 3	1	1.55
OCONEE 3	11	2.41
ZION	15	14.81