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

January 21, 1992

MEMORANDUM FOR: David J. Lange, Acting Director
Project Directorate II-3
Division of Reactor Projects I/II, NRR

FROM: Timothy A. Reed, Project Manager
Project Directoraet II-3
Division of Reactor Projects I/II, NRR

SUBJECT: SUMMARY OF MCGUIRE NUCLEAR STATION IPE/IPEEE

Duke Power Company (DPC) submitted the Individual Plant Examination (IPE) report for the McGuire Nuclear Station on November 4, 1991, in accordance with Generic Letter (GL) 88-20. The IPE included a Level 3 Probabilistic Risk Analysis (PRA) that included internal and external events. Although the IPE submittal addressed external events, it did not explicitly address GL 88-20 Supplement 4 (IPEEE) as noted in DPC's 180 day response to GL 88-20 Supplement 4 provided on December 18, 1991. In order to address the remaining GL 88-20 Supplement 4 issues, DPC is using a combined PRA/seismic margin approach to address the seismic IPEEE issue. DPC was not able to provide a finalized plan and schedule for the seismic portion of the IPEEE since their response is contingent upon the staff's issuance of the SSER on USI A-46.

DPC began staffing of personnel to work on severe accident issues in the early 1980s. An initial PRA for McGuire Nuclear Station was started in 1982 and completed in 1984. In 1988 DPC initiated a large-scale review and update of the original study. The level 3 PRA provided by DPC on November 4, 1991, is the result of that update effort. This PRA provides the primary basis for the IPE/IPEEE. As with all other IPE's received to date, RES has established a review team to review the McGuire IPE and has scheduled a review kickoff meeting for January 14, 1992, at which time a detailed review schedule will be established. RES has also established a review team for the external events portion of the McGuire level 3 PRA. An initial kickoff meeting was held December 18, 1991. Plans are to hold a second meeting in February 1992 to establish the review schedule for the external events portion.

The conclusion in the IPE report is that none of the accident sequences examined "demonstrate any unique plant vulnerability." The estimated core damage frequency (CDF) for internal and external events is 4.0E-5 and 3.4E-5, respectively, giving a total CDF estimate of 7.4E-5. No single sequence constitutes more than 25% of the total risk. The dominant internal event functional sequence (TQsU) is a reactor coolant pump (RCP) seal LOCA with a failure of injection capability. The initiator can be a tornado-induced, prolonged station blackout or a loss of all Unit 1 nuclear service water followed by a failure of containment ventilation cooling water. Either scenario results in loss of cooling water to the RCP seals, resulting in a 100 gpm/pump seal LOCA. This sequence represents about 24% of the total CDF estimate. The next highest sequence was a small break LOCA followed by

50-369

failure of long-term ECCS recirculation capability. This represents about 20% of the total risk. All other sequences are less than 10% of the total risk. A copy of the IPE Summary and conclusions is enclosed for your information.

A number of plant enhancements, including physical modifications and procedure upgrades, were taken as a result of the original McGuire PRA. Following the PRA update, sensitivity studies were conducted to evaluate the relative benefit of several potential additional enhancements. The IPE lists five specific enhancements that are presently being evaluated by DPC. These items will be discussed with DPC as the IPE review proceeds.

/s/

Timothy A. Reed, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

Enclosure:
As stated

cc: T. Murley, 12G18
F. Miraglia, 12G18
J. Partlow, 12G18
S. Varga, 14E4
G. Lainas, 14H3
R. Hernan, 14C7
W. Russell, 12G18
A. Thadani, 8E2
J. Richardson, 7D24
J. Flack - RES, NLS324
W. Beckner, 10E4
D. Wheeler, 12D22
C. Rossi, 11D4
B. Grimes, 9A2
F. Congel, 10E2
J. Roe, 10H5
A. Herdt, RII

DISTRIBUTION:
Docket File (50-269/370)
NRC PDR
PDII-3 R/F
McGuire R/F
TReed
LBerry

OFC	:LA:PDII-3	:PM:PDII-3	:D(A):PDII-3	:	:
NAME	:LBerry	:TReed:cw	:DLange	:	:
DATE	:1/10/92	:1/13/92	:1/14/92	:	:

OFFICIAL RECORD COPY
Document Name: MCGUIRE IPE

280041

DF01
11

failure of long-term ECCS recirculation capability. This represents about 20% of the total risk. All other sequences are less than 10% of the total risk. A copy of the IPE Summary and conclusions is enclosed for your information.

A number of plant enhancements, including physical modifications and procedure upgrades, were taken as a result of the original McGuire PRA. Following the PRA update, sensitivity studies were conducted to evaluate the relative benefit of several potential additional enhancements. The IPE lists five specific enhancements that are presently being evaluated by DPC. These items will be discussed with DPC as the IPE review proceeds.

Timothy A. Reed, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

Enclosure:
As stated

cc: T. Murley, 12G18
F. Miraglia, 12G18
J. Partlow, 12G18
S. Varga, 14E4
G. Lainas, 14H3
R. Hernan, 14C7
W. Russell, 12G18
A. Thadani, 8E2
J. Richardson, 7D24
J. Flack - RES, NLS324
W. Beckner, 10E4
D. Wheeler, 12D22
C. Rossi, 11D4
B. Grimes, 9A2
F. Congel, 10E2
J. Roe, 10H5
A. Herdt, RII

ENCLOSURE

Duke Power Company

MCGUIRE NUCLEAR STATION

IPE SUBMITTAL REPORT

Duke Power Company

MCGUIRE NUCLEAR STATION

IPE SUBMITTAL REPORT

November 1991

TABLE OF CONTENTS

LIST OF TABLES

1.0 INTRODUCTION

1.1 BACKGROUND

In March 1982 Duke Power Company initiated a Probabilistic Risk Assessment (PRA) Study of the McGuire Nuclear Station and this study was completed in July 1984. Subsequently, in 1988 Duke began a program to update this study to take into account a number of modifications to the plant and to take advantage of plant specific data and state-of-the-art methods. By the Duke letter of November 1, 1989, Duke informed tho NRC of the Duke plan to utilize this updated PRA to meet the requirements of Generic Letter 88-20 concerning the Individual Plant Examination (IPE). Consistent with the IPE submittal plans outlined in the November 1, 1989 Duke letter and approved by the NRC letter of January 24. 1990, Duke Power Company provides herein the complete response to GL 88-20.

This response includes this report (designated as the IPE Submittal Report) and the three-volume McGuire Nuclear Station Unit 1 Probabilistic Risk Assessment (McGuire PRA) report. To facilitate the NRC staff review, a cross-reference of the information requested in NUREG-1335 to the appropriate sections of the McGuire PRA is provided in Table 1.1-1.

The McGuire PRA is a full-scope, level 3 PRA with the analysis of external events. As such, this submittal is sufficiently responsive to GL 88-20 IPE for internal events and external events. No further effort concerning external events is considered appropriate for McGuire.

1.2 METHODOLOGY

1.2.1 Organization Elements

Duke Power Company's initial staffing to enable large scale PRA and reliability studies in-house began in 1980. A severe accident analysis group was organized and charged with the responsibility to plan, conduct, and coordinate all proposed PRA studies and to maintain and update the plant PRA models as appropriate. In addition to PRA studies, this group is also

The plant systems have been analyzed with detailed fault trees, generally to the component level. The level of detail in the model is defined by the level at which data is available. Fault trees have been developed for both front-line and support systems. A front-line system (e.g., Safety Injection System) is modeled down through its support systems (e.g., Nuclear Service Water), and the support systems are modeled down through their support systems (e.g., AC Power System). In this manner, support system fault trees are directly linked to front-line system fault trees and to each other.

The plant system models have been fully assembled into accident sequence models and solved using the CAFTA computer code. Plant-specific data has been used for many of the accident initiators, as described in Section 2.1 and Appendix C of the McGuire PRA. Plant-specific data has also been used for maintenance unavailabilities and component failure rates in many system models, as described in Appendices A and C of the McGuire PRA.

The result of these activities is a list of accident sequence cut sets. These cut sets have been analyzed for recovery and grouped by both initiator and functional sequence in Appendix D of the McGuire PRA.

The external events analysis (described in detail in Section 3.0 of the McGuire PRA) draws upon the information and logic models developed for the internal events analysis. The seismic event tree uses the fault trees and top logic and includes only those components with high random failure rates coupled with fragility information for the major components. The tornado analysis considers the same logic in terms of plant functions and systems but focuses on the effects of wind loadings and missiles. The flood and fire analysis use the same models generated for the transient event tree.

One other area of front end analysis is the development of models for the containment safeguards event tree (Section 4.0 of the McGuire PRA). The systems which affect radiological release but which are not critical to core protection are modeled here. The accident sequence cut sets for core damage are coupled with the possible containment safeguards states, resulting in plant damage states for the beginning of the back end analysis.

are defined by the CET endpoints and their paths through the tree. The MAAP code is used to determine release magnitudes by modeling the sequence defined by this path.

5. Off-site Consequence Analysis - Release category definitions and other McGuire plant-specific information are used as input to the CRAC2 computer code which calculates the public health consequences for each release category. The results are provided in the form of conditional CCDFs as well as the mean values.

Since the McGuire PRA is a full-scope, level 3 PRA, the consequence analysis goes beyond the objectives of the back end analysis requested by Generic Letter 88-20 and NUREG-1335. To assist the NRC in its review of this submittal, Table 1.2-2 provides a listing of plant parameters important to the back end analysis. A more detailed discussion of the McGuire PRA consequence analysis is included in Section 6.0 and 7.0 ι the McGuire PRA.

1.2.4 Walkdown

As part of the plant familiarization process, Duke PRA analysts perform plant walkdowns. The PRA analysts are usually guided by plant personnel, often from the Operations Group or the Design Engineering site office who have some involvement or understanding of the PRA. These walkdowns supplement the information contained in various engineering documents. Walkdowns are invaluable in determining location dependent effects such as:

- potential systems interaction and common cause failures due to flooding, fire and other externally-induced failures
- the ease or difficulty of various operator actions that may be modeled as recovery events

The plant walkdown is also used to provide a general understanding of the arrangement of plant systems.

Plant walkdowns in support of McGuire PRA activities have been performed several times during the course of the PRA effort:

Besides the technical review of the PRA, management briefings are given to apprise key management personnel of the results and conclusions.

Specifically for the McGuire IPE, the draft PRA report was reviewed by the McGuire engineering and station personnel familiar with the plant systems and/or operator actions. Subsequently, presentations were made to the McGuire station and engineering supervisory/management personnel when the IPE results and plant enhancement studies were completed. This review and dialogue facilitated the formulation and endorsements of plant enhancements discussed in Section 3 of this report.

1.2.6 Review of Industry PRAs

As discussed in Section 1.2.1 of this submittal, Duke organized a Severe Accident Analysis Group in the early 1980s. This group reviews industry and NRC studies and participates in industry organizations (such as IDCOR, EPRI and NUMARC) dealing with severe accident and PRA issues. These organizations provide a forum for exchanging information and staying abreast of the latest developments. Many of the reports listed in Attachment 2 to Generic Letter 88-20 have been reviewed for insights and lessons learned. NUREG/CR-4405, "Probabilistic Risk Assessment (PRA) Insights," has also been reviewed. A few examples are discussed below.

Duke has commented extensively on the draft NUREG-1150 analysis and its supporting documentation. By reviewing the expert judgement information provided in the documentation, insights have been gained in such areas as the potential for reactor coolant pump seal LOCAs and the likelihood of direct containment heating.

Duke was an active participant in IDCOR and reviewed in detail the many reports published as a result of this effort. In particular, the reference plant analyses provided insight into such areas as best-estimate, thermal-hydraulic success criteria.

The RSSMAP study and the NUREG-1150 study on Sequoyah have been reviewed and issues raised in these reports have been assessed for applicability to

Table 1.1-1
Cross-Reference of NUREG-1335 Table 2.1 and
McGuire IPE Submittal Reports Sections

NUREG-1335 Item	McGuire IPE Submittal Reports Sections
1. Executive Summary	
1.1 Background and Objectives	Section 1, IPESR[1] & Section 1, MPRA[2]
1.2 Plant Familiarization	" "
1.3 Overall Methodology	" "
1.4 Summary of Major Findings	Section 2, IPESR
2. Examination Description	
2.1 Introduction	----
2.2 Conformance with Generic Letter and supporting material	Section 10, IPESR
2.3 General Methodology	Section 1.2, IPESR
2.4 Information Assembly	Section 1.2.6 and Table 1.2-2, IPESR & Section 1.1, MPRA
3. Front-End Analysis	
3.1 Accident Sequence Delineation	
3.1.1 Initiating Events	Section 2.1 & 3.1, MPRA
3.1.2 Front-Line Event Trees	Section 2.2, 2.3, MPRA
3.1.3 Special Event Trees	Section 2.4, 2.5, 2.6, & 3.3, MPRA
3.1.4 Support System Event Trees	Not Applicable
3.1.5 Sequence Grouping and Back-End Interfaces	Section 1.2.3, IPESR & Section 6, MPRA
3.2 Systems Analysis	
3.2.1 System Description	Appendices A-B, MPRA
3.2.2 System Analysis	Appendices A-B, MPRA
3.2.3 System Dependencies	Appendices A-B, MPRA
3.3 Sequence Quantification	
3.3.1 List of Generic Data	Section 2.1.3 & Appendix C, MPRA
3.3.2 Plant-Specific Data and Analysis	Section 2.1.3 & Appendix A & C, MPRA
3.3.3 Human Failure Data	Section 5 & App. A & C, MPRA

1. McGuire IPE Submittal Report is abbreviated IPESR

2. McGuire Probabilistic Risk Assessment Report is abbreviated MPRA

Table 1.2-2

McGuire Consequence Analysis Data

Reactor Power	3411 MWt
Steam Generators	Vertical, U-Tube
Pressurizer PORV	3, 2.10E5 lbm/hr at setpoint pressure, 2350 psia
Pressurizer Safety Valves	3, 4.20E5 lbm/hr at setpoint pressure, 2500 psia
Core Zircaloy Mass	45352.0 lbs
Containment Type	Ice Condenser, free standing steel shell
Containment Shell	aprrox. .75 in. thick
Containment Radius	57.5 ft.
Containment Volume	1.24E6 ft^3
Containment Design Pressure	15.0 psig
Containment Ultimate Mean Failure Pressure	76 psig
Containment Cavity Floor Area	1037.7 ft^2
Containment Basemat Thickness	8.0 ft
Containment Basemat Concrete	Silicous with no carbon
Unique Features Important To The Containment Analysis	Water must accumulate to a depth of 13 ft in the lower compartment to flow to the cavity. Glow plug igniters are installed to control hydrogen.

2.0 IPE RESULTS

2.1 FRONT END RESULTS

2.1.1 Internal Event Analysis

Introduction

Table 2.1-1 displays the results of the level 1 internal events analysis in terms of functional sequences and initiators. A key is provided to assist in understanding the nomenclature. The contribution of each internal initiator group to each functional sequence is given and then summed up in the column labeled INTERNAL TOTAL for a calculated annual core-melt frequency of 4.0E-05. The total core-melt frequencies from each initiator group are given in the bottom row and correspond to the values in Table 8.1-1 of the McGuire PRA. The percent values shown in Table 2.1-1 of this submittal represent the percent contribution to the total (internal plus external) calculated annual core-melt frequency of 7.4E-05. If the contribution from any particular category is less than 1.0E-06 per year, a '<' sign has been used, with the exception of steam generator tube rupture and interfacing-systems LOCA. Only those functional sequences with frequencies greater than 1.0E-06 per year are discussed here. A detailed discussion of internal events analysis can be found in Section 2.0 of the McGuire PRA (internal flooding is treated in Section 3.0 with the external events). A detailed listing of all cut sets contributing to these functional sequences can be found in Appendix D of the McGuire PRA.

TBU

Functional sequence TBU involves a total loss of secondary side heat removal along with a failure of injection capability.

Transient Initiators - Transient initiators, such as loss of instrument air and loss of off-site power, cause a loss of main feedwater (CF). It has been conservatively assumed that CF is not recovered when its loss is due to these particular initiators, which involve support systems to CF. Historical data

(RV) System to provide backup cooling to RN loads, dominates the loss-of-RN-scenario. In both the loss-of-all-ac-power scenario and the loss-of-RN-scenario, a failure of the Standby Shutdown System to provide makeup to the RCP seals results in an RCP seal LOCA. This LOCA leakage rate is assumed to be a best estimate value of 100 gpm per pump, conservatively beginning at 15 minutes after the loss of all seal cooling. The loss of RN scenario is also followed by a failure to cross-connect Unit 2 RN to Unit 1. The CA turbine-driven pump successfully provides feedwater to the steam generators. The frequency of this sequence is 1.8E-05 per year.

TQsX

Functional sequence TQsX involves an RCP seal LOCA with failure of long-term injection or ECCS recirculation capability. Secondary side heat removal is successful. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQsU

Functional sequence TBQsU involves a failure of secondary side heat removal and an RCP seal failure along with a failure of injection capability. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQsX

Functional sequence TBQsX involves a failure of secondary side heat removal and an RCP seal failure along with a failure of long-term injection or ECCS recirculation capability. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQrU

Functional sequence TBQrU involves a failure of secondary side heat removal and a stuck-open pressurizer relief valve along with a failure of injection capability. The contribution from this functional sequence is less than 1.0E-06 per year.

MX

Functional sequence MX involves a medium break LOCA (1.5 in. to 5.0 in. dia.) along with a failure of long-term recirculation capability. The dominant failure mode of high pressure recirculation is failure of 2 or more FWST level transmitters. The failure of the level transmitters will fail the automatic realignment of the ND System to the containment emergency sump upon depletion of the FWST. High pressure injection must "piggyback" onto the ND System to function in the recirculation mode. The level transmitter failure is followed by a failure of the operating crew to detect and diagnose the situation and manually realign the emergency core cooling systems before FWST depletion leads to pump cavitation and failure. The frequency of this sequence is 1.6E-06 per year.

LU

Functional sequence LU involves a large break LOCA (> 5.0 in. dia.) along with a failure of injection capability. An ATWS event, which is assumed to rupture the Reactor Coolant System and fail or block all injection flow, contributes approximately 60 percent to the probability of this scenario. Reactor pressure vessel rupture, which is assumed to divert all injection flow through the failure in the vessel, contributes the remaining 40 percent to the probability of this scenario. The uncertain probabilities of ATWS and vessel rupture are discussed in Section 2.0 of the McGuire PRA. The assumption that injection fails with a probability of 1.0 following either event is considered conservative. The frequency of this sequence is 2.5E-06 per year.

LX

Funtional sequence LX involves a large break LOCA (> 5.0 in. dia.) followed by failure of ECCS recirculation capability. The dominant failure mode of low pressure recirculation is failure of 2 or more FWST level transmitters. The failure of the level transmitters will fail the automatic realignment of the ND System to the containment emergency sump upon depletion of the FWST. No credit is taken in this sequence for the operating crew manually realigning the system to low pressure recirculation. This is because the FWST depletes

TBU

Functional sequence TBU involves a total loss of secondary side heat removal along with a failure of injection capability. The frequency of this sequence is 1.1E-06 per year.

TBX

Functional sequence TBX involves a total loss of secondary side heat removal followed by a failure of long-term feed-and-bleed cooling. The contribution from this functional sequence is less than 1.0E-06 per year.

TBP

Functional sequence TBP involves a total loss of secondary side heat removal followed by a failure of bleed capability for feed-and-bleed cooling. The contribution from this functional sequence is less than 1.0E-06 per year.

TQsU

Functional sequence TQsU involves an RCP seal LOCA with a failure of injection capability. Secondary side heat removal is successful.

Tornado - The dominant contributor to this sequence is a tornado-induced loss of off-site power followed by random failures of both diesel generators which result in a loss of all ac power. Damage to the grid by the tornado creates a prolonged loss of off-site power (assumed to be non-recoverable during the 24-hour mission time). The CA turbine-driven pump successfully provides feedwater to the steam generators. A failure of the SSF to provide makeup to the RCP seals results in an RCP seal LOCA. This LOCA leakage rate is assumed to be a best estimate value of 100 gpm per pump, conservatively beginning at 15 minutes after the loss of all seal cooling. The frequency of this sequence is 1.8E-05 per year.

Seismic - The dominant contributor to this sequence is a seismically-induced loss of off-site power followed by random diesel generator run failures which

TBQsX

Functional sequence TBQsX involves a failure of secondary side heat removal, an RCP seal failure, and a failure of long-term injection or ECCS recirculation capability. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQrU

Functional sequence TBQrU involves a failure of secondary side heat removal, a stuck-open pressurizer relief valve, and a failure of injection capability. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQrX

Functional sequence TBQrX involves a failure of secondary side heat removal, a stuck-open pressurizer relief valve, and a failure of long-term injection or ECCS recirculation capability. The contribution from this functional sequence is less than 1.0E-06 per year.

SU

Functional sequence SU involves a small break LOCA (< 1.5 in. dia.) along with a failure of injection capability. Secondary side heat removal is available. The contribution from this functional sequence is less than 1.0E-06 per year.

SX

Functional sequence SX involves a small break LOCA (< 1.5 in. dia.) along with a failure of long-term ECCS recirculation capability. Secondary side heat removal is available. The contribution from this functional sequence is less than 1.0E-06 per year.

2.1.3 Conclusions

Internal Transients

Internal transient events have been calculated to contribute approximately 34 percent to the total annual core-melt frequency. This contribution is dominated by functional sequence TQsU with a frequency of 1.8E-05 and, consequently, is an important contributor. The predominant scenarios involve either a loss of all ac power or a loss of nuclear service water. Due to the redundancy designed into frontline systems, a failure of a common support system of this type is usually necessary for an RCP seal LOCA to develop and result in core melt.

Due to the success of secondary side heat removal, at least 3 to 4 hours would exist before the onset of core damage. This would allow time for repair of faulted equipment such as diesel generators and nuclear service water pumps. Except for the recovery of off-site power, credit for fixing failed equipment has not been taken in the McGuire PRA. In addition, a conservative approach has been taken in modeling the timing and leakage rate of an RCP seal LOCA.

The ability exists to cross-connect the Nuclear Service Water Systems of the two Units at McGuire. In addition, the Containment Ventilation Cooling Water System has the ability to feed the nuclear service water headers and, therefore, cool the loads normally cooled by nuclear service water. System availability and potential operator error are the key issues here.

The SSF design provides a totally independent means of RCP seal cooling and, thus, an additional level of RCP seal LOCA protection. The SSF can and would be utilized in both the loss of all ac power scenario and the loss of all nuclear service water scenario. The SSF, with its own power system, does not require any plant support system to operate. Operator action is necessary to initiate SSF operation.

While the calculated core-melt frequencies for these scenarios are not insignificant, it is safe to say that potentially conservative PRA modeling and the assumed operator error probabilities drive these frequencies.

pressure. Spray flow will deplete the FWST before residual heat removal entry conditions are reached, requiring ECCS recirculation. There are no cut sets with a frequency above 1.0E-08 per year which involve failure of SSHR after a LOCA. Therefore, secondary side heat removal is available for the most probable LOCA-initiated sequences.

FWST level transmitters are required to (i) alert the operators of the need to swap-over to high pressure recirculation and (ii) initiate automatic residual heat removal pump suction swapover to the containment emergency sump. If these detectors were to fail "high" or "as is", the failure would be undetectable during normal operation and the initial phases of a LOCA. This is due to the limited range of the transmitter and control room indications (0" - 160") while normal FWST level is approximately 480". A limited amount of time would be available for the operators to detect the multiple failure and manually align ECCS recirculation before pump degradation and potential failure could be expected. A more detailed discussion of this scenario can be found in Sections 2.3 and 8.1 of the McGuire PRA.

Various options are currently being investigated to determine the best way to help the operating crew respond to this scenario. These efforts to reduce the significance of this potential plant vulnerability are discussed in Section 3.

Seismic

Seismic events have been calculated to contribute approximately 19 percent to the total annual core-melt frequency. This contribution is split between functional sequences TQsU and TBQsU. A seismically-induced loss of off-site power followed by failure of the emergency diesel generators represents approximately 75 percent of the total seismically-induced core-melt frequency. At low ground accelerations, diesel failures are due to random failures (predominantly, a failure to run for their 24-hour mission time). At ground accelerations above 0.6g, the diesel failures are predominantly seismic failures (starting air tanks, dc control power). The loss of off-site power

TABLE 2.1-1
SUMMARY OF IPE RESULTS

TOTAL CORE-MELT FREQUENCY =7.4E-05

(<) MEANS LESS THAN 1.0E-06

	INTERNAL TRANSIENTS	SGTRs	LOCAs	INTERNAL FLOODING	INTERNAL TOTAL	SEISMIC	EXTERNAL FLOODING	TORNADO	FIRE	EXTERNAL TOTAL
TBU	3.9E-06 5.3%	-	-	-	3.9E-06 5.3%	<	-	<	-	1.1E-06 1.5%
TBX	<	-	-	-	<	<	-	-	-	<
TBP	<	-	-	<	<	<	-	-	-	<
TOsU	1.8E-05 24.3%	-	-	<	1.8E-05 24.3%	6.1E-06 8.2%	-	1.8E-05 24.3%	-	2.4E-05 32.4%
TOsX	-	-	-	-	-	-	-	-	-	-
TBOsU	-	-	-	-	-	7.0E-06 9.5%	-	-	<	7.2E-06 9.7%
TBOsX	-	-	-	-	-	-	-	-	-	-
TBOrU	<	-	-	-	<	<	-	<	-	<
TBOrX	-	-	-	-	-	-	-	-	-	-
SU,MU,LU	2.5E-06 3.4%	<	<	-	2.7E-06 3.6%	-	-	-	-	-
SX,MX,LX	-	<	1.5E-05 20.3%	-	1.5E-05 20.3%	-	-	-	-	-
ISLOCA	8.1E-09 <1.0%	-	-	-	8.1E-09 <1.0%	-	-	-	-	-
TOTAL	2.5E-05 33.8%	8.6E-09 <1.0%	1.5E-05 20.3%	<	4.0E-05 54.0%	1.4E-05 18.9%	-	1.9E-05 25.7%	<	3.4E-05 46.0%

2.2 CONSEQUENCE RESULTS

2.2.1 Containment

The results of the McGuire containment analysis are presented in Table 2.2-1. This table provides the percentage contribution for the six possible containment condition end states for internal and external initiators and the total for all initiators. Table 2.2-2 provides a cross reference between the containment condition end states presented in Table 2.2-1 and the release categories of the McGuire PRA.

The following insights can be drawn from Table 2.2-1.

Late Containment Failure - For sequences in which containment fails, a late containment failure is the most likely containment failure mode. Approximately 42% of the core melt frequency falls into the late containment failure release categories. Water entering the reactor cavity results in steam generation which in the absence of containment heat removal eventually results in containment overpressurization. The Refueling Water Storage Tank (FWST) can drain into the failed reactor vessel and thereby into the cavity in many sequences. For many of these sequences, containment heat removal is not available, due to a loss of ac power or the Nuclear Service Water (RN) System, and cannot be recovered prior to containment failure due to steam overpressure. If the RWST alignment did not permit this draining to occur, these sequences would mostly fall into the basemat melt-through release categories.

Containment Bypass - The probability of the containment being bypassed is low. Approximately 1.4% of the core melt frequency results in a containment bypass. This frequency would be less than .2% were it not for the occurrence of induced steam generator tube ruptures during core degradation. Starting the reactor coolant pumps per procedure following core uncovery may lead to failure of the steam generator tubes as hot gases from the core region are transported into the steam generator.

2.2.2 Source Term

The McGuire PRA containment event tree has 42 possible end points. Each of these could define a specific release category. Of these 42, only 32 end points had frequencies justifying a release category definition and discussion in Section 6.3 of the McGuire PRA report. The release categories significant to risk are described here. Their fission product release fractions and the timing of the release are presented in Table 2.2-3.

Release Categories 1.02 and 1.04 (Steam Generator Tube Rupture)

These release categories are characterized as steam generator tube ruptures, which are induced by tube overheating during core uncovery. The difference between RC 1.02 and RC 1.04 is that RC 1.02 has no ex-vessel release while 1.04 does. A steam line safety valve or drain line is assumed to remain open so that the reactor coolant system depressurizes through the rupture tube. This conservatively bounds the release fractions and allows the escape of fission products that are released ex-vessel. The dominant sequence contributing to release categories 1.02 and 1.04 is the auxiliary feedwater pump room flood. Other important sequences contributing to these release categories are a loss of the Nuclear Service Water (RN) System and a seismically induced loss of all ac power.

The releases associated with these events are large since the release is early and the containment is bypassed. Little warning time is available for the evacuation to be effective.

Release Category 2.04 (Interfacing Systems LOCA)

This release category is characterized as a containment bypass with an ex-vessel release and the release going to the Auxiliary Building. The only sequence contributing to this release category is a large LOCA at the Residual Heat Removal (ND) System heat exchanger. The location of this failure is such that no credit is taken for plate-out in the Auxiliary Building.

TABLE 2.2-1

SUMMARY OF CONTAINMENT ANALYSIS RESULTS

CONTAINMENT END STATES	INTERNAL	EXTERNAL	TOTALS
CONTAINMENT BYPASS	9.6E-07 (a) 2.4%	9.0E-08 0.3%	1.1E-06 1.4%
ISOLATION FAILURE	1.3E-07 0.3%	3.5E-07 1.0%	4.7E-07 0.6%
EARLY FAILURE	8.2E-07 2.0%	2.3E-06 6.7%	3.1E-06 4.2%
LATE FAILURE	1.4E-05 35.2%	1.7E-05 49.5%	3.1E-05 41.7%
BASEMAT MELT-THROUGH	2.4E-06 5.9%	1.4E-06 4.2%	3.8E-06 5.1%
NO CONTAINMENT FAILURE	2.2E-05 54.2%	1.3E-05 38.3%	3.5E-05 46.9%
TOTALS	4.05E-05	3.37E-05	7.41E-05

(a) PERCENTAGES APPLY FOR THE COLUMN ONLY

TABLE 2.2-3
IMPORTANT MCGUIRE PRA RELEASE CATEGORIES

CONTAINMENT FAILURE TYPE	RELEASE CATEGORY	RELEASE TIME(HRS)(a)	DURATION OF RELEASE (HRS)	WARNING TIME (HRS)	RELEASE FRACTIONS							
					Xe	I	Cs-Rb	Te-Sb	Ba	Ru	La	Sr
SGTR	1.02	3.0	1.0	2.0	1.00	4.80E-01	4.80E-01	1.70E-01	1.00E-02	2.40E-02	9.00E-05	1.00E-03
SGTR	1.04	3.0	1.0	2.0	1.00	9.10E-01	9.20E-01	5.00E-01	3.30E-02	6.40E-02	1.50E-04	4.60E-03
CONTAINMENT BYPASS	2.04	3.0	1.0	2.0	1.00	7.50E-01	7.50E-01	6.60E-01	9.80E-02	1.50E-01	2.00E-02	4.20E-02
EARLY FAILURE	5.01	3.25	0.5	2.75	1.00	1.60E-02	1.60E-02	8.00E-03	3.50E-04	9.30E-04	1.30E-05	3.00E-05
LATE CAT. FAILURE	6.02	33.0	0.5	32.5	1.00	3.20E-04	5.30E-04	1.90E-04	1.10E-03	8.80E-04	2.90E-06	2.90E-04
LATE CAT. FAILURE	6.04	33.0	0.5	32.5	1.00	3.20E-03	5.30E-03	1.90E-04	1.10E-03	8.80E-04	2.90E-06	2.90E-04

(a) THE TIME OF RELEASE IS MEASURED FROM THE TIME OF REACTOR TRIP.

2.3 RISK RESULTS

Section 8.2 of the McGuire PRA report contains a detailed discussion of the risk results. Tables 2.3-1, 2.3-2. and 2.3-3 present the results of the McGuire PRA in the form of mean values of five public risk measures. These are divided to show the contributions from the internal and external initiators as well as the total of all initiators. Also provided in the tables are the relative contributions of the various containment failure modes.

Early Fatality and Early Injury Risk

For both the internal and external initiators, the early health effects are dominated by the SGTR (induced) release categories 1.02 and 1.04.

Latent Cancer Fatalities, Thyroid Nodules and Whole-Body Person Rem

The latent cancer fatalities, thyroid nodules and whole-body person rem risks, for both internal and external initiators, are all dominated by the following release categories:

- Release category 1.02; steam generator tube rupture (induced)
- Release category 5.01; early containment failure
- Release category 6.02; late containment failure (w/o revap.)
- Release category 6.04; late containment failure (w/revap.)

TABLE 2.3-2

SUMMARY OF McGUIRE PRA RISK RESULTS FOR EXTERNAL INITIATORS

CONTAINMENT END STATES	EARLY FATALITIES / YR	EARLY INJURIES / YR	LATENT FATALITIES / YR	THYROID NODULES / YR	WHOLE-BODY PERSON-REMS / YR
CONTAINMENT BYPASS	2.1E-06 92.8% (c)	3.0E-05 97.6%	6.2E-05 7.4%	4.5E-04 21.3%	8.5E-01 6.8%
ISOLATION FAILURE	1.4E-07 6.5%	6.9E-07 2.2%	5.4E-05 6.4%	1.7E-04 8.1%	8.1E-01 6.5%
EARLY FAILURE	1.6E-08 0.7%	5.6E-08 0.2%	3.4E-04 40.9%	8.0E-04 37.6%	5.2E+00 41.8%
LATE FAILURE	0.0E+00 0.0%	0.0E+00 0.0%	3.7E-04 44.5%	6.7E-04 31.5%	5.5E+00 44.1%
BASEMAT MELT-THROUGH	0.0E+00 0.0%	0.0E+00 0.0%	5.6E-06 0.7%	2.8E-05 1.3%	7.9E-02 0.6%
NO CONTAINMENT FAILURE	0.0E+00 0.0%	0.0E+00 0.0%	1.2E-06 0.1%	3.1E-06 0.1%	2.1E-02 0.2%
TOTALS	2.23E-06 (a)	3.08E-05 (a)	8.39E-04 (b)	2.11E-03 (b)	1.25E+01 (b)

NOTE:

(a) BASED ON 5% OF EPZ NOT PARTICIPATING IN EVACUATION
FOR 0.5% NON-PARTICIPATION, EARLY FATALITIES=5.24E-07 / YR, EARLY INJURIES=2.67E-05 / YR.

(b) BASED ON 0 TO 2000 MILE POPULATION DATA
FOR 0 TO 50 MILE POPULATION, LATENT FATALITIES=5.52E-04 / YR, THYROID NODULES=1.36E-03 / YR, WHOLE-BODY PERSON-REMS=8.37E+00 / YR

(c) PERCENTAGES APPLY FOR THE COLUMN ONLY

3.0 PLANT ENHANCEMENTS

3.1 ACTIONS TAKEN DUE TO THE INITIAL STUDY

As a result of the original McGuire PRA, certain plant enhancements were implemented as risk reduction measures.

One area of enhancement pertained to plant procedure enhancements -- one dealing with a loss-of-nuclear service water event and the other dealing with a loss of all ac power event. Recognizing the importance of a loss-of-nuclear service water event, procedural guidance for operator's use was developed and implemented to better cope with the event. Also, operator actions for the loss of ac power procedure were prioritized such that the action to locally isolate the containment ventilation unit condensate drain line could be taken reliably.

Another plant enhancement related to a potential flooding condition in the auxiliary feedwater pump room. Expansion joints in the nuclear service water piping located in this room were discovered not to include a metal collar to limit the leakage. Thus, to reduce the likelihood of a large flooding event from this source, the expansion joints have been subsequently fitted with a collar to limit the leak rate.

3.2 ADDITIONAL PLANT ENHANCEMENTS

Upon completion of the McGuire PRA update, a searching examination of the results (both the core melt frequencies and fission product release potential) was made with the objective of identifying additional viable plant enhancements. This examination process consisted of first identifying potential enhancements which could potentially have an appreciable impact on core melt sequences or release potential. Sensitivity studies were then performed to determine the quantitative impact of these candidate items by varying the probability of failure of this event from the base case values. The sensitivity study results were then captured in terms of changes in core melt frequency, impact on release potential, and change in whole body person-rem. The calculated change in person-rem was multiplied by the

Reactor Coolant Pump Restart Criteria

The existing plant emergency procedure for inadequate core cooling conditions directs the operator to restart the reactor coolant pumps. If the secondary side heat removal is unavailable and the PZR PORVs are not open, the forced circulation of very hot gases from the core at high pressure could overstress the steam generator tubes, creating a containment bypass situation. Additional procedural guidance to permit pump startup only when the SG tubes are covered with a mixture level has been recommended to eliminate this concern. The procedure change process has been initiated to accomplish this enhancement.

RN Cross-Connect

In the event of a total loss of nuclear service water (RN) in one unit, the RN system from the other unit could be lined up to serve the critical loads of the affected unit by opening the RN cross-connect valves. These manual valves are normally locked-closed. There is some uncertainty on the probability of successful timely action. It has been suggested that a periodic exercising of these valves (during refueling outages) could enhance the confidence on this recovery action.

Station personnel are considering this test for implementation.

Diesel Generator Reliability

The Diesel Generator (D/G) failure data used in this study came from plant-specific experience. The values are as follows:

Maintenance unavailability	4.36E-02
Start failure probability	6.0E-03
Run failure probability	8.34E-03/hour

The sensitivity analysis involving these failure rates suggests that a modest improvement in core melt frequency could be achieved by a factor of 2 improvement in D/G reliability parameters.

- Modest benefit but ongoing awareness of effort is adequate
- Prohibitive cost in comparison to the expected benefit

Accordingly, these items were not considered further

Table 3.3-1 identifies these items.

4.0 ACCIDENT MANAGEMENT

For some time, Duke Power Company has been utilizing the insights gained from the PRA and the technology associated with severe accident analyses for a number of applications. Applications include such areas as improvement of plant procedures, emergency planning, emergency exercises, and training. Thus, many of the objectives of the accident management program are already being accomplished for McGuire although not through a formal accident management program. Highlights of specific applications follow.

Plant Procedure Improvements As part of the human interactions model for the PRA, relevant plant procedures (primarily the emergency and abnormal procedures) have been reviewed and areas of potential enhancements have been identified. Guidance in dealing with a loss of service water event and the enhancement of the station blackout procedure are examples where enhancements have been implemented. An additional procedural enhancement now being considered for implementation is the reactor coolant pump restart criteria to minimize the potential for an induced SG tube rupture.

The generic strategies identified in Generic Letter 88-20, Supplement 2, have also been reviewed. It has been found that several of these strategies are already in the existing procedures or are similar to the strategies presently under consideration for implementation as part of the overall enhancement of an accident management program, discussed later in this section.

Emergency Planning and Emergency Exercises One of the most effective applications of the Duke PRAs has been in the area of emergency planning. The plant groups responsible for developing and implementing the annual exercises have utilized the PRA for:

- risk important scenarios
- realistic and likely failure modes
- realistic timing of events and accident progression
- realistic source terms

Duke personnel are currently working with the NUMARC Severe Accident Working Group, NUMARC Joint Owners Group and the owners group on the industry program for accident management. When this generic program is completed, appropriate enhancements to existing programs will be considered for implementation.

5.0 CONTAINMENT PERFORMANCE IMPROVEMENT (CPI) PROGRAM ISSUES

In response to Supplement 3 of Generic Letter 88-20, Duke Power has reviewed the IPE results for any vulnerability to the loss of power to the hydrogen igniters.

Hydrogen control during severe accidents is accomplished at McGuire through the operation of glow plug igniters. A total of 70 igniters are located throughout the containment. Two trains of igniters, 35 igniters each, are provided so that no single failure can result in a loss of igniter capability at any installed location. The igniters are powered from 120 Vac panelboards. These panelboards receive power from the onsite emergency power system and can be energized from the control room. The emergency operating procedures call for the igniters to be energized early in an accident which indicates a high energy line break in containment.

The unavailability of the hydrogen igniters is dominated by the failure of power to the igniters. During a station blackout power would be lost to the igniters, possibly allowing high concentrations of hydrogen to accumulate. This hydrogen may ignite in an uncontrolled manner upon power restoration. In order to evaluate the potential benefit of providing a backup power supply to the igniters a sensitivity study was conducted. In this study the CET was quantified in a manner which assured that the igniters were always available.

Containment failure due to large accumulation of hydrogen in the containment is categorized as early containment failure. In the base case calculation the early containment failure accounts for about 4.2%. As seen from Table 2.3-3, the risk contributions from early containment failure are: 2.3 x E-08 early fatalities (0.1%), 4.7E-04 latent fatailities (25%), and 7.1 whole-body person-rems (26.4%). The results of the sensitivity study indicate that the mean value of the whole-body person-rem risk could be reduced by approximately 5.7 person-rem if the igniters were always available. Except, perhaps, for the severe accident management strategy, it is not expected that any cost effective means is available to dramatically improve the igniter availability for the loss of ac power sequences. The SSF has

6.0 SHUTDOWN DECAY HEAT REMOVAL ANALYSIS (formerly USI A-45)

USI A-45, entitled "Shutdown Decay Heat Removal Requirements", has as its objective the determination of the adequacy of the decay heat removal function at operating plants and possible identification of cost-effective improvements. It has been concluded by NRC that a generic resolution to the issue is not cost-effective and that resolution can only be achieved on a plant-specific basis. Therefore, NRC has subsumed A-45 into Generic Letter 88-20 and has requested an evaluation of decay heat removal vulnerabilities during power operation and hot standby. To this end, Appendix A of this submittal provides a detailed evaluation of shutdown decay heat removal for McGuire.

The calculated annual core-melt frequency due to failure of decay heat removal systems for internal initiators (including internal flooding) is 1.6E-05. The McGuire decay heat removal systems are robust and do not demonstrate any particular vulnerability to internally initiated severe accident sequences. The reliability of feedwater systems together with the capability to accomplish feed-and-bleed cooling make the McGuire decay heat removal systems sufficiently reliable.

The calculated annual core-melt frequency due to failure of decay heat removal systems for external initiators is 1.0E-05. Thus the McGuire decay heat removal systems exhibit high resistance for external events also.

It can be concluded that McGuire does not exhibit any particular vulnerability to loss of decay heat removal. The plant enhancements discussed in Section 3 would reduce the core damage potential due to failure of decay heat removal systems. Therefore, this issue should be considered resolved for McGuire.

7.0 SYSTEMS INTERACTION DUE TO INTERNAL FLOODING (USI A-17)

7.1 INTRODUCTION

Generic Letter 89-19 informs licensees that the NRC has concluded its resolution of USI A-17, "Systems Interactions in Nuclear Power Plants." The staff has identified actions to be taken by the NRC to resolve USI A-17 and has made the judgement that these actions, together with ongoing activities, should reduce the risk from adverse systems interaction. In addition to actions by the NRC, licensees are expected to take two actions: (i) consider insights from the appendix to NUREG-1174 in implementing the IPE requirement of an internal flooding assessment and (ii) continue to review information on events at operating nuclear power plants.

7.2 WATER INTRUSION AND FLOODING FROM INTERNAL SOURCES

An extensive analysis of internal flooding was performed as part of the McGuire PRA and is discussed in Section 3.3 of the PRA report. This analysis made use of experience in the flooding study initially performed in the Oconee PRA. Insights from NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants," have been reviewed. Intersystem dependencies due to both functional coupling and spatial coupling have been included as part of the analysis. Historical data on internal flooding events have been reviewed for insights and for estimating the frequency of such events. Potentially affected systems, in both the Turbine and Auxiliary Buildings, have been included in the analysis. An awareness of internal flooding and its potential effects exists both at the plant and in the general office engineering staff. **Based on the studies performed and the acti**[illegible], **this** issue should be **considered resolved for McGuire.**

7.3 REVIEW OF EVENTS AT NUCLEAR POWER PLANTS

Duke Power reviews information on events at other operating nuclear power plants through a formal Operating Experience Program. LERs, information notices and bulletins, and other reports concerning industry events are

8.0 RESOLUTION OF OTHER SAFETY ISSUES

8.1 Screening Process

PRA methodology is considered a useful tool for providing a risk-based framework for making decisions on the significance of potential safety issues, the adequacy of existing plant systems, and the benefit associated with proposed recommendations. Therefore, Duke has utilized the IPE process to assess several safety issues in an attempt to resolve them for McGuire by evaluating their effect on the calculated core-melt frequency and overall plant risk and evaluating the benefit associated with proposed recommendations. By assessing the safety issue's contribution to overall plant risk, decisions can be made on the necessity for potential plant modifications. Evaluations of Shutdown Decay Heat Removal (formerly USI A-45) and Systems Interaction Due to Internal Flooding (USI A-17) can be found in Sections 6.0 and 7.0, respectively, of this report. This section evaluates the remaining issues in NUREG-0933.

Over 700 issues/items outlined in NUREG-0933 were screened to determine those safety issues deemed amenable for resolution using PRA methodology. Utilizing the status and descriptions provided in NUREG-0933, the majority of issues was eliminated from further assessment based on the following exclusion categories:

1. Non-safety issues, including licensing issues and environmental issues
2. Regulatory impact issues (these issues deal with the potential of reducing the regulatory burden on the licensee or improving the efficiency of the NRC's regulatory guidance)
3. Issues found to be covered by other major issues
4. Resolved TMI Action Plan Items with implementation of resolution mandated by NUREG-0737
5. Issues whose potential risk reductions are small or trivial as determined by the NRC
6. Issues whose resolution resulted in no new or additional regulatory requirements
7. Issues applicable to future and new construction plants only;

At McGuire, the ESW System is referred to as the Nuclear Service Water (RN) System. The RN System is explicitly modeled in the McGuire PRA as a major support system and as a potential initiator of plant transients. The McGuire PRA results, therefore, provide information on (i) the reliability of the RN System, (ii) the contribution of a loss of RN to the overall core-melt frequency and plant risk, and (iii) the impact of the proposed GI-130 actions on both the calculated RN reliability or core-melt results.

8.2.2 McGuire RN System

McGuire Nuclear Station Units 1 and 2 each have a two-pump train RN System which provides cooling for many plant loads. The RN System is an open-loop system which normally takes suction from Lake Norman and discharges to the RC System discharge piping which leads back to the lake. These RN Systems are normally isolated between units but do contain cross-connect piping and normally shut manual valves which can be opened to align RN from one unit to the other. Normal power operations require one pump train to be in operation providing cooling for all normal essential and non-essential loads for that unit. The other RN pump train is normally in standby and would be started automatically following a blackout, a safety injection (SS) signal, starting of a motor driven Auxiliary Feedwater (CA) pump, and by procedure following indication of insufficient RN flow. The major RN cooling loads during normal operation at McGuire include the Component Cooling Water (KC) pump motors and heat exchangers, the Chemical and Volume Control System (NV) pump motors, various ventilation units, and the Reactor Coolant (NC) pump motors. Additional accident loads include cooling to the emergency diesel generators, Containment Spray (NS) heat exchangers and pump air handling units, Residual Heat Removal (ND) pump air handling units, motor driven CA pump motors, and the Safety Injection (NI) pump motors.

McGuire has several design and operational features which reduce the consequences resulting from a loss of RN initiator. First, the RN System at McGuire can get backup cooling flow from the Containment Ventilation Cooling Water (RV) System. This system is aligned to the RN non-essential header but can deliver adequate flow to the RN essential loads. The RV System contains three 3200 gpm pumps which will start automatically on low RN failure probability of 2.4E-03/demand. The frequency of a loss of RN initiator was estimated to be 3E-03/yr based on the system fault tree analysis presented in Appendix A.6 of the McGuire PRA report.

The core-melt frequency contribution resulting from the loss of RN is calculated to be approximately 1E-05/yr. These sequences contribute approximately 25% to the calculated internal core-melt frequency at McGuire. The dominant cutset for this sequence has a frequency of 6E-06/yr and involves the following basic events:

Loss of RN Initiator	3E-03
Failure to Align RV Backup	1E-01
Failure to Cross-connect RN	1E-01
Failure to Activate SSF Seal Cooling	2E-01
Total	6E-06

Two major conservatisms in the analysis include taking no credit for recovering a failed pump and assuming a NC pump seal LOCA will occur within a few minutes following a complete loss of seal cooling. In the McGuire PRA, the 0.1 failure probability for successfully using the other units RN System is dominated by the human error associated with opening two manual RN cross-connect valves in a timely manner and not by the availability of the other units RN System itself. Therefore, it does not appear that imposing new or more restrictive technical specifications on the RN System would reduce the calculated core-melt frequency attributable to loss of RN sequences. The most effective way to reduce the core-melt frequency attributable to loss of RN sequences can be attained by improving the human reliability associated with (i) aligning RV backup to RN, (ii) aligning the RN cross-connect between the units, and (iii) aligning the backup seal cooling system (SSF). To accomplish this goal, McGuire is investigating cycling the manual cross-connect valves on a periodic basis, and reviewing the adequacy of the current procedures and training practices addressing loss of RN sequences.

that a motor driven pump should not be run for excessive periods of time without motor cooling. There are also alarms that would indicate pumps overheating.

Additional information concerning the modeling of human actions can be found in Section 5 of the McGuire PRA report.

8.2.5 Conclusions

McGuire currently has in place (i) Technical Specifications covering the RN (Nuclear Service Water) System on both units in Modes 1 through 4, (ii) an independent backup source of cooling water provided from the RV System, and (iii) detailed procedural guidance for cross-connecting the RN Systems between units. Additionally, McGuire has a backup means of Reactor Coolant Pump seal cooling that is independent of RN. McGuire is also investigating other ways to improve the confidence associated with cross-connecting the RN Systems between units such as periodic testing of manual cross-connect valves and reviewing the adequacy of the current procedures and training for loss of RN sequences. Imposing new or more restrictive technical specifications is not expected to reduce the calculated core-melt frequency associated with loss of RN sequences at McGuire. Therefore, it is concluded that this issue should be considered resolved for McGuire Nuclear Station.

8.3 GI-105, INTERFACING-SYSTEMS LOCA IN PWRs

8.3.1 Introduction

This issue is concerned with evaluating the vulnerability of current PWR designs to an Interfacing-Systems LOCA (ISLOCA). ISLOCAs are of particular interest since the lost reactor coolant is released outside of containment and therefore not returned to the containment sump for recirculation. In addition, a release path to the atmosphere is established, making this event a potentially important contributor to risk. As part of this issue, the NRC is examining cost-benefit relationships in order to assess whether utilities should make changes to current procedures, testing requirements, or system designs, in order to reduce the risk associated with ISLOCAs.

The total annual frequency of an ISLOCA associated with the ND cold leg injection lines is estimated to be 6.7E-09/yr. An ISLOCA through one of these four injection paths would require the failure of two check valves and the inability of an operator to close a MOV in a timely manner. The failure of one of the upstream check valves to close would be noticeable because the associated accumulator would discharge high pressure water through the relief valves upstream of NI173A or NI178B. Additionally, the operator would have up to one hour before the FWST would be depleted to isolate the associated MOV which is designed for primary system pressure. Indications of an ISLOCA through this path could be detected by observing high ND pump discharge pressure. Emergency Procedure EP/1/A/5000/08, LOCA Outside Containment, provides guidance for isolating this and other potential ISLOCA paths.

The total annual frequency of an ISLOCA associated with the auxiliary pressurizer spray path is estimated to be 4.2E-10/yr. An ISLOCA through this path would require the failure of two check valves and a normally closed air-operated valve, 1NV840. Should this ISLOCA occur, it would take over 5 hours to deplete the FWST volume. Therefore, the operators would have sufficient time to perform an aggressive cooldown or refill the FWST prior to core damage. Due to the low probability associated with this ISLOCA path, no credit for recovery was modeled.

8.3.3 Contribution to Internally-Initiated Core Damage and Plant Risk

The estimated core damage frequency due to ISLOCAs at McGuire is 8.1E-09/yr. The ISLOCA contribution to the overall internal core-melt frequency is negligible. Based on this core-melt frequency, the internal ISLOCA contribution to latent cancer fatalities and whole-body person-rem is insignificant (less than 5%). While internal ISLOCAs contribute approximately 20% to the risk of early fatalities, the overall probability of early fatalities is extremely low. These effects are shown in the tables and figures contained in Section 8 of the McGuire PRA.

9.0 UNIT DIFFERENCES, APPLICABILITY OF IPE RESULTS

Since the McGuire PRA is based on Unit 1, an analysis was performed to determine the applicability of the PRA results to Unit 2. Appendix B summarizes the assessment relevant plant systems to confirm their similarity and to identify any differences. The review confirmed that the units are similar and their configurations are nearly identical and that the core melt frequency and risk results calculated for Unit 1 are applicable to Unit 2 also.

10.0 CONFORMANCE WITH GENERIC LETTER

The McGuire PRA study and the IPE process have resulted in a comprehensive, systematic examination of the McGuire reactors in regard to potential severe accidents. This examination has identified the most likely severe accident sequences, both internally and externally induced, with quantitative perspectives on their likelihood and fission product release potential. The results of the study have prompted certain changes in equipment, plant configuration, and enhancements in plant procedures to reduce the vulnerability of the plant to some accident sequences of concern, as discussed in Section 3.0. The examination process and the accompanying dialogue with operations, technical, and management personnel have created a level of appreciation for the varied mechanisms, complexity, and severity of this type of accident. The final results, which take into account detailed in-plant and ex-plant consequence analysis, significant plant operating experience, and plant design features as of December 1990, portray the integrated safety profile of the McGuire plant. These results confirm that the McGuire Nuclear Station poses no undue risk to the public health and safety. Accordingly, the objectives of Generic Letter 88-20 are fully satisfied.

Duke intends to continue to utilize the McGuire PRA models, results, and associated know-how, where appropriate, in evaluating operational safety issues, in optimizing design options, and as an integral element of severe accident management.

The IPE process and results have been applied to examine USI A-45 and USI A-17. It has been concluded that these issues can be considered adequately resolved for McGuire. In addition, safety issues GI-105 and GI-130 have been evaluated in relation to the IPE results, plant experience and current system configuration. The evaluation supports resolution of these issues for McGuire.

In as much as the McGuire PRA constitutes a complete level 3 PRA with a systematic treatment of internal and external events, containment response, fission product release, and risk calculations, adequate information is

APPENDIX A

A.0 SHUTDOWN DECAY HEAT REMOVAL

A.1 INTRODUCTION

USI A-45, entitled "Shutdown Decay Heat Removal Requirements", has as its objective the determination of whether the decay heat removal function at operating plants is adequate and if cost beneficial improvements can be identified. It has been concluded by the NRC that a generic resolution to the issue is not cost effective and that resolution can only be achieved on a plant-specific basis. Therefore, the NRC has subsumed A-45 into Generic Letter 88-20 and has requested an evaluation of decay heat removal vulnerabilities during power operation and hot standby. To this end, this part of the IPE response provides an evaluation of shutdown decay heat removal for McGuire Nuclear Station.

Duke Power Company is aware of the NRC case studies and has participated in industry efforts to review and resolve this issue. The McGuire PRA models all plant systems involved in the decay heat removal (DHR) function, including those that provide "feed-and-bleed" cooling. Support systems such as AC and DC power, cooling water systems, etc., which influence proper functioning of DHR systems are also modeled. Initiating events which challenge the DHR systems and thereby contribute to core damage sequences of interest have been identified and quantified. Fault tree models of the systems include the relevant equipment failure modes and human elements Particular attention is given to identify common cause failure events such as flood, fire, etc., which could impose extraordinary stress on DHR systems.

A.2 OVERVIEW OF MCGUIRE DECAY HEAT REMOVAL SYSTEMS

The frontline DHR systems consist of the following:

- Main Feedwater (CF) System and Auxiliary Feedwater (CA) System for steam generator feedwater.
- Main steam safety valves, power-operated relief valves, turbine

- Steam Line Break Outside Containment
- Feedwater Line Break Outside Containment
- Loss of Nuclear Service Water
- Loss of Component Cooling Water
- Loss of Operating 4160 V Bus
- Loss of Instrument Air
- Inadvertant Safeguards Actuation
- Loss of Vital I&C Bus
- Small Break LOCA
- Steam Generator Tube Rupture

External Initiating Events

- Seismic Events
- Fire
- Tornado
- Flooding Events

A.3.2 Accident Sequences

Accident sequences were developed for external and internal events. Section 2.0 of the McGuire PRA develops and describes accident sequences initiated by internal events. Section 3.0 of the McGuire PRA develops and describes accident sequences initiated by external events. As described in these sections, cut sets represent the final form of accident sequences. Appendix D of the McGuire PRA lists cut sets grouped by internal events, external events, and functional sequences.

There are 36 functional sequence categories. These are shown in Tables D-8 through D-43 of the McGuire PRA. The scope of the A-45 analysis is 20 of the 36 categories. These are externally and internally initiated sequences of TBU, TBX, TBP, TBQsU, TBQsX, TBQsP, TBQrU, TBQrX, SU, SX, and SGTR. Qs indicates a reactor coolant pump seal LOCA which is not in the scope of A-45. However, TBQsU, TBQsX, and TBQsP sequences would result in TBU, TBX, and TBP sequences, respectively, if the seals did not fail. Therefore, externally and internally initiated sequences of TBQsU, TBQsX, and TBQsP are grouped with the TBU, TBX, and TBP sequences for the purposes of this

Historical data has been used to apply non-recovery values to direct or independent losses of CF, such as CF pump turbine trips. The loss of CF is followed by a loss of auxiliary feedwater (CA). A major contributor to CA failure following a loss of off-site power is the failure of both emergency diesel generators (fails both motor-driven pumps) with an independent failure or unavailability of the turbine-driven pump. Another major contributor to CA failure is a loss of nuclear service water (fails cooling to both motor-driven pumps) followed by an independent failure or unavailability of the turbine-driven pump. The loss of both CF and CA in this sequence is followed by a loss of injection capability. Both the loss of all ac power and the loss of nuclear service water will fail all injection pumps (motive power and cooling, respectively). The frequency of this sequence is 3.9E-06 per year.

TBX

Functional sequence TBX involves a total loss of secondary side heat removal followed by a failure of long-term feed-and-bleed cooling. The contribution from this functional sequence is less than 1.0E-06 per year.

TBP

Functional sequence TBP involves a total loss of secondary side heat removal followed by a failure of bleed capability for feed-and-bleed cooling. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQrU

Functional sequence TBQrU involves a failure of secondary side heat removal, a stuck-open pressurizer relief valve, and a failure of injection capability. The contribution from this functional sequence is less than 1.0E-06 per year.

TBQrX

external initiator group to each functional sequence is given and then summed up in the column labeled EXTERNAL TOTAL for a calculated annual core-melt frequency of 1.0E-05 due to loss of DHR. The total core-melt frequencies from each initiator group are given in the bottom row. The percent values represent the percent contribution to the total (internal plus external) calculated annual core-melt frequency of 2.6E-05. If the contribution from any particular category is less than 1.0E-06 per year, a '<' sign has been used, with the exception of steam generator tube rupture. Only those functional sequences with frequencies greater than 1.0E-06 per year are discussed here. A detailed discussion of external events analysis can be found in Section 3.0 of the McGuire PRA. A detailed listing of all cut sets contributing to these functional sequences can be found in Appendix D of the McGuire PRA.

TBU

Functional sequence TBU involves a total loss of secondary side heat removal along with a failure of injection capability.

Seismic - The leading contributor to this sequence is a seismically-induced loss of off-site power followed by random or seismically-induced diesel generator start failures which result in a loss of all ac power. Damage to the grid by the seismic event creates a prolonged loss of off-site power (assumed to be non-recoverable during the 24 hour mission time). A seismically-induced failure of either the CA turbine-driven pump or the condensate sources results in a loss of all heat removal and a boiloff of Reactor Coolant System inventory. The seismically-induced failure of the SSF, along with the failure of all ac power at the start of the sequence, prevents the assured sources for auxiliary feedwater from being aligned. The frequency of this sequence is 7.5E-06.

TBX

Functional sequence TBX involves a total loss of secondary side heat removal followed by a failure of long-term feed-and-bleed cooling. The contribution from this functional sequence is less than 1.0E-06 per year.

A.5 DISCUSSION OF RESULTS AND ASSESSMENT OF SYSTEMS

Section 8.1.3 of the McGuire PRA presents a number of insights. Insights which are in the scope of A-45 are given below.

A.5.5 Internal Events

Internal transient events have been calculated to contribute approximately 17 percent to the total annual core-melt frequency due to loss of DHR. The dominant sequences involve a loss of secondary side heat removal followed by a failure of injection capability. Transient events leading to core melt usually involve loss of support systems. This is due to the multiple trains of both main and auxiliary feedwater, the multiple sources, steam generators and injection pathways, and the very low probability of having independent hardware failures in all of these trains. The two dominant ways to achieve total loss of DHR are 1) loss of all ac power with an independent failure of the auxiliary feedwater turbine-driven pump, and 2) loss of all nuclear service water with an independent failure of the auxiliary feedwater turbine-driven pump. Due to the low frequencies associated with these sequences, it can be concluded that these sequences do not demonstrate any unique plant vulnerability.

Steam generator tube ruptures contribute less than 1 percent to the total annual core-melt frequency due to loss of DHR. Recovery actions, possibly due to the slow progression of the sequence, significantly reduce the calculated SGTR core-melt frequency. Operators taking necessary action to cool down to residual heat removal entry conditions upon loss of high pressure injection capability, in particular, has reduced the calculated core-melt frequency. Therefore, it can be concluded that these sequences do not demonstrate any unique plant vulnerability.

LOCA events have been calculated to contribute approximately 42 percent to the total annual core-melt frequency due to loss of DHR. These events are dominated by failure of ECCS recirculation at switchover due to failure of 2 or more FWST level transmitters or operator failure. The leakage from small break LOCAs will initiate containment sprays on high containment pressure.

o[illegible]-site power is followed by a <u>demand</u> failure of the diesels, the auxiliary feedwater suction from the nuclear service water system will also be unavailable. This is because the auxiliary feedwater suction header from this source contains normally closed, motor-operated valves. Therefore, at ground accelerations of 0.4g and above, loss of all secondary side heat removal begins to contribute to the core-melt frequency.

Significant uncertainty exists concerning seismic hazard curves. The mean hazard curve generated by EPRI, specifically for the McGuire site, is used on this analysis. Use of Lawrence Livermore National Lab hazard curves would not be expected to alter the insights of the analysis.

Significant uncertainty exists concerning the seismic capacity (fragility) of many components. Seismic failures in this analysis are assumed to be catastrophic in nature and non-recoverable. In reality, it may be possible to recover some equipment in a timely manner if the actual failure were not catastrophic.

Given the extent of damage assumed to occur and the large uncertainty associated with these sequences, it is safe to say that PRA modeling assumptions are driving these results. Therefore, it can be concluded that these sequences do not demonstrate any unique plant vulnerability.

A.6 CONCLUSIONS

The McGuire decay heat removal systems are robust and do not demonstrate any particular vulnerability to either internally or externally initiated severe accident sequences. The diversity in feedwater pump trains, suction sources, and injection pathways make secondary side heat removal particularly reliable. Current efforts to investigate the reliability of entering high pressure recirculation following a LOCA will ensure the systems involved can effectively respond to multiple or common cause failures. Seismic events can cause widespread damage but the calculated failure modes, and frequencies do not indicate any unique plant vulnerability. In addition, failure of all decay heat removal capability during a seismic event requires ground accelerations significantly greater

TABLE A.4-1
SUMMARY OF A-45 RESULTS

TOTAL CORE MELT FREQUENCY =2.6E-05

(<) MEANS LESS THAN 1.0E-06

	INTERNAL TRANSIENTS	SCTRs	LOCAs	INTERNAL FLOODING	INTERNAL TOTAL	SEISMIC	EXTERNAL FLOODING	TORNADO	FIRE	EXTERNAL TOTAL
TBU	3.9E-06 15%	-	-	-	3.9E-06 15%	7.8E-06 30%	-	<	<	8.3E-06 32%
TBX	<	-	-	-	<	<	-	-	-	<
TBP	<	-	-	<	<	<	-	-	-	<
TBQrU	<	-	-	-	<	<	-	<	-	<
TBQrX	-	-	-	-	-	-	-	-	-	-
SU	-	<	<	-	<	-	-	-	-	-
SX	-	<	1.1E-05 42%	-	1.1E-05 42%	-	-	-	-	-
TOTAL	4.5E-06 17%	8.[illegible]9 [illegible]%	1.1E-05 42%	<	1.6E-05 62%	8.0E-06 31%	-	<	<	1.0E-05 38%

APPENDIX B

McGUIRE PRA UNIT DIFFERENCES

Introduction

The McGuire PRA analysis was done for Unit 1. Systems at Unit 2 have been investigated for similarity with Unit 1 to insure that the same PRA results generally apply to both units. Because of the similarities in the containments at Unit 1 and Unit 2 and because the initiating event analysis is similar, this investigation focused on the individual plant systems which are modeled in the plant fault tree.

Methodology

The unit comparison focused on differences in the system design or in component fault exposure times which would cause differences in the system fault trees. Flow diagrams were examined and compared for all mechanical systems. Where possible, system descriptions were consulted to insure that corresponding systems had similar designs, functions, and failure modes at both units. This comparison was used as a basis to determine if a further investigation was warranted. For the Containment Air Return (VX) System and the Component Cooling (KC) System, all of the modeled failure modes for electrical components were verified by a review of the electrical elementary drawings. For the Diesel Generator Load Sequencer System, similarity between units was insured by a review of the Design Basis Documents and by interviewing cognizant engineers.

For each system, fault exposure times are considered the same because of similar procedures and Technical Specifications for both units.

Other PRA aspects such as interfacing systems, LOCAs and external events were also reviewed for similarity between the units.

The following paragraphs summarize the results of the review process for unit similarity.

Safety Injection (NI) System

The mechanical system, as presented in the system description documents, is the same for both units. The electrical system descriptions were checked and were found to be essentially the same. Similar procedures exist for the systems at both units, and the same Tech. Specs. apply. There were no noted differences in the NI flow diagrams for the Units. There were no noted differences in the RN supply to NI components among the units. It is reasonable to conclude that the Unit 1 fault tree results for the NI system also apply to Unit 2.

Heating Ventilation and Air Conditioning (VC/YC) Systems

The parts of the HVAC system modeled in the PRA are shared between Units. The same fault tree for HVAC and the same results apply for Unit 1 and Unit 2.

Auxiliary Feedwater (CA) System

The mechanical system, as presented in the system description documents, is the same for both units. The electrical system descriptions are essentially the same. Similar procedures exist for the systems at both units, and the same Technical Specifications apply. No noteworthy differences have been found in the flow diagrams. Therefore, it is reasonable to conclude that the Unit 1 fault tree results for the CA system also apply to Unit 2.

Reactor Coolant (NC) System

The mechanical system, as presented in the system description documents, is the same for both units. The electrical system descriptions for Unit 1 and 2 are essentially the same. Similar procedures exist for the systems at both units, and the same Technical Specifications apply. No differences have been found in the flow diagrams. It is reasonable to conclude that the Unit 1 fault tree results for the NC system also apply to Unit 2.

of the essential headers which are unique to the units. The fact that the system is shared between units does not change the top event success criteria. It is reasonable to conclude that the Unit 1 fault tree results for the VI system apply to Unit 2 also.

Engineered Safety Features Actuation (ESFAS) System

The ESFAS model in the PRA is based on a document which applies to both units. The ESFAS system model for Unit 2 would be similar to the one for Unit 1. Any differences in the Unit ESFAS fault tree models would be in the individual actuation relays. Since ESFAS events which appear in the cut sets for the McGuire Unit 1 plant model are easily recovered and since this would be the same situation for Unit 2, ESFAS has been eliminated from a more detailed consideration of unit differences. It is reasonable to conclude that any differences in the unit ESFAS systems would have a negligible effect on the Unit 2 PRA results.

Diesel Generator and Load Sequencer System

System descriptions for systems related to the Diesel Generators are similar for both units. Similar procedures exist for these systems at both units, and the same Technical Specifications apply. Based on the similarity of the Design Basis Documentation for the unit load sequencers and on input from electrical system engineers, it is reasonable to conclude that load sequencer logic is the same for both units. It is reasonable to conclude that the Unit 1 fault tree results for the Diesel Generator and Load Sequencer System apply to Unit 2.

Vital I&C Power Supply System

System descriptions for systems included in the analysis of Vital I&C power are shared between the units or are similar. Procedures associated with these systems are the same for both units. The same Technical Specifications apply to both Units. No differences have been noted in the wiring diagrams for these systems at Unit 1 and 2. It is reasonable to

Essential A.C. Power

From an examination of the system descriptions and drawings for essential AC power, two differences were noted. One difference is that the main step up transformer for Unit 2 is 525 KV and the main step up transformer for Unit 1 is 230 KV. Since the main step-up transformers are not modeled, this difference would not lead to a difference in fault trees for the units. The other difference is that there is one less failure mode for some Unit 2 components (due to the arrangement of the 600V bus EMXH). Similar procedures exist for both units, and the same Technical Specifications apply. It is reasonable to conclude that the Unit 1 fault tree results for the essential A C. power system also apply to Unit 2.

Standby Shutdown (SS) System

The mechanical and electrical system descriptions for the SS systems are shared. The electrical controls system descriptions have been reviewed, and no noteworthy differences between Unit 1 and 2 SS electrical controls have been found. Similar procedures exist for both units and the same Tech. Specs. apply. No differences have been noted in the valve arrangements or flow paths for the unit specific parts of the SS systems. It is reasonable to conclude that the Unit 1 fault tree results for the SS system also apply to Unit 2.

Civil Structures

The general arrangement drawings have been reviewed to determine if there are any differences in McGuire Unit 2 structures which would add vulnerabilities for external events. Since the Control Room is shared between Units, the core damage sequence containing the initiating event "fire in the Control Room" applies to both units. Other plant areas in both units are also similar with respect to fire vulnerabilities. The flood analysis for Unit 1 already includes a consideration of the flood scenario for Unit 2. A core damage sequence for Unit 2 is added to the cut sets to account for this vulnerability. No structural differences have been found which would alter the results of the seismic analysis or containment

August 30, 1991

To: Distribution

Subject: McGuire Nuclear Station
Probabilistic Risk Assessment
File: MC-1535.00

Enclosed for your information is the 3-volume McGuire PRA report. This report evaluates potential plant conditions from the standpoint of severe accident vulnerability and risk, analyzes the reliabilities of plant systems designed to cope with plant accidents, and presents the calculated core damage probabilities and risk results for McGuire Unit 1.

This PRA report constitutes part of the NRC submittal for the McGuire IPE, in response to the NRC Generic Letter 88-20.

If you have any questions, please call (704) 373-4520.

P.M. Abraham

P. M. Abraham, Engineering Supervisor
Nuclear Engineering Section
Engineering Support Division

PMA/tms
Enclosure

Distribution:
R. O. Sharpe (Copy 1 & 2)
K. D. Thomas (Copy 3)
W. H. Taylor (Copy 4)
R. G. Hall (Copy 5)
R. L. Gill (Copy 6)
R. L. White (Copy 7)
C. M. Young (Copy 8)
D. M. McGinnis (Copy 9)
P. M. Abraham (Copy [illegible]
H. D. Brewer (Copy [illegible]
B. E. Busby (Copy [illegible]
G. L. Cruzan (Copy [illegible]
J. T. Holland (Copy 14)
J. A. Nash (Copy 15)
L. J. Kachnik (Copy 16)
L. J. Azzarello (Copy 17)
M. J. Barrett (Copy 18)
S. A. Deskevich (Copy 19)
P. D. Paul (Copy 20)
B. C. Carroll (Copy 21)
K. S. Canady (Copy 22)
G. B. Swindlehurst (Copy 23)
H. D. Brandes (24)
Central Records (Copy 25)
NRC Submittal (Copy 26-29)
D. E. Parnham (Copy 30)
J. R. Newton (Copy 31)
D. B. Noxon (Copy 32)