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November 27, 2000

Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
11565 Rockville Pike
Rockville, MD Washington, D.C. 20555

SUBJECT: Harris License Amendment Proceeding, Docket No. 50-400LA

Dear Licensing Board members,

On November 20, 2000, Orange County served the Board and parties with copies of some of the exhibits to its Subpart K summary, consisting of documents that the other parties had not received in discovery. Enclosed please find the remainder of the exhibits. The exhibits are provided in alphabetical order. They should be collated in alphabetical order with the exhibits that were provided on November 20, 2000.

Please note that some corrections have been made to the designation of exhibits in Appendix A to the Thompson Report. An errata sheet and a corrected copy of Appendix A are enclosed.

Finally, Orange County is enclosing an errata sheet for the Summary, as well as a corrected copy of the Summary.

Sincerely,

Diane Curran

Enclosures: as stated
cc: service list
Dr. Gordon Thompson

Template = SECY-037

SECY-02

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400 -OLA
(Shearon Harris Nuclear	)	ASLBP No. 99-762-02-LA
Power Plant)	)	
	)	

CERTIFICATE OF SERVICE

I certify that on November 27, 2000, corrected copies of Orange County's Detailed Summary of Facts, Data, and Arguments, Etc.; Orange County's Errata to Detailed Summary, Etc. Regarding Contention EC-6; and Orange County's Errata to Appendix A of Thompson Report; and corrected copy of Appendix A were served on the service list below by e-mail. In addition, hard copies were served by hand (as indicated below by an asterisk), by Federal Express (as indicated below by two asterisks) or by first class mail on:

**Secretary of the Commission
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Diane Curran

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CAROLINA POWER & LIGHT (Shearon Harris Nuclear Power Plant)	))))))	Docket No. 50-400 -OLA ASLBP No. 99-762-02-LA

CERTIFICATE OF SERVICE

I certify that on November 20 and November 27, 2000, the exhibits to Orange County's Detailed Summary of Facts, Data, and Arguments, Etc. (November 20, 2000), were served on the service list below by hand (as indicated below by one asterisk), by overnight mail (as indicated below by two asterisks), or by first-class mail:

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Diane Curran

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Novermber 27, 2000

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OFFICE OF ... RULE... ADJUD... STAFF

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400 -LA
(Shearon Harris Nuclear	)	ASLBP No. 99-762-02-LA
Power Plant)	)	
	)	

ORANGE COUNTY'S ERRATA TO DETAILED SUMMARY, ETC. REGARDING CONTENTION EC-6

Orange County hereby submits the following errata to its Detailed Summary of Facts, Data and Arguments on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact With The Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant (November 22, 2000) ("Summary"). A corrected copy of the Summary is enclosed. The errata include a Table of Contents and Table of Authorities, which have been included in the corrected Summary.

Page	Line	Change
inside cover		insert Table of Contents and Table of Authorities
1	1	change "2.113" to "2.1113"
1	4	change "January 7, 2001" to "December 7, 2000"
2	4	change "2.111(b) to "2.1111(b)"
2	7	change "Exhibit 2" to "Exhibit 1"
5	1	delete "See"

Page	Line	Change
7	17	change "Stage" to "Storage"
9	fn	change "July 20" to "July 29"
18	fn	in fourth line, insert ")" after "August 1989"
22	fn	in first line, change "40 C.F.R. § 1508.2" to "40 C.F.R. § 1502.1"
22	fn	in sixth line, change "462 U.S. 1983)" to "462 U.S. 87 (1983)
23	11	change "*Id.*" to "42 U.S.C. § 4332(C)"
23	fn	in fourth line, change "regulations re" to "regulations are"
24	fn 17	delete last sentence of indented paragraph
32	9	change "and/or" to "in"
35	14	insert "*Id.*" before "Having assumed"
36	fn	change "person" to "worker"
39	17-18	delete last two sentences in second full paragraph of Section VI

Respectfully submitted,

Diane Curran
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November 27, 2000

DOCKETED
USNRC

FILED NOVEMBER 20, 2000;
CORRECTED NOVEMBER 27, 2000

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OF...
ADJU...

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400 -LA
(Shearon Harris Nuclear	)	ASLBP No. 99-762-02-LA
Power Plant)	)	
	)	

DETAILED SUMMARY OF FACTS, DATA AND ARGUMENTS AND SWORN SUBMISSION ON WHICH ORANGE COUNTY INTENDS TO RELY AT ORAL ARGUMENT TO DEMONSTRATE THE EXISTENCE OF A GENUINE AND SUBSTANTIAL DISPUTE OF FACT WITH THE LICENSEE REGARDING THE PROPOSED EXPANSION OF SPENT FUEL STORAGE CAPACITY AT THE HARRIS NUCLEAR POWER PLANT

WITH RESPECT TO THE NEED TO PREPARE AN ENVIRONMENTAL IMPACT STATEMENT TO ADDRESS THE INCREASED RISK OF A SPENT FUEL POOL ACCIDENT (CONTENTION EC-6)

Submitted on behalf of Orange County by:

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Expert Witness for Orange County

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400 -LA
(Shearon Harris Nuclear	)	ASLBP No. 99-762-02-LA
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	)	

DETAILED SUMMARY OF FACTS, DATA AND ARGUMENTS AND SWORN SUBMISSION ON WHICH ORANGE COUNTY INTENDS TO RELY AT ORAL ARGUMENT TO DEMONSTRATE THE EXISTENCE OF A GENUINE AND SUBSTANTIAL DISPUTE OF FACT WITH THE LICENSEE REGARDING THE PROPOSED EXPANSION OF SPENT FUEL STORAGE CAPACITY AT THE HARRIS NUCLEAR POWER PLANT

WITH RESPECT TO THE NEED TO PREPARE AN ENVIRONMENTAL IMPACT STATEMENT TO ADDRESS THE INCREASED RISK OF A SPENT FUEL POOL ACCIDENT (CONTENTION EC-6)

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.1113, Orange County hereby submits a detailed written summary and sworn submission (hereinafter "Summary") of all the facts, data, and arguments which are known to the County and on which the County proposes to rely at the December 7, 2000, oral argument regarding Contention EC-6. This Summary presents Orange County's legal and factual grounds for demonstrating the existence of a genuine and material factual dispute regarding the issues raised in Contention EC-6. As demonstrated below, the NRC Staff should be prohibited from issuing an operating license amendment to Carolina Power & Light ("CP&L") for the purpose of expanding spent fuel storage capacity at the Harris nuclear power plant, unless and until it has prepared a full-scale Environmental Impact Statement ("EIS") that addresses the environmental impacts of the proposal and weighs reasonable alternatives.

As required by 10 C.F.R. § 2.1111(b), the factual assertions in this Summary are submitted under the sworn declaration of Dr. Gordon Thompson, the County's expert witness regarding to Contention EC-6. *See* Declaration of Dr. Thompson, attached as Exhibit 1. Dr Thompson's professional qualifications and experience are described in his Declaration and his Curriculum Vitae, which is an attachment to his Declaration. In addition, the technical analysis supporting Orange County's summary is contained in Dr. Thompson's report entitled *The Potential for a Large Atmospheric Release of Radioactive Material From Spent Fuel Pools at the Haris Nuclear Power Plant: the Case of a Pool Release Initiated by a Severe Reactor Accident* (Novenber 20, 2000) ("Thompson Report"), a copy of which is attached as Exhibit 2.

As detailed below, this summary demonstrates the existence of substantial and material evidence that the probability of an exothermic reaction in the spent fuel pools, leading to a massive release of radiation from the pools, is foreseeable, and may not be disregarded as a remote and speculative event.

II. STATEMENT OF THE CASE

This case raises the question of whether a severe pool accident in pools C and D of the Harris reactor is a foreseeable and plausible event, such that an Environmental Impact Statement ("EIS") must be prepared to fully evaluate the adverse impacts and weigh the costs and benefits of reasonable alternatives. The NRC Staff has prepared an

Environmental Assessment ("EA"), which claims that no EIS is necessary because the likelihood of such an accident is remote and speculative.

As demonstrated in this Summary and the attached report by Dr. Gordon Thompson, the EA is completely inadequate to justify the Staff's refusal to prepare an EIS, because it fails to take into account new information demonstrating that a spent fuel pool accident at Harris is not a remote and speculative event. Using data provided by CP&L and the NRC Staff, Dr. Thompson has provided a best estimate of the overall probability of a spent fuel pool accident which shows that such an accident is foreseeable and should be evaluated in an EIS.

The NRC Staff and CP&L will also be presenting estimates regarding the likelihood of a spent fuel pool accident at Harris. In evaluating this information, the Board must take into account the high level of uncertainty involved in the use of PRA, as well as the fact that any PRA on the seven-part scenario set forth in LBP-00-19 would take the art of risk assessment into uncharted territory that is therefore all the more uncertain. Moreover, the amount of time provided in this proceeding for such an analysis is far too short to permit the kind of "state-of-the-art" analysis contemplated by the Commission for the use of PRA in regulatory decisions. Under the circumstances, it is appropriate to require the preparation of a full-scale EIS.

The Board should also closely examine the qualitative assumptions made by the NRC Staff in support any assertion that the likelihood of a severe accident is too low to warrant the preparation of an EIS. In particular, the Board should not approve an EA that assumes that Harris workers will incur doses above regulatory limits in order to stop a severe accident from progressing, or that regulatory requirements for the safe operation of the Harris plant would be violated. To do so would unlawfully permit the trade-off of one kind of environmental harm to justify another, without taking the "hard look" required by NEPA.

Finally, the procedural posture of this case warrants the conduct of an adjudicatory hearing in order to allow a meaningful ventilation of the complex factual issues at stake here. Because of the extremely short timeframe for discovery in this expedited proceeding, none of the parties had completed their analyses before the conclusion of discovery. Therefore, Orange County has not had an opportunity to question NRC or CP&L experts or other witnesses about the results of their analyses or how they were arrived at. A summary proceeding such as this one cannot be fairly or lawfully used to cut off an intervenor's ability to probe the basis for the opposing parties' views.

III. FACTUAL AND PROCEDURAL BACKGROUND

A. Requirements of NEPA for Environmental Studies

NEPA requires federal agencies to prepare an EIS before undertaking any major federal action which may significantly affect the quality of the human environment. 42 U.S.C. § 4332(C). The NRC's implementing regulations at 10 C.F.R. § 51.20(a) also require the NRC to prepare an EIS for any licensing or regulatory action which "is a major federal action significantly affecting the quality of the human environment." Where aspects of the proposed action are addressed by a previously prepared EIS, a new EIS must be issued if there remains "major federal action" to occur, and if there is new information showing that the remaining action will affect the quality of the human environment "in a significant manner or to a significant extent not already considered."

Marsh v. Oregon Natural Resources Council, 490 U.S. 360, 374 (1989); *San Luis Obispo Mothers for Peace v. NRC*, 751 F.2d 1287, 1298 (D.C. Cir. 1984).[1]

B. The Harris License Amendment Proceeding

1. Nature of Proposed License Amendment

There are four spent fuel storage pools at the Harris nuclear power plant. Only two of the pools, designated "A" and "B," are currently in operation. At present, pool A contains 6 PWR racks with a total of 360 spaces, and 3 BWR racks with a total of 363 spaces. Pool B contains 12 PWR racks with a total of 768 spaces and 17 BWR racks with a total of 2,057 spaces. Under the present license, one additional BWR rack with a total of 121 spaces could be placed in pool B.

CP&L now seeks a license amendment to activate pools "C" and "D."[2] The purpose of the license amendment is to allow CP&L to use the Harris facility to store spent fuel generated at CP&L's one-unit Harris PWR station, its two-unit Brunswick BWR station, and its one-unit Robinson PWR station. If granted, the license amendment would allow the placement in pool C of up to 11 PWR racks with a total of 927 spaces and 19 BWR racks with a total of 2,763 spaces; and the placement in pool D of 12 PWR

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[1] *See also* 10 C.F.R. § 51.92(a), which requires supplementation where the proposed action has not been completed, if: "(1) there are substantial changes in the proposed action that are relevant to environmental concerns; or (2) There are significant new circumstances or information relevant to environmental concerns and bearing on the proposed action or its impacts." Although § 51.92 technically does not apply here, where the action proposed in the original Shearon Harris EIS has already been taken, the criteria provide applicable guidance for these circumstances.

[2] CP&L's proposed changes to its Technical Specifications are described in Enclosure 5 to the License Amendment Application.

racks with a total of 1,025 spaces. CP&L envisions this placement occurring in three campaigns in pool C, followed by two campaigns in pool D.[3]

If approved, the proposed license amendment would bring the total inventory of spent fuel assemblies that could be stored at Harris to 8,384, over a thousand more spent fuel assemblies than assumed in the 1983 Final Environmental Statement ("FES") that was prepared in connection with the Harris operating license application.[4]

The proposed license would make significant changes to the quantity of fuel now stored at Harris, as well as the method for storing the fuel. Both changes have significance with respect to the environmental impacts of the proposed license amendment. Pools C and D would have a capacity of 4,715 fuel assemblies as compared with the capacity of 3,669 fuel assemblies in pools A and B. This would result in a significant increase in the quantity of long-lived radioactive isotopes (*e.g.*, cesium-137) that could be stored at the Harris plant. An accident at pools C and D could release to the atmosphere a substantial fraction of the inventory of cesium-137 and other radioactive isotopes in these pools. *See* Thompson, Risks and Alternative Options Associated with Spent Fuel Storage at the Shearon Harris Nuclear Power Plant, Appendices D and E

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[3] Pool D will not be filled until a later "campaign," by which time CP&L will also need to have obtained a license amendment permitting it to exceed the license's current 1.0 million BTU/hour limit on the heat load in pools C and D. At that point, however, no further licensing action will be needed on the number of spent fuel assemblies permitted to be stored in pool D. The number of spent fuel assemblies permitted to be stored at the Harris site will have been previously approved in this license amendment proceeding.

[4] CP&L License Amendment Application, Enclosure 1 at 3 (December 23, 1998); NUREG-0972, Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, Docket Nos. STN 50-400 and 50-401, Carolina Power and Light Company (October 1983). It is important to note in this regard that although the FEIS assumed the storage of spent fuel at the Harris site, it did not address the environmental impacts of spent fuel storage.

(January 31, 2000) ("Thompson 1999 Report").[5] Such a release would yield consequences that would be significant in their own right, and would also be significant in comparison to the consequences of accidents at pools A and B and/or the Harris reactor.

In addition, the center-to-center distance for PWR fuel in pools C and D would be 9.0 inches instead of the 10.5 inches in pools A and B. Other factors being equal, this reduced distance would increase the propensity of pools C and D, as compared with pools A and B, to experience an exothermic reaction of fuel cladding in the event of partial or total loss of water. Given a loss of water, the conditional probability of an exothermic reaction in pools C and D would be comparable to or greater than the conditional probability of a similar reaction in pools A and B, and would be substantial over a range of pool loading patterns.[6]

2. Environmental Assessment

On December 15, 1999, the NRC Staff issued an Environmental Assessment ("EA") and Finding of No Significant Impact ("FONSI") for the CP&L license amendment application. Environmental Assessment and Finding of No Significant Impact Related to Expanding the Spent Fuel Pool Storage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432) at 10. In the EA, the NRC Staff concluded that the proposed expansion of spent fuel storage capacity at the Shearon Harris nuclear power plant will not have a significant effect on the quality of the human environment:

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[5] A copy of Dr. Thompson's 1999 Report was attached as Exhibit 2 to Orange County's Request for Admission of Late-Filed Environmental Contentions (January 31, 2000).
[6] The "conditional" probability of an accident is the probability of the accident if the occurrence of an event that could cause the accident (in this case, a loss of water) is

> The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure.

Id. at 6.

3. Orange County's intervention

On January 31, 2000, Orange County submitted a set of environmental contentions which challenged the FONSI issued in the EA. Contention EC-6 (formerly designated as Contention 1) charged that the NRC should be required to prepare an EIS for the proposed license amendment, because the proposed expansion of spent fuel pool storage capacity at Harris would significantly increase the risk of an accident at Harris. The contention identified two respects in which the risk of an accident was significantly increased: (a) CP&L would make significant changes in the physical characteristics and mode of operation of the plant that are not addressed in the EA, and (b) new information shows that spent fuel pool accident risks are higher than previously believed.

The Licensing Board admitted Contention EC-6, ruling that Orange County had posited a potential accident scenario that provided "an adequate basis to allow merits litigation on whether the sequence is not 'remote and speculative' so that a further environmental analysis of the CP&L pool expansion amendment request is required." LBP-00-19, slip op. at 13; *see also* slip op. at 16. The Board also posed three questions to the parties regarding their best estimates for the accident scenario, the effects of recent developments in probability estimation on the probabilities of the events in the scenario, and the necessary scope of the EIS should one be required. *Id.*, slip op. at 17.

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assumed.

Noting that the Applicant had previously invoked the procedures of Subpart K to 10 C.F.R. Part 2, the Board applied the procedures of Subpart K to establish an expedited schedule that included 60 days for discovery, and required the submission of legal and evidentiary summaries under 10 C.F.R. § 2.1113 within 90 days after the admission of the contention. An oral argument was set for December 7, 2000.

During the discovery period, the parties exchanged written interrogatories and document requests. Each party also took depositions of the other parties' witnesses.[7] In addition, Orange County's attorney and expert witness toured the Harris plant. The discovery process provided an opportunity for Orange County to obtain relevant documents, become familiar with the details of the Harris design and operation, and procure background information on the work that the Staff and CP&L were doing in preparation for filing their evidentiary summaries on November 20. However, none of the parties was able to complete its technical analysis by the close of discovery, and thus Orange County was unable to question either the Staff or CP&L about the results of their analyses or how those results were arrived at.

C. Use of Probabilistic Risk Assessment

1. Nature and history of PRA

The phrase "PRA techniques" refers to a wide variety of analytic models and procedures which draw upon data from experiments and from practical experience with nuclear facilities. PRA is used to quantify nuclear power plant hazards, using complex

[7] Under a July 29, 1999, Order, the parties were limited to deposing only three of the individuals identified by each of the other parties as potential affiants. Orange County deposed three out of the eight potential affiants identified by CP&L, and three out of four Staff affiants.

mathematical and phenomenological models. The methodology has been in development for almost three decades. Thompson Report at 13.

The "state of the art" in PRA is represented by NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Reactors (1990). NUREG-1150 is a Level 1 PRA which evaluates core melt and containment release frequencies for five reactors. It took ten years to produce, and cost many millions of dollars. NUREG-1150 is exemplary of the state-of-the-art in PRA, because of the depth and detail to which it examines the phenomenology of core melt accidents, because it contains uncertainty analysis, and because it was peer reviewed by a broad array of scientists. PRA techniques provide the best available methodology for estimating the overall probability of the seven-part event sequence that has been identified by the ASLB. Work on PRA development has continued since that study was completed, but subsequent PRAs have been less ambitious in their scope. *See* Thompson Report at 13.

In LBP-00-19, the Licensing Board noted the NRC's increasing reliance on PRA for regulatory decisions over the past ten years, and that the "entire trend in licensing, enforcement, inspection and the granting of amendments has swung gradually toward decision-making by probabilistic risk assessment." *Id.*, slip op. at 15. The Commission has also published a policy statement encouraging the increased use of PRA in regulatory activities. Policy Statement, Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, 60 Fed. Reg. 42,622 (August 16, 1995). Nevertheless, the Commission's policy limits the use of PRA to the "extent supported by the state-of-the-art." *Id.* In fundamental respects, the state of the art of PRA has not changed since the publication of NUREG-1150 ten years ago.

The limitations on the state of the art of PRA are discussed and summarized in Dr. Thompson's report at Section 2.3. As Dr. Thompson points out, PRAs for nuclear reactors continue to be characterized by substantial uncertainties. Moreover, PRA does not attempt to model some effects, such as acts of malice, sabotage, and degraded standards of operation. This is not because these factors make no contribution to the hazards of nuclear facility operation, but because the NRC does not know how to model them. In addition, PRAs rely heavily on expert opinion for numerous assumptions that cannot be verified. Finally, while a substantial body of knowledge has been accumulated with respect to Level 1 PRA, the results become increasingly less reliable with additional levels of analysis. For subject areas like spent fuel pool accident probability, the NRC has not accumulated anything near the level of study and understanding that it has for the phenomenology of core melt accidents. *See* Thompson Report at 13-16. These limitations on the state-of-the-art of PRA impose substantial restrictions on the degree to which the quantitative results of PRAs can be relied on for regulatory decisions, especially decisions that relax or waive safety and environmental requirements.

D. NEPA Analyses Relevant to Harris Spent Fuel Pool Expansion

1. Generic NEPA studies

Since the early 1980's, the EIS's for the licensing of all U.S. nuclear plants have considered the potential for severe accidents, without including a discussion of the potential for severe spent fuel pool accidents. This omission has been based on the findings of the Reactor Safety Study (WASH-1400).

In 1979, the NRC prepared a generic EIS on the environmental impacts of spent fuel storage, which includes a discussion of spent fuel pool accidents. *See* NUREG-0575, Handling and Storage of Spent Light Water Power Reactor Fuel (1979). In Sections 4.2.2 and 4.2.3, the GEIS addressed potential accidents, and concluded that: "The underwater storage of aged spent fuels is an operation involving an extremely low risk of a catastrophic release of radioactivity." *Id.* at 4-13. The GEIS, however, contained an extremely cursory analysis. Moreover, it contained no discussion at all of the potential for exothermic reactions under partial drain-down conditions. Since the publication of NUREG-0575 over twenty years ago, the NRC has prepared no other generic EIS which specifically examines the risks of spent fuel pool storage.

In a 1989 report, the NRC Staff summarized the Reactor Safety Study's consideration of spent fuel pool accidents, and the need for further analysis, as follows:

> "The risk of beyond design basis accidents in spent fuel storage pools was examined in WASH-1400. It was concluded that these risks were orders of magnitude below those involving the reactor core because of the simplicity of the spent fuel storage pool design: (1) the coolant is at atmospheric pressure, (2) the spent fuel is always subcritical and the heat source is low, (3) there is no piping which can drain the pool and (4) there are no anticipated operational transients that could interrupt cooling or cause criticality.
>
> The reasons for the re-examination of spent fuel storage pool accidents are twofold. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high density storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have provided evidence of the possibility of fire propagation between assemblies in an air cooled environment. Together, these two reasons provide the basis for an accident scenario which was not previously considered."[8]

Despite this recognition that pool accidents represent a new, credible accident scenario,

[8] E.D.Throm, NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools" at ES-1 (April 1989).

the NRC Staff has never undertaken any further NEPA analysis of the risks of spent fuel pool storage, nor have any of its other non-NEPA studies contained the level of analysis that has been given to reactor accidents through WASH-1400, NUREG-1150, EIS's, and IPE's.

2. FEIS for Harris operating license

In 1983, the NRC Staff prepared an EIS in connection with the proposed issuance of an operating license for the Harris nuclear power plant, Units 1 and 2.[9] The EIS examined reactor accidents only, and did not evaluate spent fuel pool accidents.

E. CP&L studies on spent fuel pool accidents

CP&L's evaluation of reactor accidents appears in CP&L's Individual Plant Examination (IPE) submittal of August 1993, and its Individual Plant Examination for External Events submittal of June 1995. Like the EIS, CP&L's IPE's did not evaluate spent fuel pool accidents. Since the publication of the IPE and IPEEE, CP&L has continued to update its risk analyses for Harris in a Probabilistic Safety Analysis ("PSA"). The PSA provides an estimate of the annual probability of core degradation for so-called "internal" initiating events, including floods, and for selected "external" initiating events, namely earthquakes and in-plant fires. In addition, the PSA estimates the annual probability and other characteristics of releases of radioactive material to the atmosphere, pursuant to core degradation.

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[9] NUREG-0972, Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, Docket Nos. STN 50-400 and 50-401,

ARGUMENT

IV. APPLICABLE STANDARDS AND WITNESS QUALIFICATIONS

A. Procedural Standards Under NRC Rules of Practice

The standard of review for this Subpart K proceeding is described in LBP-00-12, the Licensing Board's merits decision in the Subpart K proceeding for the technical phase of this proceeding. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-00-12, 51 NRC 247 (2000). Pursuant to 10 C.F.R. §§ 2.1113 and 2.1115, this proceeding provides the parties with:

> an opportunity to present facts data and arguments, by way of written summaries and sworn testimony, and an oral argument. Based on the summaries and the argument, the Commission then is to designate 'any disputed questions of fact, together with any remaining questions of law, for resolution in an adjudicatory hearing' if the Commission finds that 'there is a genuine and substantial dispute of fact which can only be resolved with sufficient accuracy by the introduction of evidence and an adjudicatory hearing,' and 'the decision of the Commission is likely to depend in whole or in part on the resolution of such dispute.'

Id., 51 NRC at 254.

The burden of demonstrating the existence of material factual disputes that must be aired in an evidentiary hearing falls on Orange County as the petitioner in this case. *See* LBP-00-12, 51 NRC at 255; Memorandum and Order (Subpart K Oral Argument Procedures) at 2 (January 13, 2000). Thus, Orange County must submit adequate evidence to show that a substantial and material dispute of fact exists between the County and CP&L and the NRC Staff regarding the need for an EIS to address the environmental impacts of spent fuel pool expansion at Harris.[10] However, the Staff and CP&L carry the

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Carolina Power and Light Company (October 1983).

[10] Orange County notes that without being able to review the legal and evidentiary summaries by the other parties, it is not possible, in this filing, to identify in

"ultimate burden" of sustaining their position that an EIS is unnecessary. *See* LBP-00-12 at 254 (as license applicant, CP&L bears "ultimate burden of proof" on the merits); *Louisiana Energy Services* (Claiborne Enrichment Center), LBP-96-25, 44 NRC 331, 338 (1996) (Staff has burden of proof in defending its own environmental studies).

B. Orange County Has Presented Evidence by a Qualified Expert

Orange County's Summary is supported by a detailed report prepared by Dr. Gordon Thompson. *See* Exhibit 2. Dr. Thompson is a highly qualified expert with respect to the technical issues in dispute in this phase of the Harris license amendment proceeding, which relate to probabilistic risk assessment, nuclear power plant design and operation, and spent fuel storage characteristics. He is qualified by "knowledge, skill, experience, training, or education" to render an expert opinion on the adequacy of probabilistic risk assessments and deterministic studies of nuclear power plant phenomena for purposes of addressing their adequacy to justify the Staff's refusal to prepare an EIS for the proposed Harris license amendment; and his expert opinion will "assist the trier of fact to understand the evidence" and to determine the facts in issue. *See* Federal Rule of Evidence 702, which was held applicable to NRC proceedings in *Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-669, 15

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detail the substantial and material facts that are in dispute, beyond contesting the conclusions of the EA. The most Orange County can do here is to identify all the facts, data, and arguments on which it intends to rely at the oral argument for the purpose of establishing any such dispute. Technical facts, data and arguments are set forth in detail in the Thompson Report. Legal arguments are applied to the facts in this summary.

NRC 453, 475 (1972).

Dr. Thompson's qualifications to testify regarding technical issues relating to nuclear power plant operation and design were at issue in the first phase of this proceeding. In LBP-00-12, the Licensing Board admitted Dr. Thompson's testimony on criticality prevention issues, but apparently decided to give it less weight than testimony by opposing parties, on the ground that "by reason of experience and training, his Thompson's] expertise relative to reactor technical issues seems largely policy-oriented rather than operational." *Id.,* 51 NRC at 267 note 9. It is appropriate to re-visit the question of Dr. Thompson's qualifications here, for two reasons. First, in focusing on Dr. Thompson's work on policy related issues, the Board overlooked his considerable knowledge of nuclear power plant design and operation. Second, the contention at issue involves new technical subjects that were not at play in the first phase of this license amendment proceeding: probabilistic risk assessment, and the phenomenology of spent fuel storage. Dr. Thompson is intimately familiar with both of these subjects, and has worked on them for many years. It is also important to note that some of Dr. Thompson's views on the severe accident risks of spent fuel storage, which were denounced as unsupported by the NRC Staff at the outset of this proceeding, have since been confirmed by the Staff.

Dr. Thompson is highly qualified, by training, knowledge, and experience, to testify in the proceeding. He has a Ph.D. in applied mathematics from Oxford University, and Bachelors' degrees in mechanical engineering and mathematics and physics from the University of New South Wales. His undergraduate and graduate work provided him with a rigorous education in scientific and mathematical methodologies and disciplines.

Dr. Thompson has also accumulated more than twenty years of professional experience, much of it in the study of nuclear facilities and their risks. As demonstrated in his attached Declaration and as detailed in his resume, this knowledge and experience go far beyond policy-oriented work. In the course of his career, Dr. Thompson has evaluated design and accident risk considerations associated with a significant array of nuclear power plants and other nuclear facilities in the U.S. and elsewhere around the world.[11] His work has included the study of high-density spent fuel storage and high-level nuclear waste management.

In addition, Dr. Thompson has spent over a year becoming closely familiar with the design and operation of the Harris nuclear power plant. His February 1999 report, Risks and Alternative Options Associated with Spent Fuel Storage at the Shearon Harris Nuclear Power Plant, reflected a reasonable degree of familiarity with the design of the Harris facility and with the accident risks posed by additional high-density spent fuel storage there.[12] His report for this Subpart K proceeding, The Potential for A Large Atmospheric Release of Radioactive Material from Spent Fuel Pools at the Harris Nuclear Power Plant (November 20, 2000) (Exhibit 2 to this Summary), demonstrates

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[11] *See* Curriculum Vitae: Gordon R. Thompson, Attachment 1 to Exhibit 1, Thompson Declaration. In making various plant-specific and generic evaluations of risks posed by nuclear facilities, Dr. Thompson has generally familiarized himself with the design of these facilities, and has also closely studied the design of specific facilities. *See* Thompson Declaration, pars. 6-8.

[12] Although CP&L claimed to identify "flaws" in this report, *see* Applicant's Response to BCOC's Late-Filed Environmental Contentions (March 3, 2000), these arguments reflect the Applicant's attempt to misconstrue and muddle the content of Dr. Thompson's report, not a lack of knowledge by Dr. Thompson. *See* Orange County's Reply to Applicant's and Staff's Oppositions to Late-Filed Environmental Contentions (March 13, 2000).

that in the past year he has gained a much higher and more detailed level of understanding of the design and operation of the facility, which is appropriate to the evidentiary phase of this proceeding, and which permits him to provide useful assistance to the Board.

Dr. Thompson is also extremely familiar with the art of probabilistic risk assessment, the ways that it can be used, and its strengths and limitations. He has personally conducted and/or participated in a number of studies which provide general analyses regarding the use of PRA.[13] Dr. Thompson's work related to PRA also includes a number of studies relating to the design and operation of individual facilities, including accident risks posed by plant operation and spent fuel pool storage. *See* Thompson Declaration, pars. 7 and 8.

Dr. Thompson's eminent qualifications are also demonstrated by the fact that his expert opinion has been accepted and adopted by government decisionmakers, including the NRC Staff. In 1979, for instance, the government of the German state of Lower Saxony accepted Dr. Thompson's findings about the potential for an exothermic reaction

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[13] *See* Thompson Declaration, par. 7. These studies include a comprehensive review and evaluation of the state-of-the-art of PRA conducted for Greenpeace International (Hirsch, et al, IAEA Safety Targets and Probabilistic Risk Assessment (Hanover, Germany; Gesellschaft fur Okologische Forschung und Beratung, August 1989)) (copy attached to Dr. Thompson's report as Thompson Rpt. Exh.: Hirsch, et al, 1989), a study of risks posed by high-density spent fuel at the Gorleben nuclear facility (Gordon Thompson et al, Report of the Gorleben International Review, Chapter 3, Potential Accidents and Their Effects (submitted to the government of Lower Saxony, March 1979)) (copy attached to Dr. Thompson's report as Thompson Rpt. Exh.: Thompson, et al, 1979); articles on the use of PRA in emergency planning (*Potential Characteristics of Severe Reactor Accidents at Nuclear Plants; The Use of Probabilistic Risk Assessment in Emergency-Response Planning for Nuclear Power Plant Accidents,* published in Golding, et al., Preparing for Nuclear Power Plant Accidents (Westview Press: 1995)) (copies attached to Dr. Thompson's report as Thompson Rpt. Exh: Golding, et al, 1995); and a study prepared for the Union of Concerned Scientists regarding the potential for escape of radioactive material from containment, Sholly and Thompson, The Source Term Debate

in high-density fuel pools. As a direct result, dry storage has been used for away-from-reactor storage of spent fuel throughout Germany.[14]

During the period 1986-1991, Dr. Thompson was commissioned by environmental groups to assess the safety of the military production reactors at the Savannah River Site, and to identify and assess alternative options for the production of tritium for the U.S. nuclear arsenal. Dr. Thompson's analyses of safety issues were recognized as accurate by nuclear safety officials at the US Department of Energy (DOE). *See* Thompson Declaration, par. 10.

In 1977, and again during the period 1996-1998, Dr. Thompson examined the safety of nuclear fuel reprocessing and liquid high-level waste management facilities at the Sellafield site in the UK. His investigation in the latter period was supported by a consortium of local governments in Ireland and the UK, and his findings were presented at briefings in the UK and Irish parliaments. As a direct result of Dr. Thompson's investigation, the UK Nuclear Installations Inspectorate (NII) required the operator of the Sellafield site to conduct extensive safety analyses. *See* Thompson Declaration, par. 10.

Although the NRC Staff has disparaged Dr. Thompson's qualifications earlier in this proceeding, the Staff now must also be included among the government entities that have accepted key elements of Dr. Thompson's views. For instance, the Staff has recently accepted Dr. Thompson's view that older fuel is more vulnerable to ignition in a

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(Cambridge, Massachusetts: UCS, January 1986)).

[14] *See* Thompson Declaration, par. 9; Ernst Albrecht (Minister-President of Lower Saxony), Declaration of the State government, Lower Saxony, West Germany Concerning the Proposed Nuclear Fuel Center at Gorleben (May 16, 1979). (An English translation of the Declaration is included as part of Thompson Rpt. Exh.: Thompson, et al, 1979).

state of partial drainage than in a state of total drainage, because convective heat transfer is suppressed by the presence of residual water at the base of the fuel assemblies. *See* Thompson 1999 Report at D-6.

A review of the positions taken by the Staff over the past year shows that the Staff has turned 180 degrees on this issue. Early in the proceeding, the NRC Staff either ignored the effects of partial drain-down, or attempted to dismiss its significance. *See,* for example, the NRC Staff's Draft Final Technical Study of Spent Fuel Accident Risk at Decommission Plants (noticed in the Federal Register at 65 F.3d Reg. 8,725 (February 22, 2,000), in which the Staff stated as follows:

> The staff has also considered a scenario with a rapid partial draindown to a level at or below the top of active fuel with a slow boiloff of water after the draindown. This could occur if a large breech (sic) occurred in the liner at or below the top of active fuel. Section 5.1 of NUREG/CR-0649 analyzes the partial draindown problem. *For the worst case draindown and a lower bound approximation for heat transfer to the water and the building the heatup time slightly less than the heatup time for the corresponding air cooled case. More accurate modeling could extend the heatup time to be comparable to or longer than the air cooled case.*

Id. at page A1-9 (emphasis added). *See also* NRC Staff Response to Intervenor's Request for Admission of Late-Filed Environmental Contentions at 21 (March 3, 2000) ("Dr Thompson's is the only opinion of which the Staff is aware that holds that fuel five years or more out of the reactor is susceptible to zircaloy/fire exothermic reaction"); *Id.* at 22 ("Dr Thompson's belief that such fuel is susceptible to exothermic reaction does not appear to be based on the scientific literature.")

In a recent meeting of the NRC's Advisory Committee on Reactor Safeguards ("ACRS"), however, the NRC Staff changed its position and conceded that the blockage of air flow caused by partial drainage of the fuel pool (*i.e.,* the "adiabiatic heatup case") would permit aged fuel to reach ignition temperatures. *See* statement by Glenn Kelly, NRC Staff, Tim Collins, NRC Staff's Deputy Director of the Division of Systems and Safety Analysis at 477th ACRS Meeting, Transcript ("Tr.") at 28-30 (November 2, 2000).[15] Moreover, the Staff now considers the probability of a fire in aged fuel to be within the same range as the probability of severe reactor accident as predicted by NUREG-1150. *Id.*, Tr. at 17-18 (Staff opinion that although the risk of a fire in fuel aged ten years is "low," it "could still be in the ball park of operating reactors."). Accordingly, the Staff's latter-day confirmation of the correctness of Dr. Thompson's views on one of the most important technical issues in this case should serve as a corrective to the doubts that the Applicant and Staff have attempted to sow regarding Dr. Thompson's qualifications.

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[15] For instance, Mr. Kelly stated:

> When we performed the thermal hydraulic analysis, we basically did it two ways. One was where we considered that we had air flow to provide oxygen to the potential oxidation of the fuel and also to provide cooling to the fuel and the other one was we assumed that there might have been flow blockage such that we had a near adiabatic heatup.
>
> In the adiabatic heatup case effectively as long as you have decay heat, you are going to eventually be able to get the fuel temperature up to whatever is your criteria . . ."

Tr. at 28. A copy of the meeting transcript is attached to Dr. Thompson's report as Thompson Rpt. Exh.: ACRS, 2000.

V. ORANGE COUNTY HAS RAISED A GENUINE AND MATERIAL DISPUTE REGARDING THE LIKELIHOOD OF A SEVERE SPENT FUEL POOL ACCIDENT AT HARRIS, SUCH THAT A HEARING MUST BE HELD TO DETERMINE WHETHER NEPA REQUIRES THE PREPARATION OF AN EIS.

A. Requirements of NEPA

1. Purpose of NEPA Analysis

NEPA is the "basic charter for the protection of the environment." 40 C.F.R. § 1500.1(1). Its fundamental purpose is to "help public officials make decisions that are based on understanding of environmental consequences, and take decisions that protect, restore, and enhance the environment." *Id.* NEPA requires federal agencies to examine the environmental consequences of their actions *before* taking those actions, in order to ensure "that important effects will not be overlooked or underestimated only to be discovered after resources have been committed or the die otherwise cast." *Robertson v. Methow Valley Citizen Council,* 490 U.S. 332, 349 (1989).

The primary method by which NEPA ensures that its mandate is met is the "action-forcing" requirement that a "detailed statement," known as an Environmental Impact Statement ("EIS"), be prepared before a federal agency takes any major action which may significantly affect the quality of the human environment. 42 U.S.C. § 4332(2)(C); 40 C.F.R. § 1502.1.[16] As the Court recognized in *Calvert Cliffs*

[16] 40 C.F.R. § 1502.1 is a regulation of the President's Council on Environmental Quality ("CEQ") for the implementation of NEPA. Although the NRC also has its own NEPA regulations, the CEQ regulations are binding on the NRC unless compliance would "be inconsistent with statutory requirements." Executive Order 11991, 3 C.F.R. 124 (1978). *See also Baltimore Gas and Electric Co. v. Natural Resources Defense Council,* 462 U.S. 87 (1983); *Andrus v. Sierra Club,* 442 U.S. 347 (1979); NRC Final Rule, Environmental Protection Regulations for Domestic Licensing and Related

Coordinating Committee v. AEC, 449 F.2d 1109, 1113 (D.C. Cir. 1971), a NEPA analysis involves "a finely tuned and systematic" balancing of "environmental amenities" against "economic and technical considerations." To "ensure that the balancing analysis is carried out and given full effect," an environmental impact statement must be "detailed" and the analysis carried out "fully and in good faith." *Id.*, 449 F.2d at 1114-15.

As required by NEPA and its implementing regulations, an EIS must describe, among other things, (1) the "environmental impact" of the proposed action, (2) any "adverse environmental effects which cannot be avoided should the proposal be implemented," (3) any "alternatives to the proposed action," and (4) any "irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented. . . ." 42 U.S.C. § 4332(C). The EIS must be circulated in draft for comment by the public and other affected agencies, in order to assure that relevant environmental information will "be made available to the larger audience that may also play a role in both the decisionmaking process and the implementation" of a proposed decision. *Robertson*, 490 U.S. at 349.

2. Decision not to prepare EIS must be supported by a "hard look"

NEPA requires that, in actions involving substantial undertakings, such as the instant proposal to substantially increase the inventory of radioactive material to be stored

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Regulator Functions and Related Conforming Amendments, 49 Fed. Reg. 9,352 (March 12, 1984) (restating Commission view that, "as a matter of law, the NRC as an independent regulatory agency can be bound by CEQ's NEPA regulations only insofar as those regulations are procedural or ministerial in nature. NRC is not bound by those portions of CEQ's NEPA regulations which have a substantive impact on the way in which the Commission performs its regulatory functions.") Orange County notes that all of the CEQ regulations cited in this brief are procedural in nature, and thus are binding on the NRC. Moreover, none of these regulations was disavowed by the Commission when at the Harris nuclear plant site, an agency may not dispense with an EIS unless and until it has prepared an Environmental Assessment ("EA") that evaluates whether an EIS is required, taking into account all relevant factors. *LaFlamme v. FERC*, 852 F.2d 389, 399 (9th Cir. 1988) (hydroelectric power plant license suspended for failure to prepare an EA). The EA must take a "hard look" at the potential environmental consequences of the action. *Maryland National Park and Planning Commission v. U.S. Postal Service*, 487 F.2d 1029, 1040 (D.C. Cir. 1973); *Foundation on Economic Trends v. Heckler*, 756 F.2d 143, 154 (D.C. Cir. 1985) (EA must "attempt to evaluate seriously the risk[s]" posed by proposed action.)[17]

Here, the EA prepared by the NRC Staff falls far short of constituting the "hard look" required by NEPA. The EA focuses on structural failure of a fuel pool, leading to total loss of water.[18] EA at 5-6. The present state of knowledge about fuel pool

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it promulgated its own set of NEPA regulations at 49 Fed. Reg. 9,352.

[17] In *Foundation on Economic Trends*, the Court affirmed an injunction prohibiting the National Institutes of Health from releasing genetically engineered recombinant-DNA-containing organisms into the environment, because the discussion of environmental impacts in the EA was too cursory to support a determination that no EIS was required. As the Court explained, NIH:

> must 'provide sufficient evidence and analysis for determining whether to prepare an environmental impact statement or a finding of no significant impact,' 40 C.F.R. § 1508.9(a)(1). Ignoring possible environmental consequences will not suffice. Nor will a mere conclusory statement that the number of recombinant-DNA-containing organisms will be small and subject to processes limiting survival.

756 F.2d at 155.

[18] In support of its limited discussion of that limited issue, the EA cites four NRC

accidents, however, is not confined to that accident scenario or the four reports cited by the NRC Staff. As Dr. Thompson demonstrates in his report, the loss of water from the Harris fuel pools is an almost certain outcome of a degraded-core accident, with containment failure or bypass, at the Harris reactor. The EA does not address this matter. In addition, Dr. Thompson's report shows that partial loss of water from a pool can be a more severe accident condition than total loss of water. The NRC Staff has conceded the correctness of Dr. Thompson's view. *See* discussion, *supra,* in Section IV.B. Thus, the EA not only fails to take a "hard look" at the questions raised by Dr. Thompson, but it does not even reflect the concerns of the NRC's own technical staff.

3. A high level of uncertainty weighs in favor of preparing an EIS

As the Court noted in *Foundation on Economic Trends*, "one of the specific criteria for determining whether an EIS is necessary is '[t]he degree to which the possible effects on the human environment are highly uncertain or involve unique or unknown risks.'" 756 F.2d at 155, *citing* 40 C.F.R. § 1508.27(b)(5). Thus, in *Blue Mountains Biodiversity Project v. Blackwood,* 161 F.3d 1208, 1213 (9th Cir. 1998), the Court found that "[a] project may have significant environmental impacts where its effects are 'highly uncertain or involve unique or unknown risks.'" *See also Morgan v. Walter,* 728 F.Supp. 1483, 1489 (D. Id. 1989).

The CEQ requirement to consider the degree of uncertainty of environmental

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reports: NUREG/CR-4982, Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82; NUREG/CR-5176, Seismic Failure and Cask Drop Analysis of the Spent Fuel Pools at Two Representative Nuclear Power Plants; NUREG/CR-5281, Value/Impact Analysis of Accident Preventative and Mitigative Options for Spent Fuel Pools; and NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82: Beyond Design Basis Accidents in Spent Fuel Pools. EA at 5-6.

impacts is particularly important in the instant case, where a high level of uncertainty is a key feature of probabilistic risk assessment, and where any new PRA performed by the NRC Staff or CP&L would take the art of PRA into a significant realm of uncharted territory. *See* Thompson Report at 13-16, 17.

4. Impacts that are not "remote and speculative" must be addressed in an EIS.

As the Board noted in LBP-00-19, all parties are in agreement that the NRC is not required to prepare an EIS for the purpose of addressing environmental impacts that are "remote and speculative." *Id.*, slip op. at 12. However, the Commission has not provided definitive guidance on what the phrase means. The most recent Commission pronouncements on this subject stem from a series of decisions in a spent fuel pool expansion case for the Vermont Yankee plant. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129 (1990); *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990). In CLI-90-4, the Commission reversed a determination by the Appeal Board that an accident with a probability of 10^{-4} is remote and speculative, and remanded for development of more information on the plausibility or probability of the accident scenario at issue. *Id.*, 31 NRC at 335. The Commission ordered that if the Appeal Board found the probability of the entire accident sequence was 10^{-4} or more, it was to return the case to the Commission; otherwise, it was to make its own decision as to whether the probability was remote and speculative or not. *Id.* at 335-36. In CLI-90-7, the Commission clarified that low probability is the "key to applying NEPA's rule of reason" test to contentions alleging adverse environmental impacts from a specified accident scenario. 32 NRC at 131.

The guidance provided by CLI-90-4 and CLI-90-7 can be summarized as follows: low probability is key to determining what impacts are remote and speculative; it is important to examine the particulars of each case; and the Commission is unwilling to hold, as a matter of law, that 10^{-4} is so low a probability as to be remote and speculative. As the Licensing Board observes, this last point suggests that a probability of 10^{-5} should not be rejected out of hand as remote and speculative.

Orange County submits that in determining what constitute "remote and speculative" environmental impacts, it is important to follow the Commission's guidance of examining the circumstances of each case independently. The Licensing Board should apply quantitative criteria cautiously, in light of relevant qualitative factors and the factual circumstances of each case. One of the most important qualitative factors that must be considered is the level of uncertainty that accompanies any PRA. *See* 40 C.F.R. § 1508.27(b)(5) and discussion in Section VI.B.3, *supra*. Before relying on a quantitative probability estimate to rule out the preparation of an EIS, the Board should consider such factors as the degree to which the estimate is affected by uncertainty. For example, it is necessary to take into account the degree to which unknown aspects of plant behavior are addressed through unverifiable judgments rather than calculations; the degree to which acts of malice, gross errors in design, unforeseen accident sequences or phenomena, or degraded standards of operation could influence the outcome of the analysis if they were considered; and the degree to which the results of the analysis depend on new and untested applications of PRA techniques. *See* Thompson Report at 17. In reflection of these uncertainties, any quantitative probability estimates should be expressed as a range of probabilities, rather than a point estimate. *Id.*

The circumstances of this case dictate that there will be a very high level of uncertainty in any probability analysis that is applied to the Harris spent fuel pools. Not only is the art of PRA generally subject to significant uncertainty, but the analysis required here breaks new ground in a number of areas. As Dr. Thompson discusses in his report and its appendices, Level 2 PRA is generally inadequate to address onsite effects of containment releases because it typically focuses on releases to the atmosphere, for purposes of modeling offsite doses. *See* Thompson Report at 18. Moreover, little work has been done to date on issues of onsite transport and distribution of radioactive material, and the complexities of the situation make analysis "exceptionally difficult." *See* Thompson Report at 18 and Appendix D at D-3 – D-4. With respect to the implications of heat transfer in spent fuel pools, the NRC Staff is still in the process of developing its understanding of the associated phenomena. *Id.* at 23, 40-41. Moreover, as Dr. Thompson concludes in his report, there is currently no technical basis for providing an estimate of uncertainty for probability calculations regarding spent fuel pool accidents at Harris. *See* Thompson Report at 42. Given this high level of uncertainty, it would not be defensible to dismiss the need for an EIS based on currently available quantitative estimates of the probability of a spent fuel pool accident at Harris.

In evaluating the adequacy of any PRA to support a decision not to prepare an EIS, the Board should also be mindful of the Commission's policy to limit the use of PRA to "the extent that it is supported by the state-of-the-art in terms of methods and data." Policy Statement, Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, Section IV, 60 Fed. Reg. 42,622 (August 16, 1995). Thus, the

Licensing Board should examine the extent to which it reflects the state-of-the-art, including depth and detail of analysis, uncertainty analysis, and peer review. As discussed in Dr. Thompson's report, given the complexity of the issues presented by the seven-step scenario posited by Orange County, it would be impossible for any party to conduct a state-of-the-art PRA on the hazards of spent fuel pool expansion in the extremely short time period allotted for this Subpart K presentation.

To the extent that the Board considers a quantitative standard for what constitutes a foreseeable accident requiring an EIS, it is clear that an accident probability of 10^{-4} would fall squarely within the range of impacts already recognized by the Commission as requiring the preparation of an EIS or the conduct of emergency planning. A degraded-core reactor accident with containment failure or bypass is recognized as a credible event by the NRC for purposes of evaluating environmental impacts in EIS's, as well as requiring emergency planning for the ten and fifty mile Emergency Planning Zones around nuclear plants. In addition, licensees are obligated to perform IPE's to examine the site-specific potential for accidents of this type. The lower bound of probability for a spent fuel pool accident is set by the probability of a degraded-core reactor accident with containment failure or bypass, because such an accident would almost certainly lead to a pool accident, as demonstrated in Dr. Thompson's report.

Moreover, a severe spent fuel pool accident probability estimate on the order of 10^{-5} should trigger an EIS, for the reasons that (a) it has a close order of magnitude to a reactor accident, which requires an EIS and emergency planning; (b) by its very nature as a new area of study, it is subject to considerable uncertainty; and (c) the environmental impacts of a large release of radioactive material from spent fuel pools, in terms of interdiction of vast areas of land, have never been evaluated by the NRC in an EIS.

5. Orange County has demonstrated the plausibility and foreseeability of a severe spent fuel pool accident at Harris.

In this proceeding, Dr. Thompson has provided a credible minimum value best estimate of overall accident risk of 1.6×10^{-5}, with a range from 0.2×10^{-5} to 1.2×10^{-4} per year. Thompson Report at 42. To make this estimate, Dr. Thompson provided a step-by-step detailed analysis, using data provided by CP&L and the NRC Staff. His analysis raises substantial and material factual disputes with the NRC Staff's EA by clearly demonstrating that the NRC Staff and CP&L have overlooked important factors which raise the probability of a severe spent fuel pool accident at Harris far above levels previously estimated by the Staff and CP&L.

It must be observed here that it is not Orange County's responsibility to "prove" that the probability of an accident at the Harris plant is above a certain level. In the first place, as Dr. Thompson asserts, it is not possible to provide a definitive calculation of any accident probability at Harris. Moreover, it is the NRC Staff who ultimately bears the burden of proving that no EIS is required here. Orange County has met its burden of going forward by setting forth significant and material evidence that throws the previous findings of the NRC Staff into contention and doubt, and demonstrates the "plausibility or probability" of a severe spent fuel pool accident, such that an EIS is warranted. The Board would have no lawful basis for refusing to order the preparation of an EIS based on this record.

6. The Board may not rule out an EIS that would address one form of environmental harm, based on an EA that assumes impacts are avoided or minimized by causing another form of environmental harm.

As discussed in the Thompson Report, there are many assumptions that go into a PRA. For purposes of evaluating the seven-step accident scenario set forth at page 13 of LBP-00-19, the analyst must make several key assumptions that have a substantial bearing on the adequacy of the analysis to satisfy the requirements for an adequate EA under NEPA. These assumptions have to do with whether workers will (a) incur harm in order to restore cooling to the spent fuel pools, or (b) violate NRC regulations in order to restore cooling to the spent fuel pools. Orange County submits that the NRC Staff may not lawfully base a decision not to prepare an EIS for the Harris license amendment on an analysis that assumes that workers would either incur harm or violate NRC safety regulations in order to minimize the probability of the accident. To allow such assumptions would violate the fundamental principles of NEPA that require the protection of the environment through detailed disclosure of any significant environmental harm that may be caused by major federal actions. *See Louisiana Energy Services* (Claiborne Enrichment Center), LBP-96-26, 44 NRC 331, 339 (1996), and cases cited therein (NEPA establishes "substantive goals for the Nation," that "the federal government should use 'all practicable means and measures' to protect the environment"); *Robertson v. Methow Valley*, 490 U.S. at 349 (NEPA's goal of protecting environment served through maximum disclosure of significant adverse environmental impacts).

a. Assumptions re harm to workers

The analysis of steps 4 and 5 in the seven-step scenario requires the determination of what constitutes an extreme dose such that CP&L personnel or other emergency workers would be precluded from re-entering the plant to perform the six backup functions for restoring cooling water to the fuel storage pools in the event that the primary cooling system fails. *See* Thompson Report, Sections 4.4 and 4.5. These allowable doses must be compared to likely radiation levels in the control room and the Technical Support Center, from which controls are taken and instructions given. It may also be necessary to compare them to likely radiation levels in the Fuel Handling Building and/or Reactor Auxiliary Building, where workers will have to enter in order to implement remedial actions. If radiation levels exceed the dose that is considered extreme enough to preclude access by workers, then it must be assumed for purposes of the analysis that remedial efforts are ineffectual and that therefore the accident will continue to progress, *i.e.*, that the probability of the next event in the sequence (inability to restart any pool cooling or makeup systems due to extreme radiation doses) is one.

The question of what constitutes a dose extreme enough to preclude extreme access is a legal issue, answerable by NRC regulations establishing occupational limits for radiation exposures at 5 rems TEDE per year. As discussed below, under these regulations, any dose exceeding 5 rems TEDE per year is expected to result in a level of harm to worker safety and health that is beyond the expected norm and that involves an assessment of trade-offs between adverse health effects to workers and the benefits achieved if the worker suffers increased exposure. It is exactly these trade-offs -- of harm to the health of workers versus the resulting benefits such as the likelihood of preventing an accident -- that must be assessed in an EIS. Accordingly, in assessing whether the plant is accessible for purposes of restoring spent fuel cooling functions, any dose above 5 rems TEDE per annum must be presumed to preclude personnel access. Otherwise, the probability analysis improperly assumes acceptance of one type of environmental harm (radiation exposure to plant workers beyond regulatory limits) as the justification for avoiding another type of environmental harm (harm to the general public and the environment caused by radiological releases from the spent fuel pools), without going through the process of fully disclosing these competing harms in an EIS. [19]

There are a number of reasons why for purposes of this analysis, a dose of 5 rems TEDE per year must be considered the upper limit of acceptable dose limits, beyond which doses are presumptively harmful. First and foremost, 5 rems TEDE per year is the occupational dose limit established by NRC standards for protection of worker safety and health. [20] *See* 10 C.F.R. § 20.1201(a)(1)(i).[21] The 5 rem standard was recommended by

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[19] Deposition testimony suggests the existence of a material dispute on this issue. During their depositions, experts for the NRC Staff and CP&L expressed differing opinions about how to answer the question of what constitutes an extreme dose sufficient to preclude access to the Harris plant. Dr. Gareth Parry, Senior Level Advisor for PRA in the Division of System Safety and Analysis of the NRC's Office of Nuclear Reactor Regulation ("NRR"), asserted that he would probably assume that doses were high enough to preclude access if they exceeded regulatory limits. Deposition of Dr. Gareth W. Parry, Transcript ("Tr.") at 91-92 (October 19, 2000). NRC witness Stephen LaVie, Health Physicist with the Office of NRR, stated that the Staff's "initial feeling" is that it is appropriate to use the U.S. Environmental Protection Agency's recommended Protective Action Guideline ("PAG") of 25 rem per accident for actions needed to save human lives. Deposition of Stephen LaVie, Tr. at 14 (October 20, 2000). Dr. Edward T. Burns, CP&L's expert on PRA, testified that there is no "firm threshold" for a dose that would preclude access to the plant, and that it is appropriate to look at the relative severity of radiation levels and make a probability calculation as to how likely a person would be to enter the radiation environment. Deposition of Dr. Edward T. Burns, Tr. at 58-59 (October 20, 2000).

[20] Although somewhat higher exposures are permitted by Part 20 regulations, these

the International Commission on Radiological Protection ("ICRP"), and was accepted by the NRC on the basis that it would maintain the annual risk of radiation-induced health damage to about 8×10^{-4}. Proposed Rule, Standards for Protection Against Radiation, 51 Fed. Reg. 1,092, 1,102 (January 9, 1986). Thus, the NRC has made a reasoned judgment that 5 rems TEDE is the maximum level of radiation that a worker can receive in a year and stay within acceptable bounds of occupational risk levels.

Second, compliance with Part 20 occupational exposure limits is assumed in the Final EIS that supported the issuance of an operating license for Harris. *See* NUREG-0972, Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, Docket Nos. STN-50-400 and STN-50-401, Carolina Power and Light Company at 5-28 (October 1993). As discussed in NUREG-0972:

> Experience shows that the dose to nuclear plant workers varies from reactor to reactor and from year to year. For environmental-impact purposes, it can be projected by using the experience to date with modern PWRs. Recently licensed 1000-Mwe

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exposures must be planned in advance, with numerous accompanying safeguards, and thus are inapplicable to accident conditions. *See* 10 C.F.R. § 20.1206(a), which permits a "planned special exposure" if it would not cause an individual to receive a dose from all planned special exposures and all doses in excess of the limits to exceed (1) the numerical values of any of the dose limits in § 20.1201(a) in any year; and (2) five times the annual dose limits in § 20.1201(a) during the individual's lifetime. Thus, they are not applicable to an unplanned severe accident situation.

Orange County recognizes that based on his professional judgment as a scientist, Dr. Thompson has applied the limits for planned special exposures in his analysis. *See* Thompson Report at 32-33. The County believes that NEPA requires setting a stricter threshold, in order to avoid hidden assumptions of environmental harm in an EA that should otherwise be disclosed in an EIS. It should be noted that the doses calculated by Dr. Thompson are far in excess of either the normal occupational limits *or* the planned occupational limits.

[21] In addition, doses must be further reduced, to the extent reasonably achievable, under the Commission's ALARA ["As Low As Reasonably Achievable] regulations. See 10 C.F.R. § 20.1101(b).

> PWRs are operated in accordance with the post-1975 regulatory requirements and guidance that place increased emphasis on maintaining occupational exposure at nuclear power plants ALARA. These requirements and guidance are outlined primarily in 10 CFR 20, Standard Review Plan (SRP) Chapter 12 (NUREG-08000), and Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclar Power Stations Will be as Low as Is Reasonably Achievable."
>
> The applicant's proposed implementation of these requirements and guidelines is reviewed by the NRC staff during the licensing process, and the results of that review are reported in the staff's Safety Evaluation Report. The license is granted only after the review indicates that an ALARA program can be implemented. In addition, regular reviews of operating plants are performed to determine whether the ALARA requirements are being met.

Id. Having assumed regulatory compliance with Part 20 in the FEIS, the Staff would have no lawful basis for now assuming that the proposed expansion of the Harris spent fuel pools poses no cognizable risk of a spent fuel pool accident because workers will be expected to incur unlawful radiation doses in order to minimize that risk.

Third, in setting Protective Action Guidelines ("PAGs") for workers during radiological emergencies, the U.S. Environmental Protection Agency ("EPA") recommends the use of a 5 rem per year "upper bound" for worker exposures during a radiological emergency.[22] *See* U.S. EPA, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents at 2-10 (October 1991). *In addition,* the EPA recommends that doses be kept "as low as reasonably achievable," *i.e.,* even lower than 5 rems per year, as is consistent with the regulation of normal occupational exposures. *Id.* The EPA's guidance makes it clear that doses above 10 rems TEDE per year are only justified by the protection of "valuable property," and doses up to 25 rems TEDE per year are only justified "for life saving activities and the protection of large populations." EPA

[22] Thus, although the EPA accepts a dose of 5 rems per accident, it also assumes that no

considers doses above 25 rems TEDE per year to be justified only under the most extreme circumstances:

> Situations may also rarely occur in which a dose in excess of 25 rem for emergency exposure would be unavoidable in order to carry out a lifesaving operation or to avoid extensive exposure of large populations. It is not possible to prejudge the risk that one should be allowed to take to save the lives of others. However, persons undertaking any emergency operations in which the dose will exceed 25 rem to the whole body should do so only on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

Id. at 2-11.

It is clear that both NRC regulations and EPA guidance establish a presumption of harm if radiation doses exceed the annual dose limit of 5 rems. Doses above 5 rems are seen by EPA as involving trade-offs, with the individual worker's life and health being off-set against the value of property, or the value of saving many lives. In other words, EPA recognizes that these exposures are hazardous to nuclear power plant workers, and that they are only justified if they would serve a greater good.

In summary, for purposes of determining whether or not the preparation of an EIS is warranted, it is appropriate and consistent with NEPA to assume that a radiation environmental yielding doses in excess of 5 rems TEDE per annum would preclude access by emergency personnel. To assume otherwise would effectively countenance one type of environmental harm (radiation exposure to plant workers beyond NRC safety limits and EPA guidance levels) in order to avoid another type of environmental harm (harm to the general public and the environment caused by radiological release from the

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worker is exposed to more than 5 rems in a year.

spent fuel pools). Such an assumption would also be grossly inconsistent with the EIS for the Harris operating license, which assumed that Harris would operate in compliance with NRC regulations.

Orange County wishes to emphasize that NEPA requires the assumption that doses above 5 rems TEDE per year preclude access in this particular legal context, which is the performance of analysis intended to evaluate whether an EIS is necessary. As discussed above, it would not be appropriate to assume, in this context, that plant workers would incur significant harm in order to maintain the probability of a spent fuel pool accident below the level that would call for an EIS. In the real-life context of an accident, it would be appropriate to assume that occupational dose limits may be exceeded. In a full-scale EIS, it may also be appropriate to examine the environmental impacts to workers of attempting to prevent the progression of a severe accident for purposes of examining the trade-offs posed by alternatives and mitigative measures. It is neither appropriate nor lawful, however, to attempt to justify the Staff's refusal to prepare an EIS for this proposed spent fuel pool expansion, based on the assumption that workers would incur unlawful and significant radiation injuries in order to prevent the accident from progressing.

b. Other assumptions of regulatory violations

As discussed above, it may not be assumed, for purposes of avoiding an EIS, that workers are exposed to environmental harm by incurring doses above occupational exposure limits. Similarly, the analysis may not assume the violation of regulations which were promulgated for the purpose of protecting protect public health and safety, and which the 1983 FEIS assumed would be met in order to maintain environmental impacts within an acceptable level.

For instance, General Design Criterion 19 of Appendix A to 10 C.F.R. Part 50 requires that the control room must be designed to prevent workers from receiving doses above 5 rems during an accident. As Dr. Thompson demonstrates in his report, radiation levels in the control room would far exceed these levels, and would, in fact, be lethal. For purposes of rationalizing the refusal to prepare an EIS, it is not lawful to assume that the requirements of GDC 19 would be violated in order to minimize the probability of an accident.

CP&L's own procedures for severe accident management appear to set up a conflict between severe accident responses and compliance with NRC safety regulations. CP&L recognizes, in its Severe Accident Management Guidelines ("SAMGs"), that in responding to a severe accident, it may be necessary to take actions which conflict with the plant's Technical Specifications.[23] Moreover, these procedures "may not have been safety reviewed." The Tech Specs are an integral part of the Harris license, and thus the 1983 FEIS necessarily assumed that they would be complied with. Non-compliance with technical specifications could raise new safety challenges, in addition to any threats stemming from the severe accident that is underway. Any assumption of regulatory violations for purposes of avoiding a severe accident must therefore be fully addressed in an EIS, rather than relied on in an EA for the purpose of avoiding an EIS.

[23] *See* CP&L, Plant Operating manual, Volume 11, Part 1, SAMG-SAMP-001, Severe Accident Management Guidelines Program Document, Rev. 2 at 3, excerpt attached to Dr. Thompson's report as Thompson Rpt. Exh. CP&L-POM.

VI. THE PROCEDURAL CIRCUMSTANCES OF THIS PROCEEDING REQUIRE THAT A HEARING BE HELD.

As discussed above in Section III.B,3, the discovery period in this expedited Subpart K proceeding was a brief 60 days. By the end of that period, none of the parties had been able to complete their analyses or report the results. Much of the work remained to be done at the time depositions were taken. Thus, it was impossible to question the NRC Staff's or CP&L's witnesses regarding the results of their analyses or how they were arrived at.

The review of the evidence presented in this Subpart K proceeding undoubtedly will raise questions about the assumptions underlying various calculations, and the methodology used. Of course such questions can't be identified with specificity at this juncture, because Orange County has not had a chance to review the presentations of the other parties. Nevertheless, at this juncture it is appropriate to raise the concern that due to the complexity of the issues raised in this proceeding, and due to the fact that none of the parties could be questioned about the results of their analyses in discovery, disagreements about the substantiality or materiality of any factual disputes between the parties should be seen in the light most favorable to Orange County.

VII. RESPONSE TO BOARD'S QUESTIONS

A. Best Estimate of Overall Probability of Sequence Set Forth in Chain of Events

In Question 1, the Board asked:

What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth in page 13 <u>supra</u>? The estimates should utilize plant-specific data where

> available and should utilize the best available generic data where generic data is relied upon.

LBP-00-19, slip op. at 17. Information responsive to this request is provided in detail in the attached report by Dr. Thompson, including the appendices. Dr. Thompson uses the best available plant specific and generic data and explains the basis for his choices of data. As discussed in Dr. Thompson's report at page 42, he has found that a minimum value for the best estimate of the overall probability of completion of the seven-part event sequence is 1.6×10^{-5} per year (point estimate), with a range from 0.2×10^{-5} to 1.2×10^{-4} per year.

B. Recent developments in the estimation of probabilities of individual events

The Board's second question asks the parties to take careful note of recent developments in the estimation of individual events in the sequence, and questions whether new data or models suggest any modification of the probability estimate set forth in NUREG-1353. In addition, the Board asks for comment on the concerns expressed in an ACRS letter of April 13, 2000. These questions are addressed in Dr. Thompson's report, Section 5 at page 54.

C. Scope of EIS Required

The Board's third question asks what is the scope of an EIS that would be required, assuming that the Board should decide that the probability of an accident cannot be dismissed as remote and speculative. Dr. Thompson provides a technical response to this question in Section 6 of his report, at page 45. This summary addresses the legal question posed by the Board, which appears to be whether the Board could somehow order that the scope of the EIS be limited to the seven-part accident scenario listed in LBP-00-19.

Orange County submits that the Board would not have that degree of authority. Once an EIS is undertaken, NEPA requires that an agency must take a "hard look" at the environmental impacts of a proposed major federal action, which includes the examination of all reasonably foreseeable and significant adverse impacts.

Moreover, the preparation of an EIS is a public process, designed to maximize public involvement in the consideration of impacts and alternatives. *See Robertson v. Methow, supra.* Thus, the EIS must be subject to public notice and comment, including the offer of an opportunity to interested members of the public to request an adjudicatory hearing on its adequacy. Any member of the public would have the right to challenge the overall adequacy of the EIS to address the adverse environmental impacts of the project. Orange County does not believe that it would be consistent with the public participation requirements of NEPA if an interested member of the public could lawfully be precluded from raising valid concerns about an EIS, based on procedural grounds relating to this proceeding.

VIII. CONCLUSION

Orange County has provided substantial and material evidence and legal arguments which demonstrate that the NRC Staff has failed to justify its refusal to prepare an Environmental Impact Statement for the proposed expansion of spent fuel pool storage capacity at the Harris reactor. Therefore, Orange County has raised a substantial and material factual dispute with the Staff, and is entitled to a hearing on Contention EC-6.

Respectfully submitted,

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November 20, 2000

Novermber 27, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400 -LA
(Shearon Harris Nuclear	)	ASLBP No. 99-762-02-LA
Power Plant)	)	
	)	

ORANGE COUNTY'S ERRATA TO APPENDIX A TO THOMPSON REPORT

Orange County hereby submits the following errata to Appendix A to the report of Dr. Gordon Thompson, which was filed in support of Orange County's Detailed Summary of Facts, Etc. (November 22, 2000). A corrected copy is enclosed.

Page	**Change**
A-3	Mark "Carr, 2000" with an asterisk
A-3	Mark "CP&L-POM" with an asterisk
A-4	Mark ""CP&L, 1993" with an asterisk; insert "(excerpts)" after "August 1993)."
A-7	With respect to "Leigh, et al, 1986, insert "(excerpts)" after "August 1986)."
A-11	Mark "NRC, 1982" with an asterisk; insert "(excerpts)" after "April 1982)."
A-14	Remove asterisk from "Thompson, 1996"

Respectfully submitted,

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THE POTENTIAL FOR A LARGE ATMOSPHERIC RELEASE OF RADIOACTIVE MATERIAL FROM SPENT FUEL POOLS AT THE HARRIS NUCLEAR POWER PLANT:
The Case of a Pool Release Initiated by a Severe Reactor Accident

A report by IRSS
20 November 2000
CORRECTED 27 NOVEMBER 2000

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TPRD/B/0072/N82

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A VIRTUAL SOURCE MODEL FOR BUILDING WAKE DISPERSION IN NUCLEAR SAFETY CALCULATIONS

By C.D. Barker

March 1982

1. INTRODUCTION

The Gaussian plume dispersion model recommended by the U.K. Atmospheric Dispersion Modelling Working Group (ADMWG, see Clarke, 1979) can be used to model the dispersion of airborne pollutants released from an isolated point source for downwind distances in the approximate range of one hundred metres to several tens of kilometres. Consequently, the model is a useful tool which can be used in the calculation of radiological doses to the public due to the atmospheric discharge of radionuclides from nuclear power stations. Very often, however, the radionuclides are discharged from large reactor buildings and so a wake dispersion model is also required in order to calculate concentrations of activity within a few hundred metres of the building where the point source model is no longer applicable.

Many simple building wake models are known to exist, but all have different limitations; the virtual source model is arguably one of the better of the simple models currently used and this report, therefore, examines how this model may best be applied, bearing in mind the particular requirements of the U.K. nuclear power industry. The advantages and limitations of the model are discussed in some detail and preliminary conclusions as to the range of applicability of the model are drawn.

2. REVIEW OF SIMPLE BUILDING WAKE DISPERSION MODELS

Mathematical models for the behaviour of an airborne pollutant in the lee of the source building can conveniently be classified into two groups; complex and simple models. The complex models (see, for example, Hirt, Ramshaw & Stein, 1978) study the air flow around buildings and attempt to solve the differential equations governing such behaviour while the simple models, in general, consist of empirical expressions. By definition, therefore, the complex models are conceptually better, but they are extremely difficult to use and, in general, they can only consider simplistic building shapes, so that their applicability to a complex site such as a nuclear power station is somewhat doubtful. The simple models, on the other hand, are not realistic in their consideration of the physical processes actually occurring and mostly ignore effects such as emission buoyancy and momentum, source location, etc. They are, however, designed for ease of application and consequently they have been more widely used.

The simple models can be further sub-divided into two classes; models based on the dimensions of the source building and models based on the Gaussian plume model. The models based on the dimensions of the source building are briefly summarised in Appendix 1.1, and are specifically limited in their range of applicability to downwind distances where the physical presence of the source building plays the major role in governing the dispersion of the plume and where atmospheric stability effects can be ignored, i.e. at downwind distances less than a few building heights. Such models cannot be simply 'merged' with medium range dispersion models and their main uses are either in predicting near wake concentrations or in approximate or 'back of the envelope' calculations of main wake concentrations where they can be quickly and simply applied to give order of magnitude estimates.

The models based on the Gaussian plume diffusion model generally consist of empirical modifications to the plume parameters (σ_y and σ_z) and/or to the effective release height of the plume (h). The modifications are usually 'designed' so that the wake model merges smoothly into a normal medium range Gaussian dispersion model and several examples of such models are briefly outlined and discussed in Appendix 1.2. The 'virtual source' and 'quadratic' models are probably the best of this type, but they must be carefully defined or else they can give predictions which are grossly at variance with observed data (by factors of 10 or more). The particular application of such models by the nuclear power industry also places specific requirements on the models and these are discussed in the next Section and a method which partially satisfies them is presented.

3. A SIMPLE MODEL FOR NUCLEAR SAFETY CALCULATIONS

3.1 Background and Special Requirements

The main application of atmospheric dispersion models (including building wake models) by the U.K. nuclear power industry is in the calculation of radiological doses to the public due to the atmospheric discharge of radionuclides from nuclear facilities. These calculations are mostly carried out in order to satisfy both the operating company and government departments that such doses are acceptably low and it is, therefore, very important that the estimated doses (and, hence, the dispersion estimates which are used) be as accurate as possible provided that they can be shown to err towards pessimism rather than towards optimism.

The practical implementation of this "accurate pessimism" criterion has been that a release of activity from an isolated stack has been modelled

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October 12, 2000

Via Federal Express
Dr. Gordon Thompson
IRSS
27 Ellsworth Avenue
Cambridge, MA 02139

Re: Docket No.50-400-OLA
Orange County's Request for Documents

Dear Dr. Thompson:

Enclosed are copies of the following numbered documents requested by Ms. Diane Curran following her review of documents produced by CP&L. These documents supplement those previously provided to you.

120001 - 1200058

Should you have any questions regarding the enclosed, please contact me at the above number, or my paralegal, Ann Fanning, at 919/546-3283.

Sincerely,

Steven Carr
Associate General Counsel

c/fax: Diane Curran (w/o enclosures)

Response to Orange County's Document Request Dated September 21, 2000

Response to Request No. 1 and 6

Notes:

1. Storage Locations are designated by pool A or B, module number, and cell number. For example, A-E1A7 means Pool A, Module E1, Cell A7. Source Reactors are designated by the following abbreviations:
   a. B1 means Brunswick Unit 1 and 7x7 or 8x8 BWR type fuel
   b. B2 means Brunswick Unit 2 and 7x7 or 8x8 BWR type fuel
   c. H1 means Harris Unit 1 and 17x17 PWR type fuel
   d. R2 means Robinson Unit 2 and 15x15 PWR type fuel
2. Burnup of the fuel is expressed in units of megawatt days of thermal energy per metric ton of initial uranium contained in the fuel.
3. The data included in the response reflects the inventory as of 9/13/2000.

STORAGE LOCATION	SOURCE PLANT	BURNUP (MwD/MTU)	REMOVAL FROM REACTOR
A-E1A7	B2	7206	9/21/77
A-E1D1	B2	6778	9/21/77
A-D1L4	B2	5210	9/15/77
A-D1L1	B2	7202	9/20/77
A-E1C7	B2	6883	9/22/77
A-E1D8	B2	6765	9/22/77
A-E1C5	B2	6982	9/15/77
A-E1E8	B2	6748	9/20/77
A-E1C4	B2	6987	9/21/77
A-E1A3	B2	7064	9/21/77
A-E1B1	B2	6907	9/20/77
A-E1C6	B2	6851	9/21/77
A-E1C1	B2	6744	9/20/77
A-E1E10	B2	6680	9/21/77
A-E1E11	B2	6699	9/21/77
A-E1D5	B2	6736	9/22/77
A-E1B5	B2	6961	9/20/77
A-D1L3	B2	7296	9/23/77
A-E1D6	B2	6758	9/24/77
A-E1D9	B2	6722	9/20/77
A-E1E9	B2	6756	9/23/77
A-E1D11	B2	6751	9/21/77
A-E1B7	B2	6986	9/20/77
A-E1D3	B2	6759	9/22/77
A-E1B11	B2	7240	9/20/77
A-E1B9	B2	6857	9/15/77
A-D1L2	B2	7004	9/20/77
A-E1D7	B2	6815	9/17/77
A-E1B8	B2	6992	9/20/77
A-E1B10	B2	7032	9/20/77
A-E1A1	B2	7297	9/22/77
A-E1C3	B2	6847	9/17/77
A-D1L5	B2	6988	9/21/77
A-E1D2	B2	6798	9/22/77
A-E1D4	B2	6729	9/24/77
A-E1C2	B2	6698	9/22/77
A-E1B6	B2	6819	9/20/77
A-E1D10	B2	6687	9/21/77
A-E1A2	B2	7005	9/21/77
A-E1A8	B2	7240	9/21/77
A-E1E7	B2	6688	9/23/77
A-E1C11	B2	6869	9/23/77
x B-B7J1	B2	19219	5/2/82
A-D1A6	B2	20702	3/17/80
x B-B7G1	B2	18989	5/2/82
x B-B2F7	B2	18861	3/17/80
x B-A1F4	B2	19216	3/20/80
A-C1L7	B2	20337	3/14/80
A-D1A11	B2	17803	3/20/80
A-D1A10	B2	20449	3/17/80
x B-B8A3	B2	22570	5/5/82

120002

A-D1C11	B2	20367	3/15/80
B-B7E8	B2	21563	5/1/82
B-B8B9	B2	19990	5/2/82
B-B7H3	B2	19988	5/1/82
A-D1A5	B2	20383	3/16/80
A-D1B8	B2	21031	3/17/80
B-A1F5	B2	18732	3/18/80
B-B8E10	B2	20710	5/1/82
A-D1B7	B2	19412	3/16/80
B-B8C2	B2	21898	3/14/80
A-C1K9	B2	20587	3/19/80
B-B6L11	B2	19959	5/3/82
A-C1J7	B2	20746	3/17/80
B-B8F6	B2	25495	5/7/82
B-B8H10	B2	25538	5/7/82
B-B8A5	B2	21389	5/7/82
A-C1L8	B2	19281	3/17/80
A-C1K10	B2	22990	5/5/82
A-C1J11	B2	19027	3/12/80
B-B7C10	B2	20692	3/20/80
B-B8H7	B2	25282	5/5/82
B-B8D1	B2	18809	3/16/80
B-B7A9	B2	25552	5/6/82
B-A1F9	B2	23007	5/5/82
A-C1J8	B2	19068	3/18/80
A-C1L11	B2	18841	3/16/80
A-C1K5	B2	21079	3/14/80
B-B6K11	B2	23262	5/6/82
B-B8E2	B2	20088	3/17/80
B-B8J3	B2	20475	3/18/80
B-A7L9	B2	20377	3/17/80
B-A1F10	B2	20414	3/21/80
B-B8B7	B2	20610	3/15/80
B-A1G3	B2	19273	3/16/80
B-A8L10	B2	18633	3/20/80
B-B6K10	B2	23444	5/6/82
B-B8E1	B2	18551	3/17/80
B-B8B3	B2	20166	5/1/82
B-B7E11	B2	18349	3/18/80
B-B8B10	B2	18923	3/14/80
B-B7A11	B2	24660	5/2/82
B-B7L1	B2	22056	5/7/82
B-B8H11	B2	24533	5/2/82
B-B8F7	B2	21319	5/7/82
B-B7A8	B2	22478	5/6/82
B-A1G4	B2	18840	3/14/80
B-A1G7	B2	20771	3/17/80
B-B8B4	B2	19795	5/2/82
B-A8L9	B2	19166	3/14/80
B-A1D5	B2	18309	3/17/80
B-B8E3	B2	20392	3/17/80
B-A1D8	B2	20450	3/18/80
B-B8C3	B2	19096	3/20/80
B-B7G2	B2	21806	5/5/82
B-B8F8	B2	25268	5/7/82

B-A1D9	B2	18755	3/16/80
B-A8L4	B2	18131	3/14/80
B-B8C8	B2	19778	5/1/82
B-B7H1	B2	19108	5/1/82
B-A1E2	B2	19414	3/15/80
B-A8L6	B2	24998	5/5/82
B-A1E3	B2	19439	3/18/80
B-B8E4	B2	19088	3/16/80
B-B7L3	B2	23398	5/7/82
B-B7E10	B2	20631	3/18/80
B-A1E7	B2	16353	3/13/79
B-B8J1	B2	21049	5/7/82
B-B8G2	B2	19015	3/21/80
B-B7G4	B2	25025	5/2/82
B-B7J3	B2	21053	5/5/82
B-A1E8	B2	23021	5/6/82
B-A8L7	B2	19646	3/20/80
B-B7B9	B2	22160	5/1/82
B-B8F10	B2	20197	3/21/80
B-B8A1	B2	20469	3/14/80
B-B7K3	B2	23565	5/2/82
B-B8G4	B2	23363	5/7/82
B-B7A7	B2	21018	5/6/82
B-B8G6	B2	23998	5/7/82
B-B6L9	B2	19592	5/2/82
B-B6K8	B2	19606	5/2/82
B-B8G5	B2	25045	5/6/82
B-B7L4	B2	25043	5/2/82
B-B8G7	B2	20077	3/21/80
B-B7G3	B2	22668	5/5/82
B-A8L2	B2	22268	5/4/82
B-A1E11	B2	18256	3/18/80
B-B8B2	B2	17208	3/14/79
B-B6L6	B2	25929	5/7/82
A-E1B4	B2	10560	9/22/77
B-B7B11	B2	16985	3/9/79
B-B7G5	B2	25598	5/7/82
B-B7B10	B2	25690	5/5/82
B-A1B9	B2	16901	3/13/79
B-A1B10	B2	17823	3/13/79
B-A1B11	B2	17077	3/14/79
B-B2D10	B2	17031	3/14/79
B-B8C1	B2	18912	3/14/80
B-A1C2	B2	16977	3/13/79
B-B8H9	B2	22428	5/2/82
B-A1B4	B2	19446	3/21/80
B-B7C8	B2	21879	5/2/82
B-A1C3	B2	17386	3/13/79
B-A8L8	B2	22346	5/1/82
A-E1A5	B2	10486	9/24/77
B-B7C9	B2	16820	3/13/79
B-B8B11	B2	17239	3/14/79
B-A1C4	B2	17365	3/11/79
B-B8B8	B2	22490	5/2/82
B-B8J2	B2	17500	3/11/79

B-A1C5	B2	17270	3/14/79
B-B8A8	B2	19339	3/17/80
B-B8F4	B2	16911	3/9/79
B-B7K4	B2	17106	3/13/79
B-A1K9	B2	18121	3/19/80
B-B8F5	B2	17197	3/13/79
B-B8C11	B2	17996	3/13/79
B-A1C6	B2	17218	3/11/79
B-B8D2	B2	16877	3/13/79
B-B8D11	B2	17451	3/13/79
B-B8G8	B2	25397	5/4/82
B-A1K6	B2	18210	3/15/80
B-A1C7	B2	17179	3/13/79
B-B7B8	B2	17118	3/11/79
B-A1C8	B2	16807	3/13/79
B-B7C11	B2	17961	3/14/79
B-A1C9	B2	17084	3/13/79
B-A1C10	B2	16823	3/11/79
B-A1C11	B2	17933	3/14/79
B-A1G11	B2	17346	3/11/79
B-B2E9	B2	18054	3/14/79
B-B8C5	B2	19039	5/1/82
B-B7F8	B2	20830	5/1/82
B-A1G8	B2	17402	3/11/79
B-B8A10	B2	16940	3/12/79
B-B8F9	B2	16355	3/14/79
B-B8H5	B2	18691	3/18/80
B-B7K1	B2	20689	5/2/82
B-B7A10	B2	21454	5/2/82
B-B8E6	B2	17120	3/12/79
B-B8C6	B2	20868	5/1/82
B-A1D2	B2	17139	3/12/79
B-A1D3	B2	16948	6/5/82
B-A1D4	B2	16541	3/12/79
B-B8H4	B2	17177	3/14/79
B-A7L10	B2	16970	3/11/79
B-B8C7	B2	16581	3/13/79
B-B8G3	B2	18244	3/17/80
B-B8E7	B2	17399	3/12/79
B-B2F4	B2	17620	3/11/79
B-B7F10	B2	21135	5/1/82
A-E1A11	B2	10472	9/22/77
B-B7B7	B2	25704	5/6/82
B-A1A2	B2	16955	3/13/79
B-B8D3	B2	19121	5/1/82
B-A1H4	B2	17568	3/13/79
B-A1A3	B2	16938	3/9/79
B-B8A4	B2	23364	5/1/82
B-A1A5	B2	16856	3/13/79
B-A1A6	B2	16660	3/13/79
B-A1A7	B2	17615	3/13/79
B-A1A8	B2	18292	3/13/79
B-B7F9	B2	21863	5/1/82
B-B7C7	B2	26501	5/5/82
B-A1A9	B2	16796	3/13/79

B-B6K7	B2	22846	5/1/82
B-A1A10	B2	17418	3/12/79
B-A1H5	B2	18028	3/14/79
B-A7L8	B2	17106	3/14/79
B-B8G1	B2	17748	3/14/79
B-B8H8	B2	16920	3/9/79
B-B8D6	B2	24539	5/4/82
B-B8H1	B2	22316	5/2/82
B-B7D8	B2	17175	3/13/79
B-B8G9	B2	25647	5/7/82
B-B8H3	B2	18966	3/18/80
B-B7J4	B2	20777	5/1/82
B-A1H9	B2	18588	3/20/80
B-A1A11	B2	17587	3/13/79
B-B7H5	B2	22310	5/2/82
A-E1C9	B2	10612	9/24/77
B-B2F5	B2	17529	3/12/79
B-A1B2	B2	16898	3/14/79
A-E1A6	B2	10416	9/22/77
B-B8G11	B2	16437	3/12/79
A-E1A10	B2	10506	9/22/77
B-A1H10	B2	23157	5/1/82
B-A1J2	B2	18977	3/20/80
B-B7H4	B2	26042	5/4/82
B-B6L8	B2	23373	5/1/82
B-B7D10	B2	18833	3/19/80
B-A1H7	B2	17332	3/14/79
B-A1B3	B2	16343	3/31/79
B-B8C9	B2	16986	3/13/79
B-B7F7	B2	16988	3/13/79
A-E1C8	B2	10566	9/24/77
B-A1B5	B2	17061	3/13/79
B-B8E5	B2	18686	3/16/80
B-B8A6	B2	22264	5/1/82
B-A1B6	B2	17411	3/12/79
B-A1J5	B2	17699	3/14/79
B-B8A9	B2	18498	3/16/80
B-A7L11	B2	16909	3/11/79
B-B8E9	B2	16912	3/13/79
B-B8J4	B2	18030	3/14/79
B-A1D1	B2	26384	5/5/82
B-A1B7	B2	17269	3/13/79
B-B8E11	B2	17767	3/14/79
A-E1B2	B2	10562	9/22/77
B-B8D9	B2	25681	5/7/82
B-B8F1	B2	17217	3/14/79
B-A1B8	B2	17521	3/11/79
A-E1A9	B2	10391	9/24/77
B-B8A11	B2	17176	3/12/79
B-B7D11	B2	18521	3/19/80
B-B8B1	B2	19805	3/14/80
B-A1J6	B2	16695	3/14/79
B-B8C10	B2	18281	3/13/79
B-A1J10	B2	17160	3/14/79
B-A1J11	B2	16946	3/10/79

A-E1B3	B2	10632	9/22/77
B-B8A7	B2	20958	5/1/82
B-B8B5	B2	19129	5/2/82
B-B8E8	B2	19754	3/19/80
B-B8F2	B2	18192	3/20/80
A-E1A4	B2	10457	9/24/77
B-A1K5	B2	17541	3/14/79
B-B8F11	B2	24598	5/4/82
B-B8H6	B2	25413	5/5/82
B-B8D4	B2	21209	5/1/82
B-B8A2	B2	17562	3/14/79
B-B8B6	B2	17470	3/13/79
B-B8D7	B2	17084	3/13/79
B-B7D9	B2	18114	3/19/80
B-B8D8	B2	22165	5/1/82
A-E1C10	B2	10580	9/24/77
B-B7E7	B2	18081	3/14/79
B-B8H2	B2	16434	3/14/79
B-A8L11	B2	21296	3/12/80
B-A8L5	B2	24782	5/4/82
B-B8D10	B2	20954	5/1/82
B-B8C4	B2	19779	3/20/80
B-B7K2	B2	25532	5/6/82
B-B7H2	B2	22669	5/1/82
B-A7L7	B2	21173	5/1/82
B-B7D7	B2	19007	3/19/80
B-B8G10	B2	22127	5/2/82
B-B8F3	B2	17173	3/14/79
B-A8L3	B2	17990	3/20/80
B-B6L5	B2	26203	5/7/82
B-B7J2	B2	25704	5/7/82
B-B8D5	B2	25452	5/4/82
B-B7E9	B2	17095	3/13/79
B-B6K9	B2	18086	5/2/82
B-B6L10	B2	18349	5/2/82
B-B7L2	B2	18339	5/2/82
B-C5E1	H1	25629	4/23/97
B-D4G6	H1	30539	4/4/94
A-D2C5	H1	17621	8/26/88
B-C5A1	H1	33428	4/23/97
B-D7E6	H1	30965	9/13/95
B-D5D1	H1	26031	11/9/89
B-D7A2	H1	25564	4/23/97
B-C5B2	H1	25530	4/23/97
B-D7B7	H1	30203	4/22/97
B-D7C2	H1	25588	4/23/97
B-D2K4	H1	31864	9/13/95
A-D2H1	H1	17509	8/27/88
B-D3K7	H1	31189	9/13/95
B-C5E8	H1	23378	4/22/97
B-C5E2	H1	23429	4/23/97
B-C4B4	H1	33412	4/23/97
B-D2D4	H1	33496	4/21/97
B-D7A7	H1	32453	4/22/97
B-D3E6	H1	31724	9/13/95

B-D7B1	H1	23253	4/23/97
B-D5K1	H1	31679	9/13/95
B-D7F7	H1	32694	4/22/97
B-D3H3	H1	31406	9/13/95
B-C5F8	H1	32459	4/22/97
A-D2F6	H1	17821	8/26/88
B-D7B6	H1	30366	4/3/94
B-D1D2	H1	23641	4/22/97
B-D3J6	H1	32787	9/13/95
B-C5D1	H1	25718	4/23/97
B-C4B7	H1	30135	4/3/94
B-D2B5	H1	23435	4/21/97
B-D3K3	H1	30658	9/13/95
B-D2F5	H1	23470	4/21/97
B-D3B4	H1	33503	4/20/97
B-D5A1	H1	30456	4/2/94
B-C5C1	H1	25678	4/23/97
B-C5C2	H1	23446	4/23/97
A-D2G4	H1	17458	8/27/88
B-C5F2	H1	25353	4/23/97
B-D5D2	H1	29017	4/3/91
B-D7G7	H1	32746	4/22/97
B-D4K2	H1	31972	9/13/95
B-D7G2	H1	25719	4/23/97
B-C5F1	H1	23454	4/23/97
B-D5B3	H1	28969	4/3/91
B-D5F7	H1	28677	4/1/91
B-D4H4	H1	28894	11/9/89
B-D4J5	H1	28398	11/8/89
B-D5G4	H1	28875	4/3/91
B-D4A7	H1	26878	11/8/89
B-D4J4	H1	28677	11/9/89
B-D5B7	H1	28984	4/1/91
B-D4D3	H1	30565	11/9/89
B-D4C7	H1	28499	11/8/89
B-D4F7	H1	37861	4/3/91
B-D5F1	H1	28818	11/8/89
B-D4A2	H1	30608	11/9/89
B-D5D7	H1	28306	4/1/91
B-D4F4	H1	28552	11/8/89
B-D5J7	H1	28942	4/2/91
B-D4E7	H1	28647	11/8/89
B-D5A7	H1	29071	4/1/91
B-D4B2	H1	28958	11/9/89
B-D4F5	H1	28607	11/9/89
B-D4A4	H1	27151	11/8/89
B-D4E5	H1	27702	11/9/89
B-D5J5	H1	28532	4/2/91
B-D4C4	H1	28926	11/8/89
B-D5F3	H1	28661	4/3/91
B-D4B7	H1	28988	11/8/89
B-D5E4	H1	28872	4/2/91
B-D5A4	H1	28239	4/2/91
B-D5H1	H1	28976	4/4/91
B-D4F3	H1	29058	11/9/89

B-D4C3	H1	28916	11/9/89
B-D4D7	H1	27009	11/8/89
B-D5J1	H1	28792	4/4/91
B-D4J7	H1	30724	11/9/89
B-D5A5	H1	28748	4/2/91
B-D5G1	H1	37974	4/3/91
B-D5E6	H1	29093	4/2/91
B-D4C2	H1	28753	11/9/89
B-D5E2	H1	29099	4/3/91
B-D5D4	H1	28750	4/2/91
B-D5C3	H1	28411	4/3/91
B-D5B1	H1	30620	11/8/89
B-D5F4	H1	28721	4/2/91
B-D5G2	H1	37810	4/3/91
B-D5G3	H1	28384	4/3/91
B-D5G7	H1	28707	4/2/91
B-D5H6	H1	37884	4/2/91
B-D4J3	H1	28984	11/9/89
B-D4G4	H1	28863	11/9/89
B-D5H5	H1	29061	4/2/91
B-D5F6	H1	28645	4/2/91
B-D5H7	H1	29338	4/2/91
B-D5G6	H1	40418	4/2/91
B-D2C5	H1	28683	11/9/89
B-C5A6	H1	38315	10/31/98
B-D2J4	H1	29754	11/9/89
B-D5B5	H1	32951	4/2/91
B-D5H2	H1	39780	4/3/91
B-D5H4	H1	34274	4/3/91
B-D5J6	H1	40451	4/2/91
B-D4E2	H1	37769	10/3/92
A-B2H1	H1	43795	4/20/00
B-D5J2	H1	40357	4/3/91
B-D2A3	H1	29901	11/9/89
B-D5A2	H1	40492	4/3/91
B-D2E4	H1	27900	11/9/89
A-D2H5	H1	20528	11/9/89
B-D2G6	H1	40683	4/3/91
B-D5D5	H1	40558	4/2/91
B-D2B3	H1	29473	11/9/89
B-D3B5	H1	32839	4/1/91
B-D5B2	H1	40183	4/3/91
B-D5F2	H1	40500	4/3/91
B-D2E6	H1	28228	11/9/89
B-D2H6	H1	40550	4/3/91
B-D5D6	H1	34407	4/2/91
B-D2C6	H1	29819	11/8/89
B-D2F6	H1	30137	11/8/89
B-D5A6	H1	34554	4/2/91
B-D2G7	H1	28305	11/8/89
B-D2J6	H1	29728	11/8/89
A-B2C2	H1	43881	4/22/00
B-D2D6	H1	29841	11/8/89
B-D2H7	H1	28648	11/8/89
B-D5E7	H1	33093	4/1/91

B-D5E5	H1	40173	4/2/91
B-D2J5	H1	28630	11/9/89
A-A1B5	H1	28207	11/9/89
B-D5D3	H1	32850	4/3/91
B-D5C4	H1	32954	4/2/91
B-D5B4	H1	32927	4/2/91
A-D2J6	H1	28111	11/9/89
B-D2J7	H1	29027	11/8/89
A-B2E6	H1	38519	4/21/00
A-B2F1	H1	43742	4/21/00
A-D2H2	H1	27778	11/9/89
B-D5G5	H1	33064	4/2/91
A-B2B4	H1	44518	4/22/00
B-D5E1	H1	40381	4/3/91
B-D2H5	H1	27964	11/9/89
B-D2A5	H1	28835	11/8/89
B-D2F4	H1	29852	11/9/89
B-D5H3	H1	32887	4/3/91
B-D5A3	H1	34471	4/3/91
B-D4B3	H1	45625	10/3/92
A-D2J1	H1	38341	10/4/92
A-A2B4	H1	45484	10/2/92
A-A2B6	H1	45274	10/2/92
B-D5C5	H1	32777	4/2/91
B-D4G2	H1	38356	10/4/92
B-D2A7	H1	38140	10/3/92
A-A2A2	H1	38009	10/2/92
A-A2A1	H1	38230	10/2/92
A-D2J2	H1	38658	10/4/92
B-D4A5	H1	45798	10/3/92
B-D4G1	H1	38579	10/3/92
B-D4H2	H1	42128	10/3/92
B-D4H7	H1	38803	10/4/92
A-A1B4	H1	38381	10/4/92
B-D2A6	H1	38319	10/3/92
B-D4B5	H1	44887	10/2/92
A-D2K1	H1	38605	10/4/92
B-D5C2	H1	32768	4/3/91
B-D5F5	H1	32510	4/2/91
B-D4J2	H1	41673	10/3/92
A-A2B2	H1	41699	10/2/92
B-D4E4	H1	41651	10/3/92
A-A1A2	H1	38174	10/4/92
B-D5C1	H1	32592	4/3/91
B-D5J4	H1	32389	4/3/91
B-D4G3	H1	45246	10/3/92
B-D4G5	H1	38517	10/3/92
B-D5B6	H1	32335	4/2/91
B-D4D5	H1	38479	10/3/92
B-D2C7	H1	38277	10/3/92
B-D5J3	H1	32628	4/3/91
B-D5C6	H1	32574	4/2/91
B-D1D4	H1	45712	10/3/92
B-D2B7	H1	38356	10/3/92
A-A2B3	H1	46030	10/2/92

B-D4H3	H1	37969	10/3/92
A-A1A3	H1	50694	10/3/92
B-D4E3	H1	38173	10/3/92
A-A1B2	H1	38377	10/4/92
A-A1B6	H1	40721	10/2/92
A-A2B5	H1	38597	10/2/92
A-A1A4	H1	49807	10/3/92
B-D1C6	H1	49787	10/3/92
B-D4D2	H1	49730	10/3/92
A-D2K2	H1	40757	10/4/92
B-D2E7	H1	38288	10/3/92
B-D2D7	H1	38514	10/3/92
A-A2B1	H1	38062	10/2/92
B-D4C5	H1	41179	10/3/92
A-D2K6	H1	40777	10/4/92
A-A1B3	H1	38706	10/4/92
B-D5K7	H1	54894	4/2/94
B-D2G3	H1	54323	4/3/94
A-A2A4	H1	37464	10/2/92
B-C5F7	H1	54478	4/2/94
B-D7C6	H1	54823	4/3/94
B-D2B6	H1	37559	10/3/92
B-D4J1	H1	37562	10/3/92
B-D5K6	H1	55129	4/3/94
B-D2J3	H1	41449	4/3/94
B-D4A1	H1	40313	10/3/92
B-D4H6	H1	41524	4/4/94
B-D4B1	H1	40472	10/3/92
B-D4D4	H1	39925	10/3/92
B-D4D1	H1	39991	10/3/92
B-D4E1	H1	39912	10/3/92
B-D4B4	H1	40274	10/3/92
B-D4G7	H1	40282	10/2/92
A-A2A6	H1	40291	10/2/92
B-D4D6	H1	41634	4/4/94
B-D2D1	H1	41448	4/2/94
B-D2D2	H1	41230	4/3/94
B-D4C6	H1	41422	4/4/94
B-D4J6	H1	40775	4/4/94
B-D2A1	H1	41222	4/2/94
B-C4A6	H1	46765	4/2/94
B-D4A6	H1	46486	4/4/94
B-D4B6	H1	46860	4/4/94
B-D4E6	H1	46088	4/4/94
B-D2B1	H1	46264	4/2/94
B-C4B6	H1	46807	4/2/94
B-D2H2	H1	46559	4/3/94
B-D4F6	H1	46741	4/4/94
B-D2C4	H1	49526	4/3/94
B-D1D6	H1	49449	4/4/94
B-D4F1	H1	40002	10/3/92
B-C4A3	H1	45744	4/4/94
B-D1D5	H1	49365	4/3/94
B-D4H1	H1	42402	10/3/92
B-C4A4	H1	45848	4/4/94

B-C5C7	H1	49446	4/4/94
B-C5A7	H1	49392	4/3/94
B-C5B7	H1	49326	4/3/94
B-D4H5	H1	40337	10/3/92
B-D4A3	H1	42765	10/3/92
B-D5E3	H1	42606	10/2/92
A-A2A5	H1	40097	10/2/92
B-D7A6	H1	45854	4/2/94
B-D4F2	H1	42540	10/3/92
B-D2J1	H1	45670	4/2/94
B-D4C1	H1	40605	10/3/92
B-D7D6	H1	49531	4/3/94
B-C5D7	H1	49359	4/4/94
B-D3F3	H1	48454	9/13/95
B-D2G2	H1	41830	4/3/94
B-D2E3	H1	41994	4/3/94
B-C4A5	H1	41984	4/2/94
B-D3C4	H1	48466	9/13/95
B-D3K6	H1	48145	9/14/95
B-D2K7	H1	48971	9/14/95
B-D3E3	H1	48500	9/14/95
B-D5K5	H1	49130	9/12/95
B-D3F7	H1	47976	9/14/95
B-D2D3	H1	42804	4/3/94
B-D2K3	H1	48422	9/13/95
B-D2F3	H1	44145	4/3/94
B-D2H3	H1	43872	4/3/94
B-D3F2	H1	48613	9/14/95
B-D2C3	H1	43578	4/3/94
B-D2E5	H1	44971	4/4/94
B-D2F2	H1	44483	4/3/94
B-D2A2	H1	44178	4/3/94
B-D2B4	H1	44536	4/3/94
B-D2H1	H1	44825	4/2/94
B-D2G4	H1	44711	4/3/94
B-D3C6	H1	48568	9/13/95
B-D2J2	H1	43987	4/3/94
B-D2G1	H1	44059	4/2/94
B-D4K7	H1	49009	9/13/95
B-D2H4	H1	44072	4/3/94
B-D2E1	H1	44724	4/2/94
B-D2D5	H1	43899	4/3/94
B-D3F1	H1	48711	9/14/95
B-D2A4	H1	44847	4/3/94
B-D2E2	H1	45041	4/3/94
B-D7E7	H1	53222	9/14/95
B-D7K1	H1	53046	9/14/95
B-D1D1	H1	53607	9/13/95
B-D7J1	H1	53317	9/14/95
B-D5K3	H1	53170	9/14/95
B-D2K6	H1	53501	9/14/95
B-D3D3	H1	53549	9/14/95
B-D3K4	H1	52829	9/14/95
B-D3G7	H1	50458	9/14/95
B-D3C7	H1	50140	9/13/95

B-D3B6	H1	50975	9/12/95
B-D3H6	H1	50316	9/14/95
B-D3G6	H1	51746	9/14/95
B-D3H7	H1	52108	9/13/95
B-D3G1	H1	52044	9/14/95
B-D2K1	H1	46190	9/14/95
B-D3D6	H1	46588	9/13/95
B-D3D4	H1	52020	9/14/95
B-D3K1	H1	52246	9/13/95
B-D3D7	H1	52240	9/13/95
B-D3G4	H1	52170	9/13/95
B-D2C2	H1	46179	4/3/94
B-D3J5	H1	46727	9/14/95
B-D2C1	H1	45944	4/2/94
B-D3E2	H1	52349	9/14/95
B-D5C7	H1	46743	9/12/95
B-D2F1	H1	45679	4/2/94
B-D2B2	H1	46122	4/3/94
B-D4K1	H1	44648	9/13/95
B-D3F4	H1	44778	9/13/95
B-D3D1	H1	45093	9/13/95
B-D3F5	H1	45247	9/13/95
B-D3K5	H1	44497	9/14/95
B-D3H1	H1	44914	9/13/95
B-D4K4	H1	44604	9/14/95
B-D4K3	H1	44653	9/14/95
B-D5K4	H1	44429	9/12/95
B-D3E5	H1	45018	9/13/95
B-D4K6	H1	44927	9/13/95
B-D3J2	H1	45042	9/13/95
B-D3E4	H1	45363	9/13/95
B-D5K2	H1	44789	9/14/95
B-D4K5	H1	44716	9/12/95
B-D2K2	H1	45304	9/13/95
B-D3K2	H1	46682	9/13/95
B-C4A7	H1	46799	9/14/95
B-D3C5	H1	46656	9/13/95
B-D3G3	H1	44963	9/13/95
B-D2K5	H1	46888	9/14/95
B-D3G2	H1	46505	9/13/95
B-D3H4	H1	45442	9/13/95
B-D3G5	H1	45640	9/13/95
B-D3J3	H1	46412	9/14/95
B-C5E7	H1	45066	9/14/95
B-D3B7	H1	45096	9/12/95
B-D3C1	H1	45325	9/12/95
B-D3D5	H1	46779	9/13/95
B-D3H5	H1	46893	9/14/95
B-D3J4	H1	45263	9/14/95
B-D3H2	H1	45784	9/13/95
B-C5D2	H1	50421	4/23/97
B-D7A1	H1	50215	4/23/97
B-D7E1	H1	49934	4/23/97
B-D7J7	H1	49875	4/21/97
B-D7D1	H1	50257	4/23/97

A-A1A6	H1	52882	4/23/97
B-D7D7	H1	49767	4/20/97
B-D7G1	H1	52245	4/23/97
B-C4A8	H1	53436	4/22/97
B-D7C1	H1	49795	4/23/97
B-C5D8	H1	52221	4/20/97
B-D2G5	H1	49686	4/21/97
B-C5C8	H1	49698	4/20/97
B-C4B5	H1	52185	4/21/97
B-D7G6	H1	50174	4/21/97
B-C4A9	H1	52997	4/23/97
B-D7C7	H1	50145	4/21/97
B-D7F1	H1	50174	4/23/97
B-D7H1	H1	52396	4/23/97
A-A1A5	H1	52787	4/21/97
B-D3D2	H1	46630	9/13/95
B-D3C3	H1	46548	9/13/95
B-D3J7	H1	47133	9/13/95
B-D3J1	H1	46780	9/13/95
B-D3F6	H1	46828	9/13/95
B-D3E7	H1	46573	9/13/95
B-D3E1	H1	46587	9/13/95
B-D3C2	H1	46610	9/13/95
B-C4B10	H1	50813	10/30/98
B-C5D3	H1	51238	11/1/98
B-D7K6	H1	50884	11/1/98
B-C5A3	H1	51371	10/30/98
B-C5B3	H1	51197	10/31/98
B-C4B9	H1	51882	10/31/98
B-C5C3	H1	50866	10/31/98
B-C4A10	H1	51622	10/30/98
B-D7D3	H1	52070	10/31/98
B-D7E4	H1	52740	10/31/98
B-D7D5	H1	52127	11/1/98
B-C5D5	H1	52194	10/31/98
B-C4B8	H1	52443	10/31/98
B-D7G5	H1	52866	10/31/98
B-C5B6	H1	51694	10/31/98
B-D7C3	H1	52619	10/31/98
B-D7G3	H1	51807	10/30/98
B-C5E5	H1	51605	11/1/98
B-D7J4	H1	52445	10/30/98
B-D7F4	H1	51909	11/1/98
B-C1B1	H1	53591	10/30/98
B-C1D1	H1	53958	11/1/98
B-C5F5	H1	53449	10/30/98
B-D7E5	H1	53525	11/1/98
B-C5E3	H1	50170	10/30/98
B-C5D4	H1	50657	11/1/98
B-D7F3	H1	49838	11/1/98
B-C5A4	H1	50170	10/31/98
B-C5B5	H1	49712	10/31/98
B-C5C4	H1	50171	10/31/98
B-D7H6	H1	49948	10/31/98
B-D7K3	H1	50781	10/30/98

B-D7H5	H1	50651	11/1/98
B-C5F6	H1	50712	10/30/98
B-D7J5	H1	50977	11/1/98
B-D7G4	H1	50666	10/30/98
B-C5B8	H1	47450	4/22/97
B-D7D2	H1	46901	4/23/97
B-D7F2	H1	46569	4/23/97
B-D2F7	H1	46953	4/21/97
B-D3B1	H1	46577	4/21/97
B-C5A2	H1	46977	4/23/97
B-C5B1	H1	46973	4/23/97
B-C5A8	H1	46916	4/21/97
B-D7F6	H1	47264	4/21/97
B-D7H2	H1	47562	4/23/97
B-C4B3	H1	47390	4/23/97
B-D3B3	H1	47000	4/21/97
B-D3B2	H1	47802	4/21/97
B-D7H7	H1	48239	4/23/97
B-D7K2	H1	47206	4/23/97
B-D1D3	H1	47689	4/21/97
A-B2K2	H1	50958	4/20/00
A-B2J4	H1	51069	4/23/00
A-B2J5	H1	50987	4/23/00
A-B2K4	H1	51014	4/21/00
A-B2K5	H1	50957	4/21/00
A-B2J3	H1	51082	4/22/00
A-B2J2	H1	51007	4/22/00
A-B2K3	H1	51024	4/20/00
A-B2H6	H1	48825	4/20/00
A-B2A6	H1	51475	4/23/00
A-B2A3	H1	48800	4/23/00
A-B2A4	H1	48847	4/23/00
A-B2G4	H1	51417	4/21/00
A-B2F6	H1	48748	4/21/00
A-B2E2	H1	48804	4/21/00
A-A2K1	H1	51445	4/23/00
A-B2B2	H1	48826	4/22/00
A-B2B1	H1	48855	4/22/00
A-B2J1	H1	51445	4/20/00
A-B2G2	H1	48786	4/20/00
B-C5D6	H1	44405	10/30/98
B-C5C6	H1	44460	11/1/98
B-C5F3	H1	44406	10/30/98
B-D7C5	H1	44477	10/31/98
A-A2K2	H1	52063	4/23/00
A-B2G5	H1	52141	4/21/00
A-A2K3	H1	52055	4/23/00
A-B2K6	H1	51966	4/20/00
A-B2K1	H1	53680	4/20/00
B-D7J6	H1	46512	10/31/98
B-C1C1	H1	53706	4/23/00
B-D7A3	H1	46447	10/30/98
B-C1A1	H1	53669	4/21/00
B-C5C5	H1	46518	10/31/98
A-A2K4	H1	53739	4/23/00

B-C5E4	H1	46473	10/30/98
A-B2B5	H1	53595	4/23/00
A-B2C6	H1	53607	4/22/00
A-B2C1	H1	53580	4/22/00
A-B2B3	H1	53654	4/22/00
B-D7H4	H1	47590	10/30/98
B-D7A4	H1	47922	10/30/98
B-D7B5	H1	47923	10/31/98
B-C5F4	H1	47569	10/30/98
B-D7B3	H1	47597	10/31/98
B-D7F5	H1	47868	10/31/98
B-C5A5	H1	47858	10/30/98
B-D7K4	H1	47531	10/31/98
B-D7A5	H1	47561	10/31/98
B-D7C4	H1	47890	10/30/98
B-C5B4	H1	47918	10/31/98
B-D7D4	H1	47608	10/31/98
B-D7B4	H1	47603	10/30/98
B-D7K5	H1	47954	10/31/98
B-D7H3	H1	47938	10/30/98
B-C5E6	H1	47564	10/30/98
A-B2B6	H1	52968	4/23/00
A-B2A2	H1	52932	4/23/00
A-B2C3	H1	52933	4/22/00
A-B2C4	H1	52982	4/22/00
A-B2A1	H1	43275	4/23/00
A-B2G6	H1	43217	4/21/00
A-B2C5	H1	43165	4/22/00
A-B2J6	H1	43196	4/20/00
A-B2F3	H1	46606	4/21/00
A-B2D5	H1	46597	4/22/00
A-B2H3	H1	46516	4/20/00
A-B2D4	H1	46557	4/22/00
A-B2D3	H1	46903	4/22/00
A-B2F2	H1	46958	4/21/00
A-B2D6	H1	46943	4/22/00
A-B2H2	H1	46897	4/20/00
A-B2F4	H1	47019	4/21/00
A-B2E3	H1	47032	4/21/00
A-B2H5	H1	46967	4/20/00
A-B2G1	H1	46987	4/20/00
A-B2E1	H1	46609	4/21/00
A-B2F5	H1	46632	4/21/00
A-B2G3	H1	46590	4/21/00
A-B2H4	H1	46556	4/20/00
A-B2D2	H1	45868	4/22/00
A-B2E4	H1	45819	4/21/00
A-B2D1	H1	45881	4/22/00
A-B2E5	H1	45930	4/21/00
A-D2A5	H1	25628	4/21/00
A-D2D4	H1	25636	4/21/00
A-D2D5	H1	25601	4/21/00
A-D2G5	H1	25588	4/20/00
A-D1E7	B1	23229	12/28/82
A-D1C10	B1	14119	6/17/80

A-D1F1	B1	23250	12/30/82
A-C1F1	B1	23194	12/28/82
A-C1J1	B1	23295	12/30/82
A-D1E10	B1	18017	12/27/82
A-D1A1	B1	23327	12/30/82
A-E1F3	B1	18021	12/29/82
A-E1F8	B1	18042	12/29/82
A-D1E11	B1	16918	12/27/82
A-C1G1	B1	23301	12/28/82
A-D1D8	B1	19763	12/27/82
A-E1H9	B1	24022	12/29/82
A-E1F6	B1	24974	12/29/82
A-D1H1	B1	22657	1/3/83
A-E1K5	B1	22521	12/28/82
A-E1K2	B1	23355	12/29/82
A-D1D9	B1	19242	12/27/82
A-D1H10	B1	21995	1/3/83
A-C1D1	B1	23216	12/28/82
A-E1G7	B1	19304	12/28/82
A-E1L1	B1	22373	12/29/82
A-E1L3	B1	22655	1/3/83
A-E1H10	B1	24166	12/28/82
A-E1L6	B1	23368	12/28/82
A-E1G3	B1	23987	12/30/82
A-E1G4	B1	24103	12/28/82
A-E1H11	B1	24019	12/29/82
A-E1G5	B1	24015	12/30/82
A-D1C6	B1	23404	12/28/82
A-D1C7	B1	22522	12/28/82
A-D1G5	B1	22057	12/29/82
A-C1E1	B1	22687	1/3/83
B-A5J6	B1	22012	1/3/83
A-E1K7	B1	22196	12/30/82
A-C1B1	B1	22602	12/28/82
B-B2E7	B1	18737	12/27/82
A-E1L4	B1	22400	12/30/82
A-E1L9	B1	22166	12/30/82
A-D1G6	B1	22074	12/29/82
A-D1G1	B1	23253	12/28/82
A-E1L5	B1	22168	12/30/82
A-D1D1	B1	23010	12/28/82
A-E1F5	B1	24785	12/30/82
A-E1G2	B1	19305	12/29/82
A-D1G7	B1	22040	1/3/83
A-D1C1	B1	23314	12/30/82
A-D1G11	B1	22038	12/29/82
A-E1J3	B1	23885	12/28/82
A-E1F1	B1	18741	12/28/82
A-D1F5	B1	22015	12/29/82
A-E1L10	B1	22141	12/30/82
A-E1F9	B1	18733	12/28/82
A-E1J9	B1	23830	12/28/82
B-A5J7	B1	22986	12/28/82
A-E1L11	B1	23357	12/29/82
A-E1K3	B1	22026	1/3/83

A-E1J2	B1	23835	12/28/82
A-E1J4	B1	23394	12/29/82
A-E1L7	B1	23379	12/28/82
A-C1L1	B1	22990	12/28/82
A-E1L8	B1	23368	12/29/82
A-E1L2	B1	22506	12/28/82
A-E1F7	B1	24785	12/30/82
A-E1F10	B1	24229	12/28/82
A-D1E1	B1	22693	1/3/83
A-E1K4	B1	22435	12/29/82
A-C1K1	B1	22987	12/28/82
A-E1F11	B1	24934	12/29/82
A-C1C1	B1	23270	12/30/82
A-E1K6	B1	22398	12/30/82
A-C1H1	B1	23351	12/29/82
A-E1J8	B1	23920	12/28/82
A-D1B1	B1	23343	12/30/82
A-E1J7	B1	23394	12/29/82
A-E1E1	B1	24318	12/28/82
A-E1K1	B1	22364	12/29/82
A-D1C2	B1	21056	1/3/83
B-A8J2	B1	17572	6/14/80
A-E1E4	B1	18939	12/29/82
B-A8H11	B1	14734	6/4/80
A-E1G6	B1	19127	12/29/82
B-A8K4	B1	17189	6/16/80
A-E1E3	B1	18512	12/26/82
B-A7J7	B1	15132	6/18/80
A-E1J11	B1	19820	12/26/82
B-A8K9	B1	13882	6/16/80
A-E1J6	B1	19910	12/26/82
B-B5D3	B1	14085	6/4/80
A-D1E2	B1	21619	12/30/82
B-B5D4	B1	14078	6/4/80
B-A8H10	B1	16831	6/16/80
B-B5D5	B1	14361	6/18/80
A-D1G2	B1	16022	6/16/80
B-A8F6	B1	14056	6/4/80
A-C1C2	B1	21515	12/29/82
B-A8F3	B1	15890	6/16/80
B-A8C3	B1	15312	6/13/80
B-B6L7	B1	15867	6/16/80
B-A1A1	B1	18758	12/27/82
A-E1F2	B1	18749	12/26/82
B-A8E4	B1	17247	6/17/80
B-A8E6	B1	17779	6/16/80
B-B5D6	B1	17722	6/17/80
A-E1K11	B1	19842	12/26/82
B-B5D7	B1	17510	6/18/80
A-D1F2	B1	21614	12/29/82
B-A8G1	B1	17504	6/16/80
A-C1K2	B1	20635	1/3/83
B-B5D8	B1	14694	6/4/80
B-A8G10	B1	15002	6/17/80
B-A8A7	B1	17923	6/16/80

A-C1B2	B1	21446	12/30/82
B-A1E1	B1	15660	6/17/80
B-A8K6	B1	12573	3/1/79
A-C1L2	B1	21785	12/29/82
B-A8D2	B1	18843	6/13/80
B-A8C7	B1	12757	2/27/79
A-E1E5	B1	18903	12/29/82
B-B5E2	B1	15994	6/18/80
A-E1G11	B1	17294	12/29/82
B-B5E3	B1	16795	6/14/80
A-C1F2	B1	21721	12/30/82
B-A1B1	B1	17298	12/27/82
A-E1K9	B1	20122	12/26/82
B-B5E4	B1	17903	6/18/80
B-A8D1	B1	12539	2/27/79
B-A8G2	B1	17819	6/16/80
B-A8B2	B1	17963	6/16/80
B-B5E5	B1	17161	6/18/80
B-B5E6	B1	17971	6/13/80
A-D1B2	B1	22030	12/30/82
B-B5E8	B1	16778	6/13/80
B-A8D5	B1	17369	6/16/80
B-A7E8	B1	16815	6/17/80
B-A8G6	B1	12043	3/1/79
A-C1H2	B1	21703	12/29/82
B-A8D11	B1	18594	6/17/80
B-B5E9	B1	16946	6/19/80
B-A8J1	B1	14058	6/4/80
B-A7F11	B1	15320	6/16/80
B-A7F9	B1	14094	6/4/80
B-A8B1	B1	16830	6/16/80
B-A8F8	B1	14028	6/4/80
B-A8F11	B1	15837	6/16/80
B-B2E6	B1	18637	12/27/82
B-B5F1	B1	17536	6/18/80
B-B2E2	B1	18310	12/27/82
B-B5B4	B1	16722	6/18/80
A-C1A2	B1	15068	6/14/80
A-D1A2	B1	22160	12/30/82
B-B5B5	B1	14220	6/14/80
A-E1H7	B1	19098	12/26/82
B-B2E8	B1	18755	12/27/82
A-C1G2	B1	22364	12/29/82
A-C1J2	B1	20965	12/30/82
A-D1D2	B1	20449	1/3/83
B-A8A1	B1	16175	6/16/80
B-A8D3	B1	17183	6/13/80
A-C1D2	B1	18023	6/13/80
B-B5B6	B1	15301	6/17/80
B-B5B7	B1	16833	6/18/80
B-A8F7	B1	17032	6/16/80
B-B5B8	B1	17409	6/18/80
B-A8F10	B1	16922	6/16/80
B-B5B9	B1	15228	6/19/80
B-A8J10	B1	15239	6/16/80

B-A8C4	B1	17340	6/15/80
B-B5C1	B1	15817	6/18/80
B-A7K9	B1	12020	2/28/79
B-A5J9	B1	19922	12/27/82
B-A5K1	B1	22477	1/3/83
B-B5C2	B1	14698	6/13/80
A-C1E2	B1	21078	1/3/83
A-D1G3	B1	22044	1/3/83
B-A8E11	B1	16102	6/17/80
B-A8J5	B1	16126	6/15/80
B-B5C3	B1	15863	6/18/80
A-E1F4	B1	17205	12/10/82
A-E1H6	B1	19619	12/26/82
B-B5C4	B1	17027	6/17/80
B-A8H9	B1	18353	6/15/80
B-A8G11	B1	15917	6/17/80
A-C1A4	B1	22549	12/31/82
B-A8K5	B1	18504	6/16/80
B-B5B3	B1	17681	6/5/85
B-A8K1	B1	15378	6/16/80
B-A8F9	B1	14676	6/4/80
B-A8H7	B1	16753	6/15/80
A-C1B4	B1	19092	6/15/80
A-C1C4	B1	21549	12/31/82
B-B2D9	B1	17184	12/27/82
B-A8B4	B1	17165	6/16/80
B-A8D10	B1	17378	6/16/80
B-B5C5	B1	15418	6/18/80
B-A8J11	B1	18258	6/16/80
B-A8J3	B1	16139	6/13/80
B-A8D7	B1	18995	6/16/80
A-C1D4	B1	22024	12/30/82
B-B2F1	B1	19043	12/27/82
A-C1E4	B1	21349	1/3/83
A-C1F4	B1	21068	12/29/82
A-E1K8	B1	19775	12/26/82
B-A8J8	B1	17568	6/16/80
B-A8J7	B1	17121	6/16/80
B-B5C6	B1	17778	6/13/80
B-B5C7	B1	18927	6/14/80
B-A7G10	B1	16546	6/5/85
B-A5K2	B1	19812	12/26/82
A-E1G1	B1	19119	12/26/82
B-B5C8	B1	17573	6/18/80
A-C1G4	B1	12854	4/24/79
B-A8K10	B1	15463	6/15/80
B-A8B3	B1	12073	2/27/79
A-C1H4	B1	20251	12/26/82
B-A5K3	B1	21297	1/3/83
B-A8B8	B1	12075	2/28/79
A-E1J1	B1	19883	12/26/82
A-E1J10	B1	19866	12/26/82
B-B5C9	B1	17003	6/18/80
B-B5D1	B1	18916	6/18/80
B-B5D2	B1	18308	6/18/80

A-C1J4	B1	21478	12/30/82
A-C1K4	B1	21832	1/3/83
B-A8E5	B1	17045	6/17/80
B-B5D9	B1	18164	6/18/80
A-C1L4	B1	22188	12/31/82
B-B2E11	B1	18946	12/27/82
A-D1A4	B1	22310	12/29/82
B-A8E2	B1	15517	6/17/80
A-E1H5	B1	19361	12/27/82
A-E1E6	B1	18371	12/26/82
B-A5K4	B1	19854	12/27/82
A-D1B4	B1	22085	12/31/82
A-D1C4	B1	21549	12/30/82
B-B5E7	B1	17546	6/13/80
A-D1D4	B1	22187	1/3/83
A-D1E4	B1	21514	12/31/82
B-A7D11	B1	17045	6/13/80
B-A7H11	B1	16348	6/17/80
B-A8E10	B1	17602	6/17/80
A-E1E2	B1	18291	12/27/82
B-A5K5	B1	19991	12/26/82
B-B2E3	B1	18408	12/26/82
B-A5K6	B1	20117	12/26/82
B-A5K7	B1	20033	12/26/82
B-A5K8	B1	19668	12/27/82
B-B2F2	B1	19141	12/27/82
A-D1F4	B1	21587	12/31/82
B-A5K9	B1	21885	1/3/83
A-D1G4	B1	21236	12/29/82
B-A7G8	B1	15545	6/18/80
A-E1G8	B1	19686	12/26/82
B-A8G5	B1	16193	6/17/80
B-A8A4	B1	12464	2/28/79
B-A8F1	B1	17551	6/17/80
B-B5H1	B1	19026	6/18/80
B-B5H2	B1	17245	6/18/80
B-A7K7	B1	12059	3/1/79
B-B5H3	B1	19068	6/18/80
A-E1G9	B1	19666	12/26/82
B-B5H4	B1	17994	6/18/80
A-E1J5	B1	19977	12/26/82
A-C1A3	B1	21781	12/30/82
B-B5H5	B1	18250	6/18/80
A-C1B3	B1	22297	12/29/82
B-A8C6	B1	12678	2/24/79
A-C1C3	B1	21040	1/3/83
B-A8G8	B1	18989	6/17/80
B-A5L1	B1	19720	12/27/82
B-A8E1	B1	17937	6/17/80
A-C1D3	B1	20422	1/3/83
B-B5H6	B1	17901	6/13/80
A-C1E3	B1	21209	12/30/82
A-C1F3	B1	20128	12/26/82
B-A8E3	B1	16284	6/8/85
A-E1K10	B1	20068	12/26/82

A-E1H3	B1	19724	12/26/82
A-C1G3	B1	21853	1/3/83
B-A7D9	B1	12191	3/1/79
A-E1H2	B1	20106	12/26/82
A-C1H3	B1	21924	12/30/82
A-C1J3	B1	20492	12/26/82
B-B2E4	B1	18542	12/27/82
B-B2F3	B1	19183	12/27/82
B-A5L2	B1	20144	12/26/82
B-B2E5	B1	18565	12/27/82
B-B5H7	B1	16317	6/17/80
B-A8H1	B1	15276	6/15/80
B-B5H8	B1	18033	6/13/80
B-B5H9	B1	17736	6/18/80
B-B5J1	B1	15314	6/18/80
B-A8H3	B1	16076	6/15/80
B-A8A3	B1	17136	6/16/80
B-B5J2	B1	17169	6/18/80
B-A8F2	B1	17783	6/16/80
B-A8C1	B1	16663	6/16/80
A-C1K3	B1	16987	6/15/80
B-A8E8	B1	17669	6/16/80
B-B5J3	B1	16881	6/19/80
A-C1L3	B1	21414	12/30/82
A-D1A3	B1	22323	12/29/82
B-B5J4	B1	15405	6/13/80
B-B2D11	B1	19354	12/27/82
B-B5J5	B1	17102	6/18/80
B-B5J6	B1	17423	6/18/80
B-A8H5	B1	16788	6/15/80
B-A8B6	B1	12756	2/28/79
A-E1G10	B1	19195	12/29/82
A-D1B3	B1	22297	12/30/82
A-D1C3	B1	21896	12/29/82
B-B5J7	B1	17775	6/18/80
B-A7G9	B1	12499	3/1/79
A-D1D3	B1	22172	12/30/82
A-D1E3	B1	22631	12/30/82
A-D1F3	B1	21865	12/30/82
B-A5L3	B1	20027	12/27/82
B-B5J8	B1	18563	6/13/80
B-B5F2	B1	18485	6/18/80
B-A7E11	B1	12634	2/27/79
A-E1H4	B1	19096	12/26/82
B-B5F3	B1	16736	6/17/80
A-D1L6	B1	22653	1/3/83
B-A8E9	B1	16978	6/17/80
B-A8L1	B1	15096	6/11/85
B-A8D8	B1	12493	2/23/79
A-D1F9	B1	12890	2/27/79
B-A8A5	B1	12432	2/26/79
B-A8H6	B1	15767	6/15/80
A-D1L7	B1	21539	1/3/83
B-B5F4	B1	16570	6/13/80
B-A8A8	B1	18489	6/16/80

B-A8K8	B1	16201	6/5/85
B-B5F5	B1	17147	6/16/80
A-D1L8	B1	21076	1/3/83
A-E1H1	B1	19983	12/26/82
B-A8K11	B1	16121	6/15/80
B-A8K2	B1	18562	6/15/80
B-A8A2	B1	17576	6/16/80
B-B5F6	B1	16895	6/18/80
B-A8C8	B1	16627	6/13/80
B-A8G3	B1	15280	6/16/80
B-B5F7	B1	15196	6/17/80
B-A8J9	B1	17131	6/16/80
B-B5F8	B1	16846	6/13/80
B-A8H8	B1	16301	6/15/80
B-B5F9	B1	16659	6/18/80
A-D1L9	B1	22301	12/31/82
B-B5G1	B1	16311	6/18/80
B-A7F7	B1	17136	6/18/80
B-A8C5	B1	16828	6/13/80
B-A8K7	B1	16413	6/14/80
B-B5G2	B1	18313	6/16/80
B-A8C2	B1	19046	6/13/80
B-A5L4	B1	19924	12/27/82
A-D1L10	B1	22203	1/3/83
A-D1L11	B1	21684	1/3/83
A-D1K1	B1	21785	1/3/83
B-A8B5	B1	16318	6/16/80
A-D1K2	B1	21348	1/3/83
B-A8D6	B1	12387	2/27/79
A-D1K3	B1	21434	12/30/82
B-A8H4	B1	16456	6/15/80
A-D1K4	B1	21187	12/29/82
B-B5G3	B1	16970	6/18/80
B-B5G4	B1	15185	6/14/80
B-B5G5	B1	19007	6/18/80
A-D1K5	B1	21797	12/30/82
B-A8D9	B1	18208	6/17/80
B-B5G6	B1	17120	6/14/80
B-A8G7	B1	12835	2/28/79
B-B5G7	B1	16568	6/18/80
B-B5G8	B1	15844	6/18/80
A-D1K6	B1	21038	1/3/83
B-B5G9	B1	15987	6/18/80
A-D1K7	B1	20532	1/3/83
B-A8F4	B1	17809	6/16/80
B-A7J10	B1	18423	6/17/80
B-B2E1	B1	18291	12/27/82
B-B5B2	B1	19735	12/26/82
B-A7D8	B1	17046	6/18/80
A-D1K8	B1	20553	1/3/83
B-A7K10	B1	15611	6/18/80
A-D1K9	B1	20631	1/3/83
B-A8H2	B1	15870	6/15/80
B-A7K8	B1	18219	6/16/80
B-A8G9	B1	12820	2/28/79

B-A8B7	B1	14934	6/16/80
B-B2E10	B1	18798	12/27/82
A-D1K10	B1	13803	6/17/80
B-A7E9	B1	19388	12/27/82
B-A8G4	B1	12375	2/28/79
B-A8E7	B1	18173	6/16/80
A-D1K11	B1	22216	1/3/83
A-E1H8	B1	19086	12/26/82
A-D1J1	B1	21486	1/3/83
A-D1J2	B1	21177	1/3/83
B-A7G7	B1	12823	2/27/79
A-D1J3	B1	20919	1/3/83
A-D1J4	B1	21643	12/31/82
B-A7J9	B1	17450	6/18/80
B-A7E7	B1	17546	6/13/80
B-A7G11	B1	16281	6/18/80
A-D1J5	B1	21096	1/3/83
A-D1J6	B1	21368	12/30/82
A-D1J7	B1	21521	1/3/83
B-A7F8	B1	19993	12/26/82
A-D1J8	B1	20828	12/30/82
B-A7H10	B1	16133	6/18/80
B-A7E10	B1	16093	6/18/80
B-A7D10	B1	16225	6/13/80
B-A8K3	B1	17744	6/16/80
A-D1J9	B1	22259	12/29/82
B-A8D4	B1	19043	6/16/80
A-D1J10	B1	21926	12/29/82
A-D1J11	B1	17705	6/4/85
A-D1H2	B1	21773	1/3/83
A-D1H3	B1	21907	1/3/83
B-A7H9	B1	14785	6/18/80
B-A7J11	B1	17336	6/17/80
B-A7H8	B1	18210	6/19/80
B-A7J8	B1	16214	6/18/80
A-D1H4	B1	20355	12/26/82
B-A8J6	B1	15412	6/8/85
A-D1H5	B1	20759	1/3/83
A-D1H6	B1	21565	12/30/82
A-D1H7	B1	21919	1/3/83
A-D1H8	B1	21758	12/30/82
A-D1H9	B1	20922	12/30/82
A-D1F10	B1	22419	1/3/83
B-A8F5	B1	18176	6/16/80
B-A8J4	B1	18165	6/16/80
A-D1F11	B1	22415	1/3/83
B-A5J8	B1	20787	1/3/83
A-D1E5	B1	21204	12/29/82
B-A7H7	B1	17620	8/23/79
B-A7F10	B1	18424	6/18/80
B-A7K11	B1	17936	6/18/80
A-D1E6	B1	22463	1/3/83
B-A8A6	B1	12800	2/26/79
B-B4B1	B2	23476	4/1/84
B-B4B4	B2	26297	4/7/84

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B-B4B5	B2	24154	4/2/84
B-B4B8	B2	26387	4/4/84
B-B4B2	B2	23533	4/1/84
A-D1B6	B2	24116	4/2/84
B-B4B9	B2	25706	4/5/84
B-B4B3	B2	22750	7/17/84
A-C1L6	B2	24098	4/3/84
B-B4B6	B2	23367	4/1/84
B-B4B7	B2	23427	4/1/84
B-B3L6	B2	23002	4/1/84
B-B3L7	B2	23927	4/1/84
B-B3L8	B2	23956	4/1/84
B-B4A8	B2	25505	4/6/84
B-B4A7	B2	27971	4/5/86
B-B3L11	B2	23437	4/1/84
B-B4A1	B2	22718	4/1/84
B-B4A5	B2	23837	4/3/84
B-B3L9	B2	27507	4/5/86
B-B4E2	B2	25101	4/6/84
B-A7H6	B2	25495	4/5/84
B-B4E3	B2	22415	4/4/84
B-B4E4	B2	26810	4/5/86
B-B4E7	B2	22653	4/1/84
B-B4E8	B2	25086	4/6/84
B-A7L4	B2	27110	4/5/86
B-B4F1	B2	26780	4/4/84
B-B4F2	B2	23414	4/1/84
B-B4F5	B2	27461	4/5/86
B-B4C7	B2	23534	4/1/84
B-B4C8	B2	25573	4/4/84
B-B7A2	B2	23896	4/2/84
B-B4C9	B2	25600	4/6/84
B-B4D1	B2	26862	4/6/84
B-B4D2	B2	23029	4/1/84
B-B4D6	B2	23916	4/1/84
B-B4D7	B2	24037	4/1/84
B-B4D10	B2	23457	4/1/84
B-A7J5	B2	22322	4/2/84
B-B7B1	B2	22346	4/3/84
B-A7K1	B2	23547	4/1/84
B-B7A5	B2	24172	4/3/84
B-A7A1	B2	25723	4/5/84
A-C1J10	B2	24180	4/5/86
B-A7L5	B2	25653	4/2/84
B-B7F2	B2	23620	4/1/84
B-A7A3	B2	26374	4/5/84
B-A7G6	B2	25547	4/3/84
B-B7F3	B2	23482	4/1/84
B-A7F5	B2	25555	4/2/84
B-B7F4	B2	24398	4/1/84
B-B7A6	B2	22764	4/4/84
B-A7L1	B2	27167	4/6/84
B-A7L6	B2	22645	4/1/84
B-B7A3	B2	22626	4/1/84
B-A7K4	B2	22920	4/1/84

B-B7D2	B2	23145	4/1/84
B-B7B4	B2	24335	4/1/84
B-A7B2	B2	26710	4/6/84
B-B7A1	B2	25516	4/6/84
B-A7H3	B2	23832	4/1/84
B-A7G1	B2	25512	4/5/84
B-B7D3	B2	23464	4/1/84
B-A7H1	B2	22314	4/1/84
B-A7J1	B2	26937	4/5/84
B-A7B4	B2	27146	4/6/84
B-A7B3	B2	25993	4/6/84
B-A7K2	B2	24981	4/6/84
B-B7F6	B2	22471	4/1/84
B-B7C6	B2	26790	4/4/84
B-B7B2	B2	24051	4/4/84
B-A7G3	B2	24941	4/5/84
B-B7D4	B2	27510	4/6/84
B-B7A4	B2	23949	4/3/84
B-B7E6	B2	23930	4/1/84
B-B7F5	B2	23527	4/1/84
B-B7E1	B2	23437	4/1/84
B-A7G2	B2	25519	4/5/84
B-B7C2	B2	23928	4/1/84
B-B7D5	B2	23034	4/1/84
B-A7G5	B2	23471	4/1/84
B-B7B3	B2	22849	4/3/84
B-B7C4	B2	23560	4/1/84
B-A7H2	B2	22863	4/1/84
B-B7D1	B2	25751	4/4/84
A-C1L10	B2	24068	4/1/84
B-A7H4	B2	24105	4/2/84
B-A7B1	B2	26371	4/4/84
B-A7L2	B2	23533	4/1/84
B-A7H5	B2	22945	4/2/84
B-A7A2	B2	26152	4/5/84
B-A7B5	B2	26439	4/6/84
B-B7E5	B2	25101	4/1/84
B-A7C2	B2	26668	4/6/84
B-B7C3	B2	25014	4/1/84
B-B7C5	B2	25552	4/1/84
B-A7F6	B2	26449	4/5/84
B-A7C3	B2	26612	4/1/84
B-B7E3	B2	24793	4/1/84
B-B7E2	B2	24299	4/1/84
B-B7B6	B2	25675	4/1/84
B-B7D6	B2	25060	4/1/84
B-A7B6	B2	26633	4/6/84
B-A7K3	B2	26464	4/6/84
B-B7B5	B2	25394	4/1/84
B-A7G4	B2	25336	4/1/84
B-A7A4	B2	26388	4/5/84
B-A7K6	B2	22715	4/2/84
B-A7J2	B2	26075	4/5/84
B-B7C1	B2	25272	4/1/84
B-A7C5	B2	25330	4/1/84

B-A7C1	B2	26420	4/6/84
B-B7E4	B2	25262	4/1/84
B-A7J4	B2	24635	4/1/84
B-A7J6	B2	22976	4/1/84
B-A7K5	B2	22957	4/1/84
B-A7A6	B2	26505	4/4/84
B-B7F1	B2	28148	4/1/84
B-A7J3	B2	25112	4/1/84
B-A7A5	B2	26649	4/7/84
B-A7L3	B2	26471	7/17/84
B-A7C4	B2	24289	4/1/84
B-B6D6	B2	21634	4/5/84
A-D1A7	B2	22281	1/28/86
A-D1B9	B2	22378	1/27/86
B-B6F7	B2	20866	4/6/84
B-B6G7	B2	22947	4/5/84
A-C1L5	B2	22109	1/30/86
A-D1B11	B2	22123	1/28/86
B-B6F6	B2	20893	4/5/84
B-B4B11	B2	22503	1/27/86
A-D1A9	B2	22496	1/28/86
B-B4B10	B2	25478	2/2/86
B-B4C1	B2	25099	2/1/86
B-A6L10	B2	23494	4/6/84
B-B4C4	B2	25082	2/1/86
B-B4C2	B2	25481	2/1/86
A-D1A8	B2	23303	1/28/86
A-D1B10	B2	23285	1/30/86
B-B6E5	B2	22217	4/4/84
B-B6A9	B2	22726	4/6/84
A-D1B5	B2	20742	1/27/86
A-C1L9	B2	20730	1/27/86
A-C1J5	B2	23333	1/28/86
B-B4C5	B2	23292	1/27/86
A-C1J9	B2	22058	1/28/86
A-C1K6	B2	22424	1/30/86
A-C1K7	B2	22465	1/28/86
A-C1K8	B2	22060	1/30/86
A-C1K11	B2	22059	1/30/86
A-C1J6	B2	22466	1/28/86
B-A1F2	B2	22426	1/30/86
B-A1F3	B2	22643	1/28/86
B-B4C6	B2	23283	1/27/86
B-A1F6	B2	23335	1/28/86
B-A1F7	B2	20729	1/28/86
B-A1F8	B2	20739	1/28/86
B-B6E7	B2	22676	4/7/84
B-B6E6	B2	22677	4/7/84
B-B6G11	B2	22197	4/6/84
B-B6D10	B2	22218	4/4/84
B-A1F11	B2	23295	1/30/86
B-B2F8	B2	23295	1/28/86
B-B4C3	B2	25480	2/2/86
B-B3L10	B2	25078	2/2/86
B-A6L11	B2	23481	6/4/84

B-B6A7	B2	23473	4/6/84
B-B4A2	B2	25098	2/2/86
B-A1G2	B2	22491	1/28/86
B-B4A6	B2	22505	1/27/86
B-B6D11	B2	20853	4/6/84
B-B6K5	B2	22956	4/6/84
B-A1G5	B2	22118	1/28/86
B-A1G6	B2	22108	1/30/86
B-B6D9	B2	22941	4/4/84
B-B6D5	B2	20839	4/5/84
B-A1D6	B2	22375	1/27/86
B-A1D7	B2	22380	1/27/86
B-B6B9	B2	21608	7/20/84
B-B6B5	B2	21641	7/20/84
B-B4A9	B2	22026	1/31/86
B-A1D10	B2	25725	1/30/86
B-A1D11	B2	25741	1/28/86
B-B2F6	B2	22019	1/30/86
B-A1E4	B2	25267	1/27/86
B-B4A3	B2	25092	1/27/86
B-B6C9	B2	23405	4/5/84
B-B6C10	B2	23409	4/5/84
B-A1E5	B2	25088	1/28/86
B-A1E6	B2	25273	1/28/86
B-A1E9	B2	24226	2/2/86
B-B6C6	B2	23992	4/5/84
B-B6D7	B2	23945	5/4/84
B-A1E10	B2	24224	2/2/86
B-A1G9	B2	25838	1/28/86
B-B6B6	B2	22777	7/14/84
B-A1G10	B2	24838	1/28/86
B-B6H10	B2	23166	4/6/84
B-B6C8	B2	23137	4/5/84
B-B2F9	B2	24833	1/27/86
B-A1H2	B2	24318	1/27/86
B-A1H3	B2	27408	1/28/86
B-A1H6	B2	27396	1/30/86
B-B2G1	B2	24310	1/28/86
B-A1H8	B2	25483	1/28/86
B-B4A4	B2	25491	1/27/86
B-B4A10	B2	24651	1/27/86
B-B6F10	B2	24467	4/6/84
B-B6G6	B2	24496	4/5/84
B-A1H11	B2	24045	1/28/86
B-B6A6	B2	23677	4/6/84
B-B6C5	B2	23646	4/6/84
B-B2F10	B2	24638	1/27/86
B-A1J3	B2	24640	1/27/86
B-A6L9	B2	23687	4/6/84
B-B6A10	B2	23693	4/6/84
B-A1J4	B2	24634	1/28/86
B-A1J7	B2	24047	1/28/86
B-B6B8	B2	24404	7/20/84
B-B4A11	B2	24651	1/27/86
B-B6F8	B2	25487	1/27/86

B-A1J8	B2	25475	1/28/86
B-A1J9	B2	24318	1/28/86
B-B1A4	B2	27393	1/30/86
B-B2F11	B2	27376	1/28/86
B-A1K2	B2	24322	1/27/86
B-A1K4	B2	24830	1/27/86
B-B6J6	B2	23097	4/6/84
B-B6C7	B2	23099	4/5/84
B-A1K7	B2	24829	1/28/86
B-B6A5	B2	22732	4/6/84
B-A1K8	B2	25816	1/28/86
B-B6B10	B2	23862	7/20/84
B-B6A11	B2	23900	7/20/84
B-B6C11	B2	23835	4/5/84
B-B6B11	B2	23332	4/6/84
B-B6B7	B2	23338	4/6/84
B-B6F11	B2	25092	1/27/86
B-B6D8	B2	22017	1/31/86
A-D1E8	B1	28875	6/13/85
A-D1E9	B1	29555	6/13/85
A-D1D5	B1	29766	6/13/85
B-B1G9	B1	27024	3/1/87
A-D1D6	B1	29921	6/13/85
A-D1D7	B1	29926	6/13/85
A-D1D10	B1	25199	6/4/85
A-D1D11	B1	24771	6/4/85
B-B1J4	B1	28597	3/1/87
B-B1E3	B1	27283	3/1/87
A-D1C5	B1	27728	6/13/85
B-B1E5	B1	27340	3/1/87
B-B1J5	B1	28495	3/1/87
B-B1E4	B1	27334	3/1/87
A-D1C8	B1	28888	6/13/85
A-D1C9	B1	28896	6/13/85
B-B1E6	B1	27345	3/1/87
B-B1J6	B1	28497	3/1/87
B-B1E7	B1	27350	3/1/87
B-A2E7	B1	27716	6/13/85
B-A2E8	B1	27722	6/13/85
B-B1E2	B1	27263	3/1/87
B-B1H1	B1	28592	3/1/87
B-A1L2	B1	24762	6/5/85
B-A1L4	B1	25194	6/4/85
B-A1L3	B1	25172	6/4/85
A-D1H11	B1	24757	6/4/85
B-B1D2	B1	29932	6/13/85
B-B1D1	B1	29919	6/13/85
B-A8A11	B1	27019	3/1/87
B-B1C11	B1	29754	6/13/85
B-B1F7	B1	29754	6/13/85
B-D8J11	B1	29549	6/13/85
B-A1F1	B1	29560	6/12/85
B-B1C3	B1	28857	6/13/85
B-B1C4	B1	28862	6/13/85
B-B1B3	B1	28247	6/13/85

B-B1B4	B1	28258	6/13/85
B-A7C10	B1	28701	2/28/87
B-A2F9	B1	27869	6/13/85
B-A2F8	B1	27857	6/12/85
B-A7A10	B1	28706	3/1/87
B-A2E6	B1	27626	6/12/85
B-B1B7	B1	28566	6/11/85
B-B1B1	B1	28224	6/12/85
B-A2C10	B1	27111	6/11/85
B-A2C9	B1	27109	6/11/85
B-B1A11	B1	28219	6/12/85
B-B1B9	B1	28568	6/11/85
B-A2E5	B1	27616	6/12/85
B-A2A3	B1	26215	6/12/85
B-B1J2	B1	28432	3/1/87
B-A2E10	B1	27810	6/12/85
B-A2D3	B1	27197	6/11/85
B-A2D2	B1	27195	6/11/85
B-A2E9	B1	27810	6/11/85
B-A1K11	B1	26197	6/11/85
B-A2D4	B1	27212	6/13/85
B-B1F10	B1	30211	6/13/85
B-B1C2	B1	28655	6/11/85
B-A2D9	B1	26536	6/13/85
B-B1D8	B1	26540	6/13/85
B-B1C1	B1	28645	6/11/85
B-B1D3	B1	30207	8/14/85
B-A2D5	B1	27214	6/13/85
B-B1D9	B1	30388	8/14/85
B-A2D8	B1	27391	6/13/85
B-B1D10	B1	30392	8/14/85
B-A2C1	B1	26935	6/11/85
B-A2B3	B1	26812	6/11/85
B-B1C7	B1	28056	6/13/85
B-B1C6	B1	28053	6/12/85
B-A2B5	B1	26815	6/11/85
B-A2C4	B1	26942	6/11/85
B-B1J11	B1	27710	3/1/87
B-A2D11	B1	27479	6/11/85
B-B1K6	B1	27872	3/2/87
B-A7A11	B1	27872	3/1/87
B-A2E1	B1	27483	6/11/85
B-A8C9	B1	27901	6/13/85
B-A2C7	B1	27054	6/11/85
B-A2C5	B1	27043	6/11/85
B-B1A10	B1	27914	6/12/85
B-B1D6	B1	30361	8/14/85
B-B1K11	B1	27966	2/28/87
B-B1D7	B1	30365	8/14/85
B-A2B1	B1	26681	6/11/85
B-A2F4	B1	27836	6/11/85
B-A1L11	B1	26013	6/11/85
B-A1L5	B1	25693	6/5/85
B-A1L7	B1	25695	6/6/85
B-A1K10	B1	26016	6/11/85

B-A2F2	B1	27827	6/11/85
B-B1F9	B1	26679	6/11/85
B-A2B2	B1	26682	6/12/85
B-A2F3	B1	27835	6/12/85
B-A1L10	B1	26013	6/12/85
B-A1L6	B1	25695	6/7/85
B-A1L8	B1	25696	6/5/85
B-A1L9	B1	26012	6/12/85
B-A2F5	B1	27842	6/12/85
B-A2A6	B1	26682	6/11/85
B-B1D4	B1	30348	8/14/85
B-B1K8	B1	27960	2/28/87
B-B1K9	B1	27962	2/28/87
B-B1D5	B1	30357	8/14/85
B-A2F11	B1	27897	6/13/85
A-D1G9	B1	27038	6/11/85
B-A2C6	B1	27044	6/11/85
B-A2F10	B1	27891	6/12/85
B-B1K1	B1	27713	2/28/87
B-A2D10	B1	27474	6/12/85
B-B1K7	B1	27877	3/1/87
B-A7A7	B1	27871	3/2/87
B-A2E2	B1	27485	8/9/85
B-B1J10	B1	27706	3/1/87
B-A2C2	B1	26936	6/12/85
B-A2B4	B1	26815	6/12/85
B-B1C5	B1	28052	6/13/85
B-B1C8	B1	28061	6/12/85
B-A2B6	B1	26820	6/11/85
B-A2C3	B1	26942	6/12/85
B-A2D6	B1	27381	6/13/85
B-A2D7	B1	27385	6/13/85
A-D1F7	B1	30381	6/13/85
B-A2D1	B1	27194	6/13/85
B-D8J9	B1	30203	6/13/85
B-B1B11	B1	28645	6/12/85
B-A2A5	B1	26532	6/13/85
B-A2A4	B1	26527	6/13/85
B-B1B10	B1	28643	6/12/85
B-B1F8	B1	30199	6/12/85
B-A2C11	B1	27190	6/13/85
B-A2A2	B1	26204	6/12/85
B-B1J3	B1	28436	3/2/87
B-A2F1	B1	27814	6/12/85
B-B1E10	B1	27186	6/12/85
B-B1K10	B1	27185	6/12/85
B-A2E11	B1	27812	6/11/85
B-B1J1	B1	28425	3/1/87
B-A2A1	B1	26199	6/11/85
B-A2E4	B1	27607	6/13/85
B-B1B8	B1	28569	6/12/85
B-B1B2	B1	28227	6/13/85
B-A2C8	B1	27105	6/12/85
B-B1G8	B1	27116	6/11/85
B-B1C9	B1	28216	6/13/85

B-B1B6	B1	28565	6/11/85
B-A2E3	B1	27606	6/12/85
B-B2B10	B1	28705	3/1/87
B-A2F7	B1	27857	6/13/85
B-A2F6	B1	27844	6/12/85
B-B1C10	B1	28233	6/13/85
B-B1B5	B1	28259	6/13/85
B-A3K5	B2	33448	1/18/88
B-A3F4	B2	32958	1/19/88
B-A3F5	B2	32970	1/18/88
B-A3K3	B2	33433	1/17/88
B-A3E10	B2	32299	1/19/88
B-A3E8	B2	32297	1/19/88
B-A3G3	B2	30657	1/17/88
B-B2L9	B2	25752	1/14/88
B-A3E6	B2	31829	3/4/88
B-A3A6	B2	26467	1/16/88
B-A3A4	B2	26456	1/15/88
B-A3E4	B2	31817	1/18/88
B-A2H3	B2	25748	1/15/88
B-A3G1	B2	30637	1/17/88
B-B3K8	B2	24779	2/2/86
B-B3K9	B2	24829	2/1/86
B-A3K10	B2	34280	1/18/88
B-B3K10	B2	24091	1/31/86
B-B3K11	B2	24203	2/1/86
B-A3A11	B2	27173	1/15/88
B-A3A10	B2	27162	1/14/88
B-B3L2	B2	24169	2/1/86
B-B3L3	B2	24137	1/31/86
B-A3L2	B2	34295	1/19/88
B-B3L4	B2	23788	1/31/86
B-B3L5	B2	23797	1/31/86
B-B3H5	B2	23140	1/31/86
B-B2L10	B2	30674	1/19/88
B-A3J11	B2	33124	1/17/88
B-A2L10	B2	25796	1/15/88
B-A2L9	B2	25792	1/16/88
B-A3F9	B2	33120	1/17/88
B-B3H7	B2	23341	2/1/86
B-A3G5	B2	30694	1/18/88
B-B1A7	B2	26052	2/2/86
B-B2K3	B2	34683	1/18/88
B-B3H8	B2	23699	1/31/86
B-B3H9	B2	23913	2/1/86
B-B3H10	B2	23274	1/31/86
B-B3H11	B2	24124	2/1/86
B-B3J1	B2	22500	2/1/86
B-B3J2	B2	22545	2/1/86
B-B3J3	B2	24129	1/31/86
B-B3J4	B2	23280	1/31/86
B-B3J5	B2	23903	1/31/86
B-B3J6	B2	23638	1/31/86
B-A3L5	B2	34705	1/17/88
B-A3H2	B2	31267	3/4/88

B-A3F1	B2	32543	1/17/88
B-B3J7	B2	24319	2/1/86
B-B3J8	B2	22439	2/1/86
B-A3K6	B2	33897	1/19/88
B-B3J9	B2	23511	2/2/86
B-A3K8	B2	33908	1/18/88
B-B3J10	B2	22438	2/1/86
B-B4E5	B2	24327	2/1/86
B-A3F2	B2	32557	1/17/88
B-A3G11	B2	31246	1/19/88
B-A3G9	B2	30915	1/17/88
B-A3G6	B2	30908	1/17/88
B-A3G7	B2	30909	1/19/88
B-A3G8	B2	30910	1/17/88
B-A3G10	B2	31241	1/19/88
B-A3F3	B2	32561	1/18/88
B-B4E6	B2	24325	2/1/86
B-A3K7	B2	33902	3/4/88
B-B4E9	B2	25146	1/31/86
B-A3H3	B2	31646	1/18/88
B-B4F7	B2	22444	2/1/86
B-B4E11	B2	24189	1/31/86
B-A3E11	B2	32539	1/17/88
B-A3H1	B2	31255	3/4/88
B-A3L4	B2	34703	1/18/88
B-B4F3	B2	23642	1/31/86
B-B4F4	B2	23902	2/1/86
B-B4F6	B2	23274	2/1/86
B-B4C10	B2	24142	2/1/86
B-B6F9	B2	22544	2/2/86
B-B4C11	B2	22500	2/1/86
B-A3K9	B2	34055	1/18/88
B-B4D3	B2	23274	2/1/86
B-B4D5	B2	23912	1/31/86
B-B4D8	B2	23703	1/31/86
B-A3L3	B2	34689	1/17/88
B-B1A6	B2	26061	2/2/86
B-B4D9	B2	23139	1/31/86
B-A3G4	B2	30691	1/19/88
B-A3F8	B2	33108	1/17/88
B-A2H5	B2	25790	1/16/88
B-A2L11	B2	25797	1/15/88
B-A3K1	B2	33125	1/18/88
B-B4D11	B2	23369	2/1/86
B-A2G7	B2	24830	3/4/88
B-B4E1	B2	23149	2/1/86
B-B3L1	B2	23801	2/1/86
B-A3L1	B2	34294	1/18/88
B-B6G9	B2	24169	2/2/86
B-A3A8	B2	27158	1/14/88
B-B1A8	B2	25844	2/2/86
B-A3A9	B2	27161	1/15/88
B-B6G8	B2	24201	2/2/86
B-B3H6	B2	24085	2/1/86
B-A3K11	B2	34281	1/17/88

B-B5K10	B1	30215	3/5/87
B-B1E1	B1	27246	3/2/87
B-A7C9	B1	30727	11/26/88
B-A7C7	B1	30788	11/26/88
B-B1G10	B1	27235	3/1/87
B-B2A11	B1	30222	3/5/87
B-B1K5	B1	27811	3/2/87
B-B5L11	B1	28876	11/28/88
B-A8B11	B1	28930	3/6/87
B-B1H4	B1	28083	3/1/87
B-B1H5	B1	28089	3/2/87
B-B1L4	B1	28910	3/5/87
B-B1H10	B1	28356	3/2/87
B-D8K11	B1	31155	3/6/87
B-B6D3	B1	30132	3/5/87
B-B1L7	B1	30139	3/5/87
B-A8A9	B1	31159	3/5/87
B-B1H9	B1	28347	3/1/87
B-A7B8	B1	29857	3/5/87
B-B1H8	B1	28166	3/1/87
B-A6G8	B1	28117	3/2/87
B-B2B8	B1	29848	3/5/87
B-A6G5	B1	30666	11/26/88
B-A6J5	B1	28873	11/28/88
B-A6F11	B1	27985	3/1/87
B-B2A10	B1	31695	3/4/87
B-A6H8	B1	31633	3/4/87
B-B2A9	B1	31622	3/3/87
B-A6H6	B1	31675	3/3/87
B-A6G7	B1	28946	11/26/88
B-A5L8	B1	30589	11/25/88
B-A5L9	B1	32181	11/25/88
B-A7B9	B1	31442	3/6/87
B-A6H9	B1	29006	3/4/87
B-A6H7	B1	28993	3/3/87
B-B2B5	B1	31441	3/5/87
B-A6F10	B1	32400	11/26/88
B-A6G9	B1	26707	3/1/87
B-B1G3	B1	26691	3/1/87
B-A6G6	B1	32091	11/26/88
B-D8K9	B1	31442	3/6/87
B-A6H11	B1	28987	3/5/87
B-B1L10	B1	28975	3/3/87
B-B2A2	B1	31452	3/6/87
B-A6G11	B1	32134	11/25/88
B-B5A1	B1	30845	11/25/88
B-A6G10	B1	28893	11/26/88
B-B1H2	B1	27987	2/28/87
B-B2B2	B1	31702	3/4/87
B-A6H10	B1	31621	3/5/87
B-B2A8	B1	31611	3/3/87
B-A6H5	B1	31672	3/3/87
B-B1L1	B1	27982	3/1/87
B-A7B10	B1	28843	11/28/88
B-A6J7	B1	30659	11/26/88

B-B2B9	B1	29854	3/5/87
B-B1H7	B1	28103	3/2/87
B-B1H6	B1	28101	3/1/87
B-B5L8	B1	29870	3/5/87
B-B1H11	B1	28359	3/1/87
B-A7D7	B1	31177	3/6/87
B-B1L5	B1	30155	3/5/87
A-D1F6	B1	30131	3/5/87
B-B2C1	B1	31151	3/5/87
B-A6K5	B1	28355	3/2/87
B-A5C11	B1	32526	3/3/87
B-A8C11	B1	28920	3/6/87
B-B1H3	B1	28083	3/2/87
A-D1F8	B1	28077	3/1/87
B-B2A7	B1	28937	3/5/87
B-A7C11	B1	28918	11/28/88
B-B1K3	B1	27803	3/2/87
B-A6J6	B1	30210	4/28/87
B-B1D11	B1	27239	3/1/87
B-A6J10	B1	30858	11/26/88
B-A6J9	B1	30774	11/26/88
B-B1G11	B1	27236	3/2/87
B-B2B1	B1	30213	4/28/87
B-B1K2	B1	27787	3/1/87
B-A6J11	B1	28842	11/26/88
B-B1L9	B1	29460	3/6/87
B-B5G11	B1	32555	3/3/87
B-B2A3	B1	29471	3/6/87
B-A6K11	B1	31098	3/4/87
B-A6L5	B1	31829	3/4/87
B-B1F2	B1	27552	2/28/87
B-B1F1	B1	27542	3/1/87
B-A6F2	B1	33382	3/5/87
B-B5L3	B1	31824	3/5/87
B-B5L2	B1	31092	3/5/87
B-A6J8	B1	28983	4/28/87
B-B1G7	B1	26969	3/1/87
B-B1L8	B1	29093	3/5/87
B-B5L7	B1	29098	3/5/87
B-B1G5	B1	26964	3/2/87
B-A6K6	B1	28983	4/27/87
B-A6K10	B1	30249	3/4/87
B-B5G10	B1	32059	3/4/87
B-B1J7	B1	27643	2/28/87
B-B1F5	B1	27635	2/28/87
B-A6C2	B1	32056	3/4/87
B-B6L3	B1	30267	3/5/87
B-B2C2	B1	29712	3/5/87
B-B5L9	B1	29709	3/5/87
B-A3L8	~~B2~~	27232	9/26/89
B-A2H10	B2	23500	1/14/88
B-A4B11	B2	26948	9/27/89
B-A3D7	B2	28438	1/17/88
B-A2G9	B2	25011	1/19/88
B-A4A1	B2	27342	9/26/89

B-A3E2	B2	28788	1/16/88
B-A2K7	B2	28120	1/18/88
B-A2K3	B2	27895	1/17/88
B-A2H2	B2	25780	1/17/88
B-A4A11	B2	27727	9/27/89
B-A4C7	B2	28244	9/26/89
B-A3B9	B2	28873	1/19/88
B-A2H7	B2	22616	1/14/88
B-A2J9	B2	23872	1/17/88
B-A3C11	B2	28264	1/17/88
B-A2K8	B2	28135	1/16/88
B-A3L10	B2	27277	9/26/89
B-A2J3	B2	23783	1/15/88
B-A3C2	B2	29288	1/18/88
B-A4B7	B2	26758	9/27/89
B-A2J7	B2	23832	1/16/88
B-A3C8	B2	29917	1/17/88
B-A3D8	B2	28441	1/17/88
B-A4F6	B2	28218	9/29/89
B-A2H8	B2	22618	1/15/88
B-A3B3	B2	28796	1/15/88
B-A4C8	B2	28267	9/26/89
B-A4B8	B2	26852	9/26/89
B-A4E3	B2	27783	9/29/89
B-A2G3	B2	24613	1/17/88
B-A3D3	B2	28283	1/17/88
B-A3B11	B2	28913	1/18/88
B-A4C10	B2	28278	9/26/89
B-A2K5	B2	28103	1/19/88
B-A2L8	B2	25791	1/17/88
B-A3C1	B2	28913	1/18/88
B-A4A5	B2	27414	9/26/89
B-A3D11	B2	28785	1/17/88
B-A2G8	B2	25006	1/19/88
B-A2J6	B2	23812	1/16/88
B-A4A6	B2	27416	9/26/89
B-A2H9	B2	22624	1/15/88
B-A2G2	B2	24447	1/19/88
B-A4B3	B2	26646	9/27/89
B-A4E2	B2	27760	9/27/89
B-A2J1	B2	23517	1/15/88
B-A2K1	B2	23916	1/17/88
B-A3B1	B2	27753	1/17/88
B-A2L1	B2	28167	1/16/88
B-A2G11	B2	25051	1/19/88
B-A3L6	B2	27220	9/26/89
B-A4A7	B2	27421	9/26/89
B-A3B8	B2	28822	1/16/88
B-A4F7	B2	28232	9/29/89
B-A4B5	B2	26717	9/27/89
B-A3J10	B2	26584	9/27/89
B-A4A4	B2	27406	9/26/89
B-A4F3	B2	28199	9/27/89
B-A3A1	B2	25808	1/18/88
B-A2J8	B2	23841	1/17/88

B-A2K2	B2	27766	1/19/88
B-A2L4	B2	28169	1/16/88
B-A4B2	B2	26616	9/26/89
B-A3B10	B2	28895	1/19/88
B-A4H7	B2	23928	1/15/88
B-A3J8	B2	26541	9/26/89
B-A3A3	B2	25814	1/18/88
B-A2G5	B2	24627	1/18/88
B-A4A2	B2	27347	9/26/89
B-A2L2	B2	28168	1/16/88
B-A3C4	B2	29336	1/18/88
B-A3J7	B2	26537	9/26/89
B-A2H1	B2	25780	1/17/88
B-A3E3	B2	28793	1/18/88
B-A3C3	B2	29289	1/18/88
B-A3C9	B2	28195	1/17/88
B-A2K9	B2	28135	1/16/88
B-A3J5	B2	26473	9/27/89
B-A2K11	B2	28149	1/16/88
B-A3B4	B2	28796	1/15/88
B-A2J11	B2	23914	1/16/88
B-A2L7	B2	28192	1/19/88
B-A3H5	B2	25767	9/27/89
B-A3D9	B2	28777	1/15/88
B-A2G6	B2	24631	1/17/88
B-A4C5	B2	27213	9/26/89
B-A2K10	B2	28149	1/16/88
B-A3C6	B2	29913	1/19/88
B-A3J6	B2	26493	9/27/89
B-A2L6	B2	28182	1/17/88
B-A3E1	B2	28786	1/16/88
B-A3B2	B2	27766	1/17/88
B-A3B5	B2	28812	1/16/88
B-A4B4	B2	26710	9/26/89
B-A3J9	B2	26571	9/27/89
B-A3B6	B2	28819	1/17/88
B-A2H4	B2	25781	1/17/88
B-A2J2	B2	23517	1/15/88
B-A3B7	B2	28819	1/18/88
B-A3D5	B2	28393	1/18/88
B-A3C7	B2	29913	1/18/88
B-A2L3	B2	28169	1/16/88
B-A3L11	B2	27338	9/27/89
B-A2J4	B2	23782	1/15/88
B-A2J10	B2	23911	1/17/88
B-A3D1	B2	28266	1/17/88
B-A4B6	B2	26732	9/26/89
B-A2L5	B2	28181	1/18/88
B-A2H11	B2	23500	1/14/88
B-A4B1	B2	26594	9/26/89
B-B2K4	B2	25774	1/16/88
B-A3F10	B2	29919	1/18/88
B-A3H7	B2	25792	9/27/89
B-A4A3	B2	27361	9/26/89
B-A4B10	B2	26934	9/27/89

B-A3D10	B2	28777	1/16/88
B-A2G1	B2	24406	1/19/88
B-A3C5	B2	29338	1/17/88
B-A4G3	B2	27745	9/29/89
B-A3D2	B2	28281	1/17/88
B-A4H11	B2	23929	1/15/88
B-A2G4	B2	24613	1/17/88
B-A4B9	B2	26872	9/26/89
B-A3D6	B2	28395	1/17/88
B-A4E11	B2	28160	9/27/89
B-A2J5	B2	23802	1/16/88
B-A3A2	B2	25810	1/17/88
B-A2G10	B2	25047	1/19/88
B-A3H6	B2	25769	9/27/89
B-A2K6	B2	28103	1/19/88
B-A4C9	B2	28270	9/26/89
B-A5K11	B1	32340	12/1/88
B-B5A11	B1	32706	12/1/88
B-B5C10	B1	32711	12/1/88
B-B5A10	B1	32319	12/1/88
B-B6F1	B1	30422	11/30/88
B-B6L1	B1	30426	11/28/88
B-A5B8	B1	31052	10/14/90
B-B2H9	B1	27523	11/25/88
B-B2J6	B1	27986	11/26/88
B-A6J2	B1	31499	11/30/88
B-A6C5	B1	29243	11/26/88
B-B2G11	B1	28762	11/26/88
B-B2H1	B1	28764	11/26/88
B-A5H2	B1	29261	11/25/88
B-A6K1	B1	31541	11/30/88
B-B6E1	B1	31591	11/30/88
B-B6H2	B1	31346	11/28/88
B-A5H11	B1	31958	12/1/88
B-B2D7	B1	27391	11/27/88
B-A5E10	B1	31969	12/1/88
B-A5G11	B1	33927	12/1/88
B-A5H3	B1	28903	11/25/88
B-B2K2	B1	28359	11/27/88
B-A6L3	B1	30775	10/16/90
A-C1H6	B1	30732	10/15/90
B-B2K1	B1	28374	11/27/88
B-A6E3	B1	28860	11/26/88
B-A5F11	B1	33726	12/1/88
B-A6H2	B1	30908	11/30/88
B-A6L1	B1	30705	11/30/88
B-B6H4	B1	29141	11/26/88
A-C1H11	B1	30438	10/15/90
B-A6D8	B1	30290	10/16/90
B-A5F9	B1	29092	11/26/88
B-B5L10	B1	29144	11/26/88
A-C1D9	B1	30365	10/16/90
B-A6D7	B1	30413	10/15/90
B-B6C4	B1	29112	11/26/88
B-A6L4	B1	30759	11/28/88

B-B6K1	B1	30741	11/28/88
B-A5D10	B1	33719	12/1/88
B-A6E5	B1	28860	11/26/88
B-B6J4	B1	28385	11/26/88
B-A6J3	B1	30805	10/15/90
B-B5K3	B1	30798	10/16/90
B-B2G6	B1	28559	11/27/88
B-B2H2	B1	28781	11/25/88
B-A5C10	B1	33716	12/1/88
B-A5K10	B1	32205	12/1/88
B-B2H10	B1	27442	11/28/88
B-B2D5	B1	27367	11/26/88
B-A5L10	B1	32003	12/1/88
B-B6A2	B1	31265	11/30/88
B-A6C4	B1	31350	11/28/88
B-B2J5	B1	27851	11/26/88
B-B6B4	B1	31484	11/28/88
B-A5G2	B1	29275	11/25/88
B-B2H6	B1	28823	11/26/88
B-B2H5	B1	28821	11/26/88
B-B6A4	B1	29337	11/26/88
B-A6J4	B1	31834	11/28/88
B-B2J2	B1	27733	11/25/88
B-B2H8	B1	27517	11/25/88
B-B2J3	B1	27753	11/26/88
B-B6B2	B1	30487	11/30/88
B-A6C3	B1	30624	11/28/88
B-B5D10	B1	32659	12/1/88
B-A5L11	B1	32751	12/1/88
B-A5A10	B1	32921	12/1/88
B-A5G10	B1	32427	12/2/88
B-B6F2	B1	29889	12/1/88
B-B2J4	B1	27486	11/30/88
B-B2H11	B1	27576	11/28/88
B-A6L2	B1	29912	11/30/88
B-B2D6	B1	27369	11/30/88
B-A5A11	B1	31112	12/1/88
B-A5J11	B1	31113	12/1/88
B-A6D1	B1	27436	11/28/88
B-A6E2	B1	29490	11/30/88
B-B6H3	B1	30819	11/27/88
B-A5F10	B1	31333	12/1/88
B-B5E10	B1	31085	12/1/88
B-B6F4	B1	30580	11/27/88
B-B6C1	B1	29526	11/30/88
B-B5B11	B1	30304	12/1/88
B-B6H1	B1	29552	11/27/88
B-B5A2	B1	28355	11/26/88
B-B2C4	B1	26774	11/26/88
B-B2J8	B1	28239	11/28/88
B-B6F3	B1	29512	11/27/88
B-B5F11	B1	30307	12/1/88
B-A5H10	B1	30284	12/1/88
B-B2J11	B1	28313	11/28/88
B-B6A3	B1	28900	11/26/88

B-B6E4	B1	28898	11/26/88
B-B2J10	B1	28273	11/26/88
B-A6K2	B1	30056	11/30/88
B-B2C8	B1	26949	11/30/88
B-A5B10	B1	30242	12/1/88
B-B2D3	B1	27154	11/25/88
B-B2G4	B1	28469	11/26/88
B-B2G7	B1	28582	11/26/88
B-B2D2	B1	27114	11/26/88
B-A6D2	B1	30200	11/30/88
B-B2C11	B1	27046	11/29/88
B-B2C7	B1	26900	11/28/88
B-A5D11	B1	30241	12/1/88
B-B2C9	B1	27007	11/26/88
B-B2H4	B1	28379	11/26/88
B-B2G9	B1	28603	11/26/88
B-B2D1	B1	27069	11/25/88
B-B6B1	B1	30436	11/30/88
B-B2C10	B1	27044	11/28/88
B-A6J1	B1	30133	11/30/88
B-B2G2	B1	28392	11/26/88
B-B2G10	B1	28707	11/26/88
B-B6C3	B1	28804	11/26/88
B-B2G3	B1	28413	11/28/88
B-A6F1	B1	30043	11/29/88
B-B5E11	B1	30433	12/1/88
B-B6K4	B1	29703	11/27/88
B-B2J9	B1	28254	11/28/88
B-B2C3	B1	26711	11/25/88
B-B2C6	B1	26891	11/26/88
B-B2J7	B1	28208	11/26/88
B-B6G3	B1	29426	11/27/88
B-B5D11	B1	30231	12/1/88
B-A6D4	B1	29718	11/28/88
B-B6K3	B1	30707	11/27/88
B-B6G1	B1	31078	11/30/88
B-A6G2	B1	31164	11/30/88
B-A6D3	B1	30770	11/27/88
B-A6C1	B1	29373	11/28/88
B-B2J1	B1	27592	11/30/88
B-A5E11	B1	31246	12/1/88
B-B5C11	B1	31057	12/1/88
B-B2D8	B1	27399	11/28/88
B-A6H1	B1	29869	11/30/88
B-B2C5	B1	27406	11/29/88
B-B2D4	B1	27360	11/28/88
B-A6G1	B1	29960	11/29/88
B-A3C10	B2	28240	1/18/88
B-A3D4	B2	28283	1/18/88
B-A2K4	B2	27941	1/19/88
B-A6G3	B1	34599	10/14/90
B-A5B9	B1	34728	10/14/90
B-B6D1	B1	33055	11/30/88
B-A6K3	B1	33138	11/28/88
B-B5B10	B1	32759	12/1/88

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B-A5J10	B1	32658	12/1/88
B-B5A3	B1	34558	1/23/89
B-B5A4	B1	34083	1/23/89
B-B5A5	B1	33786	11/30/88
B-A5B11	B1	32543	12/1/88
B-B6A1	B1	32656	11/30/88
B-B2G5	B1	28526	12/1/88
B-B2H3	B1	28794	12/1/88
B-A6E4	B1	33849	10/14/90
B-B5K11	B1	33877	10/14/90
B-A5C1	B1	33650	10/14/90
B-A6D6	B1	33717	10/14/90
B-B5F10	B1	32061	12/1/88
B-B5A6	B1	33854	1/23/89
B-B2G8	B1	28600	12/1/88
B-B6D2	B1	32387	12/1/88
B-B6D4	B1	32633	11/28/88
B-A6E1	B1	32693	11/28/88
B-B5A7	B1	33932	1/23/89
B-B5A8	B1	33928	1/23/89
B-B5A9	B1	34680	1/23/89
B-B5B1	B1	34746	11/28/88
B-B6E2	B1	32757	12/1/88
B-B6C2	B1	33045	11/30/88
B-B6G4	B1	33005	11/28/88
B-A6F5	B1	34582	10/15/90
B-B5J11	B1	34712	10/14/90
B-A4H6	B2	29691	10/6/89
B-A4H5	B2	29668	9/30/89
B-B3A4	B2	29207	9/23/91
B-B2L11	B2	30595	10/5/89
B-A4E6	B2	27899	10/2/89
B-A4E7	B2	27904	9/30/89
B-A4G2	B2	30578	10/5/89
B-B3A5	B2	29190	9/23/91
B-B3A6	B2	34224	9/25/91
B-A4C2	B2	26990	9/28/89
B-A4D9	B2	29441	10/5/89
B-A4H10	B2	29961	10/5/89
B-A4H9	B2	29944	10/5/89
B-A4D10	B2	29461	9/30/89
B-A4C3	B2	27005	9/28/89
B-B3A7	B2	34200	9/24/91
B-B3A8	B2	33567	9/25/91
B-A4K6	B2	32440	10/6/89
B-A4K4	B2	32404	10/5/89
B-B3A9	B2	33607	9/26/91
B-A4F1	B2	28179	10/5/89
B-A3J3	B2	26432	9/29/89
B-B6E8	B2	33825	4/2/94
B-B6H5	B2	33805	4/2/94
B-A3J4	B2	26455	9/27/89
B-A4F5	B2	28215	10/2/89
B-A4J11	B2	31838	10/6/89
B-B3A10	B2	31531	9/23/91

B-A4L7	B2	32740	10/1/89
B-A4L4	B2	32728	10/2/89
B-B3A11	B2	31530	9/24/91
B-A4K3	B2	31864	10/1/89
B-A4D2	B2	28529	9/28/89
B-A4A9	B2	27668	9/27/89
B-A4K1	B2	31840	10/5/89
B-B3B1	B2	30371	9/23/91
B-B3B2	B2	30351	9/23/91
B-A4J9	B2	31825	10/6/89
B-A4A10	B2	27669	9/29/89
B-A4C11	B2	28511	9/28/89
B-B3B3	B2	30828	9/23/91
B-B2K8	B2	32339	9/24/91
B-A4F8	B2	30148	10/1/89
B-A4J8	B2	30147	9/30/89
B-B2K10	B2	30815	9/23/91
B-A4D3	B2	29031	10/5/89
B-A4F9	B2	30234	10/5/89
B-B3A1	B2	33600	10/6/89
B-B6H6	B2	33257	4/3/94
B-A4L8	B2	33633	1/24/90
B-A4F11	B2	30262	9/30/89
B-A4D4	B2	29037	10/3/89
B-A4G5	B2	30766	10/6/89
B-A4L9	B2	33643	10/6/89
B-B1A2	B2	33514	10/1/89
B-A4J2	B2	30103	10/2/89
B-A4G9	B2	31315	10/6/89
B-A4G11	B2	31344	10/6/89
B-A4J6	B2	30136	10/6/89
B-B1A3	B2	33523	10/2/89
B-A4G6	B2	30822	10/6/89
B-A4H4	B2	29600	10/5/89
B-A3J2	B2	26430	9/26/89
B-B2K11	B2	31259	9/23/91
B-A4K11	B2	32607	10/5/89
B-B3A2	B2	29694	10/3/89
B-A4K10	B2	32597	10/5/89
B-B2L1	B2	31211	9/23/91
B-A3H10	B2	26419	9/27/89
B-A4H3	B2	29576	10/5/89
B-A4E8	B2	28086	10/1/89
B-A4L1	B2	32700	10/2/89
B-A3L9	B2	27244	9/27/89
B-A3L7	B2	27226	9/29/89
B-B3A3	B2	32745	9/30/89
B-A4E9	B2	28115	10/2/89
B-A4E10	B2	28128	9/30/89
B-A4L3	B2	32716	9/30/89
B-A4C6	B2	27218	9/29/89
B-A4G8	B2	30903	9/30/89
B-A4L2	B2	32707	9/30/89
B-A4E1	B2	29558	10/6/89
B-A3H9	B2	26375	9/27/89

B-B2L2	B2	31229	9/23/91
B-A4K8	B2	32549	10/5/89
B-A4H8	B2	29695	10/2/89
B-A4K9	B2	32574	10/3/89
B-B2L3	B2	31240	9/23/91
B-A4G7	B2	30842	10/5/89
B-B1A1	B2	33722	1/24/90
B-B1A9	B2	33575	9/30/89
B-A4J4	B2	30128	9/30/89
B-A4H1	B2	31348	10/6/89
B-A4G10	B2	31320	10/6/89
B-A4J3	B2	30123	9/30/89
B-B1A5	B2	33536	9/30/89
B-A4L11	B2	33699	1/24/90
B-A4D5	B2	29047	10/2/89
B-A4F10	B2	30257	10/2/89
B-B2K5	B2	33609	1/24/90
B-B6E9	B2	33230	4/4/94
B-B6J10	B2	33273	4/3/94
B-A4L10	B2	33644	1/24/90
B-A4G1	B2	30278	9/30/89
B-A4D6	B2	29084	9/30/89
B-B2L4	B2	30858	9/23/91
B-B2L5	B2	32366	9/23/91
B-A4J5	B2	30130	10/1/89
B-A4J7	B2	30145	9/30/89
B-B2L6	B2	32345	9/24/91
B-B2L7	B2	30836	9/23/91
B-A4D1	B2	28523	9/28/89
B-A4A8	B2	27663	9/29/89
B-A4K2	B2	31861	10/2/89
B-B2L8	B2	30428	9/23/91
B-B3C9	B2	30417	9/23/91
B-A4H2	B2	31802	10/1/89
B-A4D7	B2	29156	9/27/89
B-B2K6	B2	31792	10/5/89
B-B3E5	B2	31535	9/24/91
B-A4L6	B2	32736	9/30/89
B-A4L5	B2	32728	9/30/89
B-B3E6	B2	31522	9/23/91
B-A4J10	B2	31832	10/2/89
B-A4F4	B2	28205	10/5/89
B-A3H11	B2	26423	9/27/89
B-B6J7	B2	33796	4/2/94
B-B6J9	B2	33761	4/2/94
B-A3J1	B2	26426	9/29/89
B-A4F2	B2	28187	10/6/89
B-B3E7	B2	33579	9/25/91
B-A4K7	B2	32455	10/6/89
B-A4K5	B2	32436	10/5/89
B-B2K9	B2	33588	9/25/91
B-B3E8	B2	34175	9/25/91
B-A4C1	B2	26962	9/28/89
B-A4D8	B2	29439	10/2/89
B-B2K7	B2	29980	10/5/89

B-A4J1	B2	30003	10/3/89
B-A4D11	B2	29497	9/30/89
B-A4C4	B2	27046	9/28/89
B-B3E9	B2	34265	9/24/91
B-B6A8	B2	29222	9/23/91
B-A4E4	B2	27868	10/1/89
B-A4E5	B2	27888	9/30/89
B-A4G4	B2	30609	10/5/89
B-B3C10	B2	29200	9/23/91
B-B6H7	B2	32994	4/4/94
B-B6J5	B2	32973	4/3/94
B-B6F5	B2	33007	4/3/94
B-B6E10	B2	32967	4/4/94
B-B4H3	B1	32315	4/14/93
B-A5C2	B1	30754	10/19/90
A-C1F5	B1	30587	10/17/90
A-C1C10	B1	32139	4/13/93
A-C1D11	B1	32359	4/13/93
B-A5C3	B1	32468	10/18/90
A-C1G5	B1	29539	10/15/90
B-B6K2	B1	29314	10/15/90
B-A5C4	B1	32168	10/19/90
B-B5K6	B1	28739	10/17/90
B-B4H4	B1	30471	4/15/93
B-A6E9	B1	31907	10/17/90
B-A5C5	B1	34247	10/18/90
B-B6L4	B1	33198	10/17/90
B-B5L5	B1	33434	10/17/90
B-A5C6	B1	34408	10/18/90
B-A6K4	B1	31883	10/17/90
B-B4H5	B1	30733	4/16/93
B-A5C7	B1	30439	10/18/90
B-A6H4	B1	33229	10/17/90
A-C1F7	B1	33151	10/17/90
B-A6D5	B1	30273	10/18/90
B-B4H6	B1	31112	4/16/93
B-A6H3	B1	30257	10/17/90
B-A6F3	B1	32575	10/17/90
A-C1A9	B1	33679	10/17/90
A-C1B5	B1	33451	10/16/90
B-A5C8	B1	32588	10/19/90
A-C1G7	B1	30262	10/16/90
B-B4H7	B1	31162	4/14/93
B-A5G7	B1	30215	10/18/90
B-B4H8	B1	31036	4/14/93
B-A5G8	B1	32487	10/18/90
B-A6F4	B1	29749	10/15/90
B-A6G4	B1	30666	10/14/90
B-A6E6	B1	30631	10/14/90
A-C1G6	B1	29927	10/16/90
B-A5G9	B1	32233	10/18/90
B-A6F6	B1	31228	4/13/93
B-A6F7	B1	30377	10/18/90
A-C1F6	B1	31608	10/17/90
B-A6E10	B1	33114	10/17/90

B-A5H1	B1	33121	10/14/90
B-B5L4	B1	30912	10/14/90
A-C1H7	B1	31137	10/14/90
B-B4H9	B1	31994	4/14/93
B-A6D11	B1	33098	10/16/90
A-C1A11	B1	31594	10/16/90
B-A5H4	B1	33092	10/19/90
B-B4H10	B1	31495	4/14/93
B-B4H11	B1	31455	4/14/93
A-C1D5	B1	32811	10/17/90
A-C1G11	B1	28706	10/17/90
B-B4J1	B1	31569	4/13/93
B-A5H5	B1	34855	1/9/91
B-A5H6	B1	34392	10/18/90
B-B4J2	B1	32632	4/14/93
B-A6D9	B1	32862	1/13/94
B-A5H7	B1	34588	10/18/90
B-A5H8	B1	34741	1/9/91
B-B4J3	B1	31614	4/14/93
B-A5H9	B1	32389	10/18/90
B-A5J1	B1	33372	10/19/90
A-C1D10	B1	30682	10/16/90
A-C1B8	B1	27682	10/14/90
B-B4J4	B1	31980	4/13/93
A-C1H9	B1	30801	10/14/90
A-C1H8	B1	30593	10/14/90
B-B4J5	B1	32139	4/13/93
B-A6F8	B1	27694	10/14/90
B-B5K1	B1	30891	10/15/90
A-C1G8	B1	33045	10/16/90
B-A6F9	B1	32355	10/18/90
B-A5J2	B1	29609	10/18/90
B-A6E11	B1	32387	10/17/90
B-A6E8	B1	32912	10/17/90
B-A6C11	B1	30687	10/14/90
B-B4J6	B1	32157	4/13/93
B-B4J7	B1	32287	4/14/93
B-A6E7	B1	30858	10/14/90
A-C1E10	B1	33000	10/17/90
B-A6D10	B1	32657	10/16/90
B-A5J3	B1	29416	10/19/90
A-C1B6	B1	29504	10/15/90
B-B6B3	B1	31047	10/14/90
B-A6A6	B1	31278	10/14/90
B-B5H10	B1	31262	10/14/90
B-A6A8	B1	30801	10/14/90
B-A6A9	B1	29332	10/15/90
A-C1H5	B1	29447	10/15/90
A-C1A5	B1	30800	10/14/90
B-A6A7	B1	31244	10/14/90
B-B5J10	B1	31507	10/14/90
B-A5J4	B1	30824	10/14/90
B-B5K2	B1	29528	10/15/90
B-A6C9	B1	29490	10/17/90
B-B6E3	B1	32425	10/17/90

A-C1D6	B1	33074	10/17/90
B-A5J5	B1	30627	10/14/90
B-B4J8	B1	32083	4/14/93
A-C1C11	B1	32092	4/13/93
A-C1A10	B1	30614	10/14/90
B-A6B9	B1	33194	10/16/90
A-C1G9	B1	32415	10/17/90
B-A6C7	B1	29570	10/17/90
B-A5F5	B1	32397	10/18/90
A-C1B11	B1	33327	10/17/90
B-A6B6	B1	30703	10/15/90
B-A6B8	B1	27774	10/14/90
B-A5E8	B1	32028	4/13/93
B-A6B11	B1	30557	10/14/90
B-A6B7	B1	30626	10/14/90
B-A5E9	B1	32023	4/13/93
A-C1H10	B1	27714	10/14/90
B-A6A10	B1	30608	10/16/90
A-C1B10	B1	33006	10/16/90
B-B5L1	B1	32639	10/18/90
B-A5F1	B1	31724	4/14/93
B-A5F6	B1	34785	1/9/91
B-A6C10	B1	34429	10/18/90
B-B4F8	B1	32660	4/13/93
B-B4F9	B1	32691	4/13/93
B-A5F7	B1	34405	10/18/90
B-A5F8	B1	34579	1/9/91
A-C1B7	B1	31910	4/13/93
B-B4F10	B1	33967	4/15/93
B-B5K9	B1	32951	1/9/91
B-B4F11	B1	31568	4/14/93
B-B4G1	B1	31448	4/14/93
A-C1E11	B1	32977	1/9/91
A-C1G10	B1	29139	10/17/90
A-C1C9	B1	31714	10/17/90
B-A6C8	B1	33326	10/17/90
B-B4G2	B1	31752	4/14/93
B-B6J2	B1	30935	10/14/90
B-A5G1	B1	30957	10/14/90
B-B4G3	B1	31802	4/13/93
B-A6B10	B1	33131	10/16/90
A-C1C7	B1	31633	10/16/90
B-A5G3	B1	30422	10/18/90
B-B5K7	B1	31034	4/13/93
B-A5G4	B1	32394	10/19/90
B-A6A11	B1	29772	10/16/90
B-A5G5	B1	30605	10/14/90
B-B5J9	B1	30799	10/14/90
A-C1B9	B1	29731	10/15/90
B-A6C6	B1	32534	10/17/90
B-B4G4	B1	31041	4/14/93
A-C1A8	B1	30552	10/18/90
B-B4G5	B1	30882	4/14/93
A-C1C6	B1	30184	10/17/90
B-D8J10	B1	32790	10/16/90

A-C1C5	B1	33489	10/17/90
B-B5K4	B1	33345	10/16/90
A-C1D7	B1	32665	10/16/90
A-C1A1	B1	30277	10/16/90
B-A5G6	B1	30327	10/19/90
B-D8L10	B1	33487	10/16/90
B-B6L2	B1	33589	10/16/90
B-B5K8	B1	30138	10/18/90
B-B4G7	B1	30544	4/16/93
B-A8A10	B1	31959	10/17/90
B-A5L5	B1	34411	10/18/90
B-B5K5	B1	33144	10/17/90
A-C1D8	B1	33083	10/16/90
A-C1A6	B1	34562	10/18/90
B-A7C8	B1	31867	10/16/90
B-B4G8	B1	30803	4/14/93
B-A5L6	B1	32247	10/19/90
A-C1C8	B1	29429	10/15/90
B-B5E1	B1	29570	10/15/90
B-A5L7	B1	32229	10/18/90
B-B5L6	B1	28811	10/17/90
B-B4G10	B1	32174	4/13/93
A-C1A7	B1	30643	10/18/90
B-A8C10	B1	30769	10/17/90
B-B4G11	B1	31997	4/14/93
B-B3C11	B2	33145	9/26/91
B-B3D1	B2	32281	9/26/91
B-B3D2	B2	34587	9/27/91
B-B3D3	B2	34611	9/25/91
B-B3D4	B2	32271	9/27/91
B-B3D5	B2	33129	9/27/91
B-B3D6	B2	30906	9/24/91
B-B6H9	B2	31926	4/2/94
B-B3D7	B2	31447	9/27/91
B-B3D8	B2	31422	9/27/91
B-B6G5	B2	31921	4/2/94
B-B3D9	B2	30904	9/24/91
B-B3D10	B2	31576	9/26/91
B-B3D11	B2	33754	9/27/91
B-B3E1	B2	33647	9/25/91
B-B3E2	B2	33659	9/25/91
B-B3E3	B2	33756	9/26/91
B-B3E4	B2	31557	9/25/91
B-B3J11	B2	32234	9/24/91
B-B3K1	B2	32206	9/24/91
B-B3K2	B2	33329	9/26/91
B-B6H11	B2	34295	4/2/94
B-B6K6	B2	33792	4/3/94
B-B6H8	B2	33803	4/4/94
B-B6E11	B2	34349	4/2/94
B-B3K3	B2	33353	9/26/91
B-B6J8	B2	32275	4/2/94
B-B6J11	B2	33881	4/2/94
B-A7D1	B2	32214	4/2/94
B-B3B4	B2	33351	9/27/91

B-B3B5	B2	33377	9/25/91
B-B3B6	B2	31712	9/24/91
B-B3B7	B2	31751	9/24/91
B-A7D5	B2	32729	4/2/94
B-B3B8	B2	32084	9/26/91
B-B3B9	B2	34320	9/26/91
B-B3B10	B2	33607	9/27/91
B-B3B11	B2	33597	9/26/91
B-B3C1	B2	34251	9/25/91
B-B3C2	B2	32078	9/26/91
B-B3C3	B2	33170	9/26/91
B-B3C4	B2	32776	9/24/91
B-B3C5	B2	32796	9/24/91
B-A1C1	B2	33208	9/26/91
B-B3C6	B2	34856	9/25/91
B-A7F4	B2	34437	4/4/94
B-B3C7	B2	34742	9/27/91
B-B3C8	B2	34758	9/26/91
B-B3K4	B2	34782	9/26/91
B-B3K5	B2	34830	9/25/91
B-A7F1	B2	34402	4/3/94
B-B3K6	B2	34738	9/26/91
B-B3K7	B2	34773	9/26/91
B-A7D6	B2	34445	4/4/94
B-B3E10	B2	34877	9/25/91
B-B3E11	B2	33211	9/25/91
B-B3F1	B2	32807	9/24/91
B-B3F2	B2	32787	9/24/91
B-B3F3	B2	33183	9/25/91
B-B3F4	B2	32083	9/26/91
B-B3F5	B2	34236	9/25/91
B-B3F6	B2	33617	9/26/91
B-B3F7	B2	33614	9/26/91
B-B3F8	B2	34238	9/25/91
B-B3F9	B2	32089	9/26/91
B-A7E4	B2	32772	4/2/94
B-B3F10	B2	31769	9/24/91
B-B3F11	B2	31732	9/24/91
B-A7C6	B2	32752	4/2/94
B-B3G1	B2	33429	9/26/91
B-B3G2	B2	33400	9/26/91
B-A7F2	B2	32225	4/2/94
B-A7D4	B2	33815	4/2/94
B-A7F3	B2	32235	4/2/94
B-B3G3	B2	33300	9/26/91
B-A7D2	B2	34271	4/2/94
B-A7E6	B2	33757	4/3/94
B-A7E1	B2	34314	4/2/94
B-B3G4	B2	33317	9/26/91
B-B3G5	B2	32216	9/24/91
B-B3G6	B2	32197	9/24/91
B-B3G7	B2	31521	9/26/91
B-B3G8	B2	33690	9/26/91
B-B3G9	B2	33602	9/25/91
B-B3G10	B2	33605	9/25/91

B-B3G11	B2	33675	9/27/91
B-B3H1	B2	31504	9/27/91
B-B3H2	B2	30870	9/24/91
B-A7E2	B2	31892	4/2/94
B-B3H3	B2	31398	9/27/91
B-B3H4	B2	31405	9/27/91
B-A7E3	B2	31879	4/2/94
B-A7D3	B2	30832	9/24/91
B-B4E10	B2	32214	9/26/91
B-A7E5	B2	34544	9/25/91
B-B4D4	B2	32213	9/26/91
B-B4H1	B1	30993	4/18/93
B-B4H2	B1	30596	4/18/93
A-C1F9	B1	30571	4/17/93
B-B4G6	B1	32077	4/18/93
B-B4G9	B1	32326	4/17/93
A-C1F10	B1	30365	4/17/93
B-B4L4	B1	30183	4/17/93
B-B4L5	B1	29386	4/17/93
B-B4L6	B1	29593	4/17/93
B-B4L7	B1	30028	4/16/93
B-B4L8	B1	32725	4/17/93
B-B4L9	B1	32566	4/18/93
B-B4L10	B1	29453	4/16/93
B-B4L11	B1	29413	4/16/93
B-A6L8	B1	33316	1/13/94
B-A5A1	B1	31997	4/18/93
B-A5A2	B1	30580	4/15/93
B-A5A3	B1	30693	4/15/93
B-A5A4	B1	31963	4/17/93
B-A6K8	B1	33067	1/13/94
B-A5A5	B1	29863	4/17/93
B-A5A6	B1	29963	4/18/93
B-A5A7	B1	30134	4/17/93
B-A5A8	B1	30374	4/16/93
B-A5A9	B1	29754	4/18/93
B-B4J9	B1	29997	4/16/93
A-C1F11	B1	30718	4/17/93
A-C1E5	B1	30810	4/16/93
B-B4J10	B1	30730	4/18/93
B-B4J11	B1	30624	4/17/93
B-B4K1	B1	29994	4/15/93
B-B4K2	B1	31573	4/16/93
B-B4K3	B1	30328	4/17/93
B-B4K4	B1	30316	4/18/93
B-B4K5	B1	31606	4/16/93
B-B4K6	B1	30023	4/15/93
B-B4K7	B1	30213	4/17/93
B-B4K8	B1	30397	4/16/93
B-A6L7	B1	32736	1/13/94
B-B4K9	B1	32700	4/17/93
B-B4K10	B1	32433	4/18/93
B-B4K11	B1	32354	4/17/93
B-B4L1	B1	30181	4/16/93
B-B4L3	B1	30865	4/16/93

A-C1F8	B1	24724	10/17/90
B-A5F2	B1	30084	4/15/93
B-A5F3	B1	31555	4/16/93
B-A5F4	B1	30526	4/16/93
B-B4L2	B1	30363	4/16/93
B-A5C9	B1	31806	4/16/93
B-A5D1	B1	30015	4/15/93
B-A5D2	B1	31058	4/18/93
A-C1E6	B1	30817	4/16/93
A-C1E7	B1	31042	4/16/93
B-A5D3	B1	30015	4/17/93
B-A5D4	B1	29761	4/18/93
B-A5D5	B1	30138	4/17/93
B-A5D6	B1	30119	4/16/93
B-A5D7	B1	29763	4/18/93
B-A5D8	B1	30203	4/16/93
B-A6L6	B1	33152	1/13/94
B-A5D9	B1	32039	4/18/93
B-A5E1	B1	30776	4/15/93
B-A5E2	B1	30663	4/15/93
B-A5E3	B1	31954	4/17/93
B-A6K9	B1	33223	4/17/93
B-A5E4	B1	29509	4/16/93
B-A5E5	B1	29448	4/16/93
B-A5E6	B1	32661	4/17/93
B-A6K7	B1	32788	1/13/94
B-A5E7	B1	30016	4/17/93
B-A5B1	B1	29404	4/16/93
B-A5B2	B1	29379	4/16/93
B-A5B3	B1	29845	4/16/93
A-C1E8	B1	30357	4/16/93
B-A5B4	B1	32102	4/18/93
B-A5B5	B1	32348	4/17/93
A-C1E9	B1	30249	4/16/93
B-A5B6	B1	30922	4/18/93
B-A5B7	B1	30632	4/17/93
A-B1B2	R2	29546	5/4/79
A-B1B1	R2	29353	5/2/79
A-B1K2	R2	30091	5/3/79
B-C2C1	R2	30623	5/2/79
A-B1J2	R2	32670	5/3/79
B-C3D4	R2	32661	5/5/79
A-D2J4	R2	30420	5/5/79
A-B1H1	R2	30062	5/5/79
B-D6A5	R2	29713	4/29/79
B-C1B9	R2	40708	9/3/80
B-C1A3	R2	40634	9/4/80
B-D6K6	R2	29985	4/30/79
A-B1E2	R2	30090	5/4/79
B-C2F1	R2	32659	5/5/79
A-D2J5	R2	32656	5/5/79
B-C3C1	R2	30044	5/5/79
A-B1D1	R2	29603	5/4/79
B-D6J6	R2	32786	5/4/79
A-B1G2	R2	33153	5/5/79

A-A1E1	R2	29568	5/2/79
B-C3F2	R2	29840	5/3/79
A-D2E5	R2	32857	5/4/79
A-B1E1	R2	33252	5/4/79
B-C3E2	R2	29761	5/2/79
A-B1J1	R2	30503	5/5/79
B-C3D1	R2	30664	5/5/79
A-B1H2	R2	30362	5/4/79
A-B1K1	R2	33008	5/5/79
B-C3E4	R2	32766	5/5/79
A-B1F2	R2	30154	5/4/79
B-D6K7	R2	29446	4/29/79
B-C1B7	R2	40863	9/2/80
B-C2F4	R2	29916	4/30/79
B-C3D2	R2	29570	5/4/79
B-C2D3	R2	30441	4/30/79
A-B1D2	R2	32803	5/4/79
A-A1C2	R2	33170	5/4/79
A-A1D1	R2	30592	5/5/79
A-A1E2	R2	29626	5/4/79
A-D2J3	R2	29744	5/3/79
A-A1D2	R2	29845	5/3/79
B-C2E4	R2	29211	4/30/79
B-C3C2	R2	29372	5/6/79
A-B1C2	R2	29373	5/6/79
A-B1A1	R2	29680	5/6/79
B-C3E1	R2	29911	9/1/80
A-A1C1	R2	30788	8/31/80
B-D6H6	R2	30332	8/31/80
B-D6G6	R2	30105	8/29/80
B-D6D7	R2	32216	3/21/82
B-D6A6	R2	32251	9/2/80
B-C2D8	R2	32051	9/2/80
B-C2C2	R2	31981	9/2/80
B-D6D5	R2	30017	8/30/80
B-C3F3	R2	29696	9/4/80
B-C2D4	R2	33493	9/1/80
B-C2E5	R2	33274	9/1/80
A-B1C1	R2	29316	9/4/80
B-C2C9	R2	30798	8/30/80
B-C2D9	R2	30923	8/31/80
B-C2C10	R2	30056	8/29/80
B-D6F7	R2	33242	9/1/80
B-C2C4	R2	33358	9/1/80
B-C2F9	R2	29764	8/30/80
B-C4D5	R2	27443	2/25/84
B-D6J7	R2	31974	3/21/82
B-D6G5	R2	29961	9/1/80
B-C2F8	R2	32370	9/2/80
B-C2E10	R2	30231	8/31/80
B-D6F5	R2	30850	8/31/80
B-C2E9	R2	31201	8/31/80
B-C3C4	R2	29851	9/1/80
B-D6B5	R2	32348	9/2/80
A-B1A2	R2	32034	9/4/80

B-C2F10	R2	30249	8/31/80
B-C2C8	R2	32039	9/2/80
B-C2E8	R2	32114	9/2/80
B-C3F1	R2	29954	8/30/80
B-C2F5	R2	33296	9/1/80
B-C2D5	R2	33267	9/1/80
B-C2F3	R2	29942	8/30/80
B-D6H5	R2	30519	8/31/80
B-C1E5	R2	30756	8/31/80
B-C3E3	R2	29606	9/4/80
B-C2F6	R2	32923	9/1/80
B-C2C5	R2	33197	9/1/80
B-C3D3	R2	29497	9/2/80
B-D6E5	R2	29994	8/30/80
B-C4C6	R2	26798	2/26/84
B-D6C5	R2	32023	9/2/80
B-D6E6	R2	32189	9/4/80
B-D6D6	R2	31728	9/2/80
B-D6C6	R2	30062	8/30/80
B-D6B6	R2	29808	9/1/80
B-D6F6	R2	30730	8/31/80
B-C2D10	R2	30151	8/31/80
B-C4C5	R2	29558	3/20/82
B-C4D4	R2	29604	3/20/82
B-C4F6	R2	33102	2/26/84
B-C4C4	R2	32834	2/24/84
B-C3F4	R2	20298	9/5/80
B-D6B7	R2	29850	3/19/82
B-D6B4	R2	30076	3/20/82
B-C4C2	R2	30030	3/20/82
B-D6A7	R2	32689	2/26/84
B-D6D4	R2	32842	2/25/84
B-C3C3	R2	20590	9/5/80
B-C4F3	R2	29601	3/20/82
B-C4F4	R2	29753	3/20/82
B-C4E4	R2	30454	3/20/82
B-C4D3	R2	30411	3/20/82
B-C4C3	R2	29648	3/20/82
B-D6E7	R2	29873	3/20/82
B-D6F4	R2	32816	2/24/84
B-D6C4	R2	32748	2/25/84
B-D6H7	R2	30089	3/20/82
B-D6G7	R2	30232	3/20/82
B-C4E3	R2	30001	3/19/82
B-D6A4	R2	30293	3/19/82
B-C4F5	R2	32962	2/26/84
B-C4E5	R2	32729	2/25/84
B-C4F7	R2	29530	3/19/82
B-D6J5	R2	30530	3/20/82
B-D6K5	R2	30051	3/19/82
B-D6C7	R2	30228	3/19/82
B-C2E2	R2	29734	2/25/84
B-D6E4	R2	29527	2/25/84
B-C4D7	R2	30633	2/24/84
B-C4E7	R2	33927	2/26/84

B-C4D9	R2	31046	2/25/84
B-C4E9	R2	33843	2/24/84
B-C4F9	R2	32490	2/26/84
B-C2D2	R2	29654	2/24/84
B-C1E6	R2	25984	2/25/84
B-D6G4	R2	29949	2/24/84
B-D6H4	R2	29698	2/24/84
B-C1E7	R2	26080	2/24/84
B-C2F2	R2	30931	2/26/84
B-C4D6	R2	29607	2/25/84
B-C4E6	R2	29978	2/26/84
B-C1F3	R2	25878	2/16/86
B-D6G3	R2	32334	2/19/86
B-C1D6	R2	25871	2/18/86
B-D6H3	R2	31470	2/19/86
B-C3C5	R2	33196	2/17/86
B-C3D5	R2	34909	2/19/86
B-D6G1	R2	34894	2/17/86
B-D6H1	R2	33343	2/17/86
B-C3E10	R2	31570	2/16/86
B-C3E9	R2	30290	2/19/86
B-C3E8	R2	33582	2/17/86
B-C1F9	R2	34945	4/19/87
B-C3E5	R2	30444	2/16/86
B-C3F5	R2	30084	2/19/86
B-D6D3	R2	30182	2/15/86
B-C4C9	R2	30887	2/19/86
B-C3E7	R2	33971	2/16/86
B-C3E6	R2	34095	2/17/86
B-D6J1	R2	31043	2/16/86
B-C3F10	R2	33253	2/16/86
B-C3F9	R2	33673	2/18/86
B-C1F6	R2	26076	2/16/86
B-C3F8	R2	34466	2/19/86
B-C1A4	R2	35061	2/16/86
B-C1F4	R2	25994	2/18/86
B-C3F7	R2	32241	2/19/86
B-C3F6	R2	32199	2/15/86
B-C1D3	R2	25643	2/15/86
B-C1F7	R2	34890	2/19/86
B-D6A3	R2	34992	2/16/86
B-C1F5	R2	26032	2/19/86
B-C4C8	R2	33704	2/16/86
B-C4D8	R2	33241	2/19/86
B-C4E8	R2	31720	2/19/86
B-D3A5	R2	35025	4/19/87
B-C2B9	R2	35048	4/20/87
B-C4F8	R2	31524	2/16/86
B-D6J4	R2	30630	2/19/86
B-D6K4	R2	30557	2/16/86
B-C4C7	R2	29666	2/17/86
B-C2E1	R2	34360	2/17/86
B-C1F10	R2	34665	4/19/87
B-D6B3	R2	30490	2/17/86
B-D6E3	R2	30815	2/17/86

B-D6F3	R2	33269	2/16/86
B-C4C10	R2	34775	2/17/86
B-C4D10	R2	33980	2/17/86
B-C4E10	R2	32809	2/19/86
B-C4F10	R2	31005	2/17/86
B-C1D5	R2	25813	2/16/86
B-D6C3	R2	31694	2/17/86
B-C1D4	R2	25777	2/19/86
B-D6A1	R2	32325	4/20/87
B-D6B1	R2	32134	4/19/87
B-D6C1	R2	32039	4/19/87
B-C1C8	R2	33960	11/28/88
B-C1C7	R2	34493	11/27/88
B-D3A6	R2	34588	11/27/88
B-C1E8	R2	34361	11/28/88
B-D6E1	R2	32422	4/18/87
B-D6J2	R2	33222	4/19/87
B-D6K2	R2	33004	4/20/87
B-D6D1	R2	33299	4/18/87
B-D6F1	R2	32148	4/20/87
B-D6H2	R2	32227	4/18/87
B-C3C10	R2	32347	4/20/87
B-C3D10	R2	31208	4/18/87
B-C1E4	R2	34527	11/27/88
B-D3A7	R2	34622	11/28/88
B-C1D8	R2	34005	11/27/88
B-C1E3	R2	23530	4/18/87
B-D6E2	R2	30487	4/20/87
B-D6F2	R2	30914	4/18/87
B-C3C9	R2	30845	4/19/87
B-C3D9	R2	34583	4/19/87
B-D6C2	R2	34830	4/19/87
B-D6D2	R2	34864	4/19/87
B-C3C8	R2	34563	4/19/87
B-C3D8	R2	33522	4/19/87
B-D6A2	R2	34219	4/19/87
B-D6B2	R2	34235	4/20/87
B-C3C7	R2	34128	4/19/87
B-C3D7	R2	34118	4/18/87
B-D6J3	R2	34441	4/20/87
B-D6K3	R2	33573	4/20/87
B-C3C6	R2	33768	4/18/87
B-C3D6	R2	33786	4/19/87
B-C2D1	R2	33734	4/19/87
B-C2C6	R2	33541	4/19/87
B-C2D6	R2	33562	4/19/87
B-C2C3	R2	33972	4/19/87
B-C2E6	R2	33694	4/19/87
B-C2C7	R2	29701	4/19/87
B-C2D7	R2	29711	4/20/87
B-D6G2	R2	29778	4/18/87
B-C2E7	R2	30054	4/18/87
B-C2F7	R2	33506	4/19/87
B-C2E3	R2	33820	4/19/87
B-C3A1	R2	36727	10/1/90

B-C1D7	R2	34493	11/27/88
B-C2A9	R2	34577	11/27/88
B-C2A6	R2	34800	11/28/88
B-C1F8	R2	34417	11/28/88
B-C2A10	R2	34631	11/27/88
B-C3B1	R2	34756	11/28/88
B-C3B2	R2	34785	11/28/88
B-C1B8	R2	35217	11/27/88
B-C2A8	R2	33752	11/27/88
B-C1D10	R2	32631	11/27/88
B-C1E9	R2	33566	11/28/88
B-C1E10	R2	32661	11/28/88
B-C1D9	R2	33492	11/27/88
B-C2A7	R2	33206	11/27/88
B-C1C9	R2	33348	11/28/88
B-C1B5	R2	36755	11/27/88
B-C2B2	R2	36773	11/28/88
B-C1B10	R2	36031	11/27/88
B-C1C3	R2	36082	11/27/88
B-C1B4	R2	36728	11/27/88
B-C1C5	R2	36311	11/27/88
B-C2B7	R2	37118	11/28/88
B-C2B6	R2	36979	11/27/88
B-C1B3	R2	36507	11/28/88
B-C2B5	R2	36881	11/27/88
B-C1C10	R2	31831	11/28/88
B-C1C6	R2	36502	11/27/88
B-C2B4	R2	36873	11/28/88
B-C2B1	R2	38186	11/27/88
B-C1A9	R2	38043	11/28/88
B-C2B10	R2	38420	11/28/88
B-C2A1	R2	38601	11/27/88
B-C2B8	R2	37545	11/28/88
B-C1A6	R2	37672	11/27/88
B-C2B3	R2	36860	11/27/88
B-C1A5	R2	37665	11/27/88
B-C1A8	R2	38012	11/27/88
B-C2A2	R2	38722	11/28/88
B-C1A10	R2	38184	11/28/88
B-C1A7	R2	37680	11/27/88
B-C3A7	R2	21682	9/27/93
B-C3A8	R2	21470	9/25/93
B-C3A9	R2	21599	9/27/93
B-C3A10	R2	21779	9/26/93
B-C4A1	R2	23791	9/27/93
B-C4A2	R2	23471	9/25/93
B-C1A2	R2	23573	9/27/93
B-C3A3	R2	37573	9/30/90
B-C3A2	R2	37572	9/29/90
B-C3A4	R2	37676	9/30/90
B-C3A5	R2	38007	9/30/90
B-C2A4	R2	36489	9/30/90
B-C2A3	R2	36370	9/30/90
B-C3B10	R2	36611	9/30/90
B-C2A5	R2	36507	9/30/90

B-C4F2	R2	39092	9/30/90
B-C4D1	R2	39404	9/30/90
B-C4E2	R2	39367	9/30/90
B-C4E1	R2	39275	9/30/90
B-C4F1	R2	39065	9/30/90
B-C1B2	R2	39062	10/1/90
B-C1D2	R2	39047	10/1/90
B-C1C2	R2	39049	9/30/90
B-C1F2	R2	38898	9/29/90
B-C4B2	R2	38806	9/30/90
B-C1E2	R2	38915	10/1/90
B-C4B1	R2	38754	9/30/90
B-C3B5	R2	38469	9/30/90
B-C3B7	R2	38682	9/30/90
B-C3B3	R2	38357	10/1/90
B-C3B6	R2	38639	10/1/90
B-C3B4	R2	38401	9/30/90
B-C3B8	R2	38708	9/29/90
B-C3B9	R2	38719	9/29/90
B-C3A6	R2	38115	9/30/90
B-D3A3	R2	39861	9/30/90
B-D3A4	R2	39914	9/30/90
B-C1B6	R2	39722	9/30/90
B-C4D2	R2	40000	9/30/90
B-C4C1	R2	40046	9/30/90
B-D3A1	R2	39790	9/29/90
B-D3A2	R2	39813	9/29/90
B-C1C4	R2	39758	9/30/90
A-D2H4	R2	38941	4/21/92
A-D2H6	R2	38945	4/21/92
A-A2D6	R2	39328	4/21/92
A-D2K3	R2	38791	4/22/92
A-A1G2	R2	38645	4/21/92
A-D2K4	R2	38854	4/22/92
A-D2H3	R2	38932	4/21/92
A-A1H2	R2	38685	4/22/92
A-D2G6	R2	39127	4/21/92
A-D2K5	R2	38925	4/22/92
A-A2C6	R2	39179	4/22/92
A-A1K2	R2	38732	4/22/92
A-D2E4	R2	42704	4/21/92
A-D2E3	R2	42433	4/22/92
A-D2E2	R2	42004	4/22/92
A-D2D6	R2	40931	4/22/92
A-D2A6	R2	41087	4/22/92
A-D2F2	R2	41421	4/21/92
A-D2B6	R2	41028	4/21/92
A-D2G2	R2	41288	4/22/92
A-D2F1	R2	41368	4/22/92
A-D2G1	R2	41272	4/22/92
A-D2C6	R2	40980	4/22/92
A-A2F6	R2	40361	4/22/92
A-A2E6	R2	40289	4/22/92
A-A2D5	R2	40446	4/22/92
A-D2E6	R2	40736	4/22/92

A-A2F5	R2	40620	4/21/92
A-D2F5	R2	40686	4/21/92
A-A2C5	R2	40441	4/22/92
A-A2E5	R2	40569	4/22/92
A-D2G3	R2	41327	4/21/92
A-D2E1	R2	41785	4/21/92
A-A1F2	R2	38633	4/22/92
A-A1G1	R2	38563	4/21/92
A-A1F1	R2	38490	4/21/92
A-A1J2	R2	38706	4/22/92
A-D2F4	R2	41637	4/22/92
A-D2F3	R2	41509	4/22/92
B-D1E6	R2	38557	9/26/93
A-D2D1	R2	38202	9/26/93
A-D2D3	R2	38382	9/26/93
A-D2D2	R2	38331	9/26/93
B-D1G6	R2	38976	9/26/93
B-D1F6	R2	38869	9/27/93
B-D1E5	R2	39531	9/25/93
B-D1H6	R2	39064	9/27/93
B-D1J6	R2	42521	9/26/93
B-D1K6	R2	39120	9/26/93
B-D1B5	R2	39253	9/26/93
B-D1C5	R2	39508	9/26/93
B-D1G5	R2	39576	9/26/93
B-D1H5	R2	39603	9/26/93
B-D1J5	R2	39616	9/27/93
B-D1K5	R2	39620	9/26/93
B-D1F5	R2	39545	9/26/93

6.4 HABITABILITY SYSTEMS

The Control Room Habitability Systems include equipment, supplies and procedures which give assurance that the control room operators can remain in the Control Room and take effective actions to operate the nuclear power plant safely under normal conditions and maintain the facility in a safe condition following a postulated accident as required by the General Design Criterion 19 contained in Appendix A to 10 CFR 50.

The habitability systems and provisions include:

a) Control Room Air Conditioning System (which includes the Emergency Filtration System).

b) Radiation protection

c) Food and water storage

d) Kitchen and sanitary facilities

e) Breathing apparatus

The above systems and provisions provide adequate operator protection under normal and emergency operating conditions (including the design basis loss of coolant accident) and postulated release of toxic gases and smoke.

6.4.1 DESIGN BASIS

The habitability systems for the control room include shielding, air handling and filtration systems, temperature control, dehumidifiers, instrumentation to protect against airborne radioactivity, air breathing apparatus, sufficient storage for food and water, and other provisions for extended occupancy by control room personnel, including kitchen and sanitary facilities.

The bases upon which the functional design of these systems and provisions are designed include the following:

Control Room Envelope:

The control room envelope includes, in addition to the Control Room, the following auxiliary spaces:

a) Office areas

b) Relay and termination cabinet rooms

c) Kitchen and sanitary facilities

d) Component cooling water surge tank room

Period of Habitability - The period of habitability for control room operators is based on the habitability systems' capability to provide protection from the introduction into the control room envelope of airborne contaminants that present an immediate danger to life or health. The most severe hazards are posed by airborne radioactivity. After the detection of airborne radioactivity the control room envelope will be pressurized and all air will be filtered through charcoal adsorbers. This system will ensure that the control room operators will not receive doses of radiation in excess of the limits specified in GDC 19 of Appendix A to 10 CFR 50 during the time required for the safe shutdown of the plant.

Capacity - The Control Room has been designed (1) to allow continuous occupancy of five persons for a seven-day period following a design basis accident and (2) for replacement of the crews following the seven days. This includes sufficient food, water, medical supplies and sanitary facilities.

Food, Water, Medical Supplies, and Sanitary Facilities - For habitability of the Control Room during certain emergencies, a seven-day supply of food and potable water is provided within the control room area.

Basic medical supplies, kitchen and sanitary facilities are provided within the control room envelope.

Radiation Protection - The Control Room envelope has been designed to ensure continuous occupancy during normal operation and extended occupancy throughout the duration of any one of the following postulated design basis accidents:

a) Loss of coolant accident (LOCA)

b) Fuel handling accident

c) Radioactive releases due to radwaste system failure

The radiation exposures shall not exceed 5 rem whole body for the duration of any of the above accidents.

As documented in the SHNPP SER (NUREG 1038 Supplement 2), the postulated design basis LOCA event has been established as the most limiting event for demonstrating compliance with the Control Room Habitability Dose Criteria. Dose to the Control Room personnel resulting from a LOCA is discussed in Section 15.6.5.4.4.

Respiratory, Eye, and Skin Protection for Emergencies - An adequate number of respirators is provided in the Control Room for emergency use.

Habitability System Operation During Emergencies

The Control Room Air Conditioning System is safety related and designated as Safety Class 3 and Seismic Category I. The system is capable of performing its functions assuming an active component single failure.

The air conditioning system will not promote the propagation of smoke and fire from other areas in the Reactor Auxiliary Building to the control room envelope. Refer to Section 9.5.1 for a discussion of fire protection criteria for the Control Room. Provisions have been made for control room smoke purge operation, as described in Section 9.4.1.2.3.

The system has been designed to maintain the ambient temperature in the Control Room at 75 F DB and 50 percent (max.) relative humidity during normal conditions and a design basis accident.

During a postulated LOCA, the Control Room is pressurized to 1/8 in. wg. by the capability of introducing a maximum of 400 cfm outside air into the Control Room which will keep the carbon dioxide and oxygen concentrations within safe levels. Calculations of CO_2 and O_2 concentrations within the Control Room consider that the concentrations of these gases are homogenous within the control room envelope, excluding the air above the hung ceiling. Design maximum concentration of carbon dioxide is taken as 1.0 percent. Design minimum concentration of oxygen will be taken as 17 percent.

The Control Room has been designed to protect the control room operators from all design basis natural phenomena and design basis accidents.

Emergency Monitors and Control Equipment

Provisions have been made to detect radioactivity and smoke in the Control Room air intake. Following detection, the control room envelope is automatically isolated. Sensitivities of the detectors and isolation time including delays in the control circuits are designed to meet the requirements of GDC 19.

6.4.2 System Design

6.4.2.1 Control Room Envelope. The control room envelope includes those areas listed in Section 6.4.1. During an emergency, the areas which the control room operator could require access to are the Control Room, office areas, kitchen and sanitary facilities and control room emergency air intake valves located in the relay and termination and cabinet rooms.

6.4.2.2 Ventilation System Design. The control room envelope air conditioning process includes an environmental control operation and an emergency air cleanup operation. The environmental control operation is the primary function of the air conditioning system and it is accomplished by the use of redundant air conditioning trains. The Control Room will be isolated upon receipt of a safety injection signal or following a detection at the intake opening of radioactivity or smoke. Redundant, motorized butterfly valves are provided in the control room envelope outside air intake and exhaust ducts for automatic isolation of the system from the surrounding atmosphere. Consequently, the normal ventilation system is not expected to have any adverse effects on the control room habitability during a design basis accident.

Redundant trains of the Control Room Air Conditioning System are provided for the system to fulfill its essential functions. The redundant filtration train is located in a separate equipment room. The system is located within the Reactor Auxiliary Building which is designed to withstand effects of the safe-shutdown earthquake and other design basis natural phenomena.

To assure continued operation following a design basis accident, the Control Room Air Conditioning System is designed to Seismic Category I requirements. This includes equipment and ductwork up to and including the connection into the Control Room (except portions of the normal exhaust and smoke purge fans). The air intakes and exhaust of the Control Room Areas Ventilation System are tornado and missile protected.

Active system components meet the single failure criteria as described in IEEE-279-71. Refer to Table 9.4.1-4 for a single failure analysis of the Control Room Air Conditioning System.

The redundant air conditioning units are served by separate Essential Services Chilled Water Systems so that loss of one train of the chilled water systems will not affect the ability of the system to control the thermal environment in the control room envelope.

The Control Room Area Ventilation System including equipment, ductwork, valves, and air flows for both normal and emergency modes is discussed in detail in Section 9.4.1. Design data for principal components of the Control Room Area Ventilation System are presented in Table 9.4.1-1. The airflow diagram for the Control Room Area Ventilation System is shown on Figure 9.4.1-1.

The Emergency Filtration System is discussed in Section 9.4.1.2. The operational status of valves, fans and corresponding airflow rates for the Control Room Air Conditioning System and Emergency Filtration System are presented in Table 9.4.1-2. The design data is presented in Table 9.4.1-1.

The degree of compliance of the Emergency Filtration System with the requirements of Regulatory Guide 1.52 is discussed in Section 6.5.1.

The layout drawings of the Control Room showing doors, corridors, stairwells, shield walls, and the placement and type of equipment within the Control Room are shown on Figure 1.2.2-35. Elevation and plan views showing building dimensions and the location of control room air intakes are also presented on Figure 1.2.2-35.

Under a completely isolated Control Room, occupied with up to ten people, the CO_2 concentration would build up from zero to one percent in 71 hours. This buildup time is based upon a net control room envelope of 0.71 x 10^5 ft^3 which includes space above the egg crate hung ceiling and a breathing rate of 30 ft^3/hr to generate 1 ft^3/hr CO_2 per person. Considering there are no postulated design conditions which would require that the control room envelope be isolated for an extended period of time, 71 hours provides more than adequate time of the operator actions required to reestablish control room ventilation.

A ventilation rate of 3.4 cfm fresh air per person will maintain the carbon monoxide level in the control room below 0.5 percent. Since the emergency pressurization mode of the Control Room Ventilation System permits the continuous introduction of up to 400 cfm (outside air from the uncontaminated air intake) through the control room emergency air cleaning unit, there will be no excessive build up of CO_2 in the control room. The actual pressurization flow rate will be determined by testing to maintain a positive pressure differential of 1/8 inch of water gauge.

A ventilation rate of 0.5 cfm fresh air per person will maintain the oxygen level in the Control Room at 17 percent, min. The design ventilation rate capability of up to 400 cfm is therefore adequate.

Smoke purge fans are provided to expedite fire fighting efforts. Refer to Section 9.4.1.2.3 for a detailed discussion of the smoke purge operation.

Adequate bottled air capacity (of at least six hours) is readily available onsite for the five Control Room occupants to assure that sufficient time is available to locate and transport bottled air from offsite locations. This offsite supply is capable of delivering several hundred hours of bottled air to the members of the emergency crew.

6.4.2.3 Leak Tightness. The control room envelope is pressurized to 1/8 in. of water gauge differential pressure relative to the adjacent areas at all times during normal plant operation and outside air is continuously introduced to the control room envelope. During a postulated LOCA, a maximum rate of 400 cfm may be required in order to maintain 1/8 inch of water gauge. The control room is automatically isolated following a design basis radionuclide accident. In case of a radionuclide accident, the operator will re-pressurize the control room by drawing in filtered outside air through one of two emergency air intakes. The 400 cfm pressurization flow rate is approximately 0.34 volume change per hour.

All openings to the Control Room have a low leakage design. This includes doors, valves, penetrations and walls. The control room leakage rate estimate through valves, doors, penetrations and walls is shown in Tables 6.4.2-1 and 6.4.2-2. The estimate is based on AEC R&D Report NAA-SR-101000.

A maximum of 400 cfm makeup air will not make the overall doses to the control room operator exceed the radiation dose limit of General Design Criterion 19 of Appendix A to 10 CFR 50 under design basis accidents. An acceptance test is performed at startup to verify that the control room leakage rate is less than the value assumed in the analysis.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment. The following provisions are taken into consideration in the Control Room Area Ventilation System design to assure that there are no toxic or radioactive gases and other hazardous material that would transfer into the Control Room:

a) The control room envelope is pressurized to 1/8 in. w.g. relative to the adjacent areas.

b) The Control Room Area Ventilation system is independent and completely separated from other adjacent ventilation zones.

c) There is no other HVAC equipment within the Control Room envelope that serves other ventilation zones.

d) All doors, duct and cable penetrations are of low leakage design.

6.4.2.5 Shielding Design

The Control Room envelope is shielded against direct sources of radiation which are present during normal operating conditions and following a postulated accident.

There are no significant sources of direct or streaming radiation near the
control room envelope during normal operating conditions. The shielding walls
and floor provided for the accident conditions are more than sufficient to
limit the dose rates to less than 0.25 mr/hr. in the Control Room during
39 normal operation. Refer to Section 12.3.2.14 for a discussion of the control
room shielding design.

TABLE 6.4.2-1

CONTROL ROOM BUTTERFLY VALVES LEAKAGE RATE ESTIMATE

1. COMPONENTS:	Butterfly valves in: a) Exhausts b) Normal Outside Air Intake	
SIZE:	12 inch diameter (exhaust) 16 inch diameter (intake)	
QUANTITY:	Four[1] (2 valves arranged in series in each of two paths)	39
LEAK RATE AT 13.8 PSIG:	0.018 (0.024) cubic feet per day per exhaust (intake) valve[2]	
LEAK RATE AT + 1/8 INCH W.G.[3]:	0.53×10^{-6} cfm per two valves	39
2. COMPONENTS:	Butterfly valves in: a) Purge Exhausts b) Purge Make-Up	
SIZE:	30 inch diameter (exhaust) 36 inch diameter (make-up)	
QUANTITY:	Four[1] (2 valves arranged in series in each of two paths)	39
LEAK RATE AT 13.8 PSIG:	0.045 (0.054) cubic feet per day per exhaust (make-up) valve[2]	
LEAK RATE AT + 1/8 INCH W.G.[3]:	1.24×10^{-6} cfm per two valves	39
3. COMPONENTS:	Butterfly valves in: Post-Accident Air Intakes (two)	
SIZE:	12 inch diameter	
QUANTITY:	Four[1] (2 valves arranged in series in each of two paths)	39
LEAK RATE AT 13.8 PSIG:	0.018 cubic feet per day per valve[2]	
LEAK RATE AT + 1/8 INCH W.G.[3]:	0.45×10^{-6} cfm per two valves	39
TOTAL LEAKAGE TO THE OUTSIDE FROM VALVES:	1) 0.53×10^{-6} cfm 2) 1.24×10^{-6} cfm 3) 0.45×10^{-6} cfm (For conservatism, 3.0×10^{-6} cfm is used.)	39
TOTAL =	2.22×10^{-6} cfm	39

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TABLE 6.4.2-1 (cont'd)

NOTES:

1. There are a total of 12 isolation valves, two in series in each air path. However, it has been assumed that only one valve closes in each path following control room isolation. 39

2. Based on AEC R&D Report NAA-SR-101000, Reference 2, Section A-2, p III-105.

3. For control room positive pressure +1/8 inch w.g.

TABLE 6.4.2-2

SUMMARY OF MAIN CONTROL ROOM LEAK RATE CALCULATION[1]

PATH NO.	COMPONENT	UNIT	NUMBER OF UNITS	NUMBER OF REFERENCE DETAIL (1)	LEAKAGE A	COEFFICIENT B	LEAKAGE PER UNIT AP+BP1/2(2)	TOTAL CFM COMPONENT LEAKAGE
1.	Hollow metal door, metal interlocking gasketed weatherstripping, door opening in (4 single and 1 double)	3'-0x7'-0	6	ADS III-A-2	4.0	22.0	8.28	49.68
2.	Door Frames	Ft.	106	ADS I-A-7	$4x10^{-6}$	0	$5x10^{-7}$	.00006
3.	Walls	$Ft.^2$	6,000	ADS I-A-2(1)	$1x10^{-6}$	0	$1.25x10^{-7}$	.00075
4.	Slab	$Ft.^2$	10,800	ADS I-A-2(1)	$1x10^{-6}$	0	$1.25x10^{-7}$	.00135
5.	Juncture of floor slab and wall	Ft.	450	ADS I-A-3(1)	$1.6x10^{-3}$	0	$.2x10^{-4}$	.09
6.	Eave	Ft.	450	ADS I-A-5 Case 2	$6x10^{-5}$	0	$.75x10^{-5}$	.0034
7.	Corners, columns and wall joints with caulking	Ft.	340	ADS I-A-6 Case 1	$1.6x10^{-5}$	0	$.2x10^{-5}$	.0007
8.	Penetrations for electrical cables	Ft.	730	ADS III-D-1	$1.3x10^{-4}$	0	$.1625x10^{-4}$	.0118
9.	Penetrations for HVAC ducts	In. of Seal	1,040	ADS III-D-1 Case 2	$1.3x10^{-5}$	0	$.1625x10^{-5}$	.00169
10.	Isolation Butterfly Valves			ADS A-2 Case 2				$3x10^{-6}$(4)
11.	Pipe Penetrations	In. of Seal	116	ADS III-D-1 Case 2	$1.3x10^{-5}$	0	$.1625x10^{-5}$	.0002

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12.	HVAC Equipment and Ductwork (Outside of Envelope)		15.8
Subtotal (1-12)			66 x 2(1)
13.	Opening and closing of doors	Note (3)	10.00
Total			142

39

(1) Based on AEC R+D Report NAA-SR-10100
(2) Leakage estimate based on AP=0.125 in w.g.
(3) See standard review plan Section 6.4 III3d2ii
(4) See Table 6.4.2-1

9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM 39

The Control Room Area Ventilation System consists of an Air Conditioning System and an emergency filtration system to serve the Control Room.

Additional details of the Control Room Area Ventilation and habitability systems are given in Section 6.4.

9.4.1.1 Design Bases

The Control Room Air Conditioning System (CRACS) provides heating, ventilation, cooling, filtration, air intake and exhaust isolation, and 50 percent relative humidity for the control room envelope (as described in Section 9.4.1.2) during normal operation and a design basis accident. 39

Systems are designed to include the effects of the most adverse single active component failure.

The high energy piping systems outside containment are listed and described in Section 3.6A.2. Those listed systems are the CVCS, SGBS, FWS and the AFS. A discussion of the analysis is provided for those systems in Section 3.6A.2 that demonstrate that the Control Room will not be adversely affected by a pipe crack or break in an adjacent area. Figure 3.6A-2 shows the only high energy piping adjacent to the Control Room with indication of break locations and jet impingement envelopes.

The steam lines near the Control Room (elevation 305 feet) are located in the turbine building at elevation 286.00 feet and separated by the Category I wall of the RAB. The Control Room, including outside air intakes for control room ventilation will not be adversely affected by any high energy pipe breaks. The main steam line may be slightly radioactive but as discussed in Sections 15.1.5, 6.4.4 and 9.4.1 radiological impact would be limited to below acceptable levels by ventilation systems and wall shielding.

The Control Room Air Conditioning is designed:

a) To maintain the Control Room at a design temperature of 75 F dry bulb and maximum relative humidity not to exceed 50 percent, assuring personal comfort as well as suitable environment for continuous operation of controls and instrumentation.

b) To detect the introduction of radioactive material into the Control Room and automatically isolate all air intakes and exhausts upon a high radiation signal or SIS signal and to remove airborne radioactivity from the Control Room to the extent that dose to the Control Room operator following a design basis accident does not exceed the limit specified in the General Design Criteria 19. 39

c) To operate in conjunction with the Fire Detection System to remove smoke from the Control Room in the event of fire.

d) To be powered by the redundant Channel A and Channel B ESF buses.

e) To meet single component active failure in the system or a failure in a single emergency power supply coincident with loss of offsite power.

f) To meet Safety Class 3 and Seismic Category I requirements and to be tornado and missile protected. The system is physically separated from all other systems by walls and doors.

g) To permit testing, adjustment, and inspection of the principal system components on a regular basis to assure system functional reliability.

9.4.1.2 System Description. The control room envelope, which is referred to as the "Control Room," includes, in addition to the Control Room, the following auxiliary spaces.

a) Office area

b) Relay and termination cabinet rooms

c) Kitchen and sanitary facilities

d) Component cooling water surge tank room

The computer rooms, protection and control equipment rooms, communication rooms, instrument repair room, and cable spreading rooms located in the Reactor Auxiliary Building are ventilated and cooled by independent cooling systems.

The Control Room Air Conditioning System is shown on Figure 9.4.1-1 and principal system components design data are listed and described in Table 9.4.1-1.

The CRACS is designed to maintain the control room envelope at a design temperature of 75°F under normal and Design Bases Accident conditions. Space relative humidity is controlled not to rise above 50 percent. The air is cooled by a cooling coil. The chilled water supply to the cooling coil is provided by the Essential Services Chilled Water System (see Section 9.2.8). When heating is required, the air is heated by the electric heating coils to maintain the design space temperature stated above.

The Control Room Air Conditioning System consists of the following:

a) The supply system consists of two 100 percent capacity air handling units (one operating and one stand-by). The air handling units are arranged 22
in parallel. Each air handling unit includes, in the direction of air flow, a motorized isolation damper, medium efficiency filter, chilled water cooling coil, centrifugal fan, electric reheat coil, and motorized butterfly valve.

An outside air intake is provided for the supply system. The air intake is provided with two motorized isolation valves arranged in series and is protected from the tornado negative pressure effects by a self-acting tornado damper. The intake is stormproofed and missile protected.

b) The exhaust system consists of two 100 percent capacity (one operating and one stand-by) fans. Each fan is provided with an inlet and an outlet motorized damper to prevent air recirculating through the idle fan.

The exhaust system is provided with two motorized isolation valves arranged in 27
series and protected from tornado negative effects by means of a self-acting
tornado damper. The exhaust system is stormproofed and missile protected. 27

c) The smoke purge system consists of two 100 percent capacity (one operating and one stand-by) purge fans. Each purge fan is provided with a
gravity discharge damper to prevent air recirculation. The purge system inlet 27
is provided with redundant motorized isolation valves arranged in series.

d) The Emergency Filtration System consists of two 100 percent capacity filtration trains arranged in parallel. Each filtration train contains in the direction of airflow, the following components:

1) One motorized valve
2) One centrifugal fan
3) One motorized valve
4) One flow element 27
5) A Demister
6) One electric heating coil
7) HEPA pre-filter bank
8) Charcoal adsorber bank
9) HEPA after-filter bank
10) One motorized isolation valve
11) Connecting duct to other unit discharge 27

For post-accident operation, each Emergency Filtration System can be 27
independently fed from either of the emergency outside air intakes (intake points 10 or 11A) as shown on Figure 9.4.0-2.

9.4.1.2.1 Normal Operation

During normal operation, the Control Room Air Conditioning System operates in a recirculation mode with the Emergency Filtration System de-energized. The outside makeup air mixes with the returned air before it is conditioned by the air handling units. The Control Room is maintained at a slightly positive pressure with respect to the adjacent area so that the air from other sources entering the Control Room is minimized. The pressurization of the Control Room is maintained automatically by means of modulating exhaust fan dampers.

Each of the two supply air handling units are served by common supply and return ductwork with all necessary accessories to make the system complete and operable. All chilled water to the cooling coils is provided by Essential Services Chilled Water System (see Section 9.2.8).

9.4.1.2.2 Post Accident Operation

Upon receipt of a SIS or high radiation at the normal outside air intake, a Control Room Isolation signal is generated and the following functions are performed automatically:

a) All isolation valves at the normal outside air intake will close.

b) All isolation valves in the Control Room Smoke Purge System will close (these valves are normally closed).

c) Both emergency filtration units will start and their respective valves will open for full-recirculation mode.

d) All isolation valves in the Normal Exhaust System will close and the operating exhaust fan will be de-energized.

e) Dampers for lavatory and kitchen which normally by-pass the return system will open.

Following the completion of the above automatic functions, the operator will perform the following tasks:

a) Place one of the two emergency filtration trains on standby.

b) Select and open one emergency outside air intake path during radiological accident to pressurize the Control Room (these paths are normally closed).

During the radiological accident, the Emergency Filtration System will process a mixture of Control Room air and a small quantity of outside air through charcoal filters and maintain the control room envelope under positive pressure of + 1/8 in. wg. Air is continuously drawn from the supply air 40
subsystem blended with outside air, processed through the charcoal filtration system and supplied to the Control Room. A balance valve is provided at the point of initial takeoff from the supply air subsystem. 40

9.4.1.2.3 Smoke Purge Operation

In the event of a fire in any area of the control room envelope, redundant smoke detectors will shutdown the system and actuate an alarm for operator action. The operator will remote manually execute the smoke purging operation by changing the operational status of valves, dampers, and fans and the corresponding air flow rates according to those indicated in Table 9.4.1-2 when smoke is generated in the control room. 40

Thus, the airflow pattern of normal exhaust system supplemented by the purge system becomes a once-through type. An outside air purge makeup intake is located at the South Wall of the Reactor Auxiliary Building Elevation 317 ft. (intake point 6); purge vents are connected to the local stub stack at Elevation 346 ft. (release point 1). Purge fans are located outside the control room envelope boundary.

When the smoke is detected in the outside air intake, the system will automatically switch to the isolation mode.

Valve and damper position-indication lights will allow continuous monitoring of the system performance and will confirm all remote manual control actions taken.

9.4.1.3 Safety Evaluation

Continued operation of the Control Room Air Conditioning System (CRACS) during both normal and emergency conditions to maintain the Control Room temperature and humidity will be assured by the following:

a) System components are designed to meet Safety Class 3 and Seismic Category I requirements with the exception of the normal and smoke exhaust fan systems.

b) System components are separated and redundant so that a single failure of an active component in a system or the single failure of an emergency power supply coincident with loss of offsite power cannot result in loss of the system functional performance capability. A system single failure analysis is presented in Table 9.4.1-4.

c) During loss of offsite power, all active components, such as valve and damper operators, fan motors, control and instrumentation will be served by their respective redundant emergency power sources.

d) The CRACS is designed to ensure that any postulated single failure will not adversely affect the capability of the system to satisfy its design objectives. A single failure of an electric heating coil unit or cooling unit will not prevent the control room HVAC system from completing its safety function (refer to Table 9.4.1-4). Butterfly valves and/or dampers are provided to isolate flow through affected heating/cooling units as necessary. Redundant units are provided to assure adequate cooling or heating as required. Malfunctioning HVAC equipment can be readily identified and isolated from the Control Room. There is no significant effect to the control room environment from the isolated malfunctioning train. Heating and cooling equipment for the Control Room are remotely located and are not in the Control Room.

In any event, the design of the plant is such that the Control Room can be evacuated and the plant can be maintained in a safe condition from the auxiliary control panel (refer to FSAR Section 7.4.1.11). The auxiliary control panel area is serviced by totally independent HVAC units.

e) The ventilation system has sufficient redundancy to preclude inadequate heating or cooling as described in Section 7.3.1.5.7 and as shown on Figure 7.3.1-17 and Table 9.4.1-4.

The adequacy of the CRACS to limit the radiation doses to control room personnel is demonstrated in Chapter 15, Section 15.6.5.4.4.

Detection of radioactivity in the control room environment is provided by radiation monitors as described in Section 11.5 and 12.3.4. This system permits immediate and automatic isolation of the control room normal and emergency outside air intake and exhaust ducts upon receipt of a high radiation signal and enables the operator to select the least contaminated emergency outside air intake for control room pressurization. Adjustable high radiation alarms are provided to alert the operator of changes in contamination levels at both post-accident air intakes. The control room area ventilation system isolates all paths to the environment upon receipt of a high radiation signal as described in FSAR Section 6.4.3.

Smoke detectors in the Control Room will actuate an alarm so that the operator can initiate the smoke purge operation in the event of a fire.

See Section 9.5.1 for the protection of the CRACS from the effects of fire.

9.4.1.4 Inspection and Testing Requirements

The CRACS undergoes preoperational and startup tests as described in Section 14.2.12.1.58. Periodic tests are required as described in the Technical Specifications. Inservice inspection requirements are described in Section 6.6. Valve testing requirements are described in Section 3.9.6.

TABLE 9.4.1-1

DESIGN DATA FOR CONTROL ROOM AIR CONDITIONING SYSTEM COMPONENTS

A. Air Supply Units, Quantity	2 AH-15 (1A-SA) (1B-SB)	27
1. Unit Fan Section		
Quantity, Total	1	
Type	Centrifugal, Belt Driven	
Air Flow, Per Fan, acfm	14,000	27
Code	Air Movement and Control Association (AMCA) Anti-Friction Bearing Manufacturers Association (AFBMA)	
2. Motors		
Quantity, Total	1 per fan section	27
Type	20 Hp, 460 V, 60 Hz, 3 phase, Horizontal Induction Type	
Insulation	Class H Type RH	27
Enclosure	TEFC	
Code	NEMA, Class 1E	27
3. Cooling Coils		
Quantity, Total Banks	1 per unit	27
Type	Chilled water, finned tube	
Material	Copper fin on copper tube steel headers	
Code	ASME III, Code Class 3	

TABLE 9.4.1-1 (Continued)

4.	Heating Coils		
	Quantity, Total	1 per unit	27
	Type	Electric	
	Capacity (kW) Per Coil	56	
	Code	Underwriters Laboratories (UL), National Electrical Manufacturers Association (NEMA), National Electric Code (NEC), Class 1E	
5.	Medium Efficiency Filters		
	Quantity, Total Banks	1 per unit	27
	Type	Extended media	
	Material	Glass fiber	
B.	Emergency Filtration Units, Quantity	2 R2 (1A-SA) (1B-SB)	27
1.	Fans		
	Quantity, Total	1 per unit	
	Type	Centrifugal, single width, single inlet, direct drive	27
	Air Flow, Per Fan, acfm (max)	4000	
	Code	Air Movement and Control Association (AMCA), Anti-Friction Bearing Manufacturer Association (AFBMA)	

TABLE 9.4.1-1 (Continued)

2.	Motors	
	Quantity, Total	1 per fan
	Type	25 Hp, 460 V, 60 Hz, 3 phase, Horizontal Induction Type
	Insulation	Class H, Type RH
	Enclosure and Ventilation	TEFC-XT
	Code	NEMA, Class 1E
3.	HEPA Filters	
	Quantity, Total Banks	2 per unit
	Cell size	24 in. high, 24 in. wide, 11-1/2 in. deep
	Max. resistance clean, in. wg.	1.0
	Max. resistance loaded, in. wg.	2.0
	Efficiency	99.97 percent when tested with 0.3 micron DOP
	Material	Glass or glass asbestos paper separated by aluminum inserts, supported on cadmium plated steel frame
4.	Charcoal Adsorbers	
	Type	Multiple gasketless bed cells in air-tight housing
	Quantity, Total	1 per unit
	Material	Impregnated coconut shell (Meeting the requirements of ANSI-N509-1980, Table 5.1)
	Depth of bed (in.)	4 in.
	Face velocity (fpm)	40
	Average atmosphere residence time	0.25 seconds per 2 in. of adsorber bed

TABLE 9.4.1-1 (Continued)

4. **Charcoal Adsorbers (continued)**		
Adsorber capacity of iodine loading	2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon	
Efficiency:		
Elemental iodine	99% at 70% RH	
Organic iodine	99% at 70% RH	
Adsorbent Acceptance and Inplace Leak Test Criteria (See Table 6.5.1-2)	Carbon Laboratory Acceptance Testing will be performed in accordance with, and will meet the requirements of Position 6 of R.G. 1.52, Revision 2	27
	Adsorber Inplace Leak Testing will be performed in accordance with, and will meet the requirements of Position C.5.d of R.G. 1.52, Revision 2	
5. **Electric Heater**		
Quantity	1 per unit	27
Capacity (kW) Per Coil	14 sufficiently sized to reduce the relative humidity of the inlet air from 100% to 70%	
Code	Class IE	
6. **Demister**		
Quantity, Per Unit	1 bank	27
Air Flow acfm	4000	
Max. resistance clean, in. wg.	1.05	27
Max. resistance loaded, in. wg.	2.0	
Material	304 stainless steel casing and glass fiber mesh	

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TABLE 9.4.1-1 (Continued)

C.	Exhaust Systems, Quantity	1
1.	Exhaust Fans (E-9) (1A and 1B)		27
	Type	Centrifugal
	Quantity	2, One Standby
	Capacity acfm Per Fan	1000
	Motor HP	1
	Motor Code	NNS
D.	Purge Systems, Quantity	1
1.	Purge Fans (ES-1) (1A and 1B)		27
	Type	Tubular Centrifugal High Temperature
	Quantity	2, One Standby
	Capacity acfm Per Fan	13,400	27
	Motor HP	25
	Motor Code	NNS

TABLE 9.4.1-2

CONTROL ROOM AIR CONDITIONING AND EMERGENCY FILTRATION SYSTEM (1)
OPERATIONAL STATUS OF VALVES, DAMPERS, AND FANS AND CORRESPONDING AIR FLOW RATES

SYSTEMS	SIZE	VALVES DAMPERS FANS DESIGNATION	NORMAL OPERATION		PURGE OPERATION[2]		CL_2 ACCIDENT[2] OPERATION		HI RAD. SYS.[2] OPERATION	
			OPER.[3] STAT.	AIR FLOW ACFM	OPER.[3] STAT.	AIR FLOW ACFM	OPER.[3] STAT.	AIR FLOW ACFM	OPER.[3] STAT.	AIR FLOW ACFM
		AH-15 (1A-SA)	R	14000	R	14000	R	14000	R	14000
SUPPLY		CZ-D1SA-1	O	14000	O	14000	O	14000	O	14000
	36"	3CZ-B25SA-1	O	14000	O	14000	O	14000	O	14000
		AH-15 (1B-SB)	S		S		S		S	
		CZ-D2SB-1	C		C		C		C	
	36"	3CZ-B26SB-1	C		C		C		C	
O.A.I		CZ-Z1SN-1	O	1050	O	1050	O		O	
NORMAL	16"	3CZ-B1SA-1	O	1050	O	1050	C		C	
	16"	3CZ-B2SB-1	O	1050	O	1050	C		C	
O.A.I	36"	3CZ-B17SA-1	C		O	12950	C		C	
PURGE	36"	3CZ-B18SB-1	C		O	12950	C		C	
RETURN		CZ-D69SA-1	O	12950	C		O	14000	O	14000
NORMAL		CZ-D70SB-1	C		C		C		C	
		E-9 (1A-NNS)	R	1000 (max.)	R	1000	I		I	
		CZ-D6-1	O	1000 (max.)	O	1000	C		C	
EXHAUST		CZ-D12-1	M	1000 (max.)	O	1000	C		C	
		E-9 (1B-NNS)	S		S		I		I	
		CZ-D7-1	C		C		C		C	
		CZ-D13-1	C		C		C		C	
		CZ-X2SN-1	O	1000 (max.)	O	1000	O		O	
	12"	3CZ-B3SA-1	O	1000 (max.)	O	1000	C		C	
	12"	3CZ-B4SB-1	O	1000 (max.)	O	1000	C		C	
RETURN		CZ-D66SA-1	C		C		O	1050	O	1050
POST-LOCA		CZ-D61SB-1	C		C		C		C	
		ES-1 (1A-NNS)	I		R	13400	I		I	
	30"	3CZ-B13SA-1	C		O	13400	C		C	
PURGE	30"	3CZ-B14SB-1	C		O		C		C	
		ES-1 (1B-NNS)	I		S		I		I	

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TABLE 9.4.1-2 (Cont'd)

SYSTEMS	SIZE	VALVES DAMPERS FANS DESIGNATION	NORMAL OPERATION OPER.(5) STAT.	NORMAL OPERATION AIR FLOW ACFM	PURGE OPERATION(4) OPER.(5) STAT.	PURGE OPERATION(4) AIR FLOW ACFM	CL_2 ACCIDENT(2) OPERATION OPER.(5) STAT.	CL_2 ACCIDENT(2) OPERATION AIR FLOW ACFM	HI RAD. SYS.(2) OPERATION OPER.(5) STAT.	HI RAD. SYS.(2) OPERATION AIR FLOW ACFM	
											27
O.A.I.	12"	3CZ-B9SA-1	C		C		C		O	400 max	
POST ACC.	12"	3CZ-B11SA-1	C		C		C		C		
	12"	3CZ-B10SB-1	C		C		C		O	400 max	
	12"	3CZ-B12SB-1	C		C		C		C		
EMERGENCY		3CZ-V1SA-1	C		C		C		O	400 max	
SUPPLY		3CZ-V2SB-1	C		C		C		O	400 max	27
		R-2 (1A-SA)	I		I		R	4000	R	4000	
	20"	3CZ-B23SA-1	C		C		O	4000	O	4000	
EMERGENCY	20"	3CZ-B21SA-1	C		C		O	4000	O	4000	
FILTRATION	20"	3CZ-B19SA-1	C		C		O	4000	O	4000	
		R-2 (1B-SB)	I		I		S(3)		S(3)		27
	20"	3CZ-B24SB-1	C		C		C		C		
	20"	3CZ-B22SB-1	C		C		C		C		
	20"	3CZ-B20SB-1	C		C		C		C		
	*20"	3CZ-B27SAB-1	O	O	O	O	O	4000	O	3600 (min)	27

* PARTIALLY OPEN FOR BALANCING

NOTES TO TABLE 9.4.1-2

1) The table represents supplementary information to Figure 9.4.1-1.

2) Post accident operation represents the operational status of the system under operator control.

a) Depending on the reading of radiation monitors located at both air intakes, during the radiological accident the control room operator will 27
manually remotely, open the selected air intake by setting the air intake isolation valves and allowing a maximum of 400 cfm of the outside air into the control room envelope for pressurization. 27

b) All isolation valves in the post-accident outside air intakes will have leak-tight closing and variable setting capability. 27

3) After automatic start-up of both emergency filtration systems, one out of the two will be manually de-energized and placed on standby.

4) Purge operation represents the operational status of the system under operator control.

5) The notations used to designate the operational status of the tabulated components are:

Notation	Operational Status
S	Standby
R	Running (operating)
O	Open
C	Closed
M	Modulating
I	Inactive
(+) or (-)	Positive pressure (+) or Negative pressure (-) in The control room envelope.

TABLE 9.4.1-3

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TABLE 9.4.1-3

DELETED BY AMENDMENT NO. 15

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TABLE 9.4.1-3

DELETED BY AMENDMENT NO. 15

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TABLE 9.4.1-4

CONTROL ROOM AIR CONDITIONING SYSTEM SINGLE FAILURE ANALYIS

COMPONENT IDENTIFICATION	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
Normal Outside Air Intake and Exhaust Valve	Loss of power-fails to close on initiation of recirc. phase	Uncontrolled contaminated air enters Control Room	Valve position switch alarms	CRI	Redundant standby valve powered by alternate sources will be operable to isolate Control Room.
Post-Accident Outside Air Intake Valve	Loss of Power-Fails to Open	Prevent controlled pressurization	Power Failure Indicator	CRI	Operator manually hand wheel opens affected valve
	One Valve Freezes Closed	Prevent controlled pressurization	Valve position switch alarm	CRI	Redundant standby intake is available.
Supply Filter	Clogs	Reduction of supply air flow	1) Air flow switch at fan discharge	CRI	1) Redundant capacity standby unit is operable powered by alternate source.
			2) Temp. indicating controller		2) Filters are accessible for replacement.
Cooling Coil	Fails	Change in supply air temp.	Temp. indicating controller with sensor at unit discharge	CRI	Redundant capacity standby unit is operable powered by alternate source.

TABLE 9.4.1-4 (Cont'd)

COMPONENT IDENTIFICATION	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS	
Supply Fan	Fails	Loss of supply air	Power Failure Indicator	CRI	Redundant capacity standby unit is operable powered by alternate source.	27
Supply Unit Isolation Valve and Dampers	Fails to open	Loss of supply air	Flow switch at fan discharge	CRI	Redundant Supply Unit is operable powered by alternate source.	27
Zone Reheat Coil & Electric Heating Coil	Fails to automatically shut off	Increase in zone space temperature	Temp. indicating controller	CRI	Coil can be manually shut off at local control panel in Control Room area.	27
Emergency Filter Booster Fan	Fails	Loss of air flow through filter system	Flow switch at system discharge	CRI	Redundant capacity standby unit is operable powered by alternate source.	27
Isolation Valve at Emergency Filter Train	Fails to open	Loss of air flow through filter system	Flow switch at system discharge	CRI	Redundant capacity standby unit unit is operable powered by alternate source.	27
Emergency Filter Train	Clogs	Reduction of air flow through filter system	Flow switch at system discharge	CRI	Redundant capacity standby unit is operable powered by alternate source	27
Water Chiller	Fails	Loss of Cooling capacity	1) Temp. indicating controller with sensor at supply fan discharge.	CRI	Redundant capacity standby unit is operable powered by alternate source.	27

TABLE 9.4.1-4 (Cont'd)

COMPONENT IDENTIFICATION	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS	
			2) Water temp. indicator with sensor at chiller discharge	CRI	100% capacity standby system is is operable powered by alternate source.	27
Chilled Water Pump	Fails	Loss of chilled water flow	Flow switch at pump discharge	CRI	100% capacity standby system is operable powered by alternate power source.	27
Smoke Purge Air Make-up or Exhaust Valves	1. Opens inadvert.	Loss of Isolation Uncontrolled Cont. Air Enters Control Room	Valve position indicators	CRI	Redundant standby valves will assure the continued Iso. of the Control Room.	27

TABLE 9.4.1-4 (Cont'd)

COMPONENT IDENTIFICATION	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
NORMAL AND OFFSITE POWER FAILS					
Diesel Gen. to which Control Room equipment is connected.	Fails	Loss of: 1) one Control Room A/C System 2) One emergency filter system.	DG malfunction alarm, flow switch at fan discharge, flow switch in chilled water system.	CRI	1) Standby A/C System operable 2) Standby emergency filter system is operable.

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9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The information presented hereinafter in Section 9.5.1 provides a general discussion of the various fire protection systems at the Shearon Harris Nuclear Power Plant. In addition, specific reports and information have been provided to address different facets of the fire protection program in greater detail. These documents are:

a) Fire Protection Evaluation and Comparison to NUREG-0800, BTP CMEB 9.5-1, July 1981 (Response to Question 280.1, Revision 3 submitted on May 7, 1986).

b) Safe Shutdown Analysis in Case of Fire

These two sources of information are hereby incorporated by reference into the FSAR.

In addition to the above, limiting conditions for operation, action statements, and surveillance requirements for the fire protection program will be established within the Shearon Harris Plant Operating Manual. This will provide a level of protection equivalent to the following sections of the Westinghouse Standard Technical Specifications (Revision 5):

a) 3/4.7.10.1 - Fire Protection Water Supply and Distribution System

b) 3/4.7.10.2 - Preaction and Multicycle Sprinkler System

c) 3/4.7.10.3 - Fire Hose Stations

d) 3/4.7.10.4 - Yard Fire Hydrants and Hydrant Hose Houses

e) 3/4.7.11 - Fire Rated Assemblies

f) 3/4.3.3.8 - Fire Protection Instrumentation

g) 6.2.2 - Unit Staff (Fire Brigade)

The Shearon Harris Nuclear Power Plant (SHNPP) fire protection program is based on the Nuclear Regulatory Commission (NRC) guidelines, Nuclear Mutual Limited (NML) Property Loss Prevention Standards for Nuclear Generating Stations and related industry standards. With regard to NRC criteria, the SHNPP fire protection program addresses the guidelines outlined in Branch Technical Position CMEB 9.5-1, dated July 1981. Various aspects of the fire protection program are detailed, as required, to show conformance with the guidelines or to demonstrate the equivalency of alternative approaches.

General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components. They are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heattraveling screens are provided at the intake structure for the removal of larger impurities which may be present in the water. (For more details, see description of service water systems, Section 9.2.1.) Although the water supply serves as the service ultimate heat sink and also as the fire protection water supply, with sufficient capacity for both functions, fire protection system failure will not degrade the ultimate heat sink function (see Section 9.2.5).

Fire Pumps - Fire pumps and controllers are installed in accordance with NFPA 20. Water is supplied from the Auxiliary Reservoir by two 100 percent capacity outdoor type, vertical, 2,500 gpm, 125 psi fire pumps. Each fire pump is capable of delivering 3,000 gpm at approximately 110 psig, and 150 percent of rated capacity at not less than 65 percent of rated head. Each pump is also capable of delivering the design demand over the longest route of the water supply system. One electric motor driven fire pump and one diesel engine driven fire pump, suitable for outdoor operation, are installed outdoors at opposite ends of the Emergency Service Water Screening Structure which produces spatial separation in lieu of a firewall. The electric motor driven pump is UL listed. The diesel engine driven pump is FM approved. Both pump controllers are UL listed and FM approved. There are no specific requirements in NFPA 20 that electric motors for fire pumps be listed or approved by an independent laboratory and, therefore, they are not listed. Each pump has a separate intake and discharges through independent underground connections into the main fire loop (see Figure 9.5.1-1). Adequate isolation is provided between pump installations to prevent loss of service of more than one pump in event of a single fire occurrence.

The largest firewater flow and pressure requirement is 2750 gpm, 72 psi at the Turbine Building mezzanine (el. 286) South Sprinkler System. This includes 305 gpm for hose stream. This demand can be met by either of the two fire pumps.

Each fire pump provides the total fire protection water supply requirement to the fire main loop, thus required fire pump discharge capacity and pressure are available with either pump out of service.

The pump discharge connections are separated by approximately 40 ft. to prevent damage to both connections simultaneously. The fire main loop valves and fire pump discharge valves are arranged to permit discharge from either connection to the main fire loop. Sectional valves between individual pump connections are provided. No single failure or event in the Emergency Service Water Intake System will result in a failure of the fire protection water supply.

Alarms and indications of fire pump operating conditions, such as pump running, driver availability, and failure to start, are provided at the MFDIC. A common annunciation, "Fire Pump System Trouble," is provided in the Main Control Room for the Motor and Diesel Driven Fire Pump.

The fire pumps are designed for sequential automatic starting on progressive drops in fire main water pressure. The motor driven fire pump starts automatically when the pressure in the fire loop drops to 93 psig. If the pressure continues to drop, at 83 psig, the diesel driven fire pump starts automatically. Both pumps are stopped manually. The water pressure in the distribution system is maintained at approximately 105-120 psig by the 50 gpm electric motor driven jockey pump, started automatically on drop in pressure and stopped on restoration of pressure after a suitable time delay provided to prevent unnecessary operation of the fire pumps. A low fire-main pressure alarm is not provided since automatic start-up of the electric driven pump is annunciated in the main Control Room via the Communications Room.

Power for the electric motor driven fire pump is supplied from a 480V power center, which is fed from a 6.9 kV switchgear that has an alternate feed through a bus tie with another 6.9 kV switchgear. Fuel supply for the diesel engine driven fire pump, a 550 gal. No. 2 oil tank, is located outdoors about 12 feet away from the emergency service water intake screen structure, suitably protected against fire and does not expose both fire pumps to fire damage. A 12 in. dike is provided to direct oil to a pit within the dike, capable of containing the entire oil volume in case of an oil spill or tank rupture. A manual controlled drain system is provided in order to remove the spillage if required.

A pump test discharge header is provided of such capacity that the fire pumps may be given initial acceptance flow test and periodic performance tests. The discharge of flow test water is sent back to the reservoir. Water discharged from the pressure relief valves on the fire pumps and jockey pump are returned to the reservoir.

Distribution System

The fire protection water distribution system (Figures 9.5.1-1 through 9.5.1-5) consists of an underground 12 in. mechanical joint, ductile iron, cement or bituminous lined pipe loop around the main plant building complex to supply the water requirements for fire protection systems and equipment. The underground loop is cross-connected in a north-south direction through the Waste Processing and Fuel Handling Buildings. The underground loop also supplies the Reactor Auxiliary Building, the Turbine Building, and construction warehouses 6 and 9. In addition to the underground supplies, the RAB and TB are cross-connected in an east-west direction.

These cross-connections are ductile iron, cement or bituminous lined pipe for underground runs and carbon steel pipe, suitably supported, for above-ground piping within buildings. Sectional control valves are provided to assure two-directional supply to all areas.

All sectional and isolation valves in the fire suppression water supply system (except hydrant valves and inside hose connections) are either post indicator valves (PIVs) for underground piping or outside screw and yoke (OS&Y) valves for interior building piping.

The guidelines of NFPA 24 were used in the design and installation of the underground yard main fire loop. Fire protection main piping is not interconnected with any plant service or sanitary water systems.

Ductile iron, cement or bituminous lined pipe is used for the yard main fire loop to minimize the effects of tuberculation. Flushing of the system is through the yard hydrants, hose connections and suppression system drains.

Post indicator valves are provided in the distribution system as required for adequate sectionalization of loops and isolation of branch lines to facilitate system maintenance. Isolation valves are located in branch lines connecting to fire suppression systems in the buildings to avoid closing sectional valves in the main loop. Sectional isolation valves are provided in the yard loop piping to minimize the impairment of fire protection water supply if maintenance on the loop or yard hydrants becomes necessary. Sectional control valves provided in the pump discharge connections to the loop and in the yard main loop piping are positioned to assure supply of fire water systems for any area from either or both fire pumps.

Sprinkler systems and manual hose station standpipes have connections to the plant underground water main so that a single active failure or a crack in the moderate-energy line cannot impair both the primary and backup fire suppression systems for Safety-Related Areas. Headers fed from each end are used inside buildings to supply both sprinkler and standpipe systems in the Waste Processing, Fuel Handling, Turbine, and Reactor Auxiliary Buildings. They are fabricated of carbon steel piping and fittings meeting the requirements of ANSI B31.1 "Power Piping". Each sprinkler and standpipe system is equipped with an OS&Y gate valve and water flow alarm except that in the RAB the header supplying the hose standpipes is arranged so that the OS&Y gate valves in the header on each side of a standpipe must be closed to isolate the standpipe. Since this header is fed from both ends, the water supply to other standpipes served by this header is not interrupted. The Fire Hazards Analysis describes the methods used to protect safety-related equipment in each fire area from water damage.

Control and sectionalizing valves in the fire water systems are electrically supervised. The electrical supervision signal indicates on the Local Fire Detection and Control Panels and at the Main Fire Detection Information Center located in the Communication Room. A common trouble annunciation will be given in the Control Room if any valve is out of position.

Non-freeze type fire hydrants, equipped with a minimum of two 2-1/2 in. gated outlets, are installed approximately every 250 ft. along the fire main loop in the yard area around the main plant building complex and are protected from mechanical damage from vehicular traffic. Branch connections from the main loop supply hydrants, hose station and systems at outlying structures. Hose houses are installed adjacent to each hydrant and are equipped with the standard complement of 2-1/2 in. fire hose, nozzles, and hose-line equipment in accordance with NFPA 24 requirements or have an alternate mobile means of providing hose and equipment equivalent to three hose houses. A curb box valve is installed on hydrant branches.

Screw threads and gaskets for fire hose and hose line equipment are American National Fire Hose Connection. Screw threads (NST) are in accordance with NFPA No. 1963 and correspond with the fire department's threads that respond to this facility.

Manual Fire Response

Equipment used for manual fire response is described below.

a) Fire extinguishers - Fire extinguishers provided throughout the plant are UL listed and/or FM approved and labeled accordingly. Extinguishers are mounted in readily accessible locations in conformance with NFPA Standard 10.

Types of extinguishers selected are based on the nature of the fire postulated for the area, in accordance with NFPA 10, and on the unique characteristics of the fire suppression agent affecting its proper application to the fire. Considerations include quantity required in relation to the size of the anticipated fire, cleanup after use, and thermal shock or corrosive effects of the agent or its fire decomposition products, and consideration for possible adverse effects on safety-related equipment.

The following basic types of extinguishers are used:

Dry chemical - hand and wheeled - in operational areas or outdoor areas of severe fire potential,

Carbon dioxide or Halon - hand - in area of low fire hazards or containing small electrical equipment where cleanup after the fire is a major consideration, such as Control Room, laboratories and switchgear areas,

Water - hand - in areas containing ordinary combustibles such as warehouses and offices.

b) Standpipe and Hose System - Standpipe and hose systems are installed throughout the plant inside buildings to supply hose stations, suitable for safe effective use on identified hazards and involved equipment (refer to Figures 9.5.1-2 through 9.5.1-5). Sufficient hose stations are provided in each area so that all portions of the plant can be reached by effective hose streams except the tank area and Diesel Fuel Oil Storage Tank and Transfer Pumps area ESW intake and emergency screening structure which are protected by yard hydrants.

The guidelines of NFPA 14, Class 2, were followed in the design of standpipe and hose systems. Individual standpipes are minimum 4 in. diameter for multiple hose connections and 2-1/2 in. diameter for single hose connections. Hose stations are located as dictated by the fire hazard analysis to facilitate access and use for firefighting operations.

The proper type of hose nozzle supplied to each area is based on the fire hazard analysis. The usual combination spray/straight-stream nozzle are not used in areas where the straight stream can cause unacceptable mechanical damage. Hose stations are equipped with 100 ft. of 1-1/2 in. rubber lined, rubber coated hose and adjustable spray nozzles, approved for use on energized electrical equipment and cabling, stored on racks or in cabinets. Standpipe hose connections are provided in all buildings (except the Diesel Fuel Oil Storage Tank Building ESW intake and emergency screening structure).

Fire hose is hydrostatically tested in accordance with the recommendations of NFPA 1962, "Fire Hose - Care, Use, Maintenance". Hose stored in outside hose houses will be tested annually. Interior standpipe hose will be tested every three years.

The standpipe system is designed and sized to provide, to the most remote hose station, the flow rate and pressure required for effective hose streams.

Operation of a hose station associated with a particular riser is alarmed locally and alarmed and annunciated at the Main Fire Detection Information Center (MFDIC) in the Plant Communications Room and the Control Room following sensing of water flow in the standpipe riser by system flow switches.

Sectional shutoff valves provided for standpipes serving hose stations in safety related areas are located outside the safety related areas to permit access during a fire.

Portions of the standpipe and hose systems installed in the Containment, Reactor Auxiliary and Fuel Handling Buildings, as shown on (Figures 9.5.1-2 and 9.5.1-4), are designed to be operable, if needed, for manual fire control in areas required for safe plant shutdown following a safe shutdown earthquake (SSE). These portions of the standpipe system were analyzed for SSE loading and seismically supported to assure system pressure integrity. The piping and valves for these standpipes are designed to satisfy ANSI B31.1, "Power Piping."

Normally, the post-SSE standpipe hose station header is supplied from the fire protection water distribution system through seismically qualified check valves. Following an SSE event, water supply for the post-SSE portion of the standpipe system can be obtained by local operator manual positioning of valves to connect the Seismic Category I Emergency Service Water System, located in the Reactor Auxiliary Building, to the post-SSE hose standpipe header. Seismic Category I water supply is provided for the Post SSE Fire Protection Standpipe and Hose System by the emergency service water booster pumps. The ESW booster pumps are normally used following a LOCA to provide high head cooling water to the containment fan coolers. However, in the event of a Post-SSE fire, the ESW pump (A or B) and ESW booster pump would be started. This arrangement provides sufficient TDH to supply the two most remote hose stations with 75 gpm (each) of water at approximately 65 psig as discussed in NFPA. The seismic check valves prevent outflow to other portions of the fire protection water distribution system, which may have failed during the seismic event, and thus avoid loss of hose line protection after the earthquake.

c) <u>Self-Contained Breathing Equipment</u> - Breathing equipment is provided as required for protection against smoke inhalation of personnel required to be in plant areas to control fires or to continue vital plant operations.

Self-contained breathing apparatus, using full face positive pressure masks, approved by National Institute for Occupational Safety and Health (NIOSH), with a minimum capacity of one half hour, are provided for fire brigade and control room personnel.

An additional hour of air in bottles is located onsite for each self-contained 37
breathing unit, used by fire brigade and control room personnel, with an

onsite six hour supply of reserve air and refilling manifolds for recharging air bottles. The six hour reserve supply is provided from storage cylinders, with resupply from an approved breathing air compressor. The air compressor is equipped with a carbon monoxide monitor and with an air intake located away from dust, organic vapor and other contaminant sources.

d) Protective clothing - Protective clothing will be provided to members of the plant fire brigade or other designated personnel and is located in accessible locations for use of fire response personnel as developed in the Fire Protection Plan.

Instruction in the use of protective clothing and assignment to personnel is a part of the overall fire response procedures developed by plant operating groups.

e) Emergency Lighting - Redundant AC normal/emergency lighting (powered from safety-related motor control centers) is provided in areas where safety related functions are performed, in access routes to these areas, and for emergency evacuation.

Except for the control, auxiliary control and computer rooms, fixed self-contained sealed beam units with individual 8-hour minimum battery power supplies are provided in areas that must be manned for safe shutdown and for access and egress route to and from all fire areas. Emergency DC lighting, fed from the 125V station battery, provides lighting in the Control Room auxiliary control and computer rooms in the event that either train of the AC normal/emergency lighting is lost. The cable routing for the DC Emergency Lighting is included in the Safe Shutdown Analysis in Case of Fire and separated or protected in accordance with NRC position C.5.b(2) of BTP CMEB 9.5-1 (NUREG-0800), July 1981.

Suitable sealed-beam battery-powered portable hand lights are provided for emergency use by the fire brigade and other operations personnel required to achieve safe plant shutdown. Spare batteries are provided (see Section 9.5.3).

f) Emergency Communications - The sound powered telephone system for SHNPP is an independent, five-channel system consisting of master panels, remote jack stations, and sound powered headsets and wiring. The jack stations are located at control panels, relay cabinets, instrument jacks, switchgear, motor control centers, and other locations having critical system components. The sound powered telephone system requires no external power sources at SHNPP. This system may be utilized for normal or emergency communication.

A dedicated radio system for plant operation and maintenance is provided. The system consists of a base station, an interior antenna system for inside building coverage, and battery-powered hand-held portable radios. Power for the plant operations/maintenance radio system is from the non-Class 1E uninterruptible power supply. This system is totally independent from the plant security radio system with the exception of the antenna system in the plant where the O&M and security radios utilize the same antenna system operating at different frequencies and will not interfere with each other.

INFORMATION USE

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

PLANT OPERATING MANUAL

VOLUME 1

PART 2

PROCEDURE TYPE:	Plant Program
NUMBER:	PLP-201
TITLE:	Emergency Plan

HNP CONTROLLED COPY # 702

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1.0 INTRODUCTION

1.1 Authority/Requirements

The Harris Nuclear Plant (HNP) Emergency Plan and Plant Emergency Procedures have been prepared in accordance with the following requirements and guidelines:

A. Code of Federal Regulations, 10 CFR 50, Section 50.47, "Emergency Plans."

B. Code of Federal Regulations, 10 CFR 50, Section 50.54(q), "Conditions of Licenses."

C. Code of Federal Regulations, 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

D. NUREG-0654, FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980.

E. NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability", December 17, 1982.

F. NUREG/CR-4831, "State of the Art in Evacuation Time Estimate Studies for Nuclear Power Plants," March 1992.

G. FEMA Guidance Memorandum MS-1, "Medical Services," Federal Emergency Management Agency, November 13, 1986.

H. RTM-92, "Response Technical Manual" Volume 1, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C., October 1992.

I. IE Information Notice 85-55, "Revised Emergency Exercise Frequency Rule," July 15, 1985.

J. EPA-400-R-92-001, "Manual of Protective Action Guidelines and Protective Actions for Nuclear Incidents," U.S. Environmental Protection Agency, May 1992.

K. EPPOS No. 1, "Emergency Preparedness Position (EPPOS) on Acceptable Deviations from Appendix 1 of NUREG-0654 Based Upon the Staff's Regulatory Analysis of NUMARC/NESP-007, 'Methodology for Development of Emergency Action Levels'", June 6, 1995.

L. EPPOS No. 2, "Emergency Preparedness Position (EPPOS) on Timeliness of Classification of Emergency Conditions", August 17, 1995.

M. EPPOS No. 3, "Emergency Preparedness Position (EPPOS) on Requirement for Onshift Dose Assessment Capability", November 8, 1995.

N. NRC Correspondence: SECY 88-147, SECY 89-012, Generic Letter 88-20

1.2 Purpose of HNP Emergency Plan and Implementing Procedures

The purpose of the HNP Emergency Plan (E-Plan) and Implementing Procedures (Plant Emergency Procedures) is to assure that the state of on-site and off-site emergency preparedness provides reasonable assurance that adequate corrective and protective measures can and will be taken in the event of a radiological emergency at the plant. The HNP E-Plan and Implementing Procedures outline the Emergency Preparedness Program which has the following objectives:

A. Protection of plant personnel and the general public.

B. Prevention or mitigation of property damage.

C. Effective coordination of emergency activities among all organizations having a response role.

D. Early warning and clear instructions to the population-at-risk in the event of a serious radiological emergency.

E. Continued assessment of actual or potential consequences both on site and off site.

F. Effective and timely implementation of emergency measures.

G. Continued maintenance of an adequate state of emergency preparedness.

The HNP Emergency Preparedness Controlled Documents are contained in the HNP Plant Operating Manual (POM) and consist of the following parts:

- Volume 1, Part 2, Emergency Plan (PLP-201)
- Volume 2, Part 5, Plant Emergency Procedures (PEP)
- Volume 2, Part 10, Emergency Program Maintenance (EPM)

The Emergency Phone List, EPL-001, is an HNP document controlled outside the POM.

A list of documents which implement and maintain this plan can be found in Annex E.

1.3 Responsibility for Plan Development and Review

Responsibility for the HNP Emergency Plan development, review, and periodic update is assigned to the Supervisor - Emergency Preparedness who serves as the HNP Emergency Planning Coordinator.

Procedures are in place to ensure changes to the Emergency Preparedness Program are evaluated to determine whether the changes do or do not decrease the effectiveness of the plan and the plan, as changed, continues to meet the standards of 10CFR50.47(b) and the requirements of Appendix E. Changes which do result in an evaluated decrease in program effectiveness will not be implemented without prior NRC approval.

1.4 Emergency Classes

Off-normal plant conditions are classified according to four emergency classes which in order of increasing severity are Unusual Event; Alert; Site Area Emergency; and General Emergency. The emergency classes are defined in NUREG-0654, Appendix 1, as follows:

1.4.1 Unusual Event

Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs. Unusual Event is equivalent to the NRC designated class "Notification of Unusual Events."

1.4.2 Alert

Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of EPA Protective Action Guideline exposure levels. The Alert class may correspond to failure or jeopardy of one Fission Product Barrier.

1.4.3 Site Area Emergency

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA Protective Action Guideline exposure levels except near site boundary. The Site Area Emergency Class may correspond to failure or jeopardy of two Fission Product Barriers.

1.4.4 General Emergency

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels off site for more than the immediate site area. The General Emergency class may correspond to failure or jeopardy of three Fission Product Barriers.

Events that could lead to any of these emergency classifications are described in Section 4.0, "Emergency Measures and Operations."

1.5 Severe Accident Management Guidelines (SAMGs)

Severe Accident Management Guidelines are put into use when plant conditions are beyond design basis. The primary goal is to protect fission product barriers and mitigate any ongoing fission product releases, with secondary goals to mitigate severe accident phenomena and return the plant to a stable condition. The implementation of SAMGs invokes the provisions of 10 CFR 50.54(x) and (y).

1.6 Plant Site Description

The Harris Nuclear Plant (HNP) site is located in the extreme southwest corner of Wake County, North Carolina, approximately 16 miles southwest of Raleigh, which is the largest population center, and approximately 15 miles northeast of Sanford, North Carolina, in Lee County (See Figures 1.5-1 and 1.5-2). Approximate coordinates of the plant centerline are latitude 35° 38' 01" N and longitude 78° 57' 23" W. The Harris Nuclear Plant consists of one pressurized water reactor (PWR) of Westinghouse Corporation manufacture, licensed to operate at 2775 megawatts thermal (MWt). The associated net electrical output is approximately 900 megawatts electric (MWe). The major structures of HNP which contain radioactive materials are the Containment Building, Reactor Auxiliary Building, Fuel Handling Building, and the Waste Processing Building. Figure 1.5-3 shows the principle site buildings.

Figure 1.5-2 shows the Exclusion Area Boundary (EAB) and the location of the Harris Energy & Environmental Center (HE&EC) in which the Emergency Operations Facility (EOF) is located.

1.7 Plume Exposure Emergency Planning Zone (EPZ)

The Plume Exposure Emergency Planning Zone (EPZ) is defined as the area within an approximate 10-mile radius of the HNP and is referred to as the 10-Mile EPZ.

Principal exposure sources from the plume exposure pathway are (a) external exposure to gamma and beta radiation from the plume and from deposited materials and (b) exposure of the internal organs to gamma and beta radiation from inhaled radioactive gases and/or radioactive particulates. The time of potential exposure can range in length from hours to days.

Figure 1.5-2 shows the Plume Exposure EPZ in relation to the location of HNP. The Plume Exposure EPZ includes portions of the North Carolina counties of Chatham, Harnett, Lee, and Wake. Annex H, attached, shows evacuation routes and local emergency planning zone boundaries in the 10-mile EPZ.

The prevailing winds around the plant are from the southwest.

1.8 Ingestion Exposure Emergency Planning Zone

The Ingestion Exposure Emergency Planning Zone (EPZ) is defined as the area within an approximate 50-mile radius of the HNP and is referred to as the 50-Mile EPZ.

The ingestion exposure sources from the ingestion pathway are contaminated water or food, such as milk or fresh vegetables. The time of potential exposure can range in length from hours to months.

The region within a 50-mile radius of the HNP site contains both urban and rural areas with industry, farming, business, education, research, and military interests. Figure 1.5-1 shows the 50-mile Ingestion Exposure EPZ in relation to the location of the Shearon Harris Plant. The Ingestion Exposure EPZ includes the North Carolina counties of Alamance, Caswell, Chatham, Cumberland, Durham, Franklin, Granville, Guilford, Harnett, Hoke, Johnston, Lee, Montgomery, Moore, Nash, Orange, Person, Randolph, Robeson, Sampson, Vance, Wake, Wayne and Wilson.

1.9 Demographic Information

The distribution of resident population in the 10-Mile Emergency Planning Zone is presented in Table 1.8-1 and Figure 1.8-1. Special facilities within the Plume Exposure Emergency Planning Zone are depicted in Figure 1.8-2 and Table 1.8-3. The 10-Mile Emergency Planning Zone evacuation time estimates are provided in Table 1.8-2.

1.10 Supporting Emergency Plans

Emergency Plans which support this Plan are:

A. North Carolina Emergency Response Plan in Support of Shearon Harris Nuclear Power Plant, Division of Emergency Management, Department of Crime Control and Public Safety.

B. U.S. Nuclear Regulatory Commission, NUREG-0728, NRC Incident Response Plan.

C. Federal Radiological Emergency Response Plan.

D. Southern Mutual Radiological Assistance Plan.

Figure 1.5-1

Ingestion Exposure Emergency Planning Zone (50-Mile EPZ)

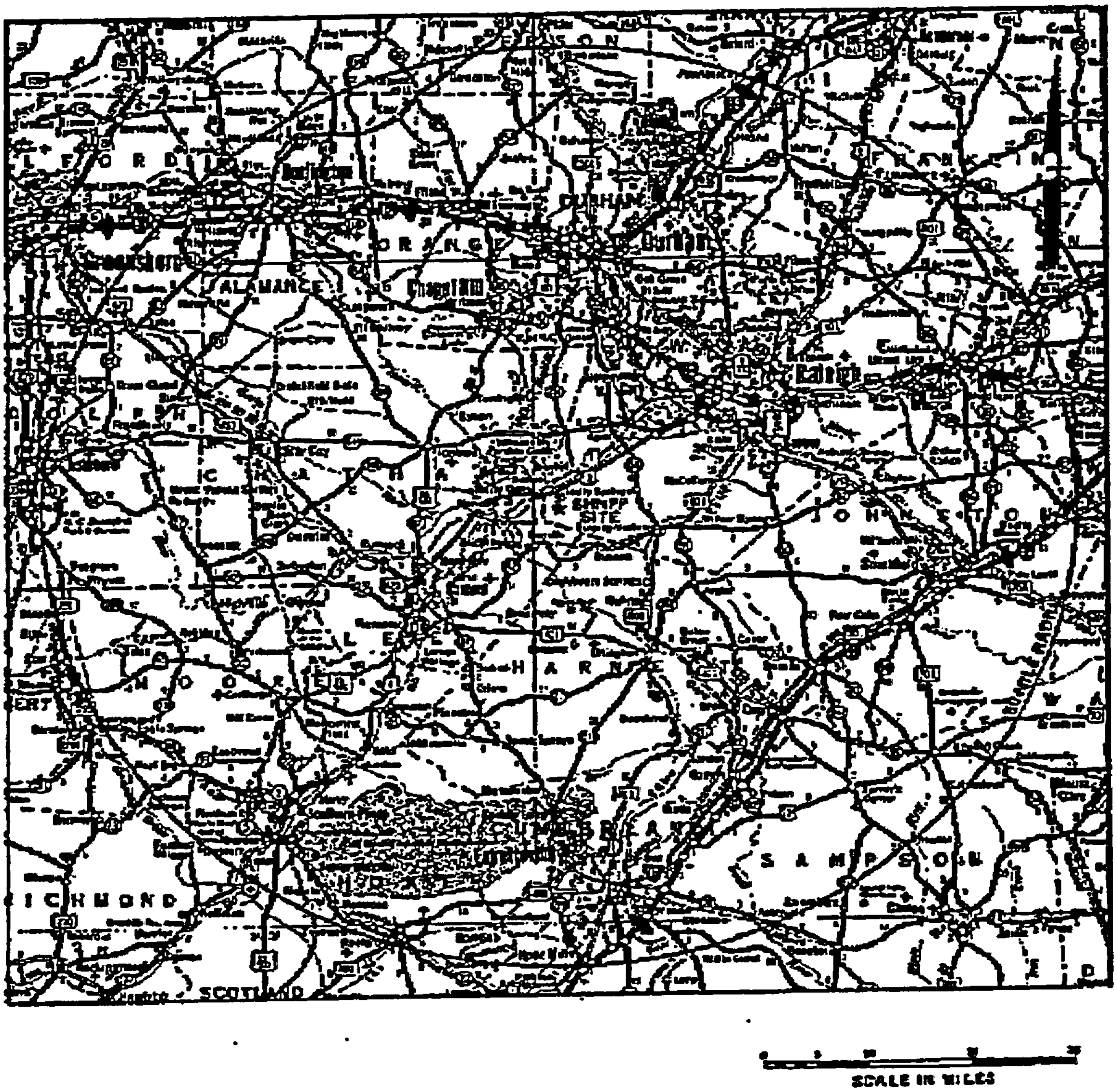

Figure 1.5-2

Plume Exposure Emergency Planning Zone (10-Mile EPZ)

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N

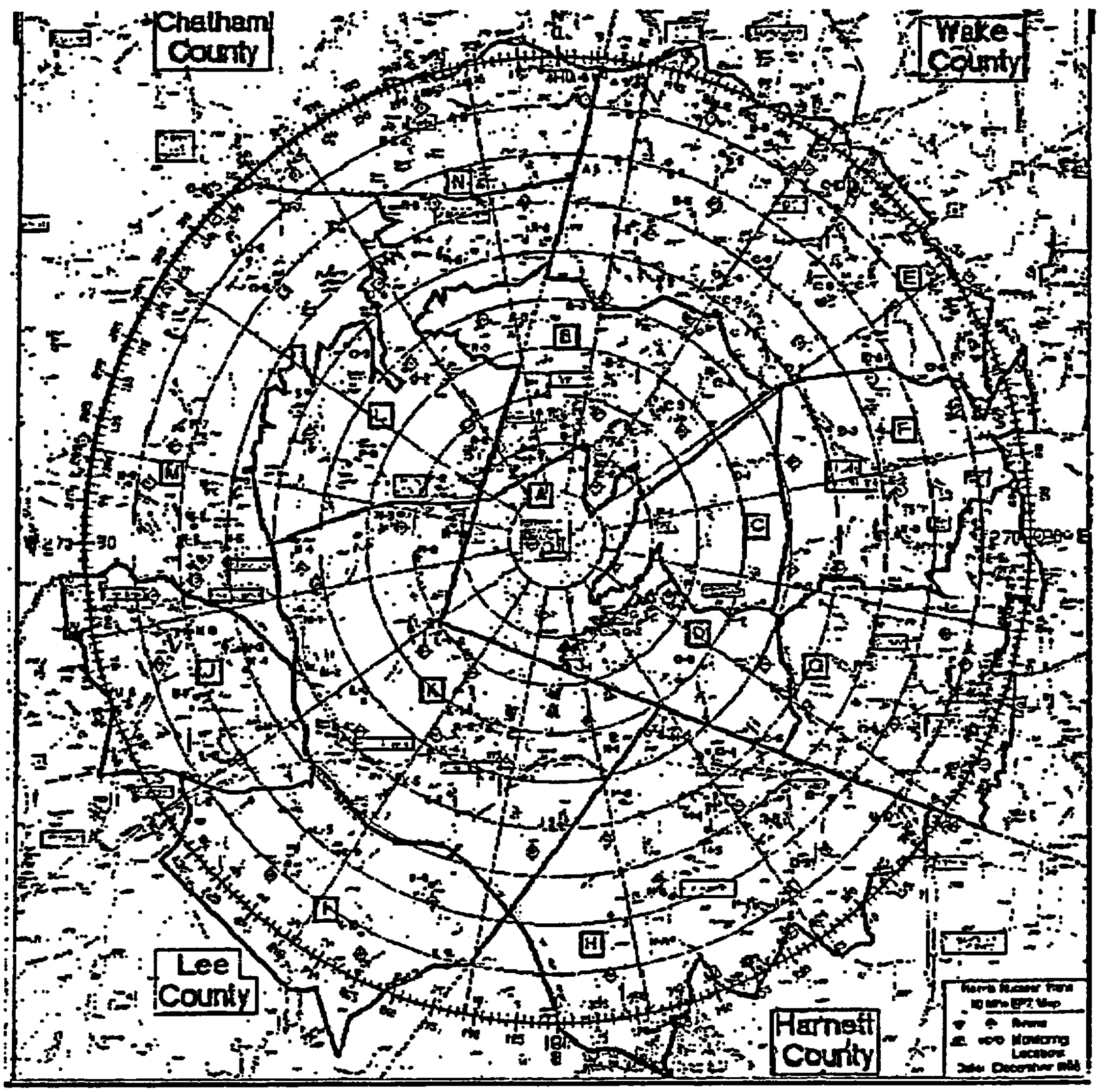
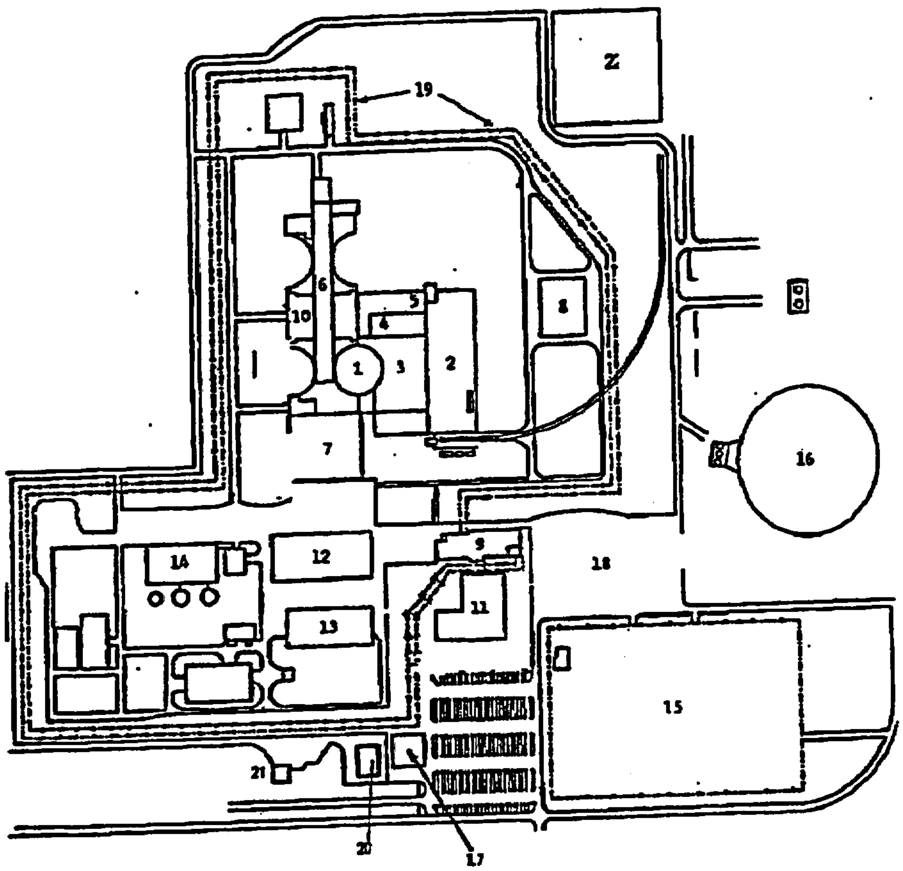


Figure 1.5-3

HNP Site Plan and Emergency Facilities



1	Reactor Containment Building
2	Turbine Building
3	Reactor Auxiliary Building
4	Operations Building
5	Main Control Room
6	Fuel Handling Building
7	Waste Processing Building (OSC in HP Tech Work Area; First Aid Room 261')
8	Diesel Generator Building
9	Security Building (Normal Evacuation)
10	K Building (TSC on 4th Floor)
11	Administration Building (Admin Building Assembly Area)
12	Service Building
13	Bulk Warehouse
14	Water Treatment Building
15	Switchyard
16	Cooling Tower
17	Helicopter Landing Zone
18	Evacuation Monitoring Area
19	Protected Area Boundary
20	Plant Access Facility
21	Mobile Equipment Shop
22	Warehouse 6

Table 1.8-1

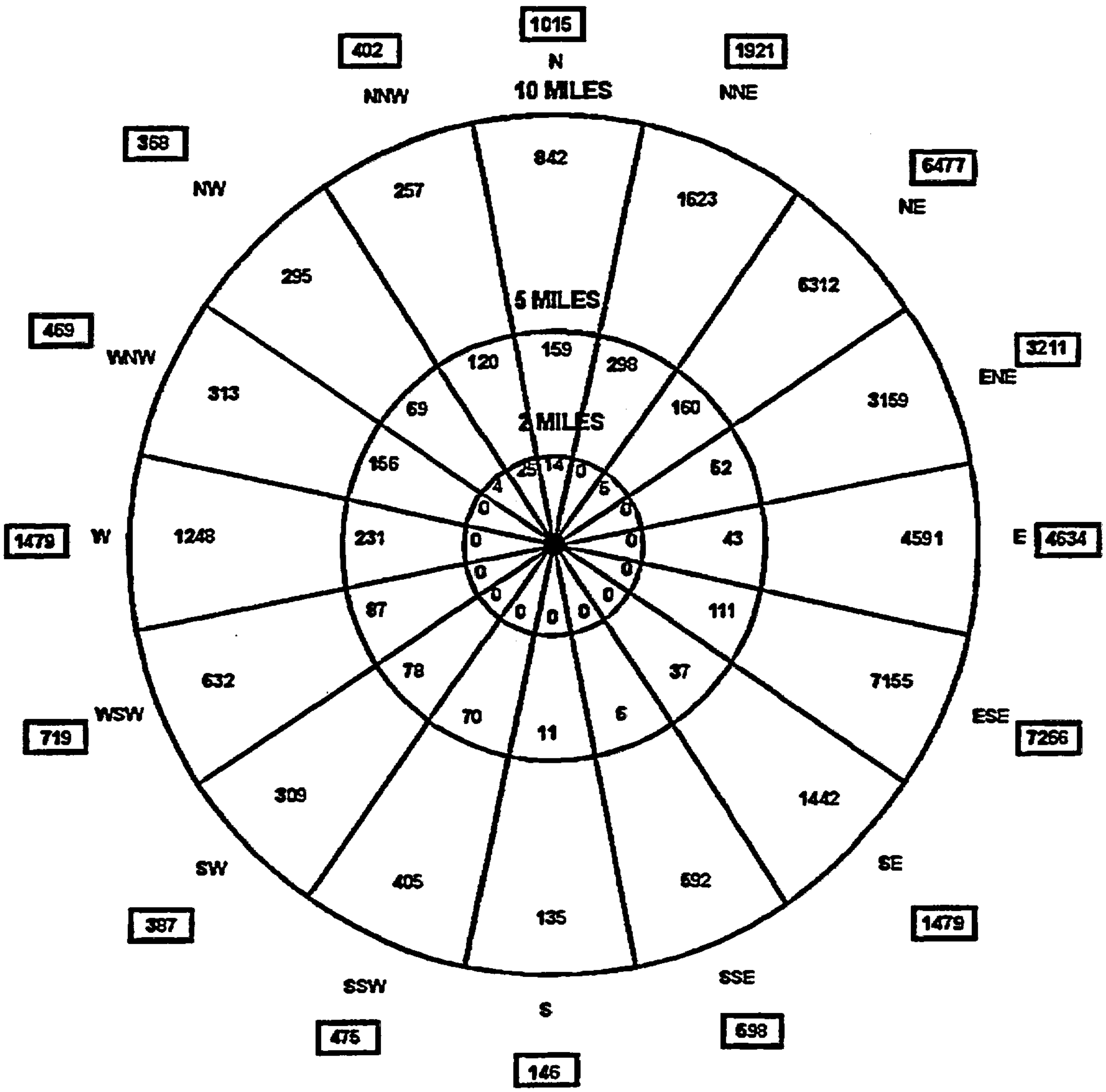
Population Data by Sub-Zone - Fall Weekday Adverse Weather**

Sub-Zone	Permanent Resident		Seasonal	Transient	Special Facility	Total Population
	Auto-Owning	Non-Auto Owning				
A	0	0	0	842	0	842
B	863	54	0	64	61	1,042
C	141	9	0	32	0	182
D	134	8	0	51	0	193
E	8,514	534	0	1,490	6,712	17,250
F	4,866	305	0	36	822	6,029
G	8,258	518	0	32	3,689	12,497
H	1,859	177	0	60	90	2,186
I	508	53	0	28	0	589
J	797	83	0	24	0	904
K	673	49	0	1,422	0	2,144
L	585	43	0	52	0	680
M	1,270	93	0	56	300	1,719
N	608	44	0	28	0	680
Full EPZ	29,076	1,970	0	4,217	11,674	46,937

** This scenario represents the largest population present within the 10-mile EPZ during an analyzed evacuation situation. This is based on the "Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone, Shearon Harris Nuclear Plant, January 1997".

Figure 1.8-1

Demographic Information by Sector

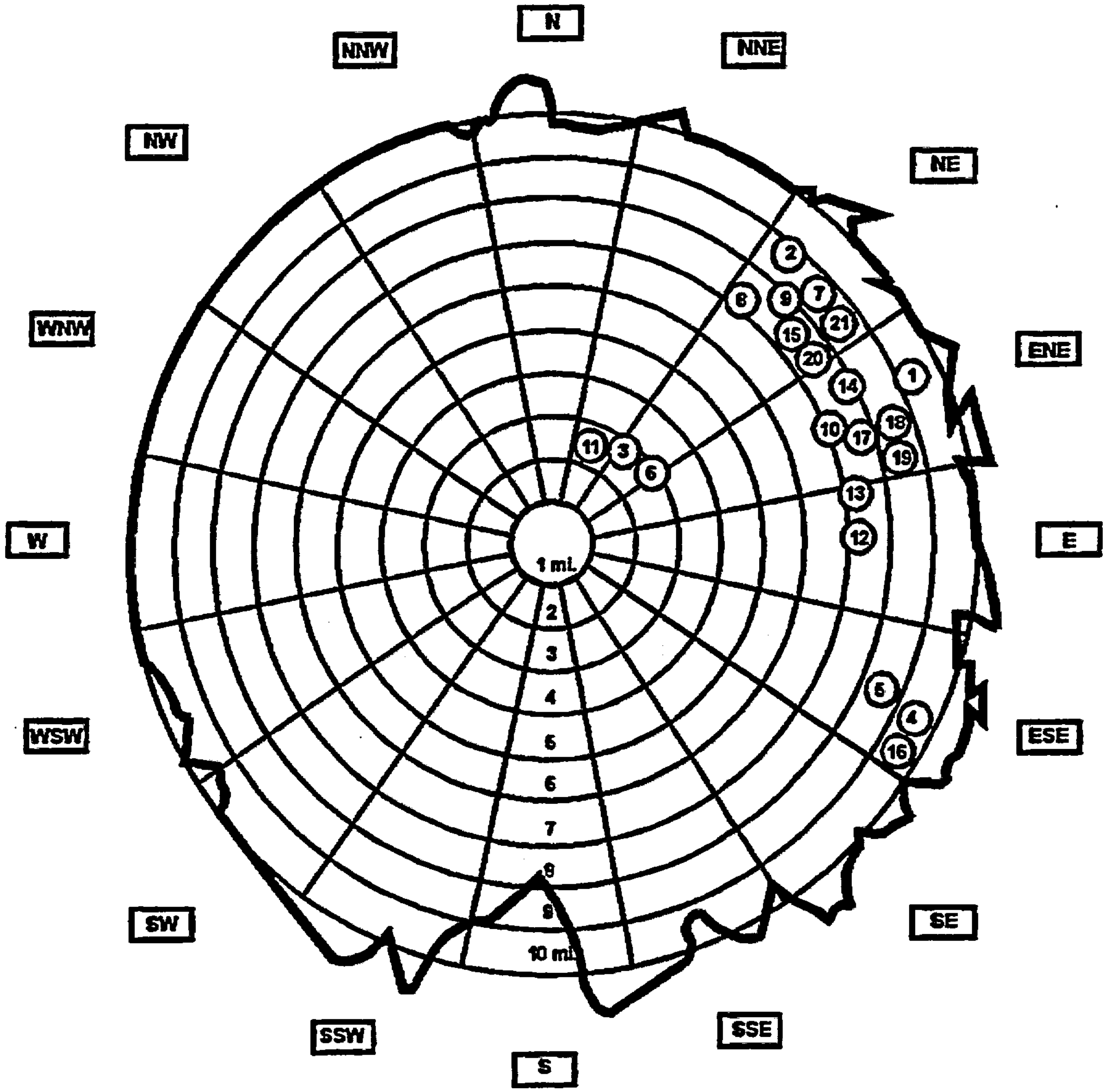


POPULATION TOTALS			
Distance (Miles)	Population	Total Miles	Total Population
0-2	48	0-2	48
2-5	1688	0-5	1736
5-10	29310	0-10	31046

NOTE: Data is based on the "Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone, Shearon Harris Nuclear Plant, January 1997".

Figure 1.8-2

Hospital and Family Care Facilities Located in the HNP 10-Mile EPZ



1 - Adam's Family Care #1 and #2
2 - Atwater's Rest Home
3 - Brown Family Care
4 - Brighton Mannor
5 - Comfort Care
6 - James Rest Home
7 - Lyles' Family Care
8 - Mary's Family Care Home
9 - Mathew's Family Care
10 - Francis Morning's Family Care
11 - Morrison's Family Care
12 - Mims Family Care Home
13 - Our Golden Ladies
14 - Rose Haven Rest Home
15 - Seagroves Family Care
16 - Southern Wake Hospital
17 - County Lane Group Home
18 - Hickory Street Group Home
19 - Creekway Group Home
20 - Olive Street Group Home
21 - Manson Street Group Home

NOTE: Wake County Department of Social Services maintains the current listing.

Table 1.8-2

HNP Plume Exposure EPZ Evacuation Time Estimates

Wind Direction (Degrees From)	Evacuation Area	Sub-zones Impacted	Summer Weekday (Good Weather)	Summer Weekend (Good Weather)	Late Fall Weekday (Adverse Weather)	Summer Evening (Good Weather)
2-Mile Radius		A	110	120	120	100
348-10	2-Mile + 5 Mile	A, D, K	130	130	130	130
	2-Mile + 10 Mile	A, D, K, H, I	190	190	190	190
11-34	2-Mile + 5 Mile	A, K	130	130	130	130
	2-Mile + 10 Mile	A, K, H, I, J	190	190	190	190
35-56	2-Mile + 5 Mile	A, K	140	140	140	140
	2-Mile + 10 Mile	A, K, I, J, M	190	190	180	180
57-79	2-Mile + 5 Mile	A, K, L	140	140	140	140
	2-Mile + 10 Mile	A, K, L, I, J, M	170	170	170	170
80-101	2-Mile + 5 Mile	A, K, L	170	140	170	140
	2-Mile + 10 Mile	A, K, L, J, M	180	170	180	170
102-124	2-Mile + 5 Mile	A, K, L	140	140	140	140
	2-Mile + 10 Mile	A, K, L, J, M, N	180	200	180	180
125-146	2-Mile + 5 Mile	A, B, L	190	190	190	190
	2-Mile + 10 Mile	A, B, L, M, N	190	190	190	190
147-169	2-Mile + 5 Mile	A, B, L	190	190	190	190
	2-Mile + 10 Mile	A, B, L, E, M, N	200	200	200	200
170-191	2-Mile + 5 Mile	A, B, L	190	190	190	190
	2-Mile + 10 Mile	A, B, L, E, M, N	200	200	200	200
192-214	2-Mile + 5 Mile	A, B	190	190	190	190
	2-Mile + 10 Mile	A, B, E, N	200	200	200	200
215-236	2-Mile + 5 Mile	A, B, C	190	190	190	190
	2-Mile + 10 Mile	A, B, C, E, F	210	200	210	200
237-259	2-Mile + 5 Mile	A, B, C	190	190	190	190
	2-Mile + 10 Mile	A, B, C, E, F, G	200	200	200	200
260-281	2-Mile + 5 Mile	A, B, C, D	190	190	190	190
	2-Mile + 10 Mile	A, B, C, D, F, G, H	260	240	300	230
282-304	2-Mile + 5 Mile	A, C, D	170	170	170	170
	2-Mile + 10 Mile	A, C, D, F, G, H	230	220	230	200
305-326	2-Mile + 5 Mile	A, C, D, K	190	190	190	190
	2-Mile + 10 Mile	A, C, D, K, F, G, H	200	200	200	200
327-347	2-Mile + 5 Mile	A, D, K	130	130	130	130
	2-Mile + 10 Mile	A, D, K, G, H, I	210	210	210	210
Full EPZ	10 Mile	Full EPZ	260	240	300	230

Evacuation times include notification and alerting of the public via primary means (15 minutes), mobilization and preparation of the public for evacuation, and evacuation to the outer boundary of all the local planning zones being evacuated.

Source: Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone: Shearon Harris Nuclear Power Plant, January 1997.

Table 1.8-3

Schools Located In the HNP 10-Mile EPZ

School	Quadrant	Distance from Plant (miles)
Apex Elementary	NE	8.5
Apex High	NE	10.0
Apex Middle	NE	8.5
Baucom Elementary	NE	9.5
Lufkin Road Middle	NE	9.0
Fuquay-Varina Middle	ESE	9.5
Fuquay-Varina High	ESE	9.5
Lincoln Heights Elementary	ESE	8.5
Moncure Elementary	W	6.5
Holly Springs Elementary	ENE	7.0
Olive Chapel Elementary	NNE	7.5
Salem Elementary	NNE	10.0

2.0 ORGANIZATION AND RESPONSIBILITIES

2.1 General

There are requirements for actions in an emergency that go beyond those encountered during routine operations. To meet these additional demands and provide an effective response to the emergency, the HNP Emergency Plan employs an organizational concept that has four features.

A. Whenever the Plan is activated, a single individual is charged with the responsibility for and the authority to direct all actions necessary to respond to the emergency.

B. The primary responsibility of the individual in charge is to assure that all emergency response functions are carried out. Upon activation of the Plan, this individual is freed of all other responsibilities and thus able to focus on managing the emergency response.

C. Specific individuals are assigned the responsibility of carrying out predefined critical actions and emergency measures.

D. There is a mechanism established to provide additional resources as necessary to respond to the emergency, which provides continuity of response on each critical action.

This concept of organization is compatible with and integrated into the normal mode of operation. The shift operating crew is routinely required to correct minor malfunctions of equipment and to diagnose the consequences of radioactivity releases. There are a number of procedures to guide operators in responding to equipment malfunctions and instrument alarms. There are also procedures to maintain effective control over contamination and radiation exposures. Emergency procedures basically involve an extension of these existing plant procedures.

2.2 Emergency Organization

The emergency response resources available to respond to an emergency consist of the personnel at the plant, at Corporate Headquarters, at other Company nuclear plants, the Harris Energy & Environmental Center and, in the longer term, at other organizations involved in the nuclear industry. Throughout CP&L there exists a staff of well-trained and experienced engineers, scientists, and technicians. These personnel represent a pool of technical expertise that can be called upon to provide additional support to the corporate emergency response and recovery organizations, if required. Corporate emergency response personnel do not receive specific training for emergency response and do not take actions which implement this emergency plan.

The plant Emergency Response Organization (ERO) is composed of a broad spectrum of personnel with specialties in operations, maintenance, engineering, radiochemistry, health physics, material control, fire protection, security, and emergency planning. The greatest number of personnel with these specialties are available during day shift operations; however, needed specialists can be recalled to the site at any time.

The first line of defense in responding to an emergency lies with the normal on-duty operating shift when the emergency begins. Shift members are assigned defined emergency response roles, as shown in Table 2.2-1, that are to be assumed whenever an emergency is declared. As additional personnel are called in to the plant, a smooth transition occurs since each individual knows ahead of time what their responsibilities will be. A current call list of ERO members is maintained in the Main Control Room and procedures are available to activate the ERO automatically or manually.

The Company is committed to providing staffing to effectively contain any emergency which might occur at its nuclear facilities. Depending on the emergency at hand, personnel with required expertise will be contacted on a priority basis as shown in Table 2.2-1. Additional personnel will be available to provide communications, on-site and off-site radiological assessment, repair and corrective actions, and technical support within a short period of time. Depending on weather conditions, 30-45 minutes should provide enough time to make the appropriate staff available to augment the plant on-shift organization. The plant ERO will continue to be augmented such that within 60-75 minutes after notification, additional personnel will be added to provide the necessary support. Additional personnel will continue to supplement the on-site ERO as necessary to meet the requirements of this Plan.

The fully augmented on-site ERO is shown in Figure 2.2-1 and personnel assignments are provided in Table 2.2-1 and/or procedures. The on-site ERO utilizes the basic plant organization structure as the principle guideline in emergency assignments. This philosophy assures whenever possible, that personnel will be performing emergency functions that are similar to their normal operating duties. Each emergency position has a succession of command from assigned, trained alternates.

2.3 Command and Control

The position of Site Emergency Coordinator is activated for command and control purposes upon declaration of an emergency. Until relieved by the Emergency Response Manager, the Site Emergency Coordinator is delegated the authority to act on behalf of the Company to manage and direct all emergency operations involving the facility. Upon activation of the Emergency Operations Facility, the Emergency Response Manager assumes responsibility of overall emergency response and performs those requirements for all off-site related activities. The Site Emergency Coordinator maintains overall on-site emergency responsibilities including emergency classification and, after EOF activation, reports to the Emergency Response Manager.

The Superintendent - Shift Operations on duty at the time the emergency is declared shall initially assume the position of Site Emergency Coordinator from the Main Control Room.

The following conditions for command and control apply:

A. If the Site Emergency Coordinator becomes incapacitated for any reason, a designated alternate shall assume the position of Site Emergency Coordinator.

B. Once the Technical Support Center is activated the position of Site Emergency Coordinator is transferred from the Main Control Room to a qualified individual in the TSC.

C. The Site Emergency Coordinator, or Emergency Response Manager after the EOF is activated, may not delegate the responsibility for notification of and making recommendations to authorities responsible for off-site measures.

D. The Site Emergency Coordinator may consult with others, but may not delegate the responsibility to determine the appropriate emergency action level for the conditions.

E. The Site Emergency Coordinator is authorized to request Federal and State assistance until the EOF is activated, whereupon such requests are made under the direction of the Emergency Response Manager.

NOTE: If deemed prudent in order to ensure an adequate response to the emergency, the Site Emergency Coordinator-MCR may direct that the TSC and/or EOF assume responsibility for any/all discrete functions prior to reaching full staffing levels or to activate only those functions which the SEC-MCR feels are necessary for an adequate emergency response.

F. The conditions for transfer of designated responsibilities from the Superintendent-Shift Operations (Site Emergency Coordinator-MCR) to the Site Emergency Coordinator-TSC and the Emergency Response Manager (EOF) are:

1) The TSC and EOF are ready to be activated and to assume emergency functions.

2) The Site Emergency Coordinator-TSC and the Emergency Response Manager have received a briefing on the status of the emergency.

2.4 Assignment of Responsibilities

All emergency response personnel with responsibilities listed in 2.4.A through 2.4.E will:

- Be trained and qualified to perform the assigned responsibilities as specified in Section 5.2.
- Be formally relieved by a qualified alternate trained for duty in the particular position before leaving that position.
- Maintain a record of activities where appropriate.

2.4.1 Main Control Room

A. Superintendent-Shift Operations: Until an emergency is declared, the Superintendent-Shift Operations has the following responsibilities relating to the Emergency Plan:

1) Direct the activities of the Operations staff.

2) Recognize an off-normal condition as indicated by instrument readings or observation.

3) Implement any Emergency Operating Procedures.

4) Determine when an Emergency Action Level has been met or exceeded, declare an emergency, and assume the position of Site Emergency Coordinator-MCR.

B. Site Emergency Coordinator-MCR: The primary person assigned to the position of Site Emergency Coordinator-MCR during the initial stages of an emergency is the Superintendent-Shift Operations. Once the Technical Support Center is activated the responsibilities of Site Emergency Coordinator-MCR are turned over to the Site Emergency Coordinator-TSC and the Emergency Response Manager in accordance with the implementing procedures.

The Site Emergency Coordinator-MCR, shall not delegate the following responsibilities:

1) Classification of the emergency.

2) Approval of required notifications made to the State/Counties and the NRC.

3) Establishment of on-site mission priorities in response to the emergency.

4) Approval of planned radiation exposures for CP&L personnel in excess of 5 Rem TEDE or entry into radiation fields greater than 25 Rem/hr.

5) Review and approval of deviations from Technical Specifications or license conditions.

6) Authorization of the administration of Potassium Iodide to on-site emergency workers.

7) Approval of Protective Action Recommendations made to the State/Counties.

8) Termination of the emergency.

C. Plant Operations Director: The Plant Operations Director, located in the Main Control Room after activation of the Technical Support Center, is responsible to the Site Emergency Coordinator-TSC for providing direction to the Main Control Room Staff, the Fire Brigade, and the First Aid Teams. The POD is trained as a SAMG decision maker whose focus is on the operational aspect of the strategy developed by the TSC.

D. Fire Brigade Team Leader: A Fire Brigade Team Leader is established on all shifts. When a fire occurs, the Fire Brigade Team Leader is the on-scene commander for fighting the fire and directs the activities of the Fire Brigade. The Fire Brigade Team Leader reports to the Site Emergency Coordinator-MCR or to the Plant Operations Director after activation of the Technical Support Center.

E. Fire Brigade: When a fire is announced, the Fire Brigade reports to the Fire Brigade Team Leader. If a fire occurs, the Fire Brigade reports to the Fire Staging Area where fire-fighting equipment is located, and then responds to the fire scene. The fire brigade is composed of on-shift personnel trained in fighting fires as described in Section 5.2.

F. First Aid Team: A First Aid Team is established on all shifts. The First Aid Team performs/coordinates emergency first aid and search and rescue activities. The First Aid Team reports to the Site Emergency Coordinator-MCR or to the Plant Operations Director after activation of the Technical Support Center.

G. Emergency Communicator-MCR: Initially filled with on-shift personnel, is appointed by and reports to the Site Emergency Coordinator-MCR and is responsible for communicating with:

1) Off-site authorities (County, State, NRC, and so forth) to perform required notifications of the declaration, upgrading, termination of an emergency prior to the activation of the TSC and EOF.

2) The plant Emergency Response Organization (during off-hours) when CP&L emergency facilities are being activated.

3) Local Immediate Response Organizations (medical, fire, law enforcement, and so forth) if their assistance is needed.

2.4.2 Technical Support Center

A. Site Emergency Coordinator-TSC: The Site Emergency Coordinator-TSC is responsible for overall command and control of the on-site response to the emergency. The Site Emergency Coordinator is also responsible for providing guidance to the Technical Analysis Director, Radiological Control Director, Communications Director, Security Director, Plant Operations Director and the Emergency Repair Director.

Upon activation of the Technical Support Center the Site Emergency Coordinator-TSC relieves the Site Emergency Coordinator-MCR of the following major responsibilities:

1) Classification of the emergency.

2) Establishment of on-site mission priorities in response to the emergency.

3) Approval of planned radiation exposures for on-site personnel in excess of 5 Rem TEDE or entry into radiation fields greater than 25 Rem/hr.

4) Review and approval of deviations from Technical Specifications or license conditions if the Site Emergency Coordinator-TSC is a Superintendent-Shift Operations, or ensure that such deviations are approved by a Superintendent-Shift Operations.

5) Authorization of the administration of Potassium Iodide to on-site emergency workers.

6) A trained SAMG decision maker whose focus is on the development and prioritization aspect of the SAMG strategy.

7) Termination of the emergency.

B. TSC-Senior Reactor Operator: The TSC-Senior Reactor Operator is located in the Technical Support Center and reports to the Site Emergency Coordinator-TSC and directs the TSC-ERFIS Operator. The TSC-Senior Reactor Operator is responsible for providing technical assistance related to plant conditions and operations and to perform monitoring and evaluations required for Severe Accident Management Guidelines.

C. TSC-ERFIS Operator: The Technical Support Center ERFIS Operator reports to the TSC-SRO and is located in the Technical Support Center. The position is responsible for providing/displaying any information from ERFIS requested by Technical Support Center personnel.

D. Technical Analysis Director: The Technical Analysis Director reports to the Site Emergency Coordinator-TSC and is located in the Technical Support Center. The Technical Analysis Director is responsible for providing direction to the Technical Support Center Accident Assessment Team, perform monitoring and evaluation required for Severe Accident Management Guidelines, and to direct AAT members to evaluate strategies that implement Severe Accident Management Guidelines.

E. TSC-Accident Assessment Team: The TSC-Accident Assessment Team reports to the Technical Analysis Director and is located in the Technical Support Center. The team is composed of a Shift Technical Advisor, Core Performance Engineer, Electrical/I&C Engineer, and Mechanical Engineer. They are responsible for providing recommendations to the Technical Analysis Director on problems as assigned.

F. Communications Director: The Communications Director, located in the Technical Support Center, reports to the Site Emergency Coordinator TSC. The Communications Director is responsible for providing direction to the Emergency Communicator-NRC, TSC-Telecomm/Computer Support, TSC Logkeeper and the Admin Team.

G. Emergency Communicator-NRC: The Emergency Communicator-NRC is located in the Technical Support Center and reports to the Communications Director. The Emergency Communicator-NRC is responsible for:

1) Generating required written notifications to the NRC in a timely manner.

2) Establishing contact with the NRC via the Emergency Notification System and providing any requested information of the status of the emergency.

H. TSC-Telecomm/Computer Support: TSC-Telecomm/Computer Support personnel are located in the Technical Support Center and report to the Communications Director. They are responsible for providing technical assistance required in the areas of telecommunications or computer support.

I. TSC Admin Team: The TSC Admin Team is located in the Technical Support Center and is composed of a Librarian and Admin Support personnel. They report to the Communications Director and are responsible for providing any documents, prints or other clerical services as requested by personnel in the Technical Support Center.

J. TSC Logkeeper: The TSC Logkeeper is located in the Technical Support Center and reports to the Communications Director. The TSC Logkeeper is responsible for recording the major activities that occur in the Technical Support Center during an emergency.

K. Radiological Control Director: The Radiological Control Director is located in the Technical Support Center and reports to the Site Emergency Coordinator-TSC. The Chemistry Coordinator and the Radiological Control Coordinator, both located in the Operations Support Center, report to the Radiological Control Director. The Radiological Control Director is responsible for:

1) Providing direction to onsite health physics and chemistry emergency response actions.

2) Ensuring that the Site Emergency Coordinator and other Directors in the Technical Support Center are kept informed of radiological/chemical conditions on and off site.

L. TSC HP Technician: The TSC HP Technician, normally located in the Technical Support Center, reports to the Radiological Control Director and is responsible for providing radiological support and monitoring activities within the TSC.

M. Security Director: The Security Director, normally located in the Technical Support Center, reports to the Site Emergency Coordinator-TSC and has the following major responsibilities:

1) Maintaining plant security in accordance with the provisions of the HNP Security Plan and Safeguards Contingency Plan.

2) Coordinating the accountability of personnel inside the Protected Area.

3) Providing Security Force personnel in support of emergency activities.

2.4.3 Operations Support Center

A. Emergency Repair Director: The Emergency Repair Director, located in the Operations Support Center, reports to the Site Emergency Coordinator-TSC. The Emergency Repair Director is responsible for providing direction to the total on-site maintenance and equipment restoration effort from the Operations Support Center.

B. Damage Control Coordinator: The Damage Control Coordinator, located in the Operations Support Center, reports to the Emergency Repair Director. The Damage Control Coordinator is responsible for providing direction to the Damage Control Team Leaders, Maintenance Planners, OSC Storekeeper and OSC Logkeeper.

C. Damage Control Team Leaders: The Damage Control Team Leaders are appointed by the Damage Control Coordinator. They are responsible to the Damage Control Coordinator for on-the-scene supervision of the Damage Control Teams to which they are assigned.

D. Damage Control Teams: The Damage Control Teams are dispatched by the Damage Control Coordinator, from their initial assembly point in the Operations Support Center, to the scene of an emergency repair or damage assessment requirement. The Damage Control Teams report to the on-scene Damage Control Team Leader and are composed of mechanical, instrument and control, and electrical maintenance personnel.

E. Maintenance Planners: Maintenance Planners, located in the Operations Support Center, report to the Damage Control Coordinator. The Maintenance Planners are responsible for developing plans for emergency repair, determining spare parts needed to make the repairs and estimating the amount of time required to perform the emergency repairs.

F. OSC Storekeeper: The OSC Storekeeper, located in the Operations Support Center, reports to the Damage Control Coordinator. The OSC Storekeeper is responsible for expediting the spare parts and tools needed in support of emergency activities.

G. OSC Logkeeper: The OSC Logkeeper, located in the Operations Support Center, reports to the Damage Control Coordinator. The OSC Logkeeper is responsible for recording the major activities that occur in the Operations Support Center during an emergency.

H. Radiological Control Coordinator: The Radiological Control Coordinator, located in the Operations Support Center, is responsible to the Radiological Control Director for providing direction to the Radiological Control Teams during an emergency.

I. Radiological Control Teams: Radiological Control Teams report to the Radiological Control Coordinator and are composed of health physics personnel. They assemble initially in the Operations Support Center and are subsequently dispatched wherever personnel radiation control and decontamination functions are needed.

J. Chemistry Coordinator: The Chemistry Coordinator, located in the OSC, is responsible to the Radiological Control Director for providing direction to the Chemistry Team during an emergency.

K. Chemistry Team: Chemistry Teams report to the Chemistry Coordinator and are composed of plant chemistry personnel. They assemble initially in the Operations Support Center and are subsequently dispatched to sampling stations, the PASS Panel, and the laboratory.

2.4.4 Emergency Operations Facility

A. Emergency Response Manager: The Emergency Response Manager, located in the Emergency Operations Facility, is responsible for overall command and control of the CP&L response to the emergency. The Emergency Response Manager is also responsible for providing guidance to the Technical Analysis Manager, Radiological Control Manager, Communications Manager, and the Admin and Logistics Manager.

Upon activation of the Emergency Operations Facility the Emergency Response Manager relieves the Site Emergency Coordinator-MCR of the following major responsibilities:

1) Approval of required notifications to the State/Counties.

2) Approval of planned radiation exposures for off-site CP&L personnel in excess of 5 Rem TEDE or entry into radiation fields greater than 25 Rem/hr.

3) Approval of the administration of Potassium Iodide to off-site HNP emergency workers.

4) Approval of Protective Action Recommendations.

5) Direct interface with offsite authorities.

6) Coordination of Dose Projection and Environmental Monitoring activities.

7) A trained SAMG decision maker whose focus is on the offsite consequences of the strategy recommended by the TSC. The ERM has the ultimate approval authority for strategy implementation.

B. EOF-Senior Reactor Operator: The EOF-Senior Reactor Operator is located in the Emergency Operations Facility and reports to the Emergency Response Manager. The EOF-Senior Reactor Operator is responsible for providing technical information and assistance related to plant conditions and operations.

C. EOF ERFIS Operator: The EOF ERFIS Operator reports to the EOF Senior Reactor Operator and is located in the Emergency Operations Facility. The position is responsible for providing/displaying any information from ERFIS requested by Emergency Operations Facility personnel.

D. Emergency Preparedness Advisor: The Emergency Preparedness Advisor, located in the Emergency Operations Facility, reports to the Emergency Response Manager in the EOF and advises the Emergency Response Manager and other Emergency Response Organization personnel on implementation of the Emergency Plan and implementing procedures.

E. News Coordinator: The News Coordinator, located in the Emergency Operations Facility, reports to the Emergency Response Manager. The News Coordinator has the responsibility for preparing and coordinating the approval of news releases.

F. Administrative and Logistics Manager: The Administrative and Logistics Manager, located in the Emergency Operations Facility, reports to the Emergency Response Manager and is responsible for direction of activities of the Administrative Team Leader and Admin Building Assembly Area Leader.

G. EOF Telecomm/Computer Support: EOF Telecommunications/Computer Support personnel are located in the EOF and report to the Administrative and Logistics Manager. They are responsible for providing technical assistance required in the areas of telecommunications or computer support.

H. Admin Team Leader: The Admin Team Leader, located in the Emergency Operations Facility, reports to the Administrative and Logistics Manager and is responsible for directing the actions of the Admin Team.

I. Admin Team: The Admin Team, located in the Emergency Operations Facility, consists of a Setup Leader, Librarian and Admin Support personnel. They report to the Admin Team Leader and are responsible for providing any documents, prints or other clerical services as requested by personnel in the Emergency Operations Facility.

J. EOF Logkeeper: The EOF Logkeeper is located in the Emergency Operations Facility and reports to the Admin Team Leader. The EOF Logkeeper is responsible for recording the major activities that occur in the Emergency Operations Facility during an emergency.

K. Assembly Area Leader: The Assembly Area Leader is responsible to the Admin and Logistics Manager, or prior to activation of this position, the Site Emergency Coordinator, for coordinating the activities in the Admin Building Assembly Area.

L. Technical Analysis Manager: The Technical Analysis Manager reports to the Emergency Response Manager and is responsible for direction of activities of the Emergency Operations Facility Accident Assessment Team.

M. EOF Accident Assessment Team: The EOF Assessment Team reports to the Technical Analysis Manager and is located in the Emergency Operations Facility. The team is composed of a Civil Engineer, Electrical Engineer, I&C Engineer, and Mechanical Engineer. They are responsible for providing recommendations to the Technical Analysis Manager on problems as assigned.

N. Radiological Control Manager: The Radiological Control Manager, located in the Emergency Operations Facility, reports to the Emergency Response Manager. The Radiological Control Manager is responsible for providing direction to the Dose Projection Team Leader, Technical Advisor and the EOF Health Physics Technician. The Radiological Control Manager is also responsible for:

1) Providing direction to offsite health physics emergency response actions.

2) Ensuring that the Emergency Response Manager and other Managers in the EOF are kept informed of radiological/chemical conditions on and off site.

O. EOF HP Technician: The EOF HP Technician, normally located in the Emergency Operations Facility, reports to the Radiological Control Manager and is responsible for providing radiological support and monitoring activities within the EOF.

P. Technical Advisor: The Technical Advisor, located in the Emergency Operations Facility, reports to the Radiological Control Manager. The Technical Advisor assists the Radiological Control Manager and staffs the HPN Line when requested by the NRC.

Q. Dose Projection Team Leader: The Dose Projection Team Leader, located in the Emergency Operations Facility, reports to the Radiological Control Manager. The Dose Projection Team Leader provides guidance to the Environmental Field Coordinator and the Dose Projection Team.

R. Dose Projection Team: The Dose Projection Team reports to the Dose Projection Team Leader and is located in the Emergency Operations Facility. The Dose Projection Team is responsible for performing source term and offsite dose calculations.

S. Environmental Field Coordinator: The Environmental Field Coordinator, located in the Emergency Operations Facility, is responsible to the Dose Projection Team Leader. The Environmental Field Coordinator is responsible for providing direction to the Environmental Monitoring Teams.

T. Environmental Monitoring Teams: Environmental Monitoring Teams report to the Environmental Field Coordinator after activation of the Emergency Operations Facility, or, prior to activation of the Emergency Operations Facility, to the Site Emergency Coordinator -MCR. Teams assemble at HE&EC and are subsequently dispatched in vehicles to the surrounding area. They are responsible for offsite plume tracking, monitoring and other sampling activities.

U. Communications Manager: The Communications Manager, located in the Emergency Operations Facility, reports to the Emergency Response Manager. The Communications Manager is responsible for providing direction to the Emergency Communicator-State/County and the Representatives to the State and County EOCs.

V. Emergency Communicator-State/Counties: The Emergency Communicator-State/Counties, located in the Emergency Operations Facility, reports to the Communications Manager. The Emergency Communicator-State/Counties is responsible for conducting timely notification and transfer of emergency information to the State and Counties.

W. Emergency Communicator-Corporate Comm/JIC: The Emergency Communicator-Corporate Comm/JIC, located in the Emergency Operations Facility, reports to the Communications Manager. The Emergency Communicator-Corporate Comm/JIC is responsible for providing information to support public information emergency response activities.

X. Representatives to the State/County EOCs: The Representatives to the State/County EOCs are located at the following:

N.C. State EOC	State Administrative Building in Raleigh, N.C., and is the principle Emergency Operations Center.
Wake County EOC	Wake County Courthouse, Raleigh, N.C.
Chatham County EOC	Law Enforcement Center, Pittsboro, N.C.
Harnett County EOC	Law Enforcement Center, Lillington, N.C.
Lee County EOC	Sanford Municipal Center, Sanford, N.C.

These representatives act as technical liaisons to facilitate communications and the coordination of information flow between the Site Emergency Coordinator or Emergency Response Manager and State/local authorities. They report to the Communications Manager in the Emergency Operations Facility.

2.4.5 Joint Information Center

A. Company Spokesperson: The Company Spokesperson, located in the Joint Information Center, reports to the Emergency Response Manager. The Company Spokesperson is responsible for providing guidance to the Company Technical Spokesperson, JIC Director, Admin Coordinator and Public Information Coordinator. The Company Spokesperson also has the following major responsibilities:

1) Maintain command and control of the Joint Information Center.

2) Coordinates and directs responses to media inquiries

3) Ensure that the composition and timeliness of CP&L News Releases are adequate.

4) Conduct periodic briefings with the news media.

5) Provide for timely exchange of information between other spokespersons.

B. Company Technical Spokesperson: The Company Technical Spokesperson, located in the Joint Information Center, reports to the Company Spokesperson. The Company Technical Spokesperson provides guidance to the Technical Specialist and has the following major responsibilities:

1) Direct the Technical Specialist to gather information from the EOF for CP&L news media briefings.

2) Provide timely and accurate technical information to the media during formal briefings.

C. Technical Specialist: The Technical Specialist, located in the Joint Information Center, reports to the Company Technical Spokesperson. The Technical Specialist is responsible for assisting the Company Technical Spokesperson in obtaining and developing technical emergency information.

D. JIC Director: The JIC Director, located in the Joint Information Center, reports to the Company Spokesperson. The JIC Director is responsible for the development and coordination of news releases and dissemination of information.

E. Administrative Coordinator: The Administrative Coordinator, located in the Joint Information Center, reports to the Company Technical Spokesperson. The Administrative Coordinator provides guidance to the Administrative Assistants and Media Badging Specialist.

F. Administrative Assistant: The Administrative Assistant, located in the Joint Information Center, reports to the Administrative Coordinator. The Administrative Assistant is responsible for providing administrative services and supplies to Joint Information Center personnel.

G. Media Badging Specialist: The Media Badging Specialist, located in the Joint Information Center, reports to the Administrative Coordinator. The Media Badging Specialist is responsible for controlling access to the Media Briefing Area and distributing information.

H. Public Information Coordinator: The Public Information Coordinator, located in the Joint Information Center, reports to the Company Spokesperson and directs the activities of the Public Information Specialists.

I. Public Information Specialist: The Public Information Specialist, located in the Joint Information Center, reports to the Public Information Coordinator. The Public Information Specialist is responsible for staffing telephone lines to respond to calls from the media and public.

2.5 Outside Organization Support

Outside organizations that support HNP in an emergency are described in Annex G.

TABLE 2.2-1

On-Shift Staffing For Emergencies

Functional Area	Major Tasks	Emergency Positions	Minimum Shift Size	Capability for Additions 30-45 min	60-75 min
1. Plant Operations and Assessment of Operational Aspects	Control Room Staff	SSO[c]	1	--	--
		USCO	1	--	--
		Control Operators	2	--	--
		Non-Licensed Operators	2[d]	--	--
2. Emergency Direction and Control	--	SEC-MCR (SSO[b])	1	--	--
		ERM[a]	--	--	1
		SEC-TSC[a]	--	--	1
3. Notification & Communication	Emergency Communicator	Plant Personnel	1	1	2
4. Radiological Assessment	Offsite Dose Assessment	Dose Projection Team Leader	--	1	--
	Offsite Surveys	Environmental Monitoring Team Personnel	--	2	2
	Onsite Surveys	Environmental Monitoring Team Personnel	--	1	1
	In-plant Surveys	Radiological Control Team Personnel	1	1	1
	Chemistry	Chemistry Team Personnel	1	--	1

(Continued on next page)

NOTES:

[a] Overall direction of facility response is assumed by the ERM when all facilities are activated. The direction of minute-to-minute facility operations remains with the SEC-TSC.

[b] On shift responsibility prior to activation of the EOF and TSC.

[c] After Activation of the EOF and TSC.

[d] One of the two non-licensed operators may be assigned to the Fire Brigade.

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TABLE 2.2-1 (continued)

On-Shift Staffing For Emergencies

Functional Area	Major Tasks	Emergency Positions	Minimum Shift Size	Capability for Additions 30-45 min	60-75 min
5. Plant Engineering Repair and Corrective Actions	Technical Support	Shift Technical Advisor	1	--	--
		Core Performance Engineering	--	1	--
		Mechanical Engineering	--	--	1
		Electrical Engineering	--	--	1
	Repair and Corrective Actions	Mechanical Maintenance	1[e]	--	2
		Electrical/I&C Maintenance	1[e]	2	1
6. In-Plant Protective Actions	Radiation Protection	Radiological Control Team Personnel	1[e] 1	2	2
7. Fire Fighting	--	--	5[f]	Local Support	
8. First Aid and Rescue Operations	--	Plant Personnel	2[e]	--	--
9. Site Access Control	Security & Accountability	Security Team Personnel	(g)	(g)	(g)
		CP&L TOTAL (Less Security):	15	11	16

NOTES:

[e] May be provided by shift personnel assigned other functions.

[f] Fire Brigade per FSAR 9.5.1

[g] Per Security Plan.

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Figure 2.2-1

On-Site Emergency Response Organization

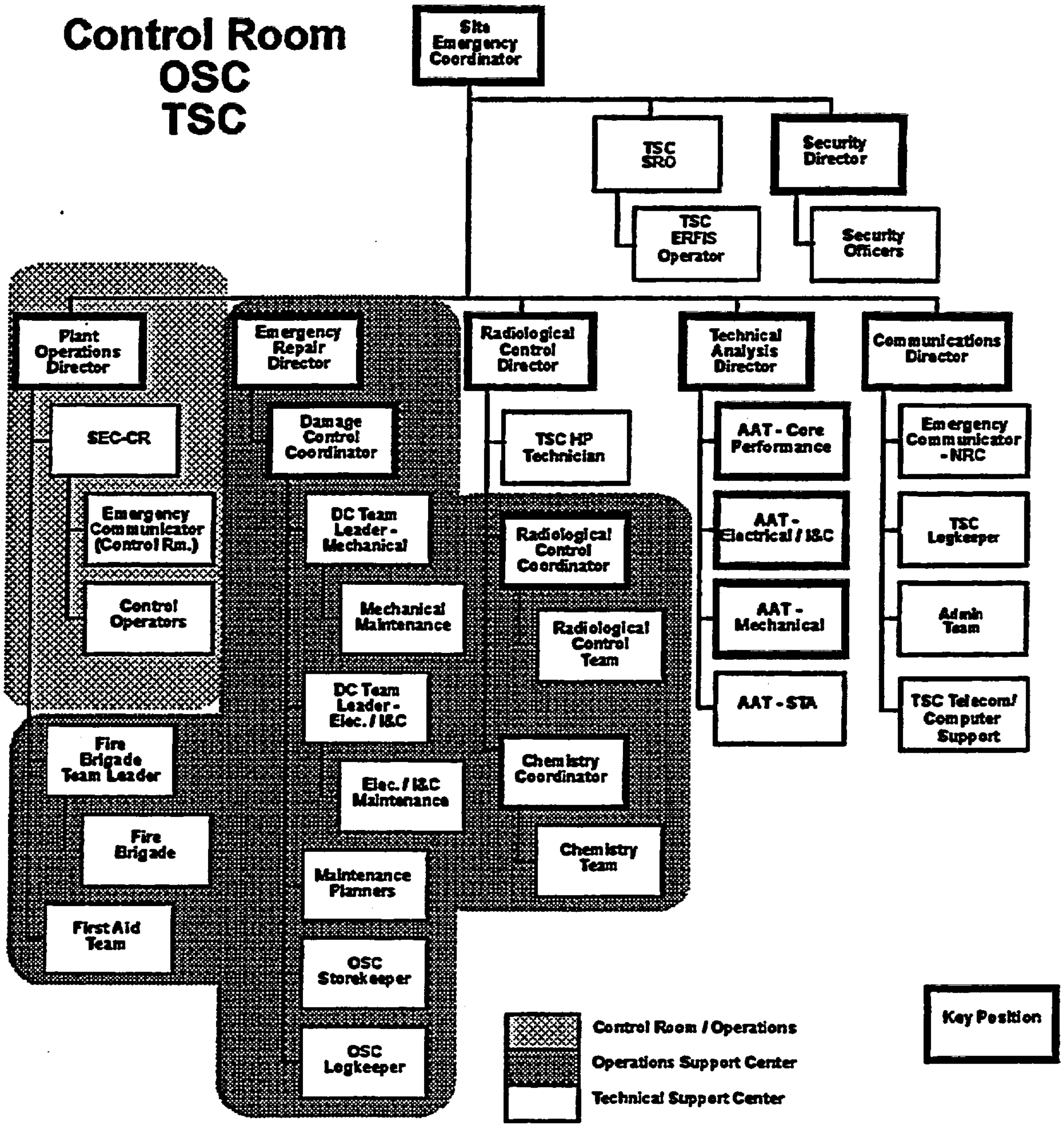
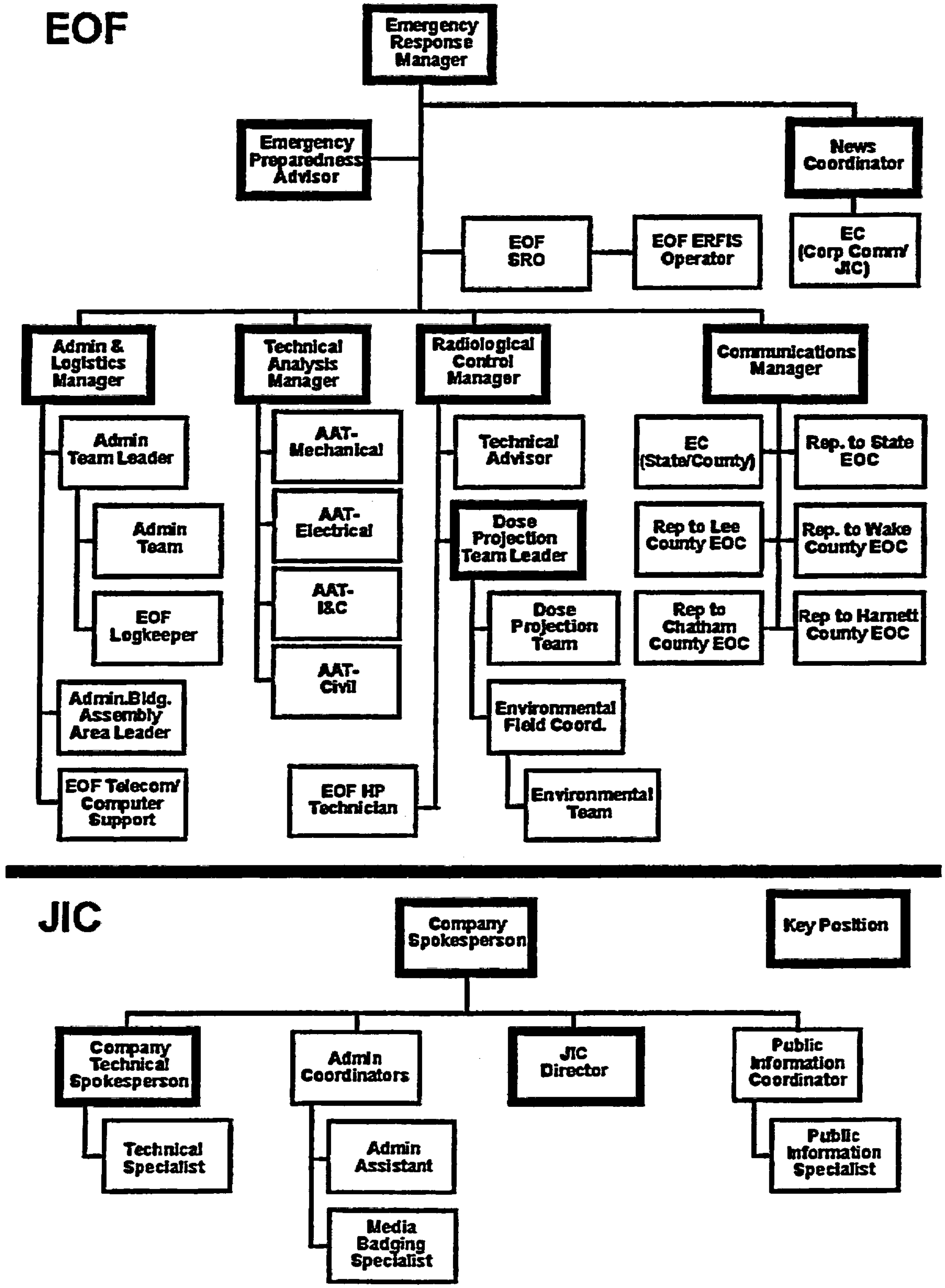


Figure 2.4-1

Off-Site Emergency Response Organization



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3.0 EMERGENCY FACILITIES, COMMUNICATIONS, AND EQUIPMENT

3.1 General

The purpose of emergency response facilities is to provide centralized locations for organized command and control of on-site activities and off-site activities performed by the Company such as environmental monitoring. Different groups within the Emergency Response Organization are provided with a location from where they may direct or perform the activities for which they are responsible while providing for coordination of activities with other organizations.

Adequate emergency facilities, communications, and equipment to support emergency response are provided and maintained. Carolina Power & Light Company (CP&L) Emergency Plans include provision for emergency response facilities as follows:

- Main Control Room (MCR)
- Technical Support Center (TSC)
- Operations Support Center (OSC)
- Emergency Operations Facility (EOF)
- Joint Information Center (JIC)

The Main Control Room is an emergency response facility that is operational on a day-to-day basis. Initially the emergency actions and in plant response would be directed by the Site Emergency Coordinator from the Main Control Room. Operations personnel would be dispatched from their work area located immediately north of the Main Control Room with assistance from on-shift health physics, maintenance, and security personnel as needed.

The facilities, other than the Main Control Room, are unmanned or used for other purposes on a day-to-day basis. In the event of an emergency, the TSC, OSC, EOF, and JIC would be activated in accordance with Section 4 of this Plan, "Emergency Measures and Operations."

In addition to the emergency response facilities, provision is made for on-site and off-site geophysical phenomena monitors (meteorological and seismic); radiological monitors; process monitors; and fire and combustion products detectors for use in initiating emergency measures and assessing the emergency. Each of these are described in subsequent paragraphs of this Chapter. Typical emergency supplies available for emergency facilities are indicated in Table 3.1-1.

3.2 Main Control Room (MCR)

3.2.1 Characteristics

A. Located in the Reactor Auxiliary Building as shown in Figure 1.5-3.

B. Main Control Room habitability and radiation protection is as described in Section 6.4 of the FSAR.

3.2.2 Functions

A. Reactor and plant control.

B. Interim location for Site Emergency Coordinator.

C. Accident recognition, classification, and mitigation.

D. Notification of off-site agencies.

E. Alerting of on-site personnel.

F. Initial dose projections.

G. Recommendations for immediate protective actions for the public.

H. Activation of HNP/CP&L emergency response facilities and recall of emergency personnel.

3.2.3 Emergency Equipment and Supplies

A. Main Control Board.

B. Emergency Response Facility Information System (ERFIS).

C. Safety Parameter Display System (SPDS is part of ERFIS).

D. Measurement and Indication of Regulatory Guide 1.97 (Rev. 2) variables (ERFIS).

E. Radiation Monitoring System (RMS).

F. Fire Detection System (adjacent room).

G. Seismic Monitoring Cabinet.

H. Gross Failed Fuel Detector Console.

I. Kitchen and sanitary facilities.

J. Reliable voice communications with the TSC, OSC, EOF, NRC Operations Centers, and State and local government 24-hour warning points.

K. See Table 3.1-1.

3.3 Technical Support Center (TSC)

3.3.1 Characteristics

A. Located within the Protected area at Elevation 324'-0" in the Fuel Handling Building, Section "K," approximately 400 feet walking distance from the Main Control Room (MCR) (primary route).

B. Protective clothing and portable breathing apparatus are kept in both the TSC and Main Control Room for personnel who must traverse between the two. Alternative paths are available that can be used based upon radiological conditions as determined by monitoring teams.

C. Exterior walls, roof, and floor are built to Seismic Category I, tornado, wind, and missile safety-related criteria.

D. Provided with radiation protection equivalent to Main Control Room habitability requirements such that the dose to an individual in the TSC for the duration of a design basis accident is less than 5 Rem TEDE. The Emergency Ventilation System includes HEPA and carbon filtration.

E. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

F. Reliable power for habitability systems and battery pack emergency lighting are provided.

G. Equipment is nonsafety-related and nonredundant.

H. Designed taking into account good human factors engineering principles.

3.3.2 Functions

A. Command center for Site Emergency Coordinator and assigned staff upon TSC activation. The TSC is officially activated by the Site Emergency Coordinator (SEC) when the necessary personnel and equipment are assembled at the TSC to carry out an emergency response function required by the emergency conditions.

B. Receives and displays plant status and parameters data on ERFIS.

C. Provides notifications to the NRC via Emergency Notification System.

D. Provides plant management and technical support to plant operations personnel.

E. Directs emergency response teams in the plant.

F. Assists the Main Control Room in accident assessment.

G. Performs emergency classification.

3.3.3 Emergency Equipment and Supplies

A. Reliable voice communications with the Main Control Room, EOF, OSC, NRC Operations Center and State and local government 24-hour warning points as described in Section 3.8 which follows.

B. Video System capable of displaying ERFIS information (such as, plant data, SPDS, and RMS) as discussed in Section 3.9.1.

C. Reference materials including Mechanical and Electrical Systems Drawings; Plant Operating Manual; FSAR; Corporate, Plant, State, and Local Emergency Plans; and a Document Services Library.

D. Decontamination and monitoring area.

E. Survey meter and area radiation monitor.

F. Fax and photocopier equipment.

G. See Table 3.1-1.

3.4 Operations Support Center (OSC)

3.4.1 Characteristics

A. Located in the Waste Processing Building inside the Protected Area (Figure 1.5-3).

B. The total area is approximately 1500 square feet in the Waste Processing Building HP Tech Work Area. This area includes a separate Command and Control area for coordinating and planning of OSC activities in addition to sufficient area for team members to standby for activities. Additional space in excess of 8500 square feet is available in adjacent offices and locker rooms to accommodate additional personnel as may be required. X

C. Alternate locations include the Turbine Building 261' North and Technical Support Center.

3.4.2 Functions

A. Assembly location for emergency teams for receipt of special equipment and assignments.

B. Dispatching of emergency teams.

3.4.3 Emergency Equipment and Supplies

A. Reliable voice communications with the Main Control Room, EOF, and TSC.

B. Supplies and equipment as shown in Table 3.1-1.

3.5 Emergency Operations Facility (EOF)

3.5.1 Characteristics

A. Located at Harris Energy & Environmental Center within 10 miles of the plant (see Figure 1.5-2).

B. Approximately 4800 square feet of space for approximately 70 persons including 14 NRC personnel.

C. Shielded to a protection factor (PF) of 5 and ventilated with an Emergency Ventilation System, with HEPA and carbon filtration, such that the total 30 day dose from all sources of a design basis accident for an individual in the EOF does not exceed 5 Rem TEDE or its equivalent to any other part of the body.

D. Structurally built in accordance with Uniform Building Code.

E. Environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment.

F. Backup power for habitability systems and battery pack emergency lighting are provided.

G. Provided with security to maintain readiness and to exclude unauthorized personnel when activated.

H. Designed taking into account good human factors engineering principles.

I. Alternate assembly area location for EOF staff is the 11th floor of Center Plaza Building in Raleigh, N.C.

3.5.2 Functions

A. Command center for Emergency Response Manager and assigned staff.

B. Upon activation, performs off-site notification, protective action recommendations, environmental monitoring, and dose projection.

C. Emergency communications systems monitoring and control.

D. Provides technical analysis and support.

E. Receives and displays plant status and parameters data on ERFIS.

F. Serves as the Recovery Center during recovery operations.

G. Primary location for writing technical news releases. The EOF may provide space for the media on a case-by-case basis, when authorized by the ERM.

3.5.3 Emergency Equipment and Supplies

A. Reliable voice communications with the TSC, Main Control Room, OSC, NRC Operations Centers and State and local government 24-hour warning points as described in Section 3.8.

B. Video system capable of displaying ERFIS information (such as, plant data, SPDS, and RMS) as discussed in Section 3.9.1.

C. Reference materials including Mechanical and Electrical Systems Drawings; Plant Operating Manual; FSAR; Corporate, Plant, State, and Local Emergency Plans.

D. Decontamination and monitoring area.

E. Survey meter and dosimetry.

F. Maps showing evacuation routes, evacuation areas, preselected radiological sampling and monitoring points, relocation centers in host areas, and shelter areas.

G. Fax and photocopier equipment.

H. Additional equipment as discussed in Section 3.8.2.

I. See Table 3.1-1.

3.6 Joint Information Center (JIC)

A. Located at the Center Plaza Building (11th floor) in downtown Raleigh, approximately 21 miles from the plant, with a media briefing room available for press conferences (located in the Raleigh Convention and Conference Center, also in downtown Raleigh).

B. Serves as the primary location for accumulating accurate and current information regarding the emergency conditions and writing non-technical news releases.

C. Provides work space and phones for public information personnel from the state, counties, NRC, FEMA, and industry-related organizations.

D. Provides telephones for use by the news media personnel.

E. Provides responses to media inquiries through media communicators who staff telephones that the media can call for information about an emergency.

F. Implements provisions for rumor control by providing a number of telephones which members of the public, who hear rumors, can call for factual information.

3.7 Non-CP&L Facilities

3.7.1 North Carolina-State Emergency Facilities

A. North Carolina State Emergency Operations Center (SEOC)

1) Assembly location for Governor, State Emergency Response Team and other officials as described in the State of North Carolina Emergency Response Plan.

2) Primary location for the coordination with federal, state, local authorities, and HNP as described in the State of North Carolina Emergency Response Plan.

3) Located at the Division of Emergency Management Headquarters, 116 W. Jones Street, Raleigh, North Carolina.

B. State Emergency Response Team (SERT)

1) A designated staff of specialists who assist State officials as described in the State of North Carolina Emergency Response Plan.

2) Located at the Division of Emergency Management Headquarters, 116 W. Jones Street, Raleigh, North Carolina.

3.7.2 County Emergency Operations Centers

A. Chatham County Emergency Operations Center (EOC)

1) Located in the Law Enforcement Center in Pittsboro

2) Functions are described in the State of North Carolina Emergency Response Plan

B. Harnett County Emergency Operations Center (EOC)

1) Located in the Harnett County Law Enforcement Building in Lillington.

2) Functions are described in the State of North Carolina Emergency Response Plan.

C. Lee County Emergency Operations Center (EOC)

1) Located in the Police Department of the Sanford Municipal Center, Sanford, N.C.

2) Functions are described in the State of North Carolina Emergency Response Plan.

D. Wake County Emergency Operations Center (EOC)

1) Located in the Wake County Courthouse in Raleigh.

2) Functions are described in the State of North Carolina Emergency Response Plan.

3.8 Communications Systems

3.8.1 Plant Communications System

A description of the plant communications systems is contained in Section 9.5.2 of the FSAR and consists of the following:

A. Private Automatic Branch Exchange (PABX) Telephone System covering the Main Control Room, TSC, EOF, and OSC.

B. Site paging system (accessed by Telephone System).

C. Sound-powered telephone system.

D. Two radio communications networks, one for security and one for operations.

E. Dedicated radio system from security center to local law enforcement agencies.

F. Plant PABX telephone system is powered from batteries charged by a rectifier.

G. Backup power is provided to fixed radio equipment.

3.8.2 Harris E&E Center PABX Telephone and Other Radio Systems

A. The Harris E&E Center (HE&EC) PABX telephone system includes:

   1) The HE&EC Private Automatic Branch Exchange (PABX) telephone system covers the Main Control Room, TSC, EOF, and OSC.

   2) An off-site Notification System (Selective Signaling System) provides communications to State and County warning points and Emergency Operations Centers from the Main Control Room, TSC, EOF, and Auxiliary Control Panel.

   3) The HE&EC PABX telephone system is powered from batteries charged by a rectifier.

B. Other radio system includes:

   1) Radio communications (separate from plant radios) with mobile and portable units used by the Environmental Monitoring Teams.

   2) Radio communications on the State Environmental Monitoring and Area B Channels.

   3) Mobile and portable radios are battery-powered.

3.8.3 Off-Site Communications Systems

A. Corporate Telephone Communications System is interconnected with plant PABX and utilizes microwave transmission equipment.

B. Commercial telephone connections to PABX, emergency telephone system, dedicated lines to emergency facilities, and lines to the Joint Information Center.

C. Load Dispatcher Radio Communications.

D. NRC Emergency Notification System (ENS) Phone.

E. NRC Health Physics Network (HPN) Phone.

3.8.4 Dialogic Communicator Automated Notification System

A computerized emergency response personnel call out computer is available to notify the CP&L Emergency Response Organization personnel and the NRC resident inspector of emergency declarations at the plant. The system provides instructions for activation of the on-site emergency facilities and the near site Emergency Operations Facility. Provisions are provided for remote activation of the system via telephone lines and for password protection from unauthorized use of the system.

3.9 Assessment Equipment

Use of the equipment described in this section during an emergency is detailed in Plant Procedures.

3.9.1 Emergency Response Facilities Information System (ERFIS) and Safety Parameter Display System (SPDS)

ERFIS receives raw data from sensors in the field and processes the data to provide meaningful information for the user. The ERFIS system consists of the following major parts: Field input multiplexer, ERFIS Host Computer, ERFIS Real Time Information Network (RTIN) Plant Server, and ERFIS-EDS personal computer work-stations. ERFIS-EDS work-stations are located in the Main Control Room, Technical Support Center (TSC), Emergency Operations Facility (EOF) and the ERFIS Computer Room. Some designated work-stations will have the capability to access the Site Business Local Area Network (LAN) when not in ERFIS-EDS mode. The TSC and EOF work-stations can be configured to run from the Simulator during drills and exercises.

The field input multiplexer obtains analog, digital, and sequence-of-events inputs from field sensors. The ERFIS Host receives these inputs, converts the raw analog inputs to engineering units, and updates the Current Value Table (CVT) at rates of 0.1 to 30 seconds. Processing consists of alarming points that exceed predefined limits, archiving input data, and performing various calculations and reports on a periodic or on-demand basis.

The ERFIS-EDS Plant Servers contain a copy of the CVT that is updated over shared memory with the ERFIS Host. The ERFIS-EDS work-stations are connected to the servers via dedicated Ethernet LANs. The work-stations communicate with the server in EDS Mode or over a serial link to the ERFIS Host in Emulation Mode. User functions use one of these two methods of communication. EDS tasks are those that run on the local work-station and get CVT data from the server. Emulation tasks use serial communication with the ERFIS Host and receive the entire display over the serial link.

There is a Primary and Backup ERFIS Host computer and a Primary and Backup ERFIs-EDS Plant Server. When a failure occurs on a primary system, an automatic failover occurs to the backup system. Failover of the ERFIS Host and ERFIS-EDS Server occur independently of each other.

The Safety Parameter Display System (SPDS) is a software subsystem of the ERFIS. The SPDS consists of a top-level display showing the status of Critical Safety Function Parameters at all times and a general display area for a summary display, graphic display of status trees, or plots of key parameters. An assigned SPDS display is provided in the Main Control Room and ERFIS terminals in any location can display SPDS with a single key stroke.

The SPDS will access all available signals and will display information related to:

A. Subcriticality

B. Core Cooling

C. Heat Sink

D. (Reactor Vessel) Integrity

E. Containment

F. (Reactor Coolant System) Inventory

Secondary displays will consist of graphic representations of the above critical safety functions and their status.

Additional detail and design criteria for the SPDS are provided in Item I.D.2 of the FSAR TMI Appendix.

3.9.2 Seismic and Hydrological Data

HNP has two distinct and separate seismic monitoring systems for the site. A seismic monitoring system, described in Section 3.7.4 of the FSAR, is located inside safety-related structures and measures horizontal and vertical acceleration. A second system, consisting of two free field strong motion detectors, is located at points on-site and must be read locally at each location. The recorded analog signal can be put on tape playback in the Main Control Room.

Offsite seismic monitoring information can be obtained from the United States Geological Survey's National Earthquake Information Center.

The design basis flood, probable maximum precipitation, and other improbable, conceivable extremes in hydrologic natural phenomena are well below any design limits for this site. Refer to FSAR Sections 2.4.2 and 2.4.3.

3.9.3 Radiological Monitoring

The Radiation Monitoring System (RMS) is a plant-wide radiation information gathering and control system encompassing the process and effluent monitors and the area and airborne monitors. Radiological monitors are provided for plant systems as described in the FSAR Sections 11.5 and 12.3.4

Effluent radiological monitors are provided for:

- Plant Vent Stacks

- Turbine Building Drains
- Tank Area Drain Transfer Pumps
- Treated Laundry and Hot Shower Tank Pumps
- Secondary Waste Sample Tank
- Main Steam Lines

The types, ranges, and locations of monitors are listed in Tables 11.5.2-1, 11.5.2-2 and 12.3.4-1 of the FSAR.

Typical portable radiation monitors and laboratory equipment are described in Section 12.5 of the FSAR.

The locations of the normal off-site and on-site environmental monitoring stations, and the location of the TLD monitoring stations are described in the Off-Site Dose Calculation Manual. Additional predetermined emergency off-site monitoring locations are contained in environmental monitoring procedures.

The Radiation Monitoring System, (RMS) provides the necessary activity or radiation levels required for determining source terms in dose projection procedures. The RMS is data linked to the ERFIS and radiation monitoring channel values are available in the TSC and EOF via ERFIS. The isotopic mix is based upon the mix discussed in EPM-600. PASS grab samples and on-site or off-site monitoring samples can then be analyzed to determine the true isotopic mix and the results used in the computerized dose projection software.

3.9.4 Normal and Post-Accident Sampling System (PASS)

The Primary Sampling System and the Secondary Sampling System are available to collect routine fluid and gaseous samples as described in FSAR 9.3.2.

The Post-Accident Sample System is provided to collect and analyze targeted fluid and gaseous samples under accident conditions within three hours of the time a decision is made to obtain the information. The PASS consists of two major components, the liquid sample system and the remote sample dilution panel, or RSDP. The RSDP's purpose is to obtain containment atmosphere samples and it relies on the containment hydrogen monitoring system (FSAR 6.2.5) to be in service to provide a pathway for sample collection.

Samples results are one of several methods used to provide information in support of core damage and offsite dose assessment activities.

3.9.5 Meteorological Instrumentation

The plant has a permanent meteorological monitoring station located within the exclusion area boundary for display and recording of wind speed, wind direction, and differential temperature for use in making off-site dose projections. Meteorological information is presented in the Main Control Room, the TSC, and the EOF by means of a computer. Additional information on the on-site meteorological monitoring system can be found in Section 2.3.3 of the FSAR.

Carolina Power & Light Company has the capability to access the National Weather Service on a 24-hour-per-day basis to provide backup should the on-site system fail. This backup source of meteorological data is the closest location which can provide reliable representative meteorological information.

Contracted weather services may be contacted during severe weather periods. They analyze national and local weather in order to provide localized weather forecasts for the System or for the HNP area as appropriate. The meteorologists can provide forecasts and current data reflecting conditions corresponding to their evaluation of weather data received from the National Weather Service and other sources. The NRC and State agencies may contact the weather service for appropriately formatted information and check meteorology data (current and forecasted) for the HNP area.

In the event that the on-site meteorological tower or monitoring instrumentation becomes inoperative and the meteorologists cannot be contacted, meteorological data may be obtained directly from the National Weather Service in Raleigh, North Carolina.

3.9.6 Field Monitoring Equipment

Field monitoring equipment will have at least the capability to detect and measure radioiodine in the vicinity of the plant site as low as 1×10^{-7} μCi/cm^3. An individual exposed to this concentration for a period of one hour would receive an exposure of about 0.2 Rem or less, a value well below Protective Action Guideline (PAG) levels (See Section 4). A standard air sampler can collect about 0.03 μCi of I-131 in 10 minutes at a concentration of 1×10^{-7} μCi/cm^3, which can easily be measured by hand survey meters that utilize probes such as the HP-210. This is a simple test that can serve as an initial check of projected releases based on plant data and can confirm that significant quantities of elemental iodine have been released (the chemical form that would pose a health hazard). More detailed measurements (such as, Sodium Iodide scintillation counters) can be brought into service to provide the longer term higher capabilities to detect and measure very low levels of contamination in the environment, as would be planned for subsequent radiation monitoring efforts.

3.9.7 Laboratory Facilities

Support of the on-site radiation monitoring and analysis effort is provided by HNP's chemistry and counting room facility. This laboratory is the central point for receipt and analysis of in-plant samples and includes equipment for chemical and radioactive analyses. Section 12.5 of the FSAR provides information on laboratory facilities. Additional facilities for counting and analyzing HNP samples can be provided by the H.B. Robinson Nuclear Plant and the Brunswick Nuclear Plant. These laboratories can act as backup facilities in the event that the plant's counting room and laboratory become unusable during an emergency.

Support of the off-site environmental radiation monitoring and analysis effort is provided by the N.C. Division of Radiation Protection's laboratory facility, both mobile and fixed and the HE&EC's chemistry and counting room facility. The State's laboratories are the central point for receipt and analysis of off-site samples when HNP is acting as a support agency to the State for ingestion pathway functions. Each lab includes equipment for chemical analyses and for analysis of radioactivity.

3.9.8 Other Plant Assessment Equipment

A. Fire Detection System (FSAR Sections 9.5.1 and 9.5.A)

B. Gross Failed Fuel Detection System

C. Security Systems (Security Plan)

D. Metal Impact Monitoring System (FSAR Section 5.4.6.4)

Table 3.1-1

Typical Emergency Supplies Available For Emergency Facilities

Supplies	MCR	TSC	JIC	OSC	EOF
7 Day supply of food and water.	✓				
Protective Clothing (Anti-Cs)	✓	✓		✓	✓
Air Sampling equipment	✓	✓		✓	✓
Full face respirators	✓	✓		✓	✓
Self-contained breathing equipment	✓	✓		✓	
High and low range portable radiation survey instruments	✓	✓		✓	✓
Emergency personnel monitoring dosimetry	✓	✓		✓	✓
Contamination control supplies such as signs, tags, rope, tape, various forms	✓	✓		✓	✓
Decontamination supplies		✓		✓	✓
Portable Communications Equipment	Radio Remotes	Radio[a] Remotes	(a)	✓	Radio[a] Remotes
Battery-Powered Lanterns		✓	✓	✓	✓
Polaroid Camera				✓	
Mechanical and electrical systems drawings, Plant Operations Manual, FSAR, Corporate, State & Local Emergency Plans		✓			✓
10-mile EPZ Area maps[c]	✓	✓	✓		✓
Copy of Plant Emergency Plan and Procedures	✓	✓	✓	✓[d]	✓
Environmental Monitoring Kits					(b)
Potassium Iodide Tablets	✓	✓		✓	✓

✓ Indicates equipment/supplies available in this facility

[a] Portable radio transceivers can be supplied to any emergency facility

[b] Stored near the Harris E&E Center

[c] Annex H of Emergency Plan in the MCR, wall maps in other facilities.

[d] Procedures Only

4.0 EMERGENCY MEASURES AND OPERATIONS

Execution of the HNP Emergency Plan involves a variety of functions including emergency classification, notification, activation, assessment, protective response actions, and recovery. Recovery is discussed in Section 6 of this Plan.

State and local governments and other agencies provide support in implementing the emergency measures in this section as shown in Table 4.0-1 and Annex G.

4.1 Emergency Classification

The four classes of emergency are Unusual Event (equivalent to NRC Notification of Unusual Event), Alert, Site Area Emergency, and General Emergency. The operating staff is provided formal training to recognize off-normal plant conditions and categorize them within the parameters of the four emergency classes.

Emergency action levels are based upon the fission product barrier concept. The three barriers that protect the public from a release of radioactive fission products (fission product barriers) are the fuel cladding, the reactor coolant system boundary, and the containment. This concept has its basis in NUREG-0654, Appendix 1 where emergency events are found that correspond to failures or jeopardy of the three basic fission product barriers. The concept used is that if any one of the fission product barriers are in jeopardy or breached, an Alert will be declared. If any combination of two barriers are either in jeopardy or breached, a Site Area Emergency is declared. If all three are in any combination of jeopardy or breach, a General Emergency is declared. The categorization of events in NUREG-0654, Appendix 1 for Unusual Events are separately evaluated as they do not directly correspond with failure or jeopardy of a fission product barrier. In addition to looking at the status of fission product barriers, the emergency action levels include the NUREG-0654 emergency action level events that are external to the plant, (natural or man-made disaster phenomena), or are not directly attributable to the condition of the reactor, (shutdown systems, fire, dose projections).

The categorization of events according to one of the four emergency classes is implemented through the Emergency Action Level (EAL) system. The system is composed of two subsystems: The Unusual Event Action Level Matrix and the EAL Network/Flow Path. The Unusual Event Action Level Matrix provides a set of plant conditions and events which coincide with the conditions associated with the Unusual Event. The Unusual Event Action Level Matrix is presented at the bottom of Figure 4.1-2. For the upper three emergency classes, the Emergency Action Level (EAL) System uses an integrated set of flowchart instructions. As with the Unusual Event Action Levels, the EAL System also associates plant conditions and events with the three upper classes of emergency, but it does so through a symptomatic (vice diagnostic) methodology using critical safety function status trees.

This allows the EAL System to interface smoothly with the Emergency Operating Procedure (EOP) Network, thus assuring the rapid and correct classification of emergencies. Figures 4.1-1 and 4.1-2 are the flowcharts which together form the EAL Network. Abnormal Operating Procedures, Functional Restoration Procedures, End Path Procedures, and Flow Path Procedures contain specific direction for using the EAL Network whenever conditions warrant. A Plant Emergency Procedure provides the

Unusual Event Action Levels, the EAL Network, the EAL Flow Path, as well as instructions for using them.

The Site Emergency Coordinator (or the Superintendent-Shift Operations when no emergency has been declared) will declare any one of the four emergency classes where EALs have been exceeded, or in their judgment, the status of the plant warrants such a declaration.

4.2 Notification

A. The warning message form to the State and Counties is contained in PEP-310 and provides information required by NUREG-0654, II.E.3 and 4. The form is approved by the Site Emergency Coordinator-MCR or Emergency Response Manager after EOF activation and provided to the appropriate Emergency Communicator (EC-Control Room or EC-State/County) as a message text.

B. The Emergency Communicator will use the electronic Notification Form on RTIN or the Selective Signaling System phone to simultaneously notify the 24-hour-per-day, manned, State and County Warning Points with the notification message. This message will be initiated to all Counties and the State within 15 minutes for all emergency classifications.

C. The North Carolina Emergency Response Plan in Support of the Harris Nuclear Power Plant describes procedures for State and Local officials to make a public notification decision promptly (within about 15 minutes) on being informed by the plant of an emergency.

D. Event notifications to the NRC will be made as soon as possible and within one hour using an NRC Event Notification worksheet or other notification message approved by the SEC-CR/ERM.

E. Plant personnel designated on the Emergency Response Organization are notified of an emergency condition by the Emergency Communicator using a computer-based automated duty roster system, or as a backup, a system of pagers and telephone call trees. These personnel are requested to be available on site to respond as directed by the Site Emergency Coordinator.

F. Personnel on site are notified by the Main Control Room using a plant Public-Address System announcement that an emergency has been declared and what actions should be taken.

G. Corporate personnel on the Emergency Response Organization will be notified of an emergency at HNP in accordance with plant emergency procedures.

H. The off-site agencies that will be notified of an emergency condition at HNP are shown in Tables 4.2-1 through 4.2-4.

I. Notifications to off-site agencies shall include a means of verification or authentication such as the use of dedicated communications networks, verification code words, or providing callback verification phone numbers.

4.3 Activation

A. Facilities are to be activated for each emergency class in accordance with Tables 4.2-1 through 4.2-4. The facilities can be declared activated when minimum staffing levels (as specified in the implementing procedures) have been met.

B. The Communications Director will verify the readiness and operability of emergency facilities in the Technical Support Center (TSC) and the Administrative and Logistics Manager will verify the readiness and operability of the Emergency Operations Facility (EOF).

C. The Emergency Repair Director will verify the readiness of the Operations Support Center.

D. The Company Spokesperson will verify the readiness of the Joint Information Center.

E. Security measures will be established for the Emergency Operations Facility upon its activation.

F. Personnel in the Emergency Response Organization will report to their preassigned locations in the emergency facilities.

4.4 Assessment Actions

4.4.1 Evaluation of Plant Conditions

A. Evaluation of plant conditions by Operations personnel is accomplished through observation of the control boards, monitoring panels, ERFIS data displays, the SPDS displays, and information provided by the Accident Assessment Teams in the TSC and EOF.

B. The Accident Assessment Teams evaluate plant conditions by using ERFIS displays, damage assessment reports, seismic data, fire reports, dose projections, and monitoring data.

C. Core damage assessment methodology is applied by the TSC Accident Assessment Team utilizing data provided from the ERFIS, the Main Control Room, the Radiation Monitoring System, and the Chemistry Team.

4.4.2 Plant Radiological Monitoring

A. The Radiation Monitoring System (RMS) will be used by Operations personnel and Radiological Control Team members to determine high radiation areas within the plant or abnormal radioactive effluents.

B. The Radiological Control Team will provide in-plant radiological measurements to supplement and confirm the RMS.

C. The Primary Sampling System may be used by the Chemistry Teams, where possible, to provide radiochemistry samples for analysis.

D. The Post Accident Sampling System provides the capability to sample reactor coolant and the containment atmosphere. Highly radioactive samples can be proportionally diluted by this system for later analysis. The Post Accident Sampling System will be operated by the Chemistry Team.

4.4.3 Dose Projection

A. Dose projections will be made to determine the off-site doses that might result from an accident and the possible need for protective action (see 4.5.1).

B. The dose projection capability on the computer can use source term data from the Radiation Monitoring System, and meteorological data from the on-site meteorological station. This system will aide personnel in the Main Control Room or EOF in determining recommendations for protective action for the public.

C. Data from the Radiation Monitoring System that is used to determine the source term for dose projections is quality tagged. If the data is off-scale, then it is suspect or bad, and the effluent radiation levels must be determined by sampling at the radiation monitor test points. The results from analyzing the samples can be entered into the dose projection program as a substitute value.

D. Radionuclide mix assumptions in EPM-600 (the accident source term) are contained in the computerized dose projection program as default values for use until actual sampling data can be substituted.

E. The National Weather Service and contracted weather sources will be contacted as needed to forecast atmospheric conditions affecting the site.

4.4.4 Environmental Monitoring

A. Environmental sampling and monitoring points are specified in environmental monitoring procedures.

B. Environmental Monitoring Teams will be activated in accordance with Table 2.2-1 and the appropriate implementing procedures. Additional teams can be called upon for support as needed.

C. The Environmental Monitoring Teams will track the plume from any radiological release by monitoring radiation levels as indicated on radiological measuring instruments and by obtaining and analyzing air samples.

D. The Environmental Monitoring Teams will aid in assessing liquid release pathways by sampling liquid effluents, such as the cooling tower blowdown.

E. Additional TLDs will be placed at various locations near the site and be periodically replaced throughout an emergency to ensure that a cumulative dose record is obtained.

4.5 Protective Actions for the Public

4.5.1 Protective Action Guides

A. Exposure guidelines for the plume pathway are based on the Environmental Protection Agency Protective Action Guides (PAGs) discussed in EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents" as follows:

IF:	THEN:
Projected dose is: < 1 Rem TEDE and < 5 Rem CDE Thyroid	No actions are necessary.
Projected dose is: ≥ 1 Rem TEDE or ≥ 5 Rem CDE Thyroid	Evacuate unless constraints make it impractical. Shelter as a minimum.

B. If projected doses exceed minimum EPA PAGs and timely evacuation is practical, then evacuation is recommended. If timely evacuation is not practical then sheltering may be recommended.

1) HNP personnel normally do not have the necessary information to determine whether off site conditions would require sheltering instead of evacuation. An effort to base Protective Action Recommendations on external factors (such as road conditions, traffic/traffic control, weather or offsite emergency response capabilities) is usually performed by the State.

2) The State may consider sheltering for doses up to 5 Rem TEDE for hazardous environmental conditions, and for doses up to 10 Rem TEDE for special populations. Hazardous environmental conditions may include the presence of severe weather or competing disasters. Special populations may include institutionalized or infirm persons.

4.5.2 Protective Action Recommendations (PARs)

A. Protective action guidelines for the plume pathway EPZ are based on NUREG-0654 Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents."

B. Plant conditions, projected dose and dose rates, and/or field monitoring data are evaluated to develop PARs for the purpose of preventing or minimizing exposure to the general public. PARs are made to the State and County agencies who are responsible for implementing protective actions for the general public within the plume exposure EPZ. PARs are approved by the Emergency Response Manager. In an emergency which requires immediate protective actions be taken prior to activation of the emergency facilities, notification approval is given by the SEC-CR directly to the State and County agencies.

C. Possible recommendations issued by HNP at a General Emergency include:

1) Evacuation of the general public within the two (2) mile radius and five (5) miles downwind. All other areas within the EPZ are sheltered (minimum PAR issued).

2) Evacuation of the general public within the five (5) mile radius and ten (10) miles downwind. All other areas within the EPZ are sheltered.

4.5.3 Ingestion Pathway Protective Measures

The responsibility for specifying protective measures to be used for the ingestion pathway rests with the State. These measures include the methods for protecting the public from exposure due to deposited radioactive materials and the consumption of contaminated water and foodstuffs.

4.5.4 Public Alerting, Warning, and Notification

Alerting, warning, and notification of the public are steps taken by government agencies to advise the public that protective actions are necessary. Alerting, warning, and notification will be provided by sounding sirens, activation of tone-activated radios within five miles of the plant, and supplemented by announcements made through radio and television (EAS), sound trucks, bullhorns, and knocking on doors. Patrol boats will be used in alerting people on Lake Jordan and Harris Lake in accordance the North Carolina Emergency Response Plan in support of the Shearon Harris Nuclear Power Plant Annexes G & J. Supplemental sirens are provided for alerting boaters on Harris Lake. Public warning when deemed necessary will be accomplished as described in the North Carolina Emergency Response Plan in Support of the Shearon Harris Nuclear Power Plant. Preplanned emergency messages and emergency instructions have been prepared and included as Annex D to that plan.

Civil defense sirens mounted on 50-foot utility poles have been installed by Carolina Power & Light Company at various locations within a 10-mile radius of the HNP.

Activation of the sirens for warning of the public will be accomplished from the county Warning Points or county Emergency Operations Centers: the Public Safety Communications Centers of Harnett and Lee Counties, the Emergency Operations Center of Chatham County, and the Raleigh Communications Center for Wake County. The sirens in each county are independently controlled by radio. The outdoor warning system provides the capability for providing an alerting signal within the 10-mile EPZ, within 15 minutes from the time the decision is made to notify the public of an emergency situation.

Activation of the tone alert radios by the National Weather Service will be accomplished after they receive a request from Wake County or the State of North Carolina. The tone alert radios provide an indoor alerting signal within a 5-mile radius of the plant.

4.6 Protective Actions for On-Site Personnel

4.6.1 On-Site Alerting, Warning, and Notification

The Plant Public-Address (PA) System will be used to alert and notify on-site personnel of an emergency condition within 15 minutes. Security personnel with portable loudspeakers may be used to augment the PA System and/or check evacuation of outlying areas, as available. The Plant PA System has the capability to transmit recognizable alarms which will alert personnel of an emergency situation, and to transmit voice communications which will notify personnel of those actions which should be taken. The Plant PA System is supplemented by the use of the normal and emergency communication systems located on site as described in Section 3.8 of this Plan.

4.6.2 Evacuation and Personnel Accountability

All personnel on-site will be accounted for within 30 minutes of the declaration of a Site Area Emergency or General Emergency and continuously thereafter during the emergency (accountability may be accomplished at any time prior to the declaration of a Site Area Emergency, if deemed appropriate). Personnel within the Protected Area will be accounted for and missing individual(s) will be identified by Security. Continuous accountability of personnel remaining inside the protected area will be maintained throughout the event. PEP-350 describes the accountability methodology. Search procedures will be implemented to locate unaccounted for persons.

Evacuation of on-site personnel can be accomplished, in accordance with PEP-350, for the Site or the Exclusion Area.

A. A Site Evacuation involves evacuation of all nonessential personnel within the Protected Area, Admin Building, parking lots, cooling tower area, sewage treatment plant, landfill, and intake structures. The site evacuation alarm will be sounded on the Plant PA system. Nonessential personnel (that is, CP&L personnel not on the ERO, ERO personnel not assigned to emergency duties and contractors) within the Protected Area will normally exit the Protected Area via the security building in accordance with normal Security procedures. Evacuating personnel may be monitored for contamination by the portal monitors as they exit the Protected Area or with portable friskers in the evacuation monitoring area, based on the situation. ERO personnel not assigned to emergency duties will travel to the HE&EC auditorium. CP&L personnel not on the ERO and contractors shall depart the site using personal transportation and follow established evacuation routes.

   Personnel without transportation will arrange for a ride from others who have space in their vehicles.

   Nonessential personnel exiting the site will be directed to either proceed to their homes or if radiological conditions warrant, reassemble at a selected off-site assembly area until off-site monitoring and decontamination stations are in place. Personnel exiting evacuated areas will be monitored and decontaminated, if necessary, at county monitoring stations.

B. An Exclusion Area Evacuation involves evacuation of all nonessential personnel and the public within the Protected Area and the site, as well as the surrounding areas controlled by CP&L within the Exclusion Area Boundary. In addition to sounding the plant evacuation alarm, personnel in outlying areas can be notified by patrol vehicles. If conditions warrant, evacuating personnel will be instructed to reassemble at the selected remote assembly area until county monitoring and decontamination stations are established.

C. Local evacuations relating to Radiation Control Areas and fire protection are conducted in accordance with plant procedures.

4.6.3 Radiological Exposure Control

A. Radiological and Contamination Control Facilities

Radiation safety controls are established 24 hours per day to contain the spread of loose surface radioactive contamination and monitor personnel exposure. CP&L contamination control limits are shown in Table 4.6-1. Emergency exposure guidance is given in Section 4.6.3.D of this plan. The radiation control facilities located in the Waste Processing Building include a contaminated laundry and storage area, clean laundry and storage area, personnel and equipment decontamination area. Additional areas where equipment is decontaminated are located in the Reactor Auxiliary Building (on the 236' and 261' levels) and at the north end of the Fuel Handling Building (on the 261' level). Radiation control and radiation control procedures are described in Section 12.5 of the FSAR.

Temporary facilities to limit contamination and exposure will be established as necessary during an emergency situation. As an example, facilities which can be used for personnel decontamination during an emergency are located near the first aid room in the Turbine Building and at the Harris Energy & Environmental Center. Radiation Control Areas can be expanded by roping off areas and/or establishing access control points to maintain personnel exposure As Low As Reasonably Achievable (ALARA).

B. Exposure Records for Emergency Workers

Emergency workers will receive self reading pocket dosimeters (SRPDs) or equivalent and TLD badges. Dose records will be maintained by the Radiological Control Coordinator in accordance with PEP-330. TLDs are read at the Harris Energy & Environmental Center. They are capable of staffing 24-hour a day.

C. Use of Protective Equipment and Supplies

During the course of an emergency, protective actions will be considered to minimize the effects of radiological exposures or contamination problems associated with personnel who must work within the affected Radiation Control Area. Measures that will be considered are:

- Use of process or engineering controls.
- Distribution of respirators.

- Use of protective clothing.

- Use of thyroid blocking agents (Potassium Iodide).

The criteria for issuance of respiratory protection and protective clothing are described in plant radiological protection procedures.

Procedures for the administration of radioprotective drugs to employees are described in the plant emergency procedures.

D. Emergency Worker Exposure

1) Dose Limits for workers in an emergency are taken from EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," U.S. Environmental Protection Agency, May 1992. Much of the discussion in this section is taken in whole from that document.

2) In emergency situations, workers may receive exposure under a variety of circumstances in order to assure protection of others and of valuable property. These exposures will be justified if the maximum risks or costs to others that are avoided by their actions outweigh the risks to which the workers are subjected (or collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved).

3) Emergency Worker Dose Limits are as follows:

Dose Limit (Rem TEDE)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Lifesaving or protection of large populations	Lower dose not practicable
> 25	Lifesaving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved.

4) Limit dose to the lens of the eye to three (3) times the above values and doses to any other organ (including thyroid, skin and body extremities) to ten (10) times the above values.

5) Routine dose limits shall not be extended to emergency dose limits for declared pregnant individuals. As in the case of normal occupational exposure, doses received under emergency conditions should be maintained as low as reasonably achievable.

6) Entry into radiation fields of greater than 25 Rem/hour or emergency exposures in excess of 5 Rem TEDE shall not be permitted unless specifically authorized by the Site Emergency Coordinator for on-site emergency workers and by the Emergency Response Manager for EOF or EOF dispatched personnel.

7) Persons undertaking any emergency operation in which the dose will exceed 25 Rem TEDE should do so only on a voluntary basis and with full awareness of the risks involved including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

8) Personnel who will receive emergency related exposure should be selected and controlled in accordance with guidelines contained in the implementing procedures.

E. Decontamination and First Aid

1) Treatment of Injured and Contaminated Persons

Personnel decontamination supplies are located near the WPB 261' First Aid Station. Personnel showers are located in the general area of the main RCA entrance (WPB 261'). Chemical decontamination agents are available from Health Physics personnel and, except in cases of severe or life-threatening injury, established decontamination procedures should be employed on site prior to medical treatment.

2) Initial First Aid

In cases of severe injury, lifesaving first aid or medical treatment will take precedence over personnel decontamination. In general, the order of medical treatment will be:

- Care of severe physical injuries or illness.
- Personnel decontamination.
- First aid to other injuries.
- Definitive medical treatment and subsequent therapy as required.

Definitive medical treatment, therapy, and evaluation may include radioprotective drugs, urinary bioassays, or whole body counts on persons suspected of inhaling or ingesting a significant amount of radioactive material or may include surveillance and therapy for persons receiving a large whole body dose.

Emergency first aid personnel are available on all shifts. Personnel who are contaminated and who require medical treatment may be treated by these personnel on the scene or at other appropriate locations.

It is anticipated that contaminated personnel will not leave the facility for medical treatment except for cases that require immediate hospitalization. Emergency treatment of contaminated personnel will normally be handled at the plant First Aid Room by personnel on the First Aid Team(s).

First Aid kits are located in various areas of the plant (see ORT-3002). The First Aid Stations/Kits contain various equipment/items necessary to treat injured personnel until off-site agencies can transport patient to appropriate treatment center, if applicable.

3) Decontamination

Radiation safety controls are established to contain the spread of loose surface radioactive contamination. Personnel and equipment leaving contaminated areas are monitored to ensure that equipment, personnel or their clothing are not contaminated. If contaminated above acceptable levels (see Table 4.6-1), they will be decontaminated in accordance with plant procedures. Supplies, instruments and equipment that are in contaminated areas or have been brought into contaminated areas will be monitored for contamination. If found to be contaminated, they will be decontaminated using normal plant decontamination techniques and facilities (discussed in Section 4.6.3.A) or may be disposed of as radwaste.

During emergency conditions, normal plant contamination control criteria will be adhered to as much as possible. Contamination control criteria for returning areas and items to normal use are contained in the plant Health Physics Procedures. These criteria are summarized in Table 4.6-1.

4) Medical Transportation

The Apex Rescue Squad, Inc. has agreed to respond to emergency calls from the plant, including transporting persons with injuries involving radioactive contamination. This service is available on a 24-hour-per-day basis. In cases not involving severe injury, one of the plant vehicles may be used to transport injured individuals. The Apex Rescue Squad, Inc. is included in Annex A, "Agreements".

In cases involving severe injury, the Superintendent - Shift Operations or Site Emergency Coordinator may bypass the Apex Rescue Squad, Inc. and directly call Carolina Air Care or Duke Life Flight and request helicopter transport of the injured.

Contaminated injured persons will be accompanied to a medical facility by a Radiological Control Team member carrying survey instrument. If possible, contaminated clothing and equipment may be removed from the patient or the patient may be wrapped in clean sheets or clothing to prevent contamination of the transporting personnel and vehicle.

Rescue vehicles have mobile communications with the Raleigh Communications Center and local receiving hospitals. The plant first aid team can communicate directly with the rescue vehicles by dialing the cellular phone located in the rescue vehicles.

F. Medical Treatment

1) Hospital Facilities

A specially designated emergency area is maintained in readiness at Rex Hospital for CP&L's use for the treatment of contaminated or overexposed patients from the plant. Although this area will be utilized by the hospital when not required by CP&L, it will be made available to CP&L when required. Equipment is available in the hospital for the emergency treatment of patients. With the facilities and equipment available, extensive decontamination and treatment of an injured patient could be performed, including any surgical treatment that may be required.

Wake Medical Center and Western Wake Medical Center serve as backup medical facilities for HNP personnel should Rex Hospital become unavailable. Wake Medical Center serves as the primary medical facility for trauma patients from HNP. Wake Medical Center, and Betsy Johnson Memorial Hospital, in Dunn, N.C., also possess the capability for the treatment of contaminated and/or overexposed members of the public.

An emergency kit is maintained at Rex Hospital, Wake Medical Center and Western Wake Medical Center containing supplies and equipment for personnel monitoring and the control of radioactive contamination. These kits contain the following:

- Low-range radiation monitoring instruments for determining contamination levels.
- Personnel monitoring equipment such as self-reading pocket dosimeters and TLDs.
- Decontamination equipment and supplies for both personnel and facility.
- Contamination control equipment and supplies such as protective clothing, signs, ropes, tags, plastic bags.

Agreements with Rex Hospital, Wake Medical Center and Western Wake Medical Center are maintained on file by HNP Emergency Preparedness. These three hospitals are listed in Annex A, "Agreements".

2) Medical Consultants

Medical assistance is available in the Raleigh area from general practitioners who have agreed to provide medical assistance for contaminated patients (See Annex A). Also, the DOE Radiological Assistance Team will provide medical assistance, if required.

G. <u>Contamination Control of Drinking Water and Food</u>

Measures will be taken to control access to potentially contaminated potable water and food supplies on site. Under emergency conditions when a release of activity has occurred, eating, drinking, smoking, and chewing will be not permitted until the facility manager has determined that it is safe to do so. If the drinking water is contaminated above acceptable levels, uncontaminated water will be brought into the plant for the personnel to drink. Emergency food supplies are stored in a secure manner (See Table 3.1-1). Packaged food is located in vending machines in lunch rooms or office areas in the Administration Building, Fuel Handling Building "K" area, Operations Building, or Service Building. If these areas become contaminated because of a release of activity, the machines will be disabled or emptied until it can be verified that the food is not contaminated or the food will be discarded. Food located in the Service Building cafeteria would be verified uncontaminated prior to use.

4.7 Fire-Fighting Assistance

Off-site fire departments will provide support as described in Annex A and Annex G.

4.8 Security Measures

Security measures during an emergency will be employed in accordance with the Plant Security Plan, implementing Security procedures, and Plant Emergency Procedures dealing with personnel accountability, egress, and ingress.

Table 4.0-1

Off-Site Agency Support Summary

Function (NUREG-0654, II.A)	Primary Responsibility	Support Responsibility
Command and Control		
On site	HNP	CP&L
Off site	State, County	FEMA
Accident Classification		
On site	HNP	N/A
Off site	N/A	N/A
Warning		
On site	HNP	N/A
Off site	County	State
Notification, Officials		
On site	HNP	CP&L
Off site	HNP	State, County, Media
Notification, Public		
On site (such as Visitors)	HNP	N/A
Off site	State, County	State
Communications		
On site	HNP	CP&L
Off site	State, County	Phone Company, CP&L
Transportation		
On site	HNP/Employees	N/A
Off site	Local Residents	State, County
Traffic Control/Security		
On site	HNP	County
Off site	County	State
Accident Assessment		
On site	HNP	CP&L, W
Off site	State	County, CP&L, FEMA, DOE
Public Information/Education		
On site	HNP, Corp Comm	NRC
Off site	State	County, Corp Comm, Media, FEMA
Protective Response		
On site	HNP	CP&L
Off site	State, County	CP&L, FEMA
Radiological Exposure Control		
On site	HNP	CP&L
Off site	State	County, FEMA, CP&L
Fire and Rescue		
On site	HNP	Local Fire & Rescue
Off site	County	State

Table 4.0-1

Off-Site Agency Support Summary (continued)

Function (NUREG-0654, II.A)	Primary Responsibility	Support Responsibility
Medical		
On site	HNP	Rescue, Hospital
Off site	County	State
Public Health & Sanitation		
On site	HNP	N/A
Off site	County	State
Social Services		
On site	N/A	N/A
Off site	County	State
Training		
On site	HNP	CP&L
Off site	County, State, CP&L	State, CP&L
Exercises		
On site	HNP	CP&L
Off site	State	County, CP&L
Reentry		
On site	CP&L	HNP, W, Raytheon
Off site	State	FEMA, County, CP&L, DOE

Notes:

CP&L - Carolina Power & Light Company

DHHS - U.S. Department of Health & Human Services

DOE - U.S. Department of Energy

Raytheon - Raytheon Engineers

FEMA - U.S. Federal Energy Management Agency

NRC - U.S. Nuclear Regulatory Commission

HNP - Harris Nuclear Plant

W - Westinghouse Electric Corporation

N/A - Not applicable

Table 4.2-1

Execution of Unusual Event

A. CLASS DESCRIPTION

This class involves events which indicate a potential degradation of the level of safety at a nuclear station.

B. RELEASE POTENTIAL

No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.

C. NOTIFY

Time frames noted below are from the time the emergency is declared.

Required Notifications

- State of North Carolina Emergency Warning Point (fifteen minutes)
- Chatham County Emergency Warning Point (fifteen minutes)
- Harnett County Emergency Warning Point (fifteen minutes)
- Lee County Emergency Warning point (fifteen minutes)
- Wake County Emergency Warning Point (fifteen minutes)
- On-site Emergency Response Organization (as specified by procedure)
- Off-site Emergency Response Organization (as specified by procedure)
- Nuclear Regulatory Commission Operations Center (one hour)

Additional Notifications as Necessary

- Raytheon Engineers
- Westinghouse Electric Corporation
- Institute of Nuclear Power Operations
- American Nuclear Insurers
- Nuclear Electric Insurance Limited (NEIL) (Fire Only)
- Department of Energy, Savannah River Operations Office

D. ACTIVATE

On-site ERO (not required, but may be staffed for support as necessary)

- Technical Support Center
- Operations Support Center

Off-site ERO (not required, but may be staffed for support as necessary)

- Emergency Operations Facility
- Joint Information Center

Request Assistance (if necessary)

- Rex Hospital
- Wake Medical Center
- Western Wake Medical Center
- Fire and Rescue Departments

Table 4.2-2

Execution of Alert

A. CLASS DESCRIPTION

This class describes events which involve actual or potential substantial degradation of the level of safety at a nuclear station.

B. RELEASE POTENTIAL

Off-site doses expected to be limited to small fractions of EPA Protective Action Guideline exposure levels.

C. NOTIFY

Time frames noted below are from the time the emergency is declared.

Required Notifications

- State of North Carolina Emergency Warning Point (Fifteen minutes)
- Chatham County Emergency Warning Point (Fifteen minutes)
- Harnett County Emergency Warning Point (Fifteen minutes)
- Lee County Emergency Warning Point (Fifteen minutes)
- Wake County Emergency Warning Point (Fifteen minutes)
- On-site Emergency Response Organization
- Off-site Emergency Response Organization
- Nuclear Regulatory Commission Operations Center (One hour)
- American Nuclear Insurers (Four hours)
- Nuclear Electric Insurance Limited (NEIL) (Fire Only)
- Institute of Nuclear Power Operations (Four hours)

Additional Notifications as Necessary

- Raytheon Engineers
- Westinghouse Electric Corporation
- Department of Energy, Savannah River Operations Office

D. ACTIVATE

On-site ERO

- Technical Support Center
- Operations Support Center

Off-site ERO

- Emergency Operations Facility
- Joint Information Center

Request Assistance (if necessary)

- Rex Hospital
- Wake Medical Center
- Western Wake Medical Center
- Fire and Rescue Departments

Table 4.2-3

Execution of Site Area Emergency

A. CLASS DESCRIPTION

This class describes events which involve major failures of plant functions needed for the protection of the public.

B. RELEASE POTENTIAL

Off-site doses not expected to exceed EPA Protective Action Guidelines exposure levels except near site boundary.

C. NOTIFY

Time frames noted below are from the time the emergency is declared.

Required Notifications

- State of North Carolina Emergency Warning Point (Fifteen minutes)
- Chatham County Emergency Warning Point (Fifteen minutes)
- Harnett County Emergency Warning Point (Fifteen minutes)
- Lee County Emergency Warning Point (Fifteen minutes)
- Wake County Emergency Warning Point (Fifteen minutes)
- On-site Emergency Response Organization
- Off-site Emergency Response Organization
- Nuclear Regulatory Commission Operations Center (One hour)
- American Nuclear Insurers (Four hours)
- Nuclear Electric Insurance Limited (NEIL) (Fire Only)
- Institute of Nuclear Power Operations (Four hours)

Additional Notifications as Necessary

- Raytheon Engineers
- Westinghouse Electric Corporation
- Department of Energy, Savannah River Operations Office

D. ACTIVATE

On-site ERO

- Technical Support Center
- Operations Support Center

Off-site ERO

- Emergency Operations Facility
- Joint Information Center

Request Assistance (if necessary)

- Rex Hospital
- Wake Medical Center
- Western Wake Medical Center
- Fire and Rescue Departments

Table 4.2-4

Execution of General Emergency

A. CLASS DESCRIPTION

This class involves events which involve actual or imminent substantial core degradation or melting with the likelihood of a related release of appreciable quantities of fission products to the environment.

B. RELEASE POTENTIAL

Doses expected to be greater than the upper EPA Protective Action Guideline exposure levels off-site for more than the immediate site area.

C. NOTIFY

Time frames noted below are from the time the emergency is declared.

Required Notifications

- State of North Carolina Emergency Warning Point (Fifteen minutes)
- Chatham County Emergency Warning Point (Fifteen minutes)
- Harnett County Emergency Warning Point (Fifteen minutes)
- Lee County Emergency Warning Point (Fifteen minutes)
- Wake County Emergency Warning Point (Fifteen minutes)
- On-site Emergency Response Organization
- Off-site Emergency Response Organization
- Nuclear Regulatory Commission Operations Center (One hour)
- American Nuclear Insurers (Four hours)
- Nuclear Electric Insurance Limited (NEIL) (Fire Only)
- Institute of Nuclear Power Operations (Four hours)

Additional Notifications as Necessary

- Raytheon Engineers
- Westinghouse Electric Corporation
- Department of Energy, Savannah River Operations Office

D. ACTIVATE

On-site ERO

- Technical Support Center
- Operations Support Center

Off-site ERO

- Emergency Operations Facility
- Joint Information Center

Request Assistance (if necessary)

- Rex Hospital
- Wake Medical Center
- Western Wake Medical Center
- Fire and Rescue Departments

Table 4.5-2

Protective Action Guides for the Ingestion Pathway

Protective Action Guide (PAG)	Projected Dose Commitment to Whole Body, Bone Marrow or any other Organ (Rem)	Projected Dose Commitment to the Thyroid (Rem)
Preventive PAG[a]	0.5	1.5
Emergency PAG[b]	5.0	15.0

[a] Preventive PAG - The projected dose commitment value at which responsible officials should take protective actions having minimal impact to prevent or reduce the radioactive contamination of human food or animal feed.

[b] Emergency PAG - The projected dose commitment value at which responsible officials should isolate food containing radioactivity to prevent its introduction into commerce and at which the responsible officials should determine whether condemnation or other disposition is appropriate.

From: Federal Register, Vol. 47, No. 205, October 22, 1982, U.S. Food and Drug Administration, Accidental Radioactive Contamination of Human Food and Animal Feeds, Recommendations for State and Local Agencies

Table 4.6-1

CP&L Area Radiation and Contamination Limits

A.	Radiation Control Area	Radiation Levels
	1. Radiation Area	5 to ≤100 mrem/hr
	2. High Radiation Area	>100 mrem/hr to ≤ 1000 mrem/hr
	3. Locked High Radiation Area	1000 mrem/hr to ≤ 500 rad/hr
	4. Very High Radiation Area	>500 rad/hr @ 1 meter
	5. Airborne Radioactivity Area	Airborne Conc. ≥ 25% of 10CFR20,App. B, Table 1 Column 3
B.	Contamination Limits	
	1. Skin contamination or personal clothing	< 100 net cpm βγ with HP210 probe or equivalent sensitivity no measurable α count rate above background
	2. Unconditional release from site for tools and equipment	No detectable α No detectable βγ above background
	3. Contamination Area	> 1000 dpm/100cm² βγ smearable and/or > 20 dpm/100cm² α

Figure 4.1-1

Emergency Action Level Flow Path, Side 1

Folded copy of Emergency Action Level Flowpath, Side 1, (Rev. 00-1) is contained in the plastic sleeve following this page.

Figure 4.1-2

Emergency Action Level Flow Path, Side 2

Folded copy of Emergency Action Level Flowpath, Side 2 (Rev. 00-1) is contained in the plastic sleeve following this page.

5.0 MAINTAINING EMERGENCY PREPAREDNESS

Emergency preparedness at HNP will be maintained by:

- Maintaining planning documents through review, updates, audits, and annual PNSC review.
- Preparing Emergency Response Organization members for proper response actions through training and retraining.
- Testing the adequacy of emergency preparedness through the use of drills and exercises.
- Inventorying and calibrating emergency equipment, supplies, and instrumentation.
- Ensuring that the public notification and alerting system is tested and maintained.
- Ensuring that the Evacuation Time Estimate is periodically reviewed for adequacy.

Each periodic requirement in this section and elsewhere in the plan and plant emergency procedures shall be performed within the specified time below:

- Annually - At least once per 366 days
- Biennially - At least once per 731 days
- Monthly - At least once per 31 days
- Quarterly - At least once per 92 days
- Semiannually- At least once per 184 days

For the above intervals, a maximum allowable extension which shall not exceed 25% of the specified interval is allowable.

This definition for periodic requirements applies to all intervals in the emergency plan and plant emergency procedures except for the biennial exercise, which is conducted every other calendar year.

5.1 Emergency Plan and Plant Emergency Procedures

5.1.1 Responsibility for the Planning Effort

The HNP Emergency Planning Coordinator is responsible for coordinating on-site and selected off-site radiological emergency response planning. The EP Coordinator is also responsible for performing the following planning functions:

A. Interfacing with federal, state, county, and local planners.

B. Revising and updating the Plan in response to action items identified during appraisals, audits, exercises, drills, and changes in regulations, hardware, and personnel.

C. Coordinating the biennial exercise and the periodic drills.

D. Identifying off-site training needs of state and local emergency support personnel and arranging for training to meet the identified needs.

E. Identifying corrective actions needed following drills and exercises, appraisals, and audits; coordinating responsibility for implementing these actions; coordinating a schedule for completion of these actions; and evaluating the adequacy of the actions taken.

F. Maintaining and negotiating agreements with state and county response agencies, federal assistance agencies, and medical and fire support agencies.

5.1.2 Emergency Plan and Plant Emergency Procedures Update and Changes

The Emergency Planning Coordinator will coordinate the updating of the Plant Emergency Plan, Plant Emergency Procedures, and Supporting Agreements as needed and will review and certify them to be current on an annual basis. The EALs shall be approved by the State of North Carolina and Wake, Chatham, Harnett, and Lee Counties annually. Plan and Procedure revisions shall be reviewed and approved in accordance with an approved plant procedure. Approved changes to the Plan will be distributed in accordance with the distribution list for the plan and procedures in a plant procedure. Revised pages will be indicated in accordance with plant procedures.

Changes to the E-Plan or PEPs shall be forwarded to the NRC within 30 days after approval.

5.1.3 Updating Telephone Listings

Updating of emergency phone listings or personnel listings is not a change to the Plan. Emergency phone listings and personnel listings shall be updated at least quarterly.

5.1.4 Plant Emergency Procedures

A list of emergency preparedness documents that support this Plan is provided in Annex E.

5.1.5 NUREG-0654 Cross-Reference

The criteria for radiological emergency response plans contained in NUREG-0654 are cross-referenced to the applicable sections of this Plan and supporting Plans in Annex D.

5.1.6 Annual Independent Audit

An independent audit of the HNP Emergency Preparedness Program will be conducted every year by the Nuclear Assessment Section. The Nuclear Assessment Section will audit the Plan, Plant Emergency Procedures, Training, Drills and Exercise, facilities and equipment for conformance with 10 CFR 50.47, 10 CFR 50.54, and 10 CFR 50 Appendix E. Written reports of the findings of these audits and reviews will be provided to Corporate Management. Written notification will be provided to the State of North Carolina and Counties of Chatham, Harnett, Lee, and Wake of the performance of the audit and the availability of the audit records for review at CP&L facilities. Each report will address the adequacy of interfaces with state and local governments, of drills and exercises, and of emergency response capabilities and procedures. The reports will be retained by Emergency Preparedness for five years. Corrective actions deemed necessary from the audit will be implemented in accordance with Section 5.1.1.E of this Plan and the site Corrective Action Program.

5.2 Emergency Response Organization Training Program

5.2.1 General Requirements

CP&L ensures the training of appropriate company personnel to support the Harris Plant Emergency Plan. Initial training and annual retraining is provided for the following categories of personnel:

A. Directors, Coordinators, and Manager in the Emergency Response Organization.

B. Personnel responsible for accident assessment.

C. Radiological monitoring teams and radiological analysis personnel

D. Damage Control Teams

E. First Aid, Search and Rescue, and Fire Brigade Teams

F. Personnel responsible for transmission of emergency information and instruction

G. Personnel responsible for communicating with the media and public

H. Offsite medical support personnel

I. Local support services personnel, including emergency management personnel

J. Police, security and offsite fire-fighting personnel who may be required to assist at the plant

Company personnel not assigned to the site are utilized and trained as members of the program.

Individuals assigned to First Aid Teams will include courses equivalent to the Red Cross Multimedia First Aid Course.

Designated ERO positions are also required to be qualified in the use of appropriate respiratory equipment.

Plant Access Training is provided to all personnel before they have unrestricted access to the Protected Area. This training includes general knowledge of alarms and actions required for non-ERO member during a declared emergency.

Site specific emergency response training shall be offered to offsite emergency organizations and local support services individuals who may be called upon to provide assistance to HNP in the event of an emergency. Training will include site access procedures and the identity (by position and title) of the individual in the HNP ERO who will control their organizations' support activities. Training for hospital personnel, ambulance/rescue, police and fire departments shall also include the procedures for notification, basic radiation protection, and their expected roles.

5.2.2 Conduct of Training

The Emergency Preparedness Unit Supervisor is responsible for the overall content and administration of the emergency plan training program.

EPM-200, ERO Training Program will include knowledge based and/or performance based training and evaluation components.

A. Knowledge based training may be provided in a classroom setting or self directed study modules and document reviews. Examination and/or interviews will be given for initial qualifications to ensure trainee has a good base knowledge of the ERO and their assigned responsibilities.

B. Performance based training and evaluations will be conducted for most ERO members (exceptions are made for pool personnel whose normal job functions closely matches their emergency functions and they are directed by qualified ERO Managers or Coordinators, such as operations, E&RC, maintenance, administrative and security pool personnel). This is done during conduct of exercises, drills or walkthroughs and documented on ERO qualification record forms.

5.2.3 Off-Site Organizations

Training of off-site organizations is described in their respective radiological emergency plans. Additional training is provided by CP&L for hospital, rescue, local law enforcement agencies, and fire personnel. Such training will include the procedures for notification, basic radiation protection, and their expected roles. For those Immediate Response Organizations who may enter the site, training by CP&L will also include site access procedures and the identity (by position and title) of the individual in the HNP organization who will control the organization's support activities. CP&L will assist these off-site organizations in performing their radiological emergency response training as related to HNP as requested.

Training of medical support personnel at the agreement hospitals will include basic training on the nature of radiological emergencies, diagnosis and treatment, and follow-up medical care.

5.2.4 Emergency Planning Coordinator and Staff Training

Training of plant emergency preparedness personnel involved in the planning effort may consist of either of the following:

A. Observing exercises at other plants.

B. Participation in emergency preparedness workshops, seminars and/or courses.

5.2.5 Public Education and Information - CP&L

Occupants in the Plume Exposure Pathway Emergency Planning Zone (EPZ) will be provided information prepared by CP&L in conjunction with the state and county agencies. This public education and information program is intended to ensure that members of the public are (1) aware of the potential for an occurrence of a radiological emergency; (2) able to recognize a radiological emergency notification; and (3) knowledgeable of the proper, immediate actions to be taken upon notification.

This will be accomplished by (1) distribution of the annual safety information calendar which contains educational information on emergency preparedness, sheltering, sirens, and radiation including telephone numbers of agencies to contact for more information; (2) annual distribution of a school brochure to school bus drivers and students; (3) availability of qualified personnel to address civic, religious, social, and occupational organizations; (4) distribution of news material to the media; and (5) periodic publication of the 10-mile EPZ newsletter, periodic not to exceed annual.

Emergency information will be made available to transient populations through the distribution of safety information brochures to commercial establishments in the 10-mile EPZ. A supply of these brochures is maintained at motels within the 10-mile EPZ.

Lake warning signs are posted at boat ramps, or access roads to boat ramps, at Harris and Jordan Lakes. These signs describe the activities which would be taken to initiate an evacuation of the lake and actions which should be taken in response to the evacuation. The posting of these signs is verified semiannually.

During an actual emergency, provisions will be established through the Joint Information Center to make available and distribute information to the news media. Provisions for a number of telephones which members of the public, who hear rumors, can call for factual information will also be implemented in the JIC when activated.

5.2.6 Public Education - State of North Carolina

The North Carolina Department of Crime Control and Public Safety has overall responsibility for maintaining a continuing disaster preparedness public education program. Such a program, prepared by the state of North Carolina, with the cooperation of the local governments and CP&L, is intended to ensure the members of the public are:

A. Aware of the potential threat of a radiological emergency;

B. Able to recognize a radiological emergency notification; and

C. Knowledgeable of the proper immediate actions (return to home, close windows and tune to an Emergency Alert System station) to be taken.

A program of this type includes education on protective actions to be taken if shelter is prescribed and the general procedures to follow if an evacuation is required. It also includes general educational information on radiation and how to learn more about emergency preparedness.

5.3 Drills and Exercises

5.3.1 Drills

Drill scenarios will be varied from year to year such that major elements of the plans and emergency organizations are tested within a 6-year period. One drill shall start between 6 p.m. and 4 a.m. once every 6 years.

EPM-210 prescribes policies and procedures for conducting the following drills:

A. Communication Drills

1) Communication from the Plant to the State and local government warning points within the plume exposure pathway Emergency Planning Zone shall be tested monthly. This shall include the transmittal of the information on an Emergency Notification Form.

2) Communications from the Main Control Room, Technical Support Center, and the Emergency Operations Facility to the NRC Headquarters Operations Center shall be tested monthly.

3) Communications between the nuclear facility, state, and local emergency operations centers, and environmental monitoring teams shall be tested annually.

4) Communications between the Main Control Room, the Technical Support Center and the Emergency Operations Facility shall be tested annually.

B. Fire Drills

Fire drills shall be conducted in accordance with Section 13.2 of the FSAR.

C. Medical Emergency Drills

A medical emergency drill involving a simulated contaminated individual with provision for participation by the local support services agencies (that is, ambulance, and off-site medical treatment facility) shall be conducted annually. The off-site portions of the medical drill may be conducted once per calendar year.

D. Environmental Monitoring Drills

Plant environs and radiological monitoring drills (on site and off site) shall be conducted annually. These drills shall include collection and analysis of all sample media (such as water, vegetation, soil, and air), and provisions for communications and record keeping.

E. Radiological Control Drills

1) Radiological Control drills shall be conducted semiannually which involve response to, and analysis of, simulated elevated airborne and liquid samples and direct radiation measurements in the environment.

2) Analysis of in-plant liquid samples with actual elevated radiation levels including use of the post-accident sampling system shall be included in Radiological Control drills annually.

F. Integrated Drills

1) Integrated training drills are conducted between biennial exercises to ensure adequate emergency response capability is maintained. An integrated drill combines principle functional areas of the on-site response which includes the management and coordination of the response, accident assessment, protective action decision-making, and plant system repair and corrective actions. Activation of all of the emergency response facilities is not necessary. Integrated drills may provide the opportunity for training for the staff.

2) At least one integrated drill is to be performed between the biennial exercises.

3) Critiques and evaluation of drills will be conducted by a qualified individual. The degree of participation by outside agencies in conducting these drills may vary and their action may actually be simulated.

5.3.2 Exercises

An exercise is an event that tests the integrated capability of major response organizations. Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, and ensure that emergency organization personnel are familiar with their duties. Procedures for the conduct of exercises are described in EPM-210. An emergency exercise involving on-site participation will be conducted at least once every other calendar year.

Partial participation exercises involving off-site agencies will be conducted at least once every other calendar year (IE Information Notice 85-55). Partial participation means appropriate off-site authorities shall actively take part in the exercise sufficient to test direction and control functions to include protective action decision making related to emergency action levels and communication capabilities among affected state and local authorities and CP&L.

Every sixth year the exercise will include the full participation of the State. These full participation exercises will include appropriate off-site local and state authorities and CP&L personnel physically and actively taking part in testing the integrated capability to adequately assess and respond to an accident at the plant. "Full participation" includes testing the major observable portions of the on-site and off-site emergency plans and mobilization of state, local, and CP&L personnel and other resources in sufficient numbers to verify the capability to respond to the accident scenario.

Exercises involving off-site agencies will simulate an emergency that results in an off-site radiological release.

The biennial exercises should be conducted during different seasons of the year and some exercises will be unannounced. .

Advance knowledge of the scenarios will be kept to a minimum to allow "free-play" decision making and to ensure a realistic participation by those involved.

Each biennial exercise plan should include the following:

- The basic objective(s) of the exercise.
- The date(s), time period, place(s), and participating organizations.
- The simulated events.
- A time schedule of real and simulated initiating events.
- A narrative summary describing the conduct of the exercise to include such things as simulated casualties, off-site fire department assistance, rescue of personnel, use of protective clothing, deployment of radiological monitoring teams, and public information activities.
- Arrangements for qualified Evaluators and Controllers.
- Critique and Evaluation Reports.

Prior to the exercise, an exercise plan will be distributed to the exercise controllers and evaluators that will include a list of performance objectives, the scenario, and a description of the expected responses.

Qualified observers from CP&L, federal, state, or local governments will observe and critique each biennial exercise in which the state and counties participate. A critique will be scheduled at the conclusion of each exercise to evaluate the ability of all participating organizations to respond. The critique will be held as soon as possible after the exercise. A formal written evaluation of the exercise will be prepared by the Emergency Planning Coordinator following the critique.

The Plant Emergency Planning Coordinator or assigned designee will determine those critique items that require corrective actions. Plant administrative controls will be utilized to ensure that corrective actions are implemented.

5.4 Maintenance and Inventory of Emergency Equipment and Supplies

5.4.1 Emergency Equipment and Supplies

A resource list of emergency equipment and supplies to be inventoried for the TSC, OSC, EOF and JIC is referenced in the emergency program maintenance procedures. This listing provides information on location and availability of emergency equipment and supplies.

An inventory of all emergency equipment and supplies is held on a quarterly basis and after use in an emergency or drill. During this inventory, radiation monitoring equipment is to be checked to verify that required calibration and location are in accordance with the inventory lists.

5.4.2 Medical Equipment and Supplies

Respiratory protection equipment, maintained for emergency purposes, is to be inspected and inventoried monthly.

At least twice each year and after use in an emergency or drill, emergency medical equipment and supplies located in the First Aid Station/Kits throughout the plant are to be inventoried, inspected, replaced, and replenished and/or resterilized as necessary. First Aid Team personnel inspect and inventory emergency medical supplies required to support a medical emergency at the plant, and plant personnel use the checklist in the applicable procedures to inspect other emergency items located in the First Aid Station/Kits.

5.4.3 Meteorological Instrumentation

Calibration of and channel checks on meteorological instrumentation are performed in accordance with Technical Specifications.

5.5 Testing and Maintenance of the Public Notification and Alerting System

5.5.1 General Description

The Public Notification and Alerting System consists of sirens located throughout the 10-mile EPZ and Tone Alert Radios distributed to households within a 5-mile radius of the HNP.

5.5.2 Siren System Testing

The sirens are tested as follows:

A. A silent test should be performed every two weeks.

B. A growl test should be performed at least once per calendar quarter.

C. A full-scale test of the system shall be conducted annually.

5.5.3 Siren System Maintenance

Maintenance of the Siren System is an ongoing process and is performed as needed based on the results of each test of the system. Records of siren maintenance are reviewed by HNP Emergency Preparedness.

5.5.4 Siren System Operability

A. The loss of all sirens within one county or the loss of 20% of the total number of sirens requires notification of the NRC within one hour.

B. The annual operability of the siren system is considered acceptable when an average of at least 90% of the siren tests for a calendar year are successful.

5.5.5 Tone Alert Radio Distribution

Tone Alert Radios are distributed to households within a 5-mile radius of the plant. The radios are tested prior to distribution and provided to each residence by a trained CP&L representative.

5.5.6 Tone Alert Radio Maintenance

A. Residences receiving a Tone Alert radio are provided with information on who to contact if the radio malfunctions.

B. CP&L annually distributes a new battery to each residence possessing a Tone Alert Radio.

C. CP&L annually distributes guidance to each residence on the purpose and operation of the Tone Alert radio.

5.5.7 Tone Alert Radio System Testing

A. The Tone Alert Radio System is tested annually.

B. An independent contractor is retained by CP&L to develop and conduct a survey to assess the effectiveness of the Tone Alert Radio System.

5.5.8 Tone Alert Radio System Operability

A. The Tone Alert Radio System is considered effective if at least 66% of those households surveyed received the test signal during the annual test.

B. The loss of either of the two National Weather Service Tone Alert Radio signal transmitters requires notification of the NRC within one hour.

5.6 Evacuation Time Estimate

The HNP Evacuation Time Estimate (ETE) (See Table 1.8-2) will be considered valid until the population within the 10-mile EPZ has increased by greater than 10% since the last ETE was determined. If the population is found to have increased by greater than 10% then a revised ETE will be established using appropriate guidance in NUREG/CR-4831, "State of the Art in Evacuation Time Estimate Studies for Nuclear Power Plants."

An ETE update should be performed every five years to ensure the adequacy of other evacuation assumptions.

6.0 RECOVERY

6.1 Recovery Planning

Recovery is defined as those steps taken to return the plant to its pre-accident condition. The overall goals of the recovery effort are to assess the in-plant consequences of the emergency and perform cleanup and repair operations. This effort includes the utilization of CP&L Corporate resources and interfacing with outside agencies. All recovery actions will be pre-planned in order to minimize radiation exposure or other hazards to recovery personnel.

Recovery from an emergency situation is guided by the following principles:

A. The protection of the public health and safety is the foremost consideration in formulating recovery plans.

B. Public officials are kept informed of recovery plans so that they can properly carry out their responsibilities to the public.

C. Periodic briefings of media representatives are held to inform the public of recovery plans and progress made.

Periodic status reports are given to company employees at other locations and to government and industry representatives.

D. The radiation doses to employees and other radiation workers are kept As Low As Reasonably Achievable (ALARA).

E. Necessary adjustments in the size and makeup of the Recovery Manager's staff are made as deemed necessary by the Recovery Manager.

The recovery organization may begin to develop plans for recovery of the facility while the emergency is still in progress. However, these efforts will not be permitted to interfere with or detract from the efforts to control the emergency situation. During the emergency phases of the incident, the recovery organization resources will be available to assist and provide support for the Site Emergency Coordinator.

6.2 Recovery Plan Activation

The Site Emergency Coordinator, with concurrence from the Emergency Response Manager, has the responsibility for determining when an emergency situation is stable and the plant is ready to enter the recovery phase. Prior to terminating an emergency and entering the recovery phase, the following conditions are considered:

A. Do conditions still meet an Emergency Action Level? If so, does it appear unlikely that conditions will deteriorate?

B. Radioactive releases are under control and are no longer in excess of Technical Specification limits.

C. The radioactive plume has dissipated and plume tracking is no longer required. The only environmental assessment activities in progress are those necessary to assess the extent of deposition resulting from passage of the plume.

D. In-plant radiation levels are stable or decreasing, and acceptable, given the plant conditions.

E. The potential for uncontrolled radioactive release is acceptably low.

F. The reactor is in a stable shutdown condition and long-term core cooling is available.

G. Containment pressure in within Technical Specification limits.

H. Any fire, flood, earthquake or similar emergency condition no longer exists.

I. All required notifications have been made.

J. Discussions have been held with Federal, State and local agencies and agreement has been reached to terminate the emergency.

K. At an Alert or higher classification, the Emergency Response Organization is in place and emergency facilities are activated.

It is not necessary that all conditions listed above be met; however, all items must be considered prior to entering the recovery phase. For example, it is possible after a severe accident that some conditions remain which exceed and Emergency Action Level, but entry into the recovery phase is appropriate.

Decisions to relax protective actions for the public will be made in accordance with the North Carolina Radiological Emergency Plan. The Recovery Manager will provide information to the appropriate state agencies to facilitate the decision.

Once the decision is made to enter the recovery phase, the extent of the staffing required for the HNP Recovery Organization is determined.

A. For events of a minor nature, (that is, for UNUSUAL EVENT classifications) the normal on shift organization is normally adequate to perform necessary recovery actions.

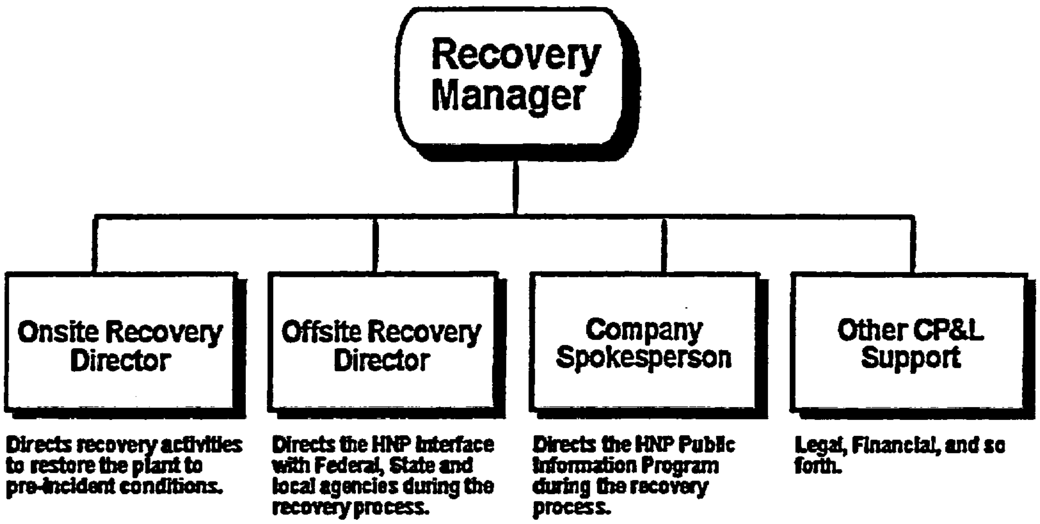
B. For events where damage to the plant has been significant, but no offsite releases have occurred and/or protective actions were not performed, (that is, for Alert classifications) the HNP Emergency Response Organization, or portions thereof, should be adequate to perform the recovery tasks prior to returning to the normal plant organization.

C. For events involving major damage to systems required to maintain safe shutdown of the plant and offsite radioactive releases have occurred, (that is, for Site Area Emergency or General Emergency classifications) the Recovery Organization is put in place.

When the decision is made to enter the recovery phase, all members of the HNP Emergency Response Organization are informed of the change. All appropriate personnel are instructed of the Recovery Organization and their responsibilities to the recovery effort. Notification of off-site organizations that the Recovery Organization is to be activated will be initiated by the Emergency Response Manager and will follow plant emergency notification procedures summarized in Section 4.2 of the Plan (except that the notification message will state that the Recovery Plan has been initiated, will list the new positions of the Recovery Organization, and the notification time limits will not be applicable).

6.3 Recovery Organization

The specific members of the Recovery Organization are selected based on the sequence of events that preceded the recovery activities as well as the requirements of the recovery phase. The basic framework of the Recovery Organization is as follows:



This organization may be modified during the recovery process to better respond to the conditions at the plant.

The state will be the lead organization for off-site recovery operations. The state's recovery organization will be in accordance with the North Carolina Emergency Response Plan.

6.4 Assignment of Responsibilities

6.4.1 Recovery Manager

The Recovery Manager is charged with the responsibility for directing the activities of the CP&L Recovery Organization. These responsibilities include:

A. Ensuring that sufficient personnel from CP&L and other organizations are available to support recovery.

B. Directing the development of a recovery plan and procedures.

C. Ensuring that adequate engineering activities to restore the plant are properly reviewed and approved.

D. Deactivating any of the HNP Emergency Response Organization which was retained to aid in recovery, in the appropriate manner.

E. Coordinating the integration of available Federal and State assistance into onsite recovery activities.

F. Coordinating the integration of CP&L support with Federal, State and local authorities into required offsite recovery activities.

G. Approving information released by the public information organization which pertains to the emergency or the recovery phase of the accident.

H. Determining when the recovery phase is terminated.

I. The Vice President-HNP, or a designated alternate is the Recovery Manager.

6.4.2 Onsite Recovery Director

The Onsite Recovery Director reports to the Recovery Manager and is responsible for:

A. Coordinating the development and implementation of the recovery plan and procedures.

B. Directing all onsite activities in support of the recovery of HNP.

C. Designating other CP&L recovery positions required in support of onsite recovery activities.

The Onsite Recovery Director position will normally be filled by the General Manager-Harris Plant or designee.

6.4.3 Offsite Recovery Director

The Offsite Recovery Director reports to the Recovery Manager and is responsible for:

A. Providing liaison with offsite agencies and coordinating CP&L assistance for offsite recovery activities.

B. Coordinating CP&L ingestion exposure pathway EPZ sampling activities.

C. Developing a radiological release report.

D. Designating other CP&L recovery positions required in support of offsite recovery activities.

The Offsite Recovery Director position will normally be filled by the Manager Plant Support Services or designee.

6.4.4 Company Spokesperson

The Company Spokesperson reports to the Recovery Manager and is responsible for:

A. Functioning as the official spokesperson to the press for CP&L on all matters relating to the accident or recovery.

B. Coordinating non-CP&L public information groups (Federal, State, County, and so forth).

C. Coordinating media monitoring and rumor control.

D. Determining what public information portions of the HNP Emergency Response Organization will remain activated.

The Company Spokesperson position will normally be filled by the Manager Communications-HNP or designee.

6.4.5 The Remainder of the HNP Recovery Organization

The remainder of the HNP Recovery Organization is established and an initial recovery plan developed at the end of the emergency phase or just after entry into the recovery phase. Consideration is given to recovery activity needs and use of the normal HNP organizations. Individual recovery supervisor may be designated in any or all of the following areas:

A. Maintenance

B. Engineering/Technical Support

C. Radiation Protection

D. Operations

E. Chemistry

F. Security

G. Quality Assurance

H. Training

I. Special Offsite Areas (Community Representatives, Environmental Samples, Investigations, and so forth)

6.5 Reentry Planning

The plans and procedures for area reentry will be developed at the time and will consider existing as well as potential conditions inside affected areas.

Prior to reentry, the Recovery Manager and staff shall:

A. Review all available radiation survey data and determine plant areas potentially affected by radiation exposure and contamination.

B. Review the radiation exposure records of personnel participating in the recovery operation and determine the need for additional personnel.

C. Review the adequacy of the radiation sampling and survey instrumentation to be used by the team (type, ranges, number, calibration, and so forth).

D. Review protective clothing, dosimetry, and respiratory protection needs.

E. Ensure appropriate communications are available.

F. Ensure all team members are briefed concerning areas to be entered, anticipated radiation levels, access control procedures, and methods and procedures that will be employed during the entry. The initial entry into the affected area should encompass the following actions:

- Conduct a comprehensive radiation survey of the plant facilities and define all radiological problem areas.
- Isolate and post with appropriate warning signs all radiation and contamination areas.
- Identify potential hazards associated with the recovery operation.

6.6 Total Population Exposure Estimates

The Radiological Control Manager will periodically update the estimate of total population exposure. The estimate will be determined from data collected in cooperation with the State.

The North Carolina Division of Radiation Protection (DRP), Department of Environment, Health and Natural Resources will be the lead state agency in the collection and analysis of radiation monitoring reports and of environmental air, foliage, food, and water samples. The DRP will be assisted by qualified personnel from HNP.

Total population exposure will be periodically determined through a variety of procedures including:

A. Examination of prepositioned TLDs.

B. Bioassay

C. Estimates based on release rates and meteorology.

D. Estimates based on environmental monitoring of food, water, and ambient dose rates.

6.7 Recovery Termination and Reporting Requirements

Responsibility for providing a closeout verbal summary and written summary to off-site authorities after the accident is the responsibility of the General Manager - Harris Plant. These summaries should be simple and in sufficient detail only to define that the accident situation is ended.

Reports to the NRC are in accordance with 10CFR50.72, 10CFR20, Subpart M, and the HNP Technical Specifications, Section 6.9.

7.0 REFERENCES

A. HNP Plant Operating Manual.

B. CP&L Radiation Control and Protection Manual.

C. Final Safety Analysis Report (FSAR), Carolina Power & Light Company, Shearon Harris Nuclear Power Plant.

D. EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," U.S. Environmental Protection Agency.

E. EPPOS No. 1, "Emergency Preparedness Position (EPPOS) on Acceptable Deviations from Appendix 1 of NUREG-0654 Based Upon the Staff's Regulatory Analysis of NUMARC/NESP-007, 'Methodology for Development of Emergency Action Levels'", June 5, 1995.

F. EPPOS No. 2, "Emergency Preparedness Position (EPPOS) on Timeliness of Classification of Emergency Conditions", August 17, 1995.

G. EPPOS No. 3, "Emergency Preparedness Position (EPPOS) on Requirement for Onshift Dose Assessment Capability", November 8, 1995.

H. NUREG-0654/FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, October 1980, Revision 1.

I. NUREG-0737, Clarification of TMI Action Plan Requirements, dated October 1980.

J. NUREG-0737, Supplement 1, Requirements for Emergency Response Capability, December 1982.

K. NUREG-0696, Functional Criteria for Emergency Response Facilities, Final Report, February 1981.

L. Title 10, Code of Federal Regulations; Part 20, Standards for Protection Against Radiation and Part 50, Licensing of Production and Utilization Facilities

M. Federal Register, Vol. 43, No. 242, December 15, 1978, U.S. Food and Drug Administration, Accidental Radioactive Contamination of Human Food and Animal Feeds.

N. Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone, Shearon Harris Nuclear Power Plant, January, 1997.

O. RTM-92.

ANNEX A

LETTERS OF AGREEMENT

This Annex contains a list of written agreements between CP&L and other organizations that may be required to provide support to the Harris Nuclear Plant in the event of an on-site radiological emergency. Copies of the original agreements are kept on file by HNP Emergency Preparedness or CP&L Contract Services.

Agreement Organization

1. Apex Volunteer Fire Department
2. Town of Holly Springs Dept. Of Public Safety Division of Municipal Fire Services
3. Apex Rescue Squad, Inc.
4. Rex Hospital
5. Wake Medical Center
6. Western Wake Medical Center
7. Douglas I. Hammer, M.D.
8. Stephen E. Johnson, M.D.
9. Institute of Nuclear Power Operations
10. National Weather Service
11. State of North Carolina - supporting emergency plan - see Annex G
12. Chatham County - supporting emergency plan - see Annex G
13. Harnett County - supporting emergency plan - see Annex G
14. Lee County - supporting emergency plan - see Annex G
15. Wake County - supporting emergency plan - see Annex G
16. Framatone
17. Atlantic Group
18. Raytheon
19. Murray and Trettle - on demand services

These agreements are maintained current through annual reconfirmation, where required, or through personal verification of current applicability where reconfirmation is not required. A copy of the EP Supervisor's annual certification that the agreements are applicable and have been reconfirmed when necessary, is kept on file by HNP Emergency Preparedness.

ANNEX B

<u>Technical Basis Of Emergency Dose Projection Program</u>

Moved the dose assessment technical basis document to EPM-600 as part of Rev. 35.

ANNEX C

Glossary Of Terms

Accident Assessment - Accident assessment consists of a variety of actions taken to determine the nature, effects, and severity of an accident and includes evaluation of reactor operator status reports, damage assessment reports, meteorological observations, seismic observations, fire reports, radiological dose projections, in-plant radiological monitoring, and environmental monitoring.

Activate - To formally put on active duty with the necessary personnel and equipment to carry out the function required, such as to activate the Technical Support Center (TSC) or the Emergency Operations Facility (EOF).

Alerting/Warning, Public - The process of signaling the public, as with sirens, to turn on their TVs or radios and listen for information or instructions broadcast by state or local government authorities on the Emergency Alert System (EAS).

Assessment Actions - Those actions taken during or after an accident to obtain and process information which is necessary to make decisions to implement specific emergency measures.

Command and Control - Exercising the authority to coordinate and utilize an organization's resources to respond to an emergency condition.

Committed Dose Equivalent (CDE) - The Dose Equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

Corrective Action - Those emergency measures taken to lessen or terminate an emergency situation at or near the source of the problem, to prevent an uncontrolled release of radioactive material, or to reduce the magnitude of a release. Corrective action includes, equipment repair or shutdown, installation of emergency structures, fire fighting, repair, and damage control.

County(ies) - When used in the context of the HNP 10-mile EPZ means Chatham, Lee, Harnett, and/or Wake County(ies).

Damage Assessment - Estimates and descriptions of the nature and extent of damages resulting from an emergency or disaster; of actions that can be taken to prevent or mitigate further damage; and of assistance required in response and recovery efforts based on actual observations by qualified engineers and inspectors.

Damage Control - The process of preventing further damage to occur and preventing the increase in severity of the accident.

Decontamination - The reduction or removal of contaminated radioactive material from a structure, area, material, object, or person. Decontamination may be accomplished by (1) treating the surface so as to remove or decrease the contamination, (2) letting the material stand so that the radioactivity is decreased as a result of natural decay, and (3) covering the contamination.

ANNEX C

Glossary Of Terms

DEM - An abbreviation standing for North Carolina Division of Emergency Management. DEM is the State agency responsible for preparing and maintaining a State Radiological Emergency Response Plan and for assembling and dispatching a State Emergency Response Team (SERT) to the scene of an emergency.

Dose Projection - The calculated estimate of a radiation dose to individuals at a given location (normally off site), determined from the source term/quantity of radioactive material (Q) released, and the appropriate meteorological dispersion parameters (χ/Q).

Dose Rate - The amount of ionizing (or nuclear) radiation to which an individual would be exposed per unit of time. As it would apply to dose rate to a person, it is usually expressed as Rem per hour or in submultiples of this unit, such as millirem per hour. The dose rate is commonly used to indicate the level of radioactivity in a contaminated area.

Dosimeter - An instrument such as a thermoluminescent dosimeter (TLD), self-reading pocket dosimeter (SRPD), or electronic dosimeter (ED) for measuring, registering, or evaluating total accumulated dose or exposure to ionizing radiation.

Drill - A supervised instruction period aimed at testing, developing, and maintaining skills in a particular operation.

Early Phase - The period at the beginning of a nuclear incident when immediate decisions for effective use of protective actions are required and must be based primarily on predictions of radiological conditions in the environment. This phase may last from hours to days. For the purposes of dose projections it is assumed to last four days.

Emergency Action Levels (EALs) - Plant conditions used to determine the existence of an emergency and to classify its severity. The conditions include specific instrument readings, alarms, and observations that in combination indicate that an emergency initiating event has occurred and therefore an appropriate class of emergency should be declared. EALs cover a broad range of events such as radioactive releases to the environment, loss of all on-site and off-site power, security threats, fire, strikes of operating employees.

Emergency Alert System (EAS) - A network of broadcast stations and interconnecting facilities which have been authorized by the Federal Communications Commission to operate in a controlled manner during a war, state of public peril or disaster, or other national emergency - as provided by the Emergency Alert System Plan. In the event of a nuclear reactor accident, instructions/notifications to the public on conditions or protective actions would be broadcast by state or local government authorities on the EAS.

Emergency Operating Procedures (EOPs) - EOPs are step-by-step procedures for direct actions taken by licensed reactor operators to mitigate and/or correct an off normal plant condition through the control of plant systems.

ANNEX C

Glossary Of Terms

Emergency Operations Center (EOC) - A facility designed and equipped for effective coordination and control of emergency operations carried out within an organization's jurisdiction. The site from which civil government officials (Municipal, County, State, and Federal) exercise direction and control in a civil defense emergency.

Emergency Operations Facility (EOF) - The EOF is a CP&L facility near the plant that is provided for the management of overall CP&L emergency response in the event of a nuclear accident at the plant. Upon activation of the EOF, it assumes for the Technical Support Center (TSC) the function of providing support to the state on off-site radiological and environmental assessments, coordination with Federal, State, and Local Government officials on recommendations for public protective actions and direction of recovery operations.

Emergency Planning Zones (EPZ) - A generic area defined about a nuclear plant to facilitate emergency planning off site. The plume exposure EPZ is described as an area with approximately a 10-mile radius and the ingestion exposure EPZ is described as an area with approximately a 50-mile radius, both of which are centered at the plant site.

Emergency Preparedness - A state of readiness that provides reasonable assurance that adequate protective measures can and will be taken upon implementation of the emergency plan in the event of a radiological emergency.

Evacuation - The urgent removal of people from an area to avoid or reduce high-level, short-term exposure usually from the plume or from deposited activity.

Evacuation, Exclusion Area - The evacuation of nonessential personnel from the Exclusion Area.

Evacuation, Local - The evacuation of personnel from a particular area, such as a room or building.

Evacuation, Site - The evacuation of nonessential personnel from the plant site.

Exercise - An event that tests the integrated capability of a major portion of the basic elements existing within emergency preparedness plans and organizations.

Exclusion Area - An Exclusion Area is an area specified for the purpose of reactor site evaluation in accordance with 10CFR100. It is an area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated release would not receive a total radiation dose to the whole body in excess of 25 Rem or a total radiation dose of 300 Rem to the thyroid from iodine exposure. The exclusion area around HNP is CP&L-owned property with a radius of approximately 7000 feet.

Fission Product Barrier - The fuel cladding, reactor coolant system boundary, or the containment boundary.

ANNEX C

Glossary Of Terms

Fission Product Barrier Status -

a. Breached - The fission product barrier is incapable of sufficiently retaining radioactive materials to protect the public.

b. Jeopardy - Conditions exist that are likely to result in fission product barrier breach, but the barrier is intact at the present time.

c. Intact - The fission product barrier retains the ability to protect the public from a harmful release of radioactive materials.

Health Physics Network (HPN) Line - In the event of a Site Area Emergency, the NRC HPN line will be activated by the NRC Operations center in Bethesda, Maryland. This phone is part of a network that includes the NRC Regional Office and the NRC Operations Headquarters in Bethesda, Maryland. This system is dedicated to the transmittal of radiological information by plant personnel to NRC Operations Center and the Regional office. HPN phones are located in the TSC and EOF.

Ingestion Exposure Pathway - The potential pathway of radioactive materials to the public through consumption of radiologically contaminated water and foods such as milk or fresh vegetables. Around a nuclear power plant this is usually described in connection with the 50-mile radius Emergency Planning Zone (50-mile EPZ).

Intermediate Phase - The period beginning after the source and releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on additional protective actions.

Joint Information Center (JIC) - An Emergency Facility activated by CP&L and staffed by CP&L, State, and County Public Information personnel. This facility serves as the single point of contact for the media and public to obtain information about an emergency.

Late Phase - The period beginning when recovery action designed to reduce radiation levels in the environment to acceptable levels for unrestricted use are commenced and ending when all recovery actions have been completed. This period may extend from months to years (also referred to as the recovery phase).

Main Control Room - The operations center of a nuclear power plant from which the plant can be monitored and controlled.

Monitoring, Environmental - The use of radiological instruments or sample collecting devices to measure and assess background radiation levels and/or the extent and magnitude of radiological contamination in the environment around the plant. This may be done in various stages such as pre-operational, operational, emergency, and post operational.

ANNEX C

Glossary Of Terms

Monitoring, Personnel - The determination of the degree of radioactive contamination on individuals, using standard survey meters, and/or the determination of dosage received by means of dosimetry devices.

Notification, Public - Public notification means to communicate instructions on the nature of an incident that prompted the public alerting/warning and on protective or precautionary actions that should be taken by the recipients of the alert. A state and local government process for providing information promptly to the public over radio and TV at the time of activating the alerting (warning) signal (sirens). Initial notifications of the public might include instructions to stay inside, close windows, and doors, and listen to radio and TV for further instructions. Commercial broadcast messages are the primary means for advising the general public of the conditions of any nuclear accident. (See Emergency Alert System.)

NRC Emergency Notification System (ENS) - The NRC Emergency Notification System hot line is a dedicated telephone system that connects the plant with NRC headquarters in Bethesda, Maryland. It is directly used for reporting emergency conditions to NRC personnel.

Off-Site - The area outside of an approximate 2500-foot radius from the plant centerline, exclusive of the area cleared for plant construction.

On-Site - The area inside of an approximate 2500-foot radius from the plant centerline, inclusive of the area cleared for plant construction, and including all permanent and temporary buildings, and the parking lots.

Operations Support Center (OSC) - An emergency response facility at the Plant to which support personnel report and stand by for deployment in an emergency situation.

Plume Exposure Pathway - The potential pathway of radioactive materials to the public through (a) whole body external exposure from the plume and from deposited materials, and (b) inhalation of radioactive materials.

Population-at-Risk - Those persons for whom protective actions are being or would be taken. In the 10-mile EPZ the population-at-risk consists of resident population, transient population, special facility population, and industrial population.

Potassium Iodide - (Symbol KI) A chemical compound that readily enters the thyroid gland when ingested. If taken in a sufficient quantity prior to exposure to radioactive iodine, it can prevent the thyroid from absorbing any of the potentially harmful radioactive iodine-131.

ANNEX C

Glossary Of Terms

Procedure, Plant Emergency (PEP) - Plant emergency procedures implement the HNP Emergency Plan and are published in Volume 2, Part 5 of the Plant Operations Manual. PEPs define the specific, step-by-step actions to be followed by the emergency organization in the process of recognizing and assessing an emergency condition, and mitigating the condition through the use of corrective and protective actions. PEPs do not include those actions taken by licensed control operators to directly control plant systems (see Emergency Instructions).

Projected Dose - An estimate of the potential radiation dose which affected population groups could receive.

Protected Area - An area of the plant site encompassed by physical barriers to which access is controlled.

Protection Factor (PF) - The relation between the amount of radiation which would be received by a completely unprotected person compared to the amount which would be received by a protected person such as a person in a shielded area. PF = Unshielded dose rate ÷ shielded dose rate.

Protective Action - Sometimes referred to as protective measure. An activity conducted in response to an incident or potential incident to avoid or reduce radiation dose to members of the public.

Protective Action Guide (PAG) - The projected dose to reference man or other defined individual from an accidental release of radioactive material at which a specific protective action to reduce or avoid that dose is warranted.

Recovery - The process of reducing radiation exposure rates and concentrations of radioactive material in the environment to levels acceptable for unconditional occupancy or use.

Release - Escape of radioactive materials into the uncontrolled environment.

Restricted Area -Any area, access to which is controlled by Carolina Power & Light Company for purposes of protection of individuals from exposure to radiation and radioactive materials.

Safety Analysis Report, Final (FSAR) - The FSAR is a comprehensive report that a utility is required to submit to the NRC as a prerequisite and as part of the application for an operating license for a nuclear power plant. The multivolume report contains detailed information on the plant's design and operation, with emphasis on safety-related matters.

ANNEX C

Glossary Of Terms

Safety-related - As used in this plan and in Plant Emergency Procedures when describing areas, equipment, systems or components, safety-related means:

1. Forming a part of the Reactor Coolant System pressure boundary, or
2. Used to mitigate the consequences of an abnormal condition, or
3. Necessary to achieve or maintain safe shutdown of the plant.

SERT - State Emergency Response Team (North Carolina). (See also DEM).

Shelter - A habitable structure or space used to protect its occupants from radiation exposure. The radiation protection factor (PF) of the shelter will vary as a function of the density of structural materials located between its occupants and the source of radiation.

Shielding - Any material or barrier that attenuates (stops or reduces the intensity of) radiation.

Source Term - Radioisotope inventory of the reactor core, or amount of radioisotope released to the environment, often as a function of time.

State - The State of North Carolina.

Technical Support Center (TSC) - A center outside of the Main Control Room in which information is supplied on the status of the plant to those individuals who are knowledgeable or responsible for engineering and management support of reactor operations in the event of an emergency, and to those persons who are responsible for management of the on-site emergency response.

Total Effective Dose Equivalent (TEDE) - The sum of external and internal ionizing radiation exposure.

Unrestricted Area - Any area to which access is not controlled by the licensee for protecting individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

ANNEX D

NUREG-0654 REV. 1 Cross-Reference

NUREG-0654, Criterion Paragraph		EP Section Number
A. Assignment of Responsibility		
A.1.a	Identify response organizations	Annex G
A.1.b	Concept of operations	2.0, 4.1, Table 4.0-1
A.1.c	Illustrate interrelationships	Figures G-1, G-2, Table 2.2 -1, Figure 2.2 -1, Figure 2.4-1
A.1.d	Individual responsible for emergency response	2.3, 2.4.1.B, 2.4.2.A, 2.4.4.A
A.1.e	Provision for 24 hours per day response	2.2, Table 2.2-1
A.3	Agreements	Annex A, Annex G
A.4	Individual responsible for resources	2.4.1.B, 2.4.2.A, 2.4.4.A
B. On-site Emergency Organization		
B.1	Plant Emergency Organization	2.0, Table 2.2-1, Figure 2.2-1, Figure 2.4-1
B.2	Assignment of Site Emergency Coordinator	2.4.1.B, 2.3, 2.4.2.A, 2.4.4.A
B.3	Line of succession	2.3, 2.4.1.B, 2.4.2.A, 2.4.4.A
B.4	Responsibilities	2.3, 2.4.1.B, 2.4.2.A, 2.4.4.A
B.5	Emergency organization and assignments	2.2, 2.4, Table 2.2-1
B.6	Interfaces - Plant, State, Local, Corp.	Figures 2.2-1, 2.4-1, Table 4.0-1, Annex G
B.7	Corporate Emergency Organization	2.2
B.7.a	Logistics support for emergency personnel	2.4.4.F
B.7.b	Technical Support - planning reentry, recovery	2.2, 6.4.5.

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NUREG-0654 REV. 1 Cross-Reference

NUREG-0654, Criterion Paragraph		EP Section Number
B.7.c	Management to Government interface	2.4.1.B, 2.4.4.A, Figures G-1 and G-2
B.7.d	Corporate news media coordination	2.4.4.E, 2.4.5
B.8	Contractor and private assistant	Annex G
B.9	Local agency services	Annexes A & G
C. Emergency Response Support and Resources		
C.1.a	Titles authorized to request federal assistance	2.3.E
C.1.b	Specific federal resources expected and delivery time.	Annex G
C.1.c	Airports, EOC, telephones, radios, available to assist federals	3.7, 3.8
C.2.b	Licensee representative to principal government EOCs	2.4.4.X
C.3	Description of available radiological labs	3.9.7
C.4	Nuclear and other facilities or organizations	2.5, Annex G
D. Emergency Classification System		
D.1	Emergency classification system and EALS, parameter values and equipment status.	4.1, Figure 4.1-1, Figure 4.1-2
D.2	Initiating conditions and FSAR accidents.	4.1, Figure 4.1-1, Figure 4.1-2
E. Notification Methods and Procedures		
E.1	Establish procedures for notification of response organization and verification.	4.2
E.2	Establish procedures for alerting, notifying, and mobilizing response personnel.	4.2, 4.3, Table 4.2-1-4
E.3	Establish content of message.	4.2
E.4	Make provision for follow-up message.	4.2

ANNEX D

NUREG-0654 REV. 1 Cross-Reference

	NUREG-0654, Criterion Paragraph	EP Section Number
E.6	Provide for alerting and notifying public.	4.5.4, Annex H
E.7	Provide narrative for public messages on protective actions.	4.5.4
F. Emergency Communications		
F.1	Establish organizational titles and alternate means of primary and backup communications	2.4.1.G, 2.4.2.F-G, 2.4.4.U-W, 3.8, 4.1
F.1.a	Provide Telephone link and alternate for 24-hour notification to state and local agencies	2.4.1.8, 2.4.4.V, 3.8, 4.2, Annex G
F.1.b	Provide for communications with contiguous state/local agencies	2.4.1.G, 2.4.4.V, 3.8
F.1.c	Provide for communications with Federal agencies	2.4.1.G, 2.4.2.G, 3.8
F.1.d	Communication between plant, EOF, state and local EOCs and RM teams	3.8
F.1.e	Provide for alerting and activating emergency personnel	3.8, 4.2
F.1.f	Communication between NRC, EOF and environmental monitoring teams	3.8
F.2	Communication link for fixed and mobile medical	3.8, 4.6.3.E.4)
F.3	Conduct periodic testing of communication system	5.3.1.A
G. Public Education and Information		
G.1	Disseminate, annually, educational information to public	5.2.5
G.2	Disseminate, annually, educational information for transient population	5.2.5
G.3.a	Designate contacts and space for media	3.6
G.3.b	Provide space for media at the EOF	3.5.2.
G.4.a	Designate a spokesperson	2.4.5.A

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NUREG-0654 REV. 1 Cross-Reference

NUREG-0654, Criterion Paragraph		EP Section Number
G.4.b	Provide for timely exchange of information between spokespersons	2.4.5.A
G.4.c	Provide for coordinated rumor control	3.6.F
G.5	Provide annual training for media	5.2.5
H. Emergency Facilities and Equipment		
H.1	Establish TSC and OSC	3.3, 3.4
H.2	Establish EOF	3.5
H.4	Provide for timely activation of facilities	4.2, 4.3
H.5.a	Identify & establish on-site geophysical phenomena monitors	3.9.2, 3.9.5
H.5.b	On-site radiological monitors: process, area, emergency.	3.9.3, 3.9.6
H.5.c	On-site Process monitors: reactor coolant pressure, temperature, and so forth	3.9.1
H.5.d	On-site fire and combustion products detectors	3.9.8
H.6.a	Provide access to off-site geophysical monitors	3.9.2, 3.9.5
H.6.b	Access to off-site radiological monitors and sampling	3.9.3, 3.9.6, 3.9.7
H.6.c	Access to off-site laboratories: fixed or mobile	3.9.7
H.7	Provide for radiological monitoring equipment off-site	3.5.3, 3.9.3, Table 3.1-1, 4.4.4
H.8	Provide meteorological instrumentation and procedures	3.9.5
H.9	Provide for OSC and special equipment in the OSC	3.4, Table 3.1-1
H.10	Inspect emergency equipment and supplies	5.4.1
H.11	Identify emergency kits by general category	Table 3.1-1

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NUREG-0654 REV. 1 Cross-Reference

NUREG-0654, Criterion Paragraph		EP Section Number
H.12	Establish point near EOF for receipt of environmental monitoring data	3.9.7

I. Accident Assessment

I.1	Identify plant system and effluent parameters and instruments values	3.91, 4.1, Figures 4.1-1, Figure 4.1.-2
I.2	Post-accident sampling, radiation monitors, and so forth	3.9.3, 4.4.2.
I.3.a	Establish methods and techniques to determine source terms	4.4
I.3.b	Methods to determine magnitude of release	4.4
I.4	Establish relationships for effluent monitor readings	4.4
I.5	Capability to acquire and evaluate meteorological data	3.9.5
I.6	Procedure for assessment when instruments off-scale	4.4.3
I.7	Describe capability and resources for environmental monitoring	2.4.4.T, Table 2.2-1, Table 3.1-1, 4.4.4
I.8	Assessment of (radiological) environmental hazards from liquid or gas	2.4.4.T, 3.9.6, 3.9.7, 4.2, 4.4.4, Table 2.2-1
I.9	Detect and measure radioiodine in the10-Mile EPZ.	3.9.6, 4.4.4
I.10	Procedure for dose or dose rate projection	4.4.3

J. Protective Response

J.1.a	Establish means and time to warn on-site employees and individuals in the exclusion area not on the ERO	4.6.1, 4.6.2
J.1.b	Establish means and time to warn on-site visitors or visitors in the exclusion area	4.6.1, 4.6.2
J.1.c	Establish means and time to warn contractor/construction personnel	4.6.1

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NUREG-0654 REV. 1 Cross-Reference

NUREG-0654 Criterion Paragraph		EP Section Number
J.1.d	Est. means and time to warn others	4.5.4, 4.6.1
J.2	Evacuation routes and transportation for on-site people	4.6.2
J.3	Provide radiation monitoring for people in J.2	4.6.2
J.4	Provide decon capability at J.3 location	4.6.2, 4.6.3.A
J.5	Account for personnel, ascertain missing individuals within 30 minutes of start of emergency and account for on-site persons continuously thereafter.	4.6.2
J.6.a	For individuals remaining or arriving - respiratory protection	Table 3.1-1, 4.6.3.C
J.6.b	For individuals remaining or arriving - protective clothing	Table 3.1-1, 4.6.3.C
J.6.c	For individuals remaining or arriving - radioprotective drugs	Table 3.1-1, 4.6.3.C
J.7	Recommendations To Local Government	4.5
J.8	Evacuation time estimates - 10-Mile EPZ	1.7, Table 1.8-2
J.10.a	Maps-Evac. routes, areas, rad. Sampling and monitoring points, reception and shelter areas	Annex H, 4.4.4, Annex G
J.10.b	Map-Population by Sectors and local zones	1.8, Table 1.8-1, Figure 1.8-1
J.10.c	Means for notifying transient and resident population	4.5.4
J.10.m	Bases for recommended protective actions; shelter, evac. time	4.5.1, 4.5.2, Table 4.5-2
K. Radiological Exposure Control		
K.1.a	Exposure guidelines - removal of injured persons	4.6.3.D
K.1.b	Exposure guidelines - performing corrective actions	4.6.3.D

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NUREG-0654, Criterion Paragraph		EP Section Number
K.1.c	Exposure guidelines - performing assessment actions	4.6.3.D
K.1.d	Exposure guidelines - providing first aid	4.6.3.E
K.1.e	Exposure guidelines - personnel decontamination	4.6.3.E
K.1.f	Exposure guidelines - providing ambulance service	4.6.3.B, 4.6.3.E
K.1.g	Exposure guidelines - medical treatment	4.6.3.F, 4.6.3.E
K.2	On-site radiation protection program- emergency	4.6.3.D
K.3.a	Dosimetry - 24-hour capability	4.6.3.B
K.3.b	Emergency worker dosimeters and dose records	4.6.3.B
K.5.a	Decontamination guides - action levels	4.6.3.E., Table 4.6-1
K.5.b	Means for decontamination and waste disposal	4.6.3.A, 4.6.3.E, Table 4.6-1
K.6.a	Contamination control - access control	4.6.3.E, Table 4.6-1
K.6.b	Contamination control - drinking water and food	4.6.3.G
K.6.c	Criteria for return to normal use- areas, items	4.6.3.E.3
K.7	Decontamination - relocated on-site personnel	4.6.2, 4.6.3.A, 4.6.3.E.3
L. Medical and Public Health Support		
L.1	Local and backup hospital for evaluation of radiation exposure - adequately prepared	4.6.3.F.1, Annex A
L.2	On-site first aid capability	2.4.1.F, 4.6.3.E.2
L.4	Transportation - victims of radiation accident	4.6.3.E.4, Annex A
M. Recovery and Reentry Planning and Post-Accident Operations		
M.1	Plans and procedures - relaxation of protective measures	6.4, 6.5, 6.6
M.2	Recovery organization	6.2, 6.3

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NUREG-0654, Criterion Paragraph		EP Section Number
M.3	Means for alerting recovery personnel	4.2, 6.4
M.4	Method for periodically estimating population dose	6.6
N. Exercises and Drills		
N.1.a	Conduct annual exercise - off-site release	5.3.2
N.1.b	Verify capability to respond - Evaluate, Critique	5.3.2
N.2.a	Conduct communication drills to test communications with: State & local in10-mi. EPZ monthly; Federal & State in50-mi. EPZ quarterly; plant, State & local EOCs and field assessment teams annually.	5.3.1.A
N.2.b	Conduct fire drills per plant tech specs	5.3.1.B
N.2.c	Medical emergency drill, contaminated individual & participation by local ambulance & off-site medical facility annually.	5.3.1.C
N.2.d	Plant environs and radiological monitoring drills - annually	5.3.1.D
N.2.e.(1)	HP drill, semi-annual, response to , analysis of, simulated airborne and liquid	5.3.1.E
N.2.e.(2)	HP drill, annual, analysis of actual elevated liquid samples, including use of PASS	5.3.1.E
N.3.a	Plans/Scenario content - objectives and evaluation criteria	5.3.2
N.3.b.	Plans/Scenario content -dates, time period, place, and participating organization	5.3.2
N.3.c	Plans/Scenario content - simulated events	5.3.2
N.3.d	Plans/Scenario content - time	5.3.2
N.3.e	Plans/Scenario content - narrative summary	5.3.2
N.3.f	Plans/Scenario content - official observers	5.3.2

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NUREG-0654 REV. 1 Cross-Reference

NUREG-0654, Criterion Paragraph		EP Section Number
N.4	Provision for critiques	5.3.2
N.5	Provision for identifying areas of improvement and assigning responsibility for corrective action	5.3.2
O. Radiological Emergency Response Training		
0.1	Assure the training of individuals who maybe called on to assist in an emergency	5.2
0.1.a	Provide training for off-site emergency organizations	5.2.1, 5.2.3
0.2	Training to include practical drills - on-site organization	5.2.1
0.3	First aid team training to include Multimedia	5.2.1
0.4.a	Training of response organization directors	5.2.1
0.4.b	Training of accident assessment personnel	5.2.1
0.4.c	Training of radiological monitoring and analysis personnel	5.2.1
0.4.d	Training of police, security, fire-fighting personnel	5.2.1
0.4.e	Training of repair and damage control teams	5.2.1
0.4.f	Training of first aid and rescue personnel	5.2.1
0.4.g	Training of local support service/CD	5.2.1, 5.2.3
0.4.h	Training of medical support personnel	5.2.1, 5.2.3
0.4.i	Training of headquarters support personnel	5.2.2
0.4.j	Training of emergency communicators	5.2.1
0.5	Provide initial and annual retraining	5.2.1
P. Responsibility for the Planning Effort		
P.1	Provide training for emergency planners	5.2.4
P.2	Identify titles responsible for planning	1.3, 5.1.1

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NUREG-0654 REV. 1 Cross-Reference

NUREG-0654, Criterion Paragraph		EP Section Number
P.3	Designate Emergency Planning Coordinator	1.3, 5.1.1
P.4	Annually review and update plans and agreements	5.1.2
P.5	Distribute emergency plan; identify revisions	5.1.2
P.6	List other supporting plans	1.10
P.7	List and cross-reference procedures for implementing the plan	5.1.4, Annex E
P.8	Provide Table of Contents and NUREG cross-reference	pgs. 2-7, 5.1.5, Annex D
P.9	Arrange independent review every 12 months	5.1.6
P.10	Provide for quarterly updating of telephone numbers	5.1.3

ANNEX E

List Of Emergency Preparedness Documents

Document Type		Plan Section
Emergency Plan Implementing Procedures (PEPs)		
PEP-110	Emergency Classification and Protective Action Recommendations	4.1, 4.5, 4.5.1-2
PEP-230	Control Room Operations	2.4.1, 4.6.1
PEP-240	Activation and Operation of the Technical Support Center	2.4.2, 4.8
PEP-250	Activation and Operation of the Joint Information Center	2.4.5
PEP-260	Activation and Operation of the Operations Support Center	2.4.3
PEP-270	Activation and Operation of the Emergency Operations Facility	2.4.4
PEP-310	Notifications and Communications	4.2, 4.3
PEP-330	Radiological Consequences	2.4.3.I, 2.4.4.T, 4.4.2, 4.4.4, 4.6.3, 4.6.3 E-F, 4.7
PEP-340	Dose Assessment	2.4.4, 4.4.3,
PEP-342	Core Damage Assessment	2.4.4.T, 4.4.1, 4.4.3
PEP-350	Protective Actions	2.4.1.E-G, 4.6.1-2,
PEP-430	Routine Maintenance and Testing of the Dialogic System	4.2.f
PEP-500	Recovery	6.1-5
Emergency Program Maintenance and Administration (EPMs)		
EPM-100	EP Program Administration	5.0, 5.1.1-2, 5.1.6, 5.3.1-2
EPM-200	ERO Training Program	5.2.1-3
EPM-210	EP Drill and Exercise Program	5.3
EPM-211	EP Scenario Development Guidelines	5.3
EPM-400	Public Notification and Alerting System	5.5
EPM-410	Communication and Facility Performance Tests	5.0, 5.5.1-4
EPM-420	Emergency Equipment Inventory	5.0, 5.4.1
EPM-500	Public Education and Information Program	5.2.5
EPM-600	Dose Assessment Technical Basis	2.4.4Q and R, 4.4.3,
Other Documents		
EPL-001	Emergency Phone List	5.1.3, Annex G

ANNEX F

The warning message form used to notify the State and Counties is provided in Annex F of the North Carolina Emergency Response Plan and is included as a form in PEP-310, "Notifications and Communications."

ANNEX G

Interfacing Information From Supporting Emergency Plans

1.0 General

The material in this Annex is included as general information on how supporting emergency plans interface with the HNP Emergency Plan. The information is presented in a similar format as the basic Plan. Emergency organization interfaces, based on levels of activation, are depicted in Figures G-1 and G-2. A summary of organizations expected to support emergency response is contained in Table G.1-1.

2.0 Coordination with Participating State and Local Government Agencies

2.1 State of North Carolina Governor's Office

The Governor has the authority to direct and control the State Emergency Management Program. During a declared State of Disaster, the Governor has the authority to utilize all available state resources reasonably necessary to cope with emergencies. The Governor's representatives coordinate as necessary with Carolina Power & Light Company (CP&L) and with local government officials.

2.2 North Carolina Department of Crime Control and Public Safety

The Department of Crime Control and Public Safety functions as the State of North Carolina Emergency Planning Coordinator. In that capacity, the Department has overall management responsibility for North Carolina's radiological emergency response planning, development, updating, and coordination with CP&L. The Department coordinates emergency response activities for the State of North Carolina and other government emergency response agencies.

The Department, through its State Highway Patrol, provides the initial 24-hour emergency notification point for the State.

2.3 North Carolina Division of Emergency Management

The Division of Emergency Management (DEM) is the responsible organization within the N.C. Department of Crime Control and Public Safety to prepare and maintain a State Radiological Emergency Response Plan for HNP in coordination with the Department of Environment, Health and Natural Resources and other interested agencies. The DEM is the lead response agency within State government and coordinates the activities of the State Emergency Response Team (SERT) at the State Emergency Operations Center (SEOC) in Raleigh. Personnel within the SEOC will confer with CP&L to determine appropriate emergency response activities which should be taken to protect the health and safety of the public.

Interfacing Information From Supporting Emergency Plans

2.4 Division of Radiation Protection

The Division of Radiation Protection (DRP), within North Carolina Department of Environment, Health and Natural Resources, will be the lead agency in the collection and analysis of radiation monitoring reports and of environmental air, foliage, food, and water samples. The DRP will be assisted by qualified personnel from HNP.

2.5 Chatham County Emergency Operations

Chatham County Emergency Operations has the following responsibilities:

- Develop and maintain Chatham County's Plan to Support the Harris Nuclear Power Plant.
- Coordinate emergency response matters between the State, County, CP&L, and local government agencies.
- Operate the county warning point (Communications Center) on a 24-hour basis. The Communications Center is manned continuously by a Public Safety Dispatcher.
- Coordinate the protective response operations required by the Chatham County Plan to Support the Harris Nuclear Power Plant during an emergency.

2.6 Harnett County Emergency Services

Harnett County Emergency Services has the following responsibilities:

- Develop and maintain the Harnett County's Plan to Support the Harris Nuclear Power Plant.
- Coordinate emergency response matters between the State, County, CP&L, and local government agencies.
- Coordinate the protective response operations required by the Harnett County Plan to Support the Harris Nuclear Power Plant during an emergency.

2.7 Harnett County Sheriff's Department

The Sheriff's Department operates the county warning point on a 24-hour basis. The county warning point is the Sheriff's Department communications center which is manned continuously by a Public Safety Dispatcher.

2.8 Lee County Emergency Services

Lee County Emergency Services has the following responsibilities:

- Develop and maintain the Lee County Plan to Support the Harris Nuclear Power Plant.

- Coordinate emergency response matters between the State, County, CP&L, and local governmental agencies.
- Coordinate the protective response operations required by the Lee County Plan to Support the Harris Nuclear Power Plant during an emergency.

2.9 Lee County Sheriff's Department

The Sheriff's Department operates the county warning point on a 24-hour basis.

The county warning point is the Lee County communications center which is manned continuously by a Public Safety Dispatcher.

2.10 Wake County Emergency Management

- The Wake County Emergency Management has been assigned the following responsibilities:
- Develop and maintain Wake County's Plan to Support the Harris Nuclear Power Plant.
- Coordinate emergency response matters between the State, County, CP&L, and local government agencies.
- Coordinate the protective response operations required by the Wake County Plan to Support the Harris Nuclear Power Plant during an emergency.

2.11 Raleigh Communications Center

The Raleigh City Communications Center provides emergency telephone notification service and serves Wake County and all municipalities within the county as the 24-hour warning point. The warning point is manned continuously by a Public Safety Dispatcher.

3.0 Coordination With Federal Agencies and Other States

3.1 Department of Energy, Savannah River Operations Office

The role of the Department of Energy is described in the Federal Radiological Emergency Response Plan published in the Federal Register, Volume 50, No. 217, November 8, 1985.

3.2 Federal Emergency Management Agency (FEMA)

The role of the Federal Emergency Management Agency (FEMA) is described in the Federal Radiological Emergency Response Plan.

Interfacing Information From Supporting Emergency Plans

3.3 Nuclear Regulatory Commission (NRC)

The NRC provides at least one resident inspector at HNP. Upon notification by CP&L, the NRC provides additional technical advice, technical assistance, and personnel per NUREG-0728, "Report to Congress, NRC Incident Response Plan," and NUREG-0845, "Agency Procedures for the NRC Incident Response Plan." The NRC Operations Center will be notified of radiation incidents in accordance with 10 CFR 50.72 using the Emergency Notification System (ENS) phone.

3.4 Weather Service

The National Weather Service at the Raleigh-Durham Airport, Raleigh, North Carolina, will provide meteorological information during emergency situations, if required. Data available will include existing and forecasted surface wind directions, wind speed with azimuth variability, and ambient surface air temperature.

4.0 Contracted Services

A number of active contracts are maintained in order to ensure continuing access to qualified personnel when and if they are needed to supplement CP&L resources. These contracts provide the capability of obtaining, on an expedited basis, additional maintenance support personnel (such as mechanics, electricians, and I&C Technicians), other technical personnel (such as HP and Chemistry Technicians), and engineering and consulting services. For example, contracts are maintained with Westinghouse, Atlantic Group, and Raytheon Engineers (the NSSS vendor, constructor, and architect-engineer respectively for HNP). A contract is maintained with the Framatone Technologies for analysis of in-plant radioactive samples and one is maintained with Murray and Trettle Weather Services, Inc. which provides localized weather forecasts for the system or for HNP area as requested.

5.0 Industry Resource Support

American Nuclear Insurers (ANI) would assist the affected utility by managing the insurance claims generated by the public who may be affected by an offsite radiological event.

Nuclear Electric Insurance Limited (NEIL) would assist the affected utility in determining the damage to equipment on site and managing the insurance claims made by the utility for the loss of the generation of power due to an emergency.

One of INPO's roles is to assist the affected utility in applying the resources of the nuclear industry to meet the needs of the emergency. When notified of an emergency situation, INPO will provide emergency response in accordance with the INPO Emergency Response Plan at the request of the utility. Utility emergency response planning includes notification to INPO, via the emergency telephone number, of events classified Alert or higher.

ANNEX G

Interfacing Information From Supporting Emergency Plans

INPO is able to provide the following emergency support functions:

- Identifying sources of emergency manpower
- Dissemination of technical information concerning the incident to member utilities and participants
- Analysis of operational aspects of the incident.

To support these functions, INPO maintains the following emergency support capabilities:

- Dedicated emergency call number capable of reaching INPO staff and activating INPO support functions 24 hours per day.
- Designated INPO representative(s) who can be dispatched to the utility to coordinate INPO support activities and information flow.
- An Emergency Response Center available for operation 24 hours per day.

An INPO duty person will respond to the call, and the Emergency Response Center at INPO will be activated as necessary.

If requested by the utility or when deemed appropriate, one or more suitably qualified members of the INPO staff will report to the Emergency Response Manager and assist in coordinating INPO's response to the emergency, as follows:

- Staffing a position responsible to the appropriate utility manager as liaison for all INPO matters.
- Working with INPO personnel in Atlanta to coordinate requests for assistance, INPO response, and related communications.
- Assisting the utility as requested in the use of industry information systems (such as NETWORK) concerning accident status and related information of interest to other utilities.
- Ensuring that emergency information released by the INPO liaison is cleared through appropriate utility channels.

An INPO representative could be dispatched on approximately a four-hour notice. On-site activities, when undertaken, will be approved by the President of INPO and coordinated with the affected utility through the on-site INPO representative.

Carolina Power & Light Company is a signatory to the mutual assistance agreement developed by INPO for utilities in the nuclear industry.

ANNEX G

Interfacing Information From Supporting Emergency Plans

6.0 Local Services Support

HNP is equipped and staffed to cope with many types of emergency situations. However, if a fire, medical, or other type of incident occurs that requires outside assistance, such assistance is available as shown on Table 4.0-1.

6.1 Medical Assistance

Medical assistance is available through agreements with the following organizations as described in Section 4.6.3.7 of this plan. HNP agreements with the listed agencies are on file at CP&L. Annex A lists each agreement:

- Local area physicians
- Rex Hospital
- Wake Medical Center
- Western Wake Medical Center

6.2 Ambulance Service

HNP maintains a contract for support services with Apex Rescue Squad, Inc. as described in Section 4.6.3.6.4 of this plan. Annex A lists this agreement.

6.3 Fire Assistance

Agencies with fire protection resources in the vicinity of HNP are as follows:

- Apex Volunteer Fire Department
- Town of Holly Springs Dept of Public Safety Division of Municipal Fire Services
- Other Wake County Fire Departments

The Apex Volunteer Fire Department is the primary fire protection response agency for HNP and will coordinate assistance activities, through a County-wide Mutual Aid Agreement of the other area Fire Departments. HNP agreement with Holly Springs is on file at CP&L. Annex A lists each agreement.

7.0 General Public

Protective actions which should be taken by the general public will be provided by State and local government agencies. Carolina Power & Light Company will make recommendations to these government agencies as discussed in Section 4 of this Plan.

ANNEX G

Interfacing Information From Supporting Emergency Plans

7.1 Evacuation

In the event that evacuation of the plume exposure pathway EPZ is required, the evacuation routes shown in Annex H will be used by on-site personnel and the public.

The time required to evacuate personnel from the plume exposure pathway EPZ varies depending on whether a part of the EPZ is to be evacuated or all of it, or prevailing weather conditions, as provided in Table 1.8-2.

7.2 Shelter

The decision to evacuate or remain (in shelter) should be based on an evaluation of many factors including the protection afforded by dwellings, public fallout shelters, and other structures that might provide protection from surface deposited radionuclides and from a gamma cloud source in the plume exposure pathway EPZ.

The locations of public shelters are depicted in Annex H.

Table G.1-1

Organizations Participating in Emergency Response

Organization	Contact	Location for Response	Approximate Response Time	Agent for Initial Notification
HNP	Site Emergency Coordinator	Control Room	5 Minutes	Superintendent - Shift Operations
Corporate Communications	On-call Corporate Communications	14th floor, CPB	1-2 Hours	On-call Corp. Communications
Nuclear Regulatory Commission	1. Emergency Office (HQ) 2. Base Team Mg (Reg.)	NRC Ops. Ctr Incident Response Center	Immediate Immediate	HQ Duty Officer Regional Duty Officer
Nuclear Regulatory Comm. (Site Team)	1. Director-Site Team OPs 2. Interim Director	EOF, New Hill EOF, New Hill	5-8 Hours 60-75 Minutes	Dir. of Site Team OPs Resident Inspector
State Emergency Response Team	SERT Coordinator	Division Emer. Management Hqtrs, Raleigh	2 Hours	Highway Patrol Communications Center
Chatham County EOC	County Board Chairman	County Law Enforcement Center	1 1/4 - 2 Hours	County Communications Center, Pittsboro
Harnett County EOC	County Board Chairman	County Law Enforcement Bldg.	1 1/4 - 2 Hours	Sheriff's Department, Lillington
Lee County EOC	County Board Chairman	Sanford Municipal Center, Sanford	1-3 Hours	Lee County Sanford Municipal Center, Sanford
Wake County EOC	County Board Chairman	County Courthouse, Raleigh	1-2 Hours	Raleigh Comm. Center
Apex Rescue Squad	Captain	HNP	30-45 Minutes	Raleigh Comm. Center
Apex Volunteer Fire Department	Captain	HNP	20 minutes	Raleigh Comm. Center

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Table G.1-1

Organizations Participating in Emergency Response

Organization	Contact	Location for Response	Approximate Response Time	Agent for Initial Notification
Holly Springs Dept. Of Public Safety Division of Municipal Fire Services	Fire Chief	HNP	30-45 Minutes	Raleigh Comm. Center
Atlantic Group	Designated Staff	HNP	3-5 Hours	District Manager
Framatone Technologies	Designated Staff	Alliance Research Center, Indiana	24 hour	Lynchburg, VA
National Weather Service	Designated Staff	Raleigh, NC	phone contact	Raleigh, NC
Raytheon Engineers	Manager of Projects	HNP	8-16 Hours	Princeton, NJ
Rex Hospital	Emergency Room	Rex Hospital, Raleigh	30-45 Minutes	Rex Emergency Room or Raleigh Comm. Center
Wake Medical Center (WMC)	Emergency Room	WMC, Raleigh	30-45 Minutes	WMC Emergency Room or Raleigh Comm. Center
Western Wake Medical Center (WWMC)	Emergency Room	WWMC, Cary	20-30 Minutes	WWMC Emergency Room or Raleigh Comm. Center
Westinghouse	Emergency Response Director	Command Center Monroeville, PA	8-16 Hours	Regional Service Manager, Southern Service Region, Atlanta

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Figure G-1

<u>Emergency Response Organization Interfaces, TSC and EOF Not Activated</u>

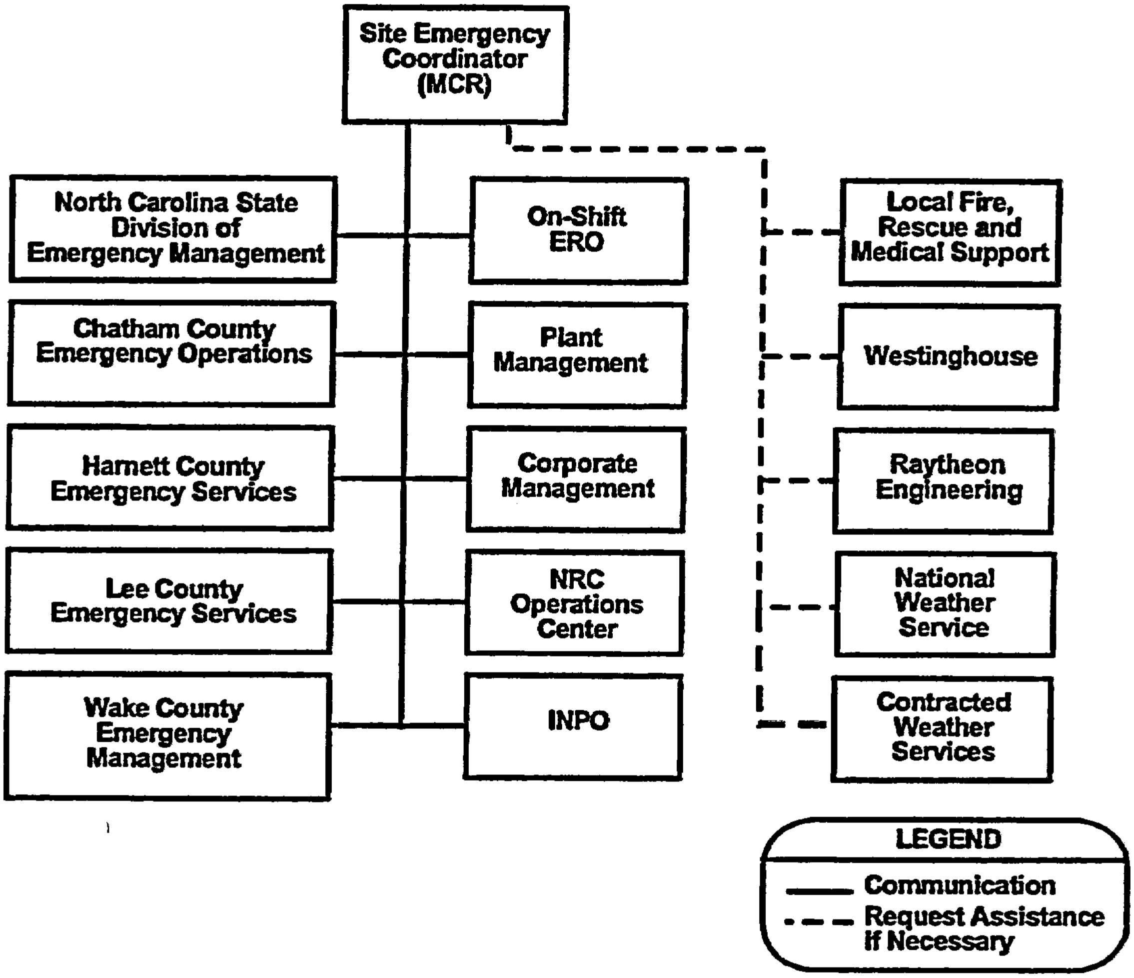

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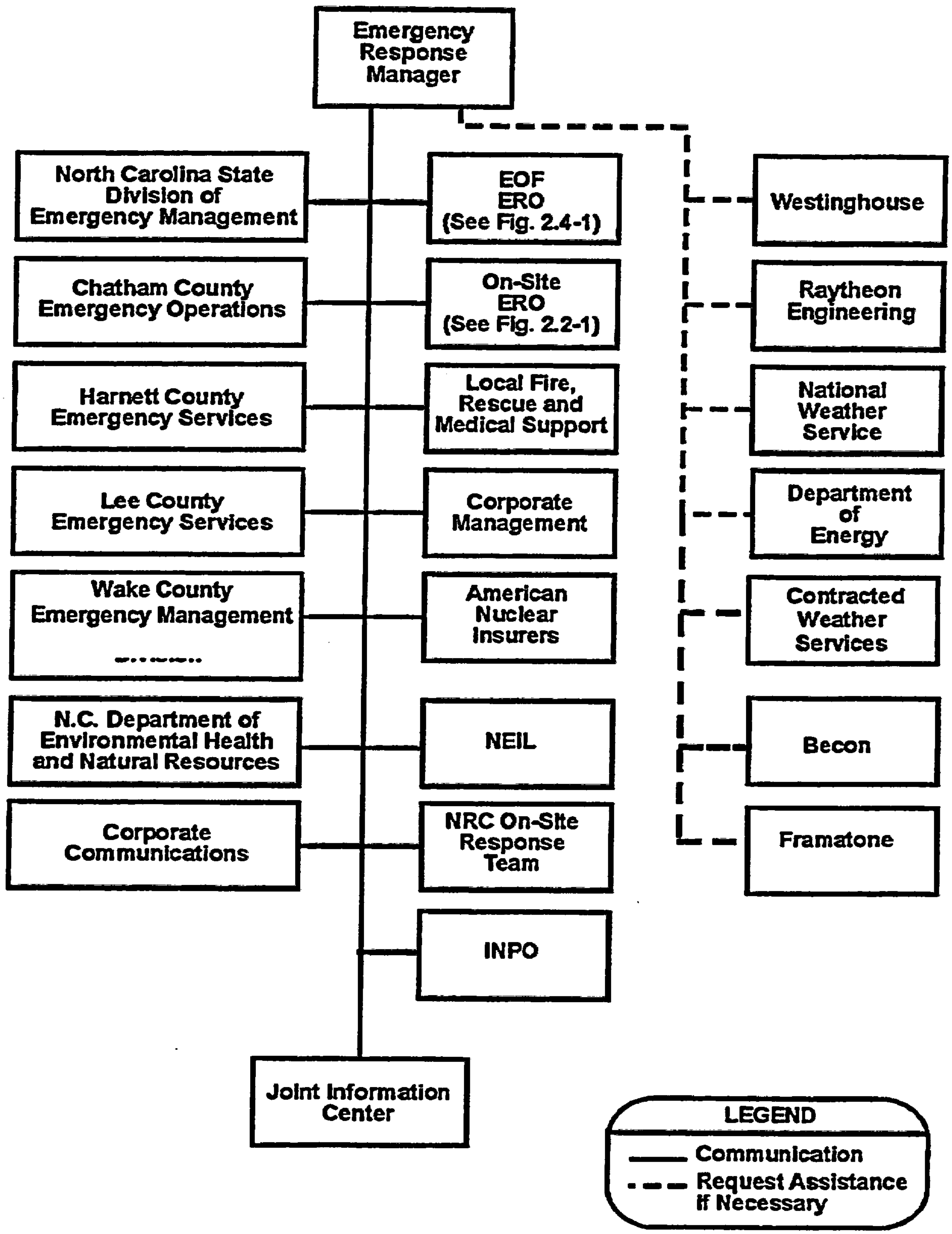
Figure G-2

Emergency Organization Interfaces, TSC and EOF Activated



Annex H

Harris Nuclear Plant - Operations Map

A folded copy of the "Harris Nuclear Plant - Operations Map", from Annex I of the "North Carolina Emergency Response Plan in Support of the Shearon Harris Nuclear Power Plant", is contained in the plastic sleeve following this page.

Revision Summary for PLP-201, Rev. 39

The primary reason for revision is to reflect the removal of SEC-MCR alternates. Previously, SROs were listed as assigned alternates, now only the SSOs are assigned to SEC-MCR. Additionally , corrections were made to Table 1.8-1 that came from a recent review of the ETE study. Updates to 10-mile EPZ schools were added, the limitation of use of only the NCDRP mobile lab was expanded to include both mobile and fixed facilities. Wording for the Weather Center was changed to reflect the NWS and contracted services. First Aid locations and supplies are referenced back to ORT-3002 and not itemized in the Emergency Plan. Wording for treatment, storage, etc. made more generalized for the procedure to govern. Off hours demonstration is no longer restricted to an exercise, it may be performed in a drill. Dr. Denton, an additional physician for the plan, has been removed from the Emergency Plan. PEP-400 has been deleted and EPM-100 has taken its place.

Page 14 Figure 1.5-3	Deleted reference to First Aid Room in Turbine Building
Page 15 Table 1.8-1	Column total for transient changed from 6951 to 4217 and the column total for special facility should be 11,674 not 1439.
Page 19	Changed Lufkin Road 6th Grade/Year Round and Lufkin Road 9th Grade Traditional to Lufkin Road Middle School and added Salem Elementary School
Page 23	Deleted the statement, "The assigned alternates are on-shift Licensed Senior Control Operators as designated in accordance with operations procedures."
Page 48	N.C. Department of Radiological Protection's mobile laboratory has been corrected to Division of Radiation Protection laboratories, both mobile and fixed facility. Changed the specific reference to the State's mobile lab to the more general reference of 'State's laboratories'. Removed an 'and' in Each lab [and] includes equipment for chemical analyses and for analysis of radioactivity.
Page 50 Table 3.1-1	Deleted the line referencing first aid kits in the facilities
Page 54 4.4.3.E.	Changed 'The Weather Center will be contacted as needed to forecast atmospheric conditions affecting the site.' To 'The National Weather Service and contracted weather sources will be contacted as needed to forecast

	atmospheric conditions affecting the site.'
Page 60 4.6.3.E	Clarified that personnel decontamination supplies are located in WPB 261' and deleted references to the specific locations and contents of First Aid Kits.
Page 80	Clarified statement concerning scenarios varying year to year to encompass drill scenarios and relocated statement from Section 5.3.2, Exercises, to Section 5.3.1, Drills.
Page 82 5.4.2.	Reworded reference to First Aid Room to a more general reference to First Aid Stations/Kits.
Page 91	Removed Dr. Robert Denton, M.D. as supporting the E-Plan. He has chosen not to continue his support of the Harris Plant.
Page 110	PEP-400 was deleted and EPM-100 replaced it

Shearon Harris Nuclear Power Plant
Unit No. 1

Individual Plant Examination
Submittal

CAROLINA POWER & LIGHT COMPANY

August 1993

Figure 3–32
Steam Generator PORV System

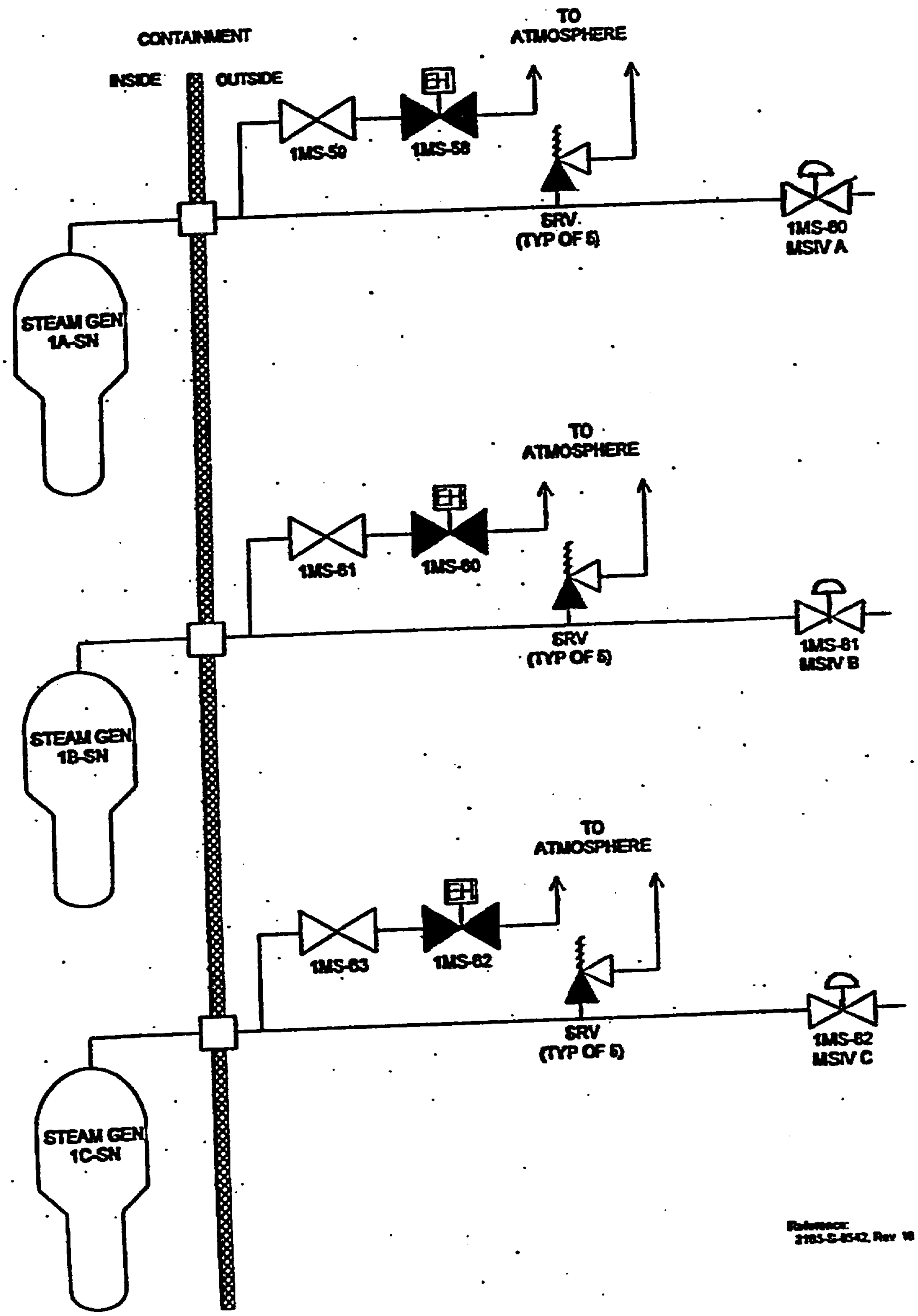


Table 4–1 (cont.)
SHNPP Design Information

Number of Safety Valves per steam generator	5
Lowest SV Setpoint, psia	1,185
High Pressure Injection System	
Number of Pumps	3
Capacity, gpm @ psia	650 @ 1340 (150 @ 2500)
Shutoff Head, psia	>2700
Refueling Water Storage Tank (minimum), gal	432,000
Low Pressure Injection System (RHR)	
Number of Pumps	2
Capacity, gpm @ psia	3,750 @ 104
Shutoff Head, psia	147
Accumulators	
Number of Accumulators	3
Pressure, psia	680
Water Capacity (Nominal), ft^3	1,012
Auxiliary Feedwater System	
Type Drive	Motor
Number of Pumps	2
Capacity, gpm	450
Type Drive	Turbine
Number of Pumps	1
Capacity, gpm	900
Containment (Large Dry)	
Inside Diameter, ft	130
Maximum Inside Height, ft	220
Free volume, ft^3	2.1×10^6
Design Leak Rate, %volume/day	0.1

Thompson Rept. Exh.: CP&L, 1995a
(excerpts)

Probabilistic Safety Assessment

Section 3

INITIATING EVENTS IDENTIFICATION

(Revision 0 prepared by SAROS, Inc.)

Carolina Power and Light Company

Shearon Harris Nuclear Power Plant

Revision 1

October 1995

Multiplying this value by one year (8760 hours) yields a value of 5.1E-4/yr. Given there are three valves of concern (1CC-219 1CC-230, 1CC-241 and 1CC-294) the frequency of relief valve LOCAs is 2.0E-3/yr. Given the operators have 10 minutes to respond and would receive indications such as low CCW surge tank level and low pump suction pressure, an operator action is included for the potential to isolate the non-essential header flow prior to pump failure. This is OPER-53, and is discussed in Appendix E.

Quantifying the fault tree model, the frequency of this initiating event is found to be 3.9E-3/yr.

3.2.8 Fire Initiating Events

An analysis of the impact of fires on SHNPP is documented in Appendix P. Six fire events, two control room fires and four switchgear room fires, were found to significantly impact core damage risk, in that the frequency of core damage exceeded a threshold of 1E-6/year. These six fire initiators are identified along with their frequency of occurrence and plant effects in Appendix P.

3.2.9 Seismic Events

No seismic PRA analysis has been conducted for SHNPP, due to low expected risk from these events. In order to include the contribution of seismic events in the overall analysis, a review of the seismic analysis for SHNPP completed to comply with Generic Letter 88-20.

The SHNPP licensing basis identifies an operating basis earthquake (OBE) as 0.075g and a safe shutdown earthquake (SSE) as 0.15g. A seismic margins study concluded that the plant is able to withstand at least a 0.30g event, the review level earthquake (RLE) without sustaining damage to the core due to equipment failures. Components required for safe shutdown were verified to be seismically ruged with exceptions identified for further evaluation. These evaluations (see Appendix P) identified the following:

- RHR heat exchangers acceptable up to 0.30g
- low voltage switchgear acceptable up to 0.35g
- other components with failures > 0.40g

Further, based on a paper by O'Hara et. al.[27], the low end of a plant's core damage median capacity is at least 2.67 times the SSE. For SHNPP, this corresponds to 0.4g. Using this information, a simplified seismic model can be developed with the following assumptions:

- Any earthquake > 0.4g directly causes core damage (failure of all PSA functions)
- Any earthquake > OBE fails nonsafety-related components and trips the plant
- Any earthquake > SSE fails components not specifically evaluated in the seismic margins analysis
- Any earthquake > 0.3g fails the RHR heat exchangers (and flowpaths)
- Any earthquake > 0.35g fails low voltage switchgear

Two seismic events are defined. First, any earthquake with a magnitude between 0.075g and 0.4g (i.e., above the OBE but less than the core damage median capacity earthquake) is T23. Second, above 0.4g is T24. For equipment which fails at an intermediate level, conditional probabilities were developed to account for the conditional probability of seismic events above the threshold for component failure.

Earthquake frequency data was taken form EPRI–NP–6395–D[28] for the SHNPP site. A log curve was fitted to the data and used to estimate the frequency of various magnitude earthquakes, as shown in Figure 3–8.

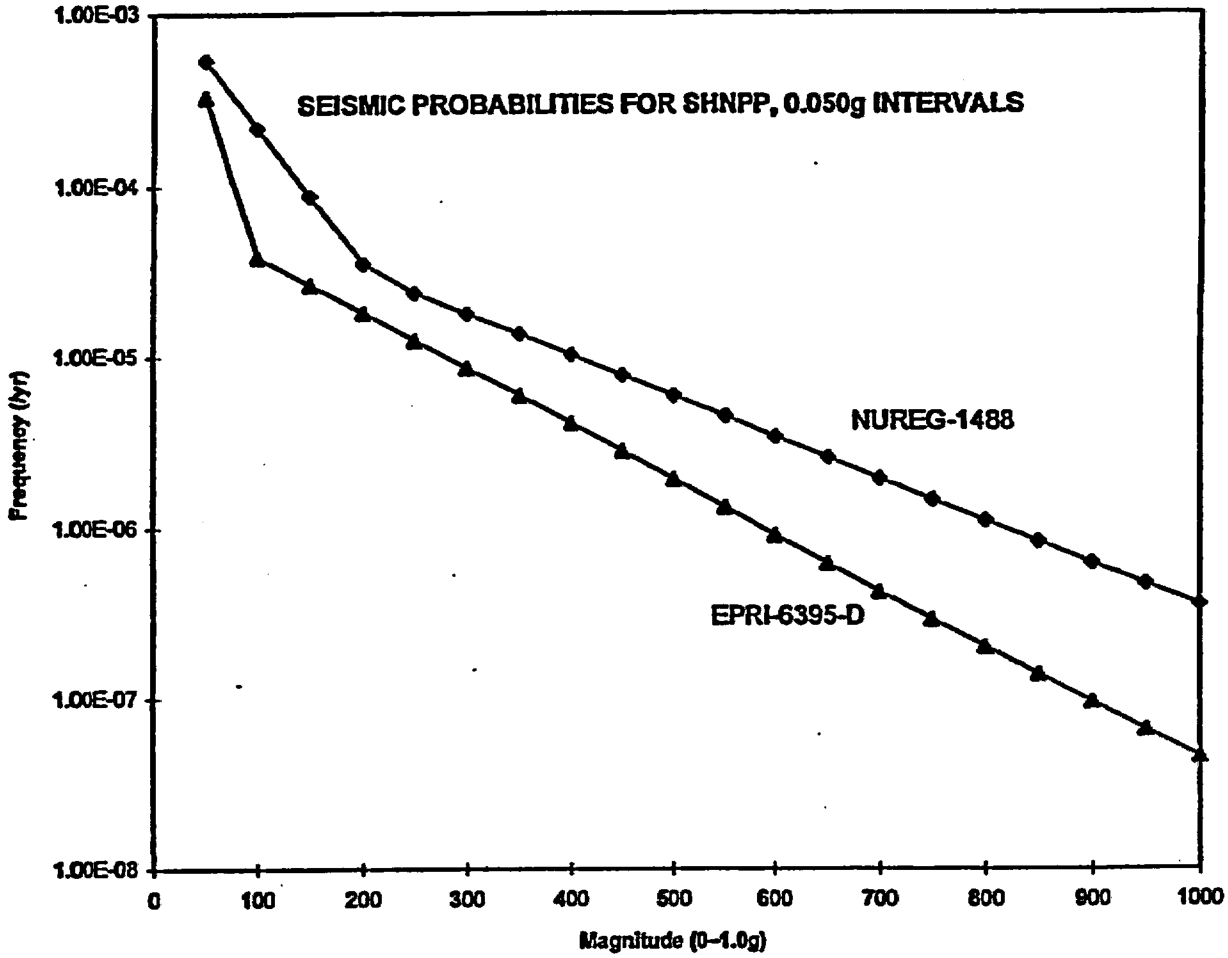


Figure 3–8 Seismic Event Frequencies for SHNPP

There were three calculations made to arrive at the initiating event frequencies and the conditional probabilities. First, the frequency of T24, a seismic event > 0.4g, was taken directly from the graph as 4.3E–6/yr. Next, the frequency of T23, a seismic event > 0.075g but less than 0.4 g was found by subtracting the frequency of T24 from the frequency of seismic events > 0.075g, or 4.7E–5/yr – 4.3E–6/yr, giving 4.3E–5/yr. Conditional probabilites were calculated by determining the frequency of a seismic event between the failure magnitude and 0.4g, then dividing by the frequency of the OBE. Three conditional probabilities were required:

• X–SEISSSE (Seismic event > SSE, 0.15g)	0.53
• X–SEISRLE (Seismic event > 0.3g)	0.11
• X–SEISLVS (Seismic event > 0.35g)	0.045

These conditional probabilities are placed in the system fault trees along with the T23 initiator to propagate seismic failures of equipment.

An effort to estimate uncertainty in the seismic data yielded very high range factors. Therefore, 10.0 was chosen as the nominal uncertainty (i.e., order of magnitude).

3.2.10 Initiating Events and Event Frequencies

Based on the above assessment, the transient initiating events and associated event frequencies for the SHNPP PSA were developed. Table 3–17 presents the transient initiating event listing.

Table 3–17
SHNPP Transient Initiating Event Frequencies of Occurrence

Identifier	Transient Initiating Event	Frequency (/yr)	Range Factor
T_1	Reactor trip	0.65	2.3
T_2	Reactor trip with RCS pressure challenge	6.9E–2	5.0
T_3	Turbine trip	0.65	2.3
T_4	Loss of main feedwater	0.39	3.1
T_{4P}	Partial loss of main feedwater	0.65	2.3
T_5	Loss of offsite power	3.1E–2	5.6
T_6	Secondary line break between the MFW isolation valves and the MSIVs	2.0E–2	5.0
T_7	Reactor trip with safety injection	7.4E–2	5.0
T_8	Inadvertent containment phase B isolation	2.8E–2	5.0
T_9	Loss of normal service water	5.1E–2[2]	N/A[3]
T_{10}	Loss of NSW return valves	9.6E–5[2]	N/A[3]
T_{11}	Loss of component cooling water	3.9E–3[2]	N/A[3]
T_{12A}	Loss of 6.9 kV bus 1A–SA	2.0E–3[2]	N/A[3]
T_{12B}	Loss of 6.9 kV bus 1B–SB	2.0E–3[2]	N/A[3]
T_{13}	Loss of instrument air	0.13	3.1
T_{14A}	Loss of DC bus DP–1A–SA	2.9E–3[2]	N/A[3]
T_{14B}	Loss of DC bus DP–1B–SB	2.9E–3[2]	N/A[3]
T_{15}	Loss of DC non-safety bus DP–1A	2.9E–3[2]	N/A[3]
T_{16}	Loss of CVCS/SI	8.9E–3[2]	N/A[3]
T_{17}	Fire—6.9 kV Bus 1A–SA	7.3E–4	Note 4
T_{18}	Fire—6.9 kV Bus 1B–SB	7.3E–4	Note 4
T_{19}	Fire—6.9 kV Bus 1A–SA unsuppressed	5.2E–4	Note 4
T_{20}	Fire—6.9 kV Bus 1B–SB unsuppressed	5.2E–4	Note 4
T_{21}	Fire—Control Room Isolation and Annunciation Cabinets	5.6E–5	Note 4
T_{22}	Fire—Control Room requiring ACP shutdown	2.0E–4	Note 4
T_{23}	Safe Shutdown Earthquake (0.15g)	4.3E–5	10.0[1]
T_{24}	Limiting Earthquake (0.4g)	4.3E–6	10.0[1]

[1] Estimated range factor.
[2] Frequencies presented for information only, fault tree model is imbedded in overall model.
[3] Uncertainty of individual components propagated in model calculation.
[4] Uncertainties for fire initiating events has not been evaluated.

Probabilistic Safety Assessment

Section 6

CORE DAMAGE RESULTS ASSESSMENT

Carolina Power and Light Company

Shearon Harris Nuclear Power Plant

Revision 1

October 1995

6.1.1 Sequence T–Q–U–B CDF = 1.69×10^{-5} per year (30.92% of total)

This sequence represents a transient-induced LOCA with a failure of high pressure ECCS and a failure of secondary-side heat removal. Seismic events, which are of sufficient magnitude to fail the safety-related charging/safety injection pumps and the component cooling water pumps and thus interrupt reactor coolant pump (RCP) seal cooling, are placed into this category, and account for more than 40% of the total of this sequence. Internal flooding scenarios, which also fail RCP seal cooling, are another significant contributor, at just under 30% of the total. Another 17% of the total can be attributed to fires in either the main control room or the switchgear rooms.

The remaining cutsets are typically a loss of nonsafety-related DC power, which disables the automatic fast bus transfer of offsite power from the unit auxiliary transformers to the startup transformers. This results in a demand for the emergency diesel generators. If the diesels fail, then offsite power may be realigned manually. Without restoration of power, a loss of RCP seal cooling occurs, leading to seal leakage; the mitigating systems for a small LOCA are unavailable due to the power failure, and secondary-side heat removal fails once the 1B–SB battery is depleted, failing the turbine-driven AFW pump.

6.1.2 Sequence T–B–X CDF = 1.02×10^{-5} (18.71%)

This sequence represents a transient with secondary-side heat removal failed. Feed-and-bleed cooling is established but fails at or during recirculation. More than 60% of these sequences involve train B AC or DC power failures, with subsequent failure of the A train AFW and ECCS equipment. Train B power failures are more significant than train A since these also fail the turbine-driven AFW pump, which relies upon train B DC power for control power. Loss of main feedwater as an initiator is also significant, representing about 20% of these sequences. Switchgear room fires also contribute to this sequence.

The dominant cause of failure of high pressure ECCS during recirculation is from the MOVs in the CSIP suction from the RHR pumps. Failure of the containment recirculation sump valves are also significant.

6.1.3 Sequence T–B–U CDF = 9.26×10^{-6} (16.91%)

This sequence represents a transient with secondary-side heat removal failed. Feed-and-bleed cooling is not established, and core damage occurs.

Nearly a third of this sequence is due to a control room fire with failure to achieve shutdown remotely. Loss of 6.9 kV bus 1B–SB and loss of main feedwater are dominant initiators.

High pressure ECCS failures are dominated by failures in the CSIP suction from the RWST, due to MOV failures. The testing procedures for the MOVs do not independently verify flow through each of the two parallel valves, so the failure probability is high.

About 17% of the sequences are due to human error, either failure to initiate feed-and-bleed cooling, or failure to manually actuate AFW or main feedwater.

6.1.4 Sequence SBO (Station Blackout) CDF = 7.89 x 10^{-6} (14.41%)

This sequence represents a loss of offsite power with subsequent failure of the emergency diesel generators and failure to restore offsite power. An RCP seal LOCA will occur due to loss of both seal injection and thermal barrier heat exchanger cooling from CCW.

Early sequences, involving start failures of the diesels or their support equipment, account for more than 60% of these sequences. Longer term failures are less important due to the likelihood of restoring offsite power.

No one failure mode is dominant; however, failure of load shed to occur on the safety busses is significant, as well as diesel generator HVAC failures. Fuel oil system failures account for about 15% of the sequence total; these sequences are recoverable by establishing an alternate source of fuel oil. However, no credit is taken for this recovery.

6.1.5 Sequence T–K–M–C CDF = 3.34 x 10^{-6} (6.09%)

This sequence represents an ATWS sequence with loss of main feedwater and inadequate Doppler reactivity feedback to allow control of the RCS pressure transient (occurs in first 15% of cycle). RCS pressure increases to 3200 psig, at which point vessel failure occurs, and ECCS is assumed to be unavailable due to damage to the ECCS check valves from the high pressure condition.

Common-cause failure of the reactor trip breakers is the dominant cause of the ATWS, accounting for 92% of all ATWS events. The remaining 8% are caused by mechanical binding of the control rods. This low probability event disables all reactivity control with control rods, requiring functioning of emergency boration systems.

Transients which cause a loss of main feedwater included turbine trip, loss of feedwater, loss of instrument air, and spurious SI.

Probabilistic Safety Assessment

Section 9

SOURCE TERMS AND RELEASE CATEGORIES

(Revision 0 prepared by SAROS, Inc.)

Carolina Power and Light Company

Shearon Harris Nuclear Power Plant

Revision 1

October 1995

Table 9–4
Release Category Source Terms

Release Category	Source Terms												
	Nobles %	CsI %	TeO_2 %	SrO %	MoO_2 %	CsOH %	BaO %	La_2O_3 %	CeO_2 %	Sb %	Te_2 %	UO_2,ACT %
RC–1	100	2.72	0.0	2.4×10^{-7}	3.2×10^{-6}	2.50	2.0×10^{-6}	2.0×10^{-5}	2.0×10^{-5}	0.31	8.1×10^{-4}	Note 2
RC–1A	100	4.0	0.0	1.8×10^{-7}	1.8×10^{-6}	3.74	2.1×10^{-6}	1.6×10^{-5}	1.6×10^{-5}	0.087	8.7×10^{-4}	Note 2
RC–1B[1]	100	2.72	0.0	2.3×10^{-4}	3.2×10^{-6}	2.50	3.0×10^{-4}	1.5×10^{-4}	2.3×10^{-4}	0.31	0.718	Note 2
RC–1BA	100	4.0	0.0	6.2×10^{-4}	0.077	2.50	0.002	2.7×10^{-5}	3.8×10^{-5}	0.046	0.31	Note 2
RC–2	100	0.47	0.0	1.6×10^{-4}	0.0018	0.39	0.0035	0.013	0.018	0.15	0.0	0.0
RC–2B	100	4.1	0.0	0.250	0.0073	4.56	0.118	.341	.462	3.40	12.42	0.0
RC–3	100	0.067	0.0	3.0×10^{-5}	2.2×10^{-4}	0.069	2.6×10^{-4}	6.8×10^{-4}	6.8×10^{-4}	0.006	3.2×10^{-5}	Note 2
RC–3B	100	1.82	0.0	9.9×10^{-4}	0.021	1.32	0.031	0.26	0.26	0.33	0.006	Note 2
RC–4	100	0.22	0.0	4.7×10^{-5}	2.3×10^{-4}	0.176	2.6×10^{-4}	1.0×10^{-5}	1.0×10^{-5}	0.013	0.0	0.0
RC–4C	100	3.7	0.0	8.35×10^{-4}	0.0047	3.39	0.005	0.0013	0.0013	0.50	0.0	0.0
RC–5	100	59.05	0.0	0.011	0.28	52.6	0.067	9.2×10^{-5}	9.6×10^{-5}	3.3	0.009	Note 2
RC–5C	100	59.05	0.0	0.011	0.28	52.6	0.067	9.2×10^{-5}	9.6×10^{-5}	3.3	0.009	Note 2
RC–6	100	0.0014	0.0	6.4×10^{-5}	0.0078	0.019	0.0002	2.7×10^{-6}	3.8×10^{-6}	0.0046	0.031	Note 2
RC–7	100	0.014	0.0	6.4×10^{-4}	0.078	0.191	0.002	2.7×10^{-5}	3.8×10^{-5}	0.046	0.31	Note 2

1 Some values (volatile source terms) taken from the sequence for release category 1 since it predicts a higher release.

2 This release fraction is smaller than 1.0×10^{-8}%.

Harris Nuclear Plant Probabilistic Safety Assessment

Appendix L—Summary Document

Abstract

The purpose of this document is to provide an overview of the technology of probabilistic safety assessment (PSA), and to summarize the details, results and potential applications of the Harris Nuclear Plant (HNP) PSA model. The current model is based on the plant configuration as of June 1997, and includes both a level 1 (core damage end state) and a level 2 (fission product release end state) analysis. The original analysis was submitted to the Nuclear Regulatory Commission (NRC) on August 20, 1993, to meet the requirements of Generic Letter 88-20 for an individual plant examination (IPE). Additional analyses were submitted to the NRC on June 30, 1995, to meet the requirements of Generic Letter 88-20 Supplement 4 for an IPE of external events (IPEEE). The complete PSA report is voluminous, addressing details which are likely to be of interest only to analysts and reviewers. This document provides a concise summary of the important features and conclusions of the HNP PSA analysis, and provides to those who use the results of the PSA a general understanding of the process and its applications.

October 1998

Prepared By: Concurrence via telecon Steven A. Laur for
Steven L. Mabe

Reviewed By: Steven A. Laur
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Approved By: Steven A. Laur 10-5-98
Steven A. Laur

Probabilistic Safety Assessment Unit
Nuclear Engineering Department

VIII. Applications of PSA

Although development of a PSA was used to satisfy Generic Letter 88–20, there are benefits which can be obtained beyond simply satisfying a regulatory requirement. By applying the knowledge and insights gained from the HNP PSA to the various activities supporting plant operation, CP&L would be able to enhance plant safety, reduce operating and maintenance expenses and increase capacity factors. These could be accomplished using some of the applications discussed in this section.

A criticism of the use of PSA in decision making is the presence of uncertainties in the model. There are two important points relevant to this criticism. First, while uncertainties may shade the exact meaning of the absolute CDF calculated, the relative risk calculated for different accident sequences, or for different plant systems and components, represent the true value of the PSA, and these are less impacted by the existing uncertainties. Second, the alternative to the use of PSA is to base decisions on each individual's assessment of the situation, which can vary considerably depending upon the person's background and biases. PSA provides a logical structural basis for decisions, and provides common ground for discussions relevant to plant safety.

The NRC has recognized the value of PSA by issuing policy endorsing the use of PSA in various applications. The NRC has demonstrated its willingness to approve licensing actions which blend the probabilistic analyses of PSA with deterministic (licensing basis) evaluations and engineering judgment. The NRC has also promulgated rules and regulations which are in part risk based, such as maintenance rule and RG 1.174. The end result of the NRC use of PSA will be to place the responsibility of ensuring public health and safety more in the control of its licensees, allowing utility resources to be focused on real safety issues. It also places a burden on plant management and personnel to acquire an understanding of PSA fundamentals and of the plant-specific PSA results and insights applicable to their facilities.

The Institute of Nuclear Power Operations (INPO) has also recognized the potential application of PSA and has recommended that plant management and key personnel who will make use of PSA in the decision making process should be properly trained to understand the technology and its uses. They note that PSA is one of many inputs in the plant management processes, and that to properly make use of the information one must understand the model limitations and uncertainties. They further observe that PSA provides insights into severe accident risk, and that other aspects of nuclear safety must also be considered.

The Nuclear Energy Institute (NEI, formerly NUMARC), with involvement of the NRC, has developed and continues to develop generic guidance on PSA. The Westinghouse Owners' Group is also involved in PSA applications.

Harris Nuclear Plant
Probabilistic Safety Assessment

HNP

Appendix A.12
Safety-Related DC Power System
(D)

CP&L

61864

flow control valves will fail open, and the governor will fail "fully open", resulting in an overspeed trip of the pump.

The battery depletion time for the vital 125 VDC batteries as modeled in the PSA is taken to be 4 hours.

Depletion of the vital batteries is modeled in the DC power system fault tree by flag events. The timing of battery depletion and shedding of DC loads in order to prolong DC operation are considered in the study as a sensitivity issue and as a potential recovery action, respectively.

- *Effect of a loss of DC power on operation of switchyard breakers*

The discussion of subtle interactions provided in NUREG/CR–4550, Volume 1 describes an incident at a plant at which vital loads are normally powered from the unit transformer, and control power for bus switching operations is supplied from the vital DC system. This configuration leads to the possibility that loss of a vital DC bus could trip the plant and the unit generator, and at the same time impair the ability to control the breakers which permit switchover to the offsite power supply.

Although at SHNPP AC power is normally supplied by the unit transformers, similar to the referenced plant, control power to the 230 kV switchyard breakers is supplied from two switchyard DC systems, each consisting of a 125 VDC battery and battery charger. These systems are independent of the safety-related DC distribution system, and therefore the potential for a similar subtle interaction to occur does not exist at the SHNPP.

A.12.7.3 System Level Insights

The important system insights gained from fault tree analysis of the DC power system are as follows:

- Control power to the turbine driven AFW pump is supplied by vital DC bus DP–1B–SB, which is in turn supplied by vital motor control centers 1B21–SB and 1B31–SB, and vital battery 1B–SB. In the event of a loss of all AC power (station blackout) the only source of DC control power to the turbine driven pump would be vital battery 1B–SB. In the event of a prolonged station blackout, the battery would be depleted and control power to the turbine-driven pump lost. With control power unavailable, the pump governor would fail "full open," resulting in an overspeed trip of the pump. A discussion on battery depletion is included in section A.12.7.2.

A.12.7.4 Sensitivity Issues

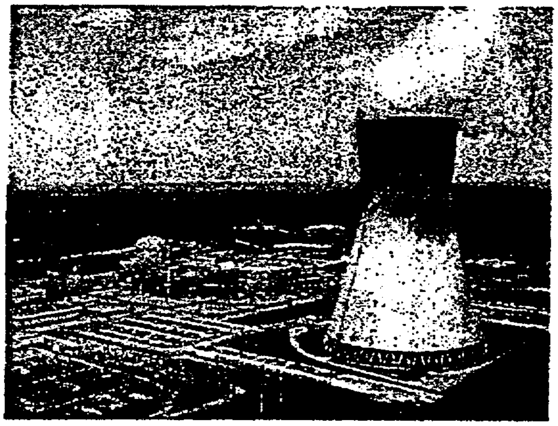
Aspects of the DC power model which may be especially sensitive to changes in assumptions, equipment failure rate data, or operator intervention are:

- The timing of battery depletion. Currently it is assumed that the vital batteries will last for 4 hours with no load shedding. Experience at other plants is that the batteries may not last for their rated duration. Conditions related to temperature, water level in the batteries, and load drains on the battery may all affect battery performance. The assumptions of battery life stated above is one area for sensitivity analysis.

To prolong DC power operation the operators may choose to shed loads from the DC power buses. If loss of DC power is an important contributor, the use of load shedding to prolong battery life may be a consideration and could be evaluated as a sensitivity issue.

Carolina Power and Light Company RSC 99-12

Shearon Harris Nuclear Plant Probabilistic Safety Assessment



Assessment of Harris Fuel Handling Building Operations

Revision 0
May 1999

Principal Analyst
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Presented to:
Carolina Power and Light Company
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80178

A cask drop during the move from the decontamination enclosure to the cask loading pool due to a failure of the cask links or improper attachment or a failure of the 150-ton crane during the cask lift would result in the potential for cask failure and a loss of boundary control. However, the cask is capable of withstanding such a drop without failure (Reference 4). However, all but four bolts are detensioned prior to movement. Analysis indicates that it is possible that the cask seal may burp upon impact, there is a slight potential for a small release to the fuel handling building (Reference 4). The release would be far less than allowed by 10 CFR 100 does limits, however the analysis of the event is included in the analysis since it is a function of the number of fuel shipments.

Passing a spent fuel cask over a fuel pool is prevented by physical stops located on the crane and by electrical interlocks on crane movement. Therefore, the current physical limitations preclude an accidental movement of a cask over any fuel storage pool.. However, if the interlocks were to fail, it is possible that the cask could pass over the fuel pool "D", but this would also require an operator error to follow movement procedures. The combination of component and operator failure is considered very remote. However, the potential for it to occur in combination with a cask drop is retained in the analysis due to the potential of the cask to strike the fuel stored in the pool and result in multiple fuel element failure and associated radionuclide release. The release would occur underwater and cooling would most likely be retained. However, the magnitude of the plenum release is sufficient to warrant further consideration.

Another consideration is the potential for a single fuel element to be stuck in the cask and failing when the fuel handling bridge crane attempts a lift. Procedural guidance is designed to limit this occurrence, but it cannot be ruled out. The release would be from a single fuel element and would occur underwater. These two factors would greatly reduce the significance of any release, however the event is retained due to the number of lifts having an impact on the frequency of release.

Once lifted, the fuel element is transported down the transfer canal to its storage location by the fuel handling bridge crane. During this time, the fuel element could drop due to either a crane failure or a failure of the latching mechanism. The drop could fail the fuel element or it could occur while over a spent fuel rack and potentially fail additional fuel elements. Analysis indicates that the drop of a fuel element would not result in a failure of the fuel pool liner.[5]

The dimensions of the fuel storage racks are not the same for BWR and PWR fuel elements. The BWR rack dimensions are smaller for each fuel assembly. If the operators attempted to place a PWR fuel element into a BWR rack, there is some small potential for fuel element damage if the fuel element were to be compressed and fail. The design of the fuel handling bridge crane, however, includes an automatic stop on slack cable. Therefore, the postulated event could not occur and it is excluded from the analysis.

Once all fuel elements are offloaded, the cask must be moved back to the railcar for transport. This again provides an opportunity for a combination of electrical interlock failure and an operator error combined with a crane failure resulting in a drop of the cask into fuel storage pool

Thompson Rept. Exh.: EPA, 1991 (excerpt)

PB92-164763

MANUAL OF PROTECTIVE ACTION GUIDES

AND

PROTECTIVE ACTIONS

FOR NUCLEAR INCIDENTS

Office of Radiation Programs
United States Environmental Protection Agency
Washington, DC 20460

Revised 1991

CHAPTER 2

Protective Action Guides for the Early Phase of an Atmospheric Release

2.1 Introduction

Rapid action may be needed to protect members of the public during an incident involving a large release of radioactive materials to the atmosphere. This chapter identifies the levels of exposure to radiation at which such prompt protective action should be initiated. These are set forth as Protective Action Guides (PAGs) for the general population. Guidance for limiting exposure of workers during such an incident is also provided. This guidance applies to any type of nuclear accident or other incident (except nuclear war) that can result in exposure of the public to an airborne release of radioactive materials.

In the case of an airborne release the principal relevant protective actions are evacuation or sheltering. These may be supplemented by additional actions such as washing and changing clothing or by using stable iodine to partially block uptake of radioiodine by the thyroid.

The former Federal Radiation Council (FRC), in a series of recommendations issued in the 1960's, introduced the concept of PAGs and issued guides for avoidance of exposure due to ingestion of strontium-89, strontium-90, cesium-137, and iodine-131. Those guides were developed for the case of worldwide atmospheric fallout from weapons testing, and are appropriate for application to intake due to long term contamination from such atmospheric releases. That is, they were not developed for protective actions relevant to prompt exposure to an airborne release from a fixed facility. The guidance in this chapter thus does not supersede this previous FRC guidance, but provides new guidance for different exposure pathways and situations.

2.1.1 Applicability

These PAGs are expected to be used for planning purposes: for example, to develop radiological emergency response plans and to exercise those plans. They provide guidance for response decisions and should not be regarded as dose limits. During a real incident, because of characteristics of the incident and local conditions that cannot be anticipated, professional judgment will be required in their application. Situations could occur, for example, in which a nuclear incident happens when environmental conditions or other constraints make evacuation impracticable. In these situations, sheltering may be the protective action of choice, even at projected doses above the PAG for evacuation. Conversely, in some cases evacuation may be useful at projected doses below the PAGs. Each case will require judgments by those responsible for decisions on protective actions at the time of an incident.

The PAGs are intended for general use to protect all of the individuals in an exposed population. To avoid social and family disruption and the complexity of implementing different PAGs for different groups under emergency conditions, the PAGs should be applied equally to most members of the population. However, there are some population groups that are at markedly different levels of risk from some protective actions -- particularly evacuation. Evacuation at higher values is appropriate for a few groups for whom the risk associated with evacuation is exceptionally high (e.g., the infirm who are not readily mobile), and the PAGs provide for this.

Some incidents may occur under circumstances in which protective actions cannot be implemented prior to a release (e.g., transportation incidents). Other incidents may involve only slow, small releases over an extended period, so that the urgency is reduced and protective action may be more appropriately treated as relocation (see Chapter 4) than as evacuation. Careful judgment will be needed to decide whether or not to apply these PAGs for the early phase under such circumstances.

The PAGs do not imply an acceptable level of risk for normal (nonemergency) conditions. PAGs also do not represent the boundary between safe and unsafe conditions; rather, they are the approximate levels at which the associated protective actions are justified. Furthermore, under emergency conditions, in addition to the protective actions specifically identified, any other reasonable measures available should be taken to reduce radiation exposure of the general public and of emergency workers. These PAGs are not intended for use as criteria for the ingestion of contaminated food or water, for relocation, or for return to an area contaminated by radioactivity. Separate guidance is provided for these situations in Chapters 3 and 4.

2.1.2 Emergency Planning Zones and the PAGs

For the purpose of identifying the size of the planning area needed to establish and test radiological emergency response plans, emergency planning zones (EPZs) are typically specified around nuclear facilities. There has been some confusion among emergency planners between these EPZs and the areas potentially affected by protective actions. It is not appropriate to use the maximum distance where a PAG might be exceeded as the basis for establishing the boundary of the EPZ for a facility. For example, the choice of EPZs for commercial nuclear power facilities has been based, primarily, on consideration of the area needed to assure an adequate planning basis for local response functions and the area in which acute health effects could occur.[1] These considerations will also be appropriate for use in selecting EPZs for most other nuclear facilities. However, since it will usually not be necessary to have offsite planning if PAGs cannot be exceeded offsite, EPZs need not be established for such cases.

2.1.3 Incident Phase

The period addressed by this chapter is denoted the "early phase." This is somewhat arbitrarily defined as the period beginning at the projected (or actual) initiation of a release and extending to a few days later, when deposition of airborne materials has ceased and enough information has become available to permit reliable decisions about the need for longer term protection. During the early phase of an incident doses may accrue both from airborne and from deposited radioactive materials. Since the dose to persons who are not evacuated will continue until relocation can be implemented (if it is necessary), it is appropriate to include in the early phase the total dose that will be received prior to such relocation. For the purpose of planning, it will usually be convenient to assume that the early phase will last for four days -- that is, that the duration of the primary release is less than four days, and that exposure to deposited materials after four days can be addressed through other protective actions, such as relocation, if this is warranted. (Because of the unique characteristics of some facilities or situations, different time periods may be more appropriate for planning purposes, with corresponding modification of the dose conversion factors cited in Chapter 5.)

2.2 Exposure Pathways

The PAGs for members of the public specified in this chapter refer only to doses incurred during the early phase. These may include external gamma dose and beta dose to the skin from direct exposure to airborne materials and from deposited materials, and the committed dose to internal organs from inhalation of radioactive material. Exposure pathways that make only a small contribution (e.g., less than about 10 percent) to the dose incurred in the early phase need not be considered. Inhalation of resuspended particulate materials will, for example, generally fall into this category.

Individuals exposed to a plume may also be exposed to deposited material over longer periods of time via ingestion, direct external exposure, and inhalation pathways. Because it is

[1]The development of EPZs for nuclear power facilities is discussed in the 1978 NRC/EPA document "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" NUREG-0396. EPZs for these facilities have typically been chosen to have a radius of approximately 10 miles for planning evacuation and sheltering and a radius of approximately 50 miles for planning protection from ingestion of contaminated foods.

usually not practicable, at the time of an incident, to project these long-term doses and because different protective actions may be appropriate, these doses are not included in the dose specified in the PAGs for the early phase. Such doses are addressed by the PAGs for the intermediate phase (see Chapters 3 and 4).

The first exposure pathway from an accidental airborne release of radioactive material will often be direct exposure to an overhead plume of radioactive material carried by winds. The detailed content of such a plume will depend on the source involved and conditions of the incident. For example, in the case of an incident at a nuclear power reactor, it would most commonly contain radioactive noble gases, but may also contain radioiodines and radioactive particulate materials. Many of the these materials emit gamma radiation which can expose people nearby, as the plume passes. In the case of some other types of incidents, particularly those involving releases of alpha emitting particulate materials, direct exposure to gamma radiation is not likely to be the most important pathway.

A second exposure pathway occurs when people are directly immersed in a radioactive plume, in which case radioactive material is inhaled (and the skin and clothes may also become contaminated), e.g., when particulate materials or radioiodines are present. When this occurs, internal body organs as well as the skin may be exposed. Although exposure from materials deposited on the skin and clothing could be significant, generally it will be less important than that from radioactive material taken into the body through inhalation. This is especially true if early protective actions include washing exposed skin and changing clothing. Inhaled radioactive particulate materials, depending on their solubility in body fluids, may remain in the lungs or move via the bloodstream to other organs, prior to elimination from the body. Some radionuclides, once in the bloodstream, are concentrated in a single body organ, with only small amounts going to other organs. For example, if radioiodines are inhaled, a significant fraction moves rapidly through the bloodstream to the thyroid gland.

As the passage of a radioactive plume containing particulate material and/or radioiodine progresses, some of these materials will deposit onto the ground and other surfaces and create a third exposure pathway. People present after the plume has passed will receive exposure from gamma and beta radiation emitted from these deposited materials. If large quantities of radioiodines or gamma-emitting particulate materials are contained in a release, this exposure pathway, over a long period, can be more significant than direct exposure to gamma radiation from the passing plume.

2.3 The Protective Action Guides

The PAGs for response during the early phase of an incident are summarized in Table 2-1. The PAG for evacuation (or, as an alternative in certain cases, sheltering) is expressed in terms of the projected sum of the effective dose equivalent from external radiation and the committed effective dose equivalent incurred from inhalation of radioactive materials from exposure and intake during the early phase. (Further references to dose to members of the public in this Chapter refer to this definition, unless otherwise specified.) Supplementary guides are specified in terms of committed dose equivalent to the thyroid and dose equivalent to the skin. The PAG for the administration of stable iodine is specified in terms of the committed dose equivalent to the thyroid from radioiodine. This more complete guidance updates and replaces previous values, expressed in terms of whole-body dose equivalent from external gamma exposure and thyroid dose equivalent from inhalation of radioactive iodines, that were recommended in the 1980 edition of this document.

2.3.1 Evacuation and Sheltering

The basis for the PAGs is given in Appendix C. In summary, this analysis indicates that evacuation of the public will usually be justified when the projected dose to an individual is one rem. This conclusion is based primarily on EPA's judgment concerning acceptable levels of risk of effects on public health from radiation exposure in an emergency situation. The analysis also shows that, at this radiation dose, the risk avoided is usually much greater than the risk from evacuation itself. However, EPA recognizes the uncertainties associated with quantifying risks associated with these levels of radiation exposure, as well as the variability of risks associated with evacuation under differing conditions.

Some judgment will be necessary when considering the types of protective actions to be implemented and at what levels in an emergency situation. Although the PAG is expressed as a range of 1-5 rem, it is emphasized that, under normal conditions, evacuation of members of the general population should be initiated for most incidents at a projected dose of 1 rem. (It should be recognized that doses to some individuals may exceed 1 rem, even if protective actions are initiated within this guidance.) It is also possible that conditions may exist at specific facilities which warrant consideration of values other than those recommended for general use here.[3]

Sheltering may be preferable to evacuation as a protective action in some situations. Because of the higher risk associated with evacuation of some special groups in the population (e.g. those who are not readily mobile), sheltering may be the preferred alternative for such groups as a

[3]EPA, in accordance with its responsibilities under the regulations governing radiological emergency planning (47FR10758; March 11, 1982) and under the Federal Radiological Emergency Response Plan, will consult with Federal agencies and the States, as requested, in such cases.

Table 2-1 PAGs for the Early Phase of a Nuclear Incident

Protective Action	PAG (projected dose)	Comments
Evacuation (or sheltering[a])	1-5 rem[b]	Evacuation (or, for some situations, sheltering[a]) should normally be initiated at 1 rem. Further guidance is provided in Section 2.3.1
Administration of stable iodine	25 rem[c]	Requires approval of State medical officials.

[a]Sheltering may be the preferred protective action when it will provide protection equal to or greater than evacuation, based on consideration of factors such as source term characteristics, and temporal or other site-specific conditions (see Section 2.3.1).

[b]The sum of the effective dose equivalent resulting from exposure to external sources and the committed effective dose equivalent incurred from all significant inhalation pathways during the early phase. Committed dose equivalents to the thyroid and to the skin may be 5 and 50 times larger, respectively.

[c]Committed dose equivalent to the thyroid from radioiodine.

protective action at projected doses up to 5 rem. In addition, under unusually hazardous environmental conditions use of sheltering at projected doses up to 5 rem to the general population (and up to 10 rem to special groups) may become justified. Sheltering may also provide protection equal to or greater than evacuation due to the nature of the source term and/or in the presence of temporal or other site-specific conditions. Illustrative examples of situations or groups for which evacuation may not be appropriate at 1 rem include: a) the presence of severe weather, b) competing disasters, c) institutionalized persons who are not readily mobile, and d) local physical factors which impede evacuation. Examples of situations or groups for which evacuation at 1 rem normally would be appropriate include: a) an incident which occurs at night, b) an incident which occurs when children are in school, and c) institutionalized persons who are readily mobile. Evacuation seldom will be justified at less than 1 rem. The examples described above regarding selection of the most appropriate protective action are intended to be illustrative and not exhaustive. In general, sheltering should be preferred to evacuation whenever it provides equal or greater protection.

No specific minimum level is established for initiation of sheltering. Sheltering in place is a low-cost, low-risk protective action that can provide protection with an efficiency ranging from zero to almost 100 percent, depending on the circumstances. It can also be particularly useful to assure that a population is positioned so that, if the need arises, communication with the population can be carried out expeditiously. For the above reasons, planners and decision makers should consider implementing sheltering at projected doses below 1 rem; however, implementing protective actions for projected doses at very low levels would not be reasonable (e.g. below 0.1 rem). (This guidance should not be construed as establishing an additional lower level PAG for sheltering.) Sheltering should always be implemented in cases when evacuation is not carried out at projected doses of 1 rem or more.

Analyses for some hypothesized accidents, such as short-term releases of transuranic materials, show that sheltering in residences and other buildings can be highly effective at reducing dose, may provide adequate protection, and may be more effective than evacuation when evacuation cannot be completed before plume arrival (DO-90). However, reliance on large dose reduction factors for sheltering should be accompanied by cautious examination of possible failure mechanisms, and, except in very unusual circumstances, should never be relied upon at projected doses greater than 10 rem. Such analyses should be based on realistic or "best estimate" dose models and include unavoidable dose during evacuation. Sheltering and evacuation are discussed in more detail in Section 5.5.

2.3.2 Thyroid and Skin Protection

Since the thyroid is at disproportionately high risk for induction of nonfatal cancer and nodules, compared to other internal organs, additional guidance is provided to limit the risk of these effects (see footnote to Table 2-1). In addition, effective dose, the quantity used to express the PAG, encompasses only the risk of fatal cancer from irradiation of organs within the body, and does not include dose to skin. Guidance is also provided, therefore, to protect against the risk of skin cancer (see Table 2-1, footnote b).

The use of stable iodine to protect against uptake of inhaled radioiodine by the thyroid is recognized as an effective alternative to evacuation for situations involving radioiodine releases when evacuation cannot be implemented or exposure occurs during evacuation. Stable iodine is most effective when administered immediately prior to exposure to radioiodine. However, significant blockage of the thyroid dose can be provided by administration within one or two hours after uptake of radioiodine. If the administration of stable iodine is included in an emergency response plan, its use may be considered for exposure situations in which the committed dose equivalent to the thyroid can be 25 rem or greater (see 47 FR 28158; June 29, 1982).

Washing and changing of clothing is recommended primarily to provide protection from beta radiation from radioiodines and particulate materials deposited on the skin or clothing. Calculations indicate that dose to skin should seldom, if ever, be a controlling pathway. However, it is good radiation protection practice to recommend these actions, even for alpha-emitting radioactive materials, as soon as practical for persons significantly exposed to a contaminating plume (i.e., when the projected dose from inhalation would have justified evacuation of the public under normal conditions).

2.4 Dose Projection

The PAGs are expressed in terms of projected dose. However, in the early phase of an incident (either at a nuclear facility or other accident site), parameters other than projected dose may frequently provide a more appropriate basis for decisions to implement protective actions. When a facility is operating outside its design basis, or an incident is imminent but has not yet occurred, data adequate to directly estimate the projected dose may not be available. For such cases, provision should be made during the planning stage for decisions to be made based on specific conditions at the source of a possible release that are relatable to ranges of anticipated offsite consequences. Emergency response plans for facilities should make use of Emergency Action Levels (EALs)[4], based on in-plant conditions, to trigger notification of and recommendations to offsite officials to implement prompt evacuation or sheltering in specified areas in the absence of information on actual releases or environmental measurements. Later, when these data become available, dose projections based on measurements may be used, in addition to plant conditions, as the basis for implementing further protective actions. (Exceptions may occur at sites with large exclusion areas where some field and source data may be available in sufficient time for protective action decisions to be based on environmental measurements.) In the case of transportation accidents or other incidents that are not related to a facility, it will often not be practicable to establish EALs.

The calculation of projected doses should be based on realistic dose

[4]Emergency Action Levels related to plant conditions at commercial nuclear power plants are discussed in Appendix 1 to NUREG-0654 (NR-80).

models, to the extent practicable. Doses incurred prior to initiation of a protective action should not normally be included. Similarly, doses that might be received following the early phase should not be included for decisions on whether or not to evacuate or shelter. Such doses, which may occur from food and water, long-term radiation exposure to deposited radioactive materials, or long-term inhalation of resuspended materials, are chronic exposures for which neither emergency evacuation nor sheltering are appropriate protective actions. Separate PAGs relate the appropriate protective action decisions to those exposure pathways (Chapter 4). As noted earlier, the projection of doses in the early phase need include only those exposure pathways that contribute a significant fraction (e.g., more than about 10 percent) of the dose to an individual.

In practical applications, dose projection will usually begin at the time of the anticipated (or actual) initiation of a release. For those situations where significant dose has already occurred prior to implementing protective action, the projected dose for comparison to a PAG should not include this prior dose.

2.5 Guidance for Controlling Doses to Workers Under Emergency Conditions

The PAGs for protection of the general population and dose limits for workers performing emergency services are derived under different assumptions. PAGs consider the risks to individuals, themselves, from exposure to radiation, and the risks and costs associated with a specific protective action. On the other hand, workers may receive exposure under a variety of circumstances in order to assure protection of others and of valuable property. These exposures will be justified if the maximum risks permitted to workers are acceptably low, and the risks or costs to others that are avoided by their actions outweigh the risks to which workers are subjected.

Workers who may incur increased levels of exposure under emergency conditions may include those employed in law enforcement, fire fighting, radiation protection, civil defense, traffic control, health services, environmental monitoring, transportation services, and animal care. In addition, selected workers at institutional, utility, and industrial facilities, and at farms and other agribusiness may be required to protect others, or to protect valuable property during an emergency. The above are examples - not designations - of workers that may be exposed to radiation under emergency conditions.

Guidance on dose limits for workers performing emergency services is summarized in Table 2-2. These limits apply to doses incurred over the duration of an emergency. That is, in contrast to the PAGs, where only the future dose that can be avoided by a specific protective action is considered, all doses received during an emergency are included in the limit. Further, the dose to workers performing emergency

Table 2-2 Guidance on Dose Limits for Workers Performing Emergency Services

Dose limit[a] (rem)	Activity	Condition
5	all	
10	protecting valuable property	lower dose not practicable
25	life saving or protection of large populations	lower dose not practicable
>25	lifesaving or protection of large populations	only on a voluntary basis to persons fully aware of the risks involved (See Tables 2-3 and 2-4)

[a]Sum of external effective dose equivalent and committed effective dose equivalent to nonpregnant adults from exposure and intake during an emergency situation. Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value. These limits apply to all doses from an incident, except those received in unrestricted areas as members of the public during the intermediate phase of the incident (see Chapters 3 and 4).

services may be treated as a once-in-a-lifetime exposure, and not added to occupational exposure accumulated under nonemergency conditions for the purpose of ascertaining conformance to normal occupational limits, if this is necessary. However, any radiation exposure of workers that is associated with an incident, but accrued during nonemergency operations, should be limited in accordance with relevant occupational limits for normal situations. Federal Radiation Protection Guidance for occupational exposure recommends an upper bound of five rem per year for adults and one tenth this value for minors and the unborn (EP-87). We recommend use of this same value here for the case of exposures during an emergency. To assure adequate protection of minors and the unborn during emergencies, the performance of emergency services should be limited to nonpregnant adults. As in the case of normal occupational exposure, doses received under emergency conditions should also be maintained as low as reasonably achievable (e.g., use of stable iodine, where appropriate, as a prophylaxis to reduce thyroid dose from inhalation of radioiodines and use of rotation of workers).

Doses to all workers during emergencies should, to the extent practicable, be limited to 5 rem. There are some emergency situations, however, for which higher exposure limits may be justified. Justification of any such exposure must include the presence of conditions that prevent the rotation of workers or other commonly-used dose reduction methods. Except as noted below, the dose resulting from such emergency exposure should be limited to 10 rem for protecting valuable property, and to 25 rem for life saving activities and the protection of large populations. In the context of this guidance, exposure of workers that is incurred for the protection of large populations may be considered justified for situations in which the collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved.

Situations may also rarely occur in which a dose in excess of 25 rem for emergency exposure would be unavoidable in order to carry out a lifesaving operation or to avoid extensive exposure of large populations. It is not possible to prejudge the risk that one should be allowed to take to save the lives of others. However, persons undertaking any emergency operation in which the dose will exceed 25 rem to the whole body should do so only on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

Tables 2-3 and 2-4 provide some general information that may be useful in advising emergency workers of risks of acute and delayed health effects associated with large doses of radiation. Table 2-3 presents estimated risks of early fatalities and moderately severe prodromal (forewarning) effects that are likely to occur shortly after exposure to a wide range of whole body radiation doses. Estimated average cancer mortality risks for emergency workers corresponding to a whole-body dose equivalent of 25 rem are given in Table 2-4, as a function of age at the time of exposure. To estimate average cancer mortality for moderately higher doses the results in Table 2-4 may be increased linearly. These values were calculated using a life table analysis that assumes the period of risk continues for the duration of the worker's lifetime. Somewhat smaller risks of serious genetic effects (if gonadal tissue is exposed) and of nonfatal cancer would also be incurred. An expanded discussion of health effects from radiation dose is provided in Appendix B.

Some workers performing emergency services will have little or no health physics training, so dose minimization through use of protective equipment cannot always be assumed. However, the use of respiratory protective equipment can reduce dose from inhalation, and clothing can reduce beta dose. Stable iodine is also recommended for blocking thyroid

Table 2-3 Health Effects Associated with Whole-Body Absorbed Doses Received Within a Few Hours[a] (see Appendix B)

Whole Body Absorbed dose (rad)	Early Fatalities[b] (percent)	Whole Body Absorbed dose (rad)	Prodromal Effects[c] (percent affected)
140	5	50	2
200	5	100	15
300	50	150	50
400	85	200	85
460	95	250	98

[a]Risks will be lower for protracted exposure periods.

[b]Supportive medical treatment may increase the dose at which these frequencies occur by approximately 50 percent.

[c]Forewarning symptoms of more serious health effects associated with large doses of radiation.

Table 2-4 Approximate Cancer Risk to Average Individuals from 25 Rem Effective Dose Equivalent Delivered Promptly (see Appendix C)

Age at exposure (years)	Appropriate risk of premature death (deaths per 1,000 persons exposed)	Average years of life lost if premature death occurs (years)
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15
50 to 60	3.5	11

uptake of radioiodine in personnel involved in emergency actions where atmospheric releases include radioiodine. The decision to issue stable iodine should include consideration of established State medical procedures, and planning is required to ensure its availability and proper use.

References

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EP-87 U.S. Environmental Protection Agency. Radiation Protection Guidance to Federal Agencies for Occupational Exposure. Federal Register, 52, 2822; January 27, 1987.

NR-80 U.S. Nuclear Regulatory Commission. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. NUREG-0654, U.S. Nuclear Regulatory Commission, Washington, (1980).

Landmark Perspective

Acute Radiation Syndrome*

Stuart C. Finch, MD

THE ATOMIC bomb detonations of Hiroshima and Nagasaki in 1945 abruptly awakened the world to the realities of the nuclear age. Radiologists, radiation physicists, and some physicians were aware of the sickness that frequently accompanied the use of x-ray therapy, but few knew the constellation of devastating events that may occur after whole-body exposure to excessive amounts of ionizing radiation. Although the majority of acute deaths in Hiroshima and Nagasaki were due to burns and other forms of physical trauma, at least a third of the victims probably died of radiation sickness, with many more developing varying degrees of acute systemic illness secondary to radiation exposure. The Japanese quickly recognized the peculiar effects that resulted from atomic bomb exposure, but not until observations were reported in the medical literature by several astute American military physicians did the acute radiation syndrome become known throughout the scientific world.

See also p 661.

The LANDMARK ARTICLE by Paul D. Keller published in *JAMA* in 1946 and reprinted in this issue of THE JOURNAL probably was the first article published in a major medical journal that identified and organized the salient clinical and laboratory features of the acute radiation syndrome in proper sequence. Keller succinctly summarized the outstanding clinical signs and symptoms of the syndrome and enumerated the major hematologic changes. His notations on the locations of the Japanese survivors when the bombings occurred in relationship to the development of acute effects and survival were in recognition of the importance of early dosimetric estimations. Almost half of the survivors were said to have been located in wooden structures at a distance of 500 m (1650 ft) or less from one of the hypocenters. It is known that there were few survivors in this proximal region. Most of the survivors probably either were adequately shielded or were more distally located. At 2000 m (6600 ft) from the hypocenters, whole-body absorbed radiation was a maximum of 1 or 2 rad so that acute radiation symptoms beyond that distance would not have occurred. This information, even though somewhat misleading, only enhances the value of the Keller article because it illustrates the difficult problem of establishing accurate radiation dose estimates for the survivors of radiation accidents or nuclear warfare. Despite these problems, Keller was able to recognize the clinical manifestations of mechanical and thermal trauma and separate them from the acute radiation effects, which he noted to vary in severity with distance from the bomb hypocenters in the two cities. Subsequent definitions of the acute radiation syndrome in the medical literature represent embellishments and refinements of the syndrome as described by Keller.

Historical Perspective

It is almost always possible in medical history to find partial or complete descriptions of disorders that antedate publication of a classic article. Some of the toxic effects of radiation exposure were recognized in the late 19th century. In 1912, Gauss and Lembeke[1] described a disorder known as "Röntgenkater." This has been defined as "Röntgen-hang-over" and equated to a hangover from alcohol. In 1918, Béclère[2] noted that after radiation exposure patients may develop "*mal des irradiations pénétrantes*," which in English is best defined as radiation sickness. The importance of the hematologic depression with radiation therapy was perhaps best appreciated in the mid-1930s in a form of whole-body radiation therapy known as "Mallet's" therapy. The limits of this therapy were governed by the occurrence of leukopenia and thrombocytopenia.[3] Some patients developed severe agranulocytosis and thrombocytopenia associated with bleeding and fatal outcome.

Following the cessation of hostilities with Japan, a group of American physicians from the Army, Navy, and Manhattan Project joined with Japanese physicians, nurses, and medical students for medical follow-up of the atomic bomb survivors of Hiroshima and Nagasaki. This group was known as the Joint Commission. Their observations during a two-month period in the fall of 1945 resulted in an important monograph and several comprehensive articles in which the acute radiation syndrome was described in detail.[4-6] Hachiya's diary of the immediate aftermath of the bombing of Hiroshima also clearly defined many of the components of the acute radiation syndrome.[7] In the late 1940s and early 1950s a series of excellent articles by Cronkite, Hempelmann, and others described in more detail and greater perspective the characteristics and importance of the acute radiation syndrome in relationship to radiation dosimetry, prognosis, and treatment.[8-13] During the late 1950s and early 1960s, a number of lucid descriptions of the clinical aspects of several radiation accidents were published.[14-19] Some excellent publications have been devoted to the management of the clinical problems associated with radiation accidents.[20-23] Several other noteworthy monographs have related the clinical features of the acute radiation syndrome to its histopathology.[4,5,22,23] Three Mile Island failed to elicit publication of any important articles on the acute radiation syndrome, not surprising since no acute radiation illness was experienced in association with that accident. Chernobyl was a different story. Although Soviet scientists have published little concerning the acute medical consequences of that disastrous event in the open medical literature, there have been some interesting and informative discussions concerning the resultant injuries.[24-26] Chernobyl has again alerted the medical profession to the need for a complete understanding of the diagnosis and proper treat-

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*A commentary on Paul D. Keller: A clinical syndrome following exposure to atomic bomb explosions. *JAMA* 1946;131:504-506.

ment of persons with the acute radiation syndrome.

Clinical Features of the Acute Radiation Syndrome

There are several reasons for stressing the importance of understanding the early clinical features of the acute radiation syndrome. Most important is that of predicting the eventual outcome to provide a guideline to therapy. It is advantageous if reliable estimations of the extent of radiation exposure can be rapidly established for each person. Various experts in the field, however, have repeatedly stressed that accurate radiation dose information is difficult to determine when an accident occurs. Even when it is determined, the information may be of limited value. Often it is impossible to establish the duration of exposure and the amount of shielding after the chaotic events of an accident. The exact type of radiation exposure may be unclear. Radiation dosimeters frequently are improperly located or fail because of physical damage. Importantly, the sensitivity to exposure may vary so that fixation on the establishment of an exact dose is much less important than are the clinical manifestations of the radiation injury.

The acute radiation syndrome for those who survive involves a series of clinical events that vary in severity, duration, and timing, depending on the extent of tissue injury. Its four classic clinical stages commence with an early prodromal phase of nausea and vomiting, lasting from a few hours to one or two days. Next is a latent stage of days to several weeks' duration, when the individual feels quite well. The third stage usually begins during the third to fifth week with abrupt onset and continuation of moderate to severe gastrointestinal tract disturbances, bleeding, infections, and epilation. The fourth stage—recovery—may take weeks to months.

Generally, the larger the exposure dose the more rapid the onset and the more severe the clinical manifestations. Radiation and thermal burns and other forms of physical injury may complicate the clinical presentation, and the various stages of the syndrome may overlap. Clinical manifestations also will depend on the type and distribution of the absorbed radiation. Neutrons are more damaging than are gamma rays, and the systemic effects of whole-body radiation will be more severe than for radiation exposure to separate segments of the body. Abdominal radiation will result in more acute effects than will an equivalent amount of irradiation to the chest or the extremities. Ingram et al[illegible] have emphasized that the usual clinical pattern associated with accidental occupational radiation injuries is oversimplified and that often there are atypical presentations due to partial body exposure. Nevertheless, most authorities agree that careful attention to the clinical manifestations of the acute radiation syndrome constitutes the best guide to prognosis and therapy.

Manifestations of damage to the central nervous system (CNS), gastrointestinal tract, or bone marrow may dominate the clinical picture at various times, depending on the amount and type of radiation. Although there is little direct evidence for humans, there is little doubt that massive amounts of radiation will produce a series of problems that have been named the "CNS syndrome."[illegible] It is characterized by the rapid onset of apathy, lethargy, and prostration, frequently followed by seizures ranging from muscle contractions to grand mal convulsions, ataxia, and death. The early occurrence of severe CNS problems may rapidly yield to intractable hypotension, arrhythmias, and shock before death occurs. Most experimental evidence suggests that the CNS complications probably are secondary to vascular lesions so that the syndrome is more properly called the "neurovascular syndrome."[illegible]

Loss of gastrointestinal epithelium in association with agranulocytosis will result in a series of clinical events that have been called the "gastrointestinal syndrome."[illegible] The syndrome usually begins at any time from a few days to a few weeks after the prodromata with anorexia, nausea, vomiting, diarrhea, and fever. Persistent diarrhea often becomes bloody, and there may be progression to abdominal distention, loss of peristalsis, dehydration, circulatory collapse, and death. The major associated clinical problems are severe systemic infection with enteric organisms, electrolyte disturbances, and hypovolemic shock.

The hematologic changes that develop following acute exposure often are referred to as the "hemopoietic syndrome."[illegible] The earliest change is a fall in the absolute peripheral lymphocyte count. This commences in the first few hours and continues for several days to levels commensurate with the amount of radiation exposure within certain limits. Reduced lymphocyte levels may persist for several weeks. There often is a prompt increase in the leukocyte count during the first few days, then a leveling off for a few more days, following which the granulocyte count will continue to fall with maximum leukopenia developing in two to five weeks. Large doses of radiation may result in severe granulocytopenia within the first seven to ten days, a poor prognostic indicator. Recovery may take several weeks to months. The platelet count usually begins to fall one to two weeks after exposure, with maximum depression in four to five weeks. Massive radiation exposure doses may cause severe thrombocytopenia to develop much earlier. It may take several months before the platelet counts return to normal. Usually there is a slow decline in the erythrocyte count associated with reticulocytopenia, the extent of which depends on the amount of radiation exposure and the severity of the acute radiation syndrome.[illegible] Anemia may develop much more rapidly with blood loss from the gastrointestinal tract or into tissues.

It is convenient to categorize exposed individuals into groups according to the extent of total-body radiation exposure.[illegible] Those who have received superlethal amounts in the range of 50 Gy (5000 rad) or more usually die within 24 to 48 hours. Death usually is attributed to the neurovascular syndrome. Exposure to lesser amounts of superlethal radiation may, in a matter of minutes up to a few hours, be associated with a severe prodromal phase of intractable vomiting, diarrhea, fever, dehydration, and coma leading to death in a few days. Others may recover transiently from the early illness onset only to develop the gastrointestinal syndrome in four or five days[22]; death usually supervenes before the major peripheral blood consequences of bone marrow depression are observed. There are no known effective forms of therapy for persons who have sustained such extensive amounts of radiation damage. Fluids, the liberal use of analgesics, and other empirical symptomatic therapy constitute the most humane modalities of medical management.

The next category extends broadly from about 5 to 20 Gy (500 to 2000 rad). Survival may be possible under the most optimal circumstances. Early CNS and cardiovascular complications may be as severe as those following massive exposures, but a less severe and more protracted course is expected. Individuals first experience a prodromal phase lasting one or two days with variable amounts of anorexia, nausea, vomiting, sweating, fatigue, and prostration. This is followed by a latent period of a week or two of relative well-being. The acute illness phase then supervenes as the result of intestinal mucosal damage and severe bone marrow depression. Agranulocytosis invariably results in the development of buccal and pharyngeal ulcerations, bacteremia, and many other types of infection. Thrombocytopenia will be associated with bleeding into the skin, mucous membranes, and gastrointestinal tract. Some or all manifestations of the gastrointestinal syndrome may develop. Scalp epilation occurs relatively late. Damage to bone marrow stem cells is so severe that marrow recovery may not occur for weeks or months. Survival depends on the use of maximum supportive therapy including the administration of fluids, electrolytes, antibiotics, and platelet and red blood cell transfusions. Transplantation with matched allogeneic bone marrow probably is indicated in this exposure range because of the expectation of either irreversible or prolonged bone marrow stem cell damage.

Exposure in the 2- to 5-Gy (200- to 500-rad) range is lethal if untreated, but many will survive if they receive optimal therapy. The manifestations of exposure may be similar to those noted previously, but they will be more delayed and less severe. The bone marrow depression phase usually lasts three or four weeks. If recovery occurs, it will begin about the sixth week. In the next several weeks, the hematologic picture improves and good health eventually is restored. This category of individuals also will require maximum supportive therapy for the treatment of bacterial infections, bleeding, electrolyte disturbances, and fluid loss. Bone marrow transplantation is not recommended for several reasons. Donor marrow would not engraft without further immunosuppression, which would greatly complicate the clinical course. Furthermore, most patients will survive without bone marrow transplantation if medical management is optimal.

Survival with little or no therapy is almost invariable for persons exposed in the range of 1 to 2 Gy (100 to 200 rad) or less. It has been stated that about 15% of those exposed to radiation will develop signs or symptoms at 1 Gy (100 rad).[16] Usually there is little or no vomiting or diarrhea, but mild late signs and symptoms, similar to those seen in the more seriously ill patients, may develop. Serial blood cell counts will show the same changes that develop with greater exposure, but in a much milder form and at a slightly slower rate of occurrence.

Biological Dosimetry

It is imperative that the effective biologic radiation dose be established for each exposed person at the earliest possible time for triage and subsequent therapy. If, for example, bone marrow transplantation is needed, it is important to complete tissue typing and cross matching before severe peripheral lymphocytopenia develops. The clinical symptoms on the first day of severe nausea, vomiting, and diarrhea will identify the seriously ill, but lesser symptoms may be misleading in a setting of chaos and emotional stress. The accelerated appearance of the gastrointestinal and hematologic syndromes indicate high-dose exposure, but earlier estimates of tissue damage should be established before those problems develop if lives are to be saved. The severity and rapidity of development of lymphopenia is dose related up to 2 to 3 Gy (200 to 300 rad), but for larger exposure doses it is not helpful because few lymphocytes are present in the peripheral blood.[4,8,10-12,14,20] The early leukocytosis is inconsistent and is not dose related, but the subsequent rate of granulocyte decline and severity of agranulocytosis and thrombocytopenia are probably the most reliable early biologic indexes of the extent of radiation damage.[11,18,28] Reticulocytopenia, however, also has been reported to be a reliable indicator of the severity of marrow damage.[28] A person with no peripheral blood granulocytes, platelets, or reticulocytes at 15 days would have a bad prognosis.[11] Anemia is not a good index of marrow damage due to the long lifespan of the erythrocyte and the high frequency of bleeding. Bone marrow aspirations have limited quantitative relationships to exposure, but if performed in various sites may indicate the extent of marrow damage or evidence of early recovery. The type and extent of radiation-induced chromosome aberrations in the peripheral blood lymphocytes provide another early index of the amount and distribution of radiation received,[29-31] although the reliability of dose estimates from these findings has been questioned.[32] Radiation exposure is rapidly followed by amylasemia and the appearance of certain tissue breakdown products in the urine,[18,33] but the usefulness of these findings in terms of biologic dosimetry is uncertain. Electron spin resonance spectroscopy[34] and measurements of neutron-induced total-body ^{24}Na[15,35] are valuable techniques for dose assessment, but they have limited application. Measurements of radiation-induced somatic mutations at the hypoxanthine guanine phosphoribosyl transferase and glycophorin A loci show great promise as biologic dosimeters for the estimation of radiation dose.[36,37]

Chernobyl

The 31 deaths resulting from the Chernobyl nuclear reactor accident[24-26] approximately equaled the total number of accidental radiation deaths occurring in the world during the previous 42 years. About half of the Chernobyl victims had moderate to severe thermal or radiation burns of the skin; 203 persons were said to have had some level of radiation sickness that eventually required hospitalization. Most of the fatalities occurred among the 129 patients hospitalized on the basis of initial signs and symptoms within the first two days. All patients were classified according to four degrees of illness severity during the first three days.

The 29 persons in the highest illness category were estimated to have received 6 Gy (600 rad) or more of radiation exposure. Within the first half hour headaches, fever, and vomiting developed followed by rapid progression of severe lymphopenia within the first six days. Shortly thereafter, the patients developed severe gastroenteritis and all of the clinical consequences of profound granulocytopenia and thrombocytopenia; 13 persons in this group received allogeneic bone marrow transplants and eight fetal liver cell transplants. Identification of patients for transplantation was made during the first 36 hours, but final decision was not made until several days later when the chromosome data corroborated large-dose exposure. Two persons survived bone marrow transplantation, and both eventually rejected the graft. There were no survivors among the eight persons who received fetal liver cells.

At Chernobyl there were 23 persons with six deaths in the third level of illness who were estimated to have received about 2 to 6 Gy (200 to 600 rad) of radiation. All in this group and the groups with less exposure had progressively longer latent periods and less severe clinical manifestations. There were 53 persons in the group at the second level of severity estimated to have received from 2 to 4 Gy (200 to 400 rad). Those in the lowest exposure level (0.8 to 2.1 Gy [80 to 210 rad]) developed only mild initial symptoms several hours after exposure, slight lymphopenia during the first few days, and only modest thrombocytopenia and granulocytopenia at about four weeks.

Physical trauma played a large role in complicating classification and clinical sequelae at Chernobyl, but the overall experience was important for several reasons. Radiation monitoring information from dosimeters was found to be of little value. Soviet scientists believed the best early biologic indicators of dose were the length of the asymptomatic latent period and severity of early symptoms, the rate of decline in lymphocyte count, and the number and distribution of dicentric chromosomes in peripheral lymphocytes. Changes in peripheral granulocytes, platelets, and reticulocytes corresponded closely to the magnitude of chromosomal aberrations, but overall, the chromosomal information was believed to be the better index of total-body exposure. External body radioactive contamination hampered whole-body counting, but limited information on internal contamination was obtained by blood and urine monitoring for radioactivity. Increased concentrations of serum and urine amylase at 36 to 48 hours were proportionate to the extent of exposure. At one to two weeks there were many serum biochemical changes that reflected varying degrees of azotemia and hepatic dysfunction. Reduction in the vitamin K-dependent clotting factors contributed to bleeding. Many of the disturbances lasted up to a month or longer.

There is no specific therapy for acute radiation illness so that patient treatment is largely supportive.[a] At Chernobyl there was success with the use of triple broad-spectrum antibiotic therapy for fever and bacteremia associated with agranulocytosis. Antiviral and antifungal agents were successfully employed when indicated. Platelet transfusions were effective for thrombocytopenia, and packed red blood cells were given for anemia, which tended to be more severe than anticipated. Most controversial was the use of bone marrow transplantation. Finding suitable donors, determining histocompatibility when the recipient had few circulating lymphocytes, the best time for transplantation, and identifying of suitable recipients were a few of the many problems encountered. Graft-vs-host disease and making the patients even more ill also were major concerns. The overall marrow transplantation results suggest that it will be of benefit to few exposed persons in similar types of future radiation accidents. The accident also illustrates the extreme difficulties in the treatment of about 200 radiation-exposed persons of whom about 25% are critically ill with burns and the acute radiation syndrome.

The observations by Keller in 1946 remain important in terms of dose assessment and patient management. The lessons of Chernobyl, however, emphasize our continued poor understanding of the pathogenesis, prognosis, and management of the acute radiation syndrome. There is urgent need for renewed research and planning in the field of radiation accidents if we are to continue to accept the nuclear option as an alternate source of energy.

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RISK MANAGEMENT ACTIONS TO ASSURE CONTAINMENT EFFECTIVENESS AT SEABROOK STATION

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EXECUTIVE SUMMARY

The ongoing risk management program at Seabrook Station began in 1982 with the initiation of a full-scope, Level 3 PRA called the SSPSA (Reference ES-1). The SSPSA, which was completed in December 1983, concluded that the risk to public health and safety around Seabrook Station was very low and that the plant met the NRC safety goals for individual and societal risk by wide margins. In an update to the SSPSA, reported in PLG-0432 (Reference ES-2) and PLG-0465 (Reference ES-3), it was concluded that a 1-mile evacuation zone at Seabrook Station would result in lower risk levels than was thought could be achieved for all U.S. plants with a 10-mile evacuation zone based on the results of NUREG-0396 (Reference ES-4). Brookhaven National Laboratory conducted a review of these studies for the NRC (Reference ES-5).

In their review, BNL correctly determined that these favorable conclusions from the Seabrook Station PSA studies follow from the results showing that the Seabrook Station containment is exceptionally strong and that the frequency of accidents with large, early releases is extremely low. Based on this insight, NRC and BNL focused their reviews on aspects of the PSA studies that influenced the estimation of the frequency of large, early releases. Although BNL agreed that the Seabrook Station containment has an exceptionally high pressure capacity and that containment bypass events had been adequately considered, they and the NRC staff raised two issues that had not been explicitly addressed and that could conceivably influence the SSPSA results for early release frequency. These issues, which had been subsequently addressed in NUREG-1150 (Reference ES-6), are direct containment heating and induced steam generator tube rupture.

The DCH issue stems from a concern that early containment pressure loads during high pressure core melt scenarios might be higher than previously calculated. This is due to the postulated event of rapid heat transfer to the containment resulting from the high pressure ejection of finely dispersed core debris into and out of the reactor cavity. This pressure load could combine with other pressure loads, such as those due to RCS depressurization and hydrogen burns, to elevate the peak pressure at the time of reactor vessel melt-through in contrast with the levels previously calculated by safety assessment experts.

The ISGTR issue stems from a concern that, during high pressure core melt scenarios with no secondary side cooling of the steam generators, the steam generator tubes might fail due to thermal creep rupture prior to the time of reactor vessel melt-through. This would create a potential bypass condition.

In support of the NRC and BNL review effort, New Hampshire Yankee provided analytic and experimental evidence to support its technical position that the risk of early containment failure or bypass due to DCH and ISGTR are insignificant for Seabrook Station. Although there were no errors or specific problems identified with this evidence, NRC and BNL concluded that more work was needed to provided a convincing case that DCH and ISGTR are insignificant risk contributors at Seabrook Station.

In this study, the risk significance of ISGTR and DCH are examined in greater detail to help resolve the residual uncertainties that remain from the BNL review. Although NHY is no longer pursuing a reduced EPZ, they agreed to perform this study to better understand the safety significance of these issues for Seabrook Station and to assist them in their continuing efforts in evaluating emergency planning options. The basis for a probabilistic evaluation of DCH and ISGTR at Seabrook Station is set forth in Sections 1 through 7 of this report. The probabilistic evaluation itself is presented in Section 8 and the conclusions in Section 9. The major elements of this study include:

- A sequence-by-sequence examination of SSPSA scenarios to determine their applicability to the DCH and ISGTR issues.

- A systematic evaluation of the current procedures for their ability to minimize the likelihood of a high pressure core melt and to ensure that steam generator cooling is maintained, thereby reducing the risk of DCH and ISGTR.

- The development of new procedures and plant modifications to reduce the risk of DCH and ISGTR. These include the expanded use of the pressurizer PORVs to depressurize the RCS and the use of fire water pumps to feed the steam generators.

- The evaluation of PORV operability during degraded core conditions.

- The evaluation of potential negative effects of new procedures to reduce the risk of DCH and ISGTR.

- An event tree analysis of a class of accident sequences known as TMLB' to evaluate the potential for DCH and ISGTR and the merits of actions to mitigate them at Seabrook Station. This analysis includes a probabilistic quantification of uncertainties associated with DCH and ISGTR phenomena.

- Detailed thermal hydraulic analysis of the RCS and containment during high pressure core melt scenarios, using MAAP 3.0B model of Seabrook Station. This analysis includes benchmarking the MAAP code to other analytic and experimental data and a range of sensitivity cases to support the event tree quantification of uncertainty.

- A reexamination of the risk significance of DCH and ISGTR at Seabrook Station.

The following conclusions were reached in this study:

- Given a high pressure core melt sequence, no operator actions to restore cooling of the steam generators, and no actions to depressurize the RCS, the conditional mean frequency of the possible end states are as follows.

End State	Conditional Mean Frequency, Given a High Pressure Core Melt
Success (no melt)	0
Low Pressure Core Melt with Intact Containment	.5
High Pressure Core Melt with Intact Containment	.5
Containment Bypass due to ISGTR	1×10^{-3}
Containment Failure due to DCH	5×10^{-4}
Total	1.000

In this analysis, the low pressure core melt with intact containment end state results from the case in which the RCS hot leg piping fails before vessel melt-through. The conditional median frequencies of DCH and ISGTR for this case are both zero.

- A conservative assessment of the mean frequency of early containment failure or bypass due to ISGTR at Seabrook Station is 6×10^{-10} per year. The median corresponding to this mean is 0. Even without any credit for RCS depressurization according to existing or new procedures, the mean frequency of ISGTR is very low, a value of 3×10^{-8} per year.

- A conservative assessment of the mean frequency of containment failure due to DCH at Seabrook Station is 8.8×10^{-8} per year. The median corresponding to this mean is 0. Even without any credit for RCS depressurization, the mean frequency of DCH is very low, a value of 2.8×10^{-7}. These results for DCH are based, in part, on the BNL assessment of the pressure capacity of the Seabrook containment and DCH pressure loads from current NRC contractor estimates in support of NUREG-1150.

- This detailed examination of the DCH and ISGTR issues at Seabrook Station upholds the principal conclusions of PLG-0465 and PLG-0432 regarding the risk reduction benefit of evacuation and comparisons with risk acceptance criteria.

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Bounding Probabilistic Safety Assessment Probabilities by Reality

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ABSTRACT

The investigation of the failure in systems where failure is a rare event makes the continual comparisons between the developed probabilities and empirical evidence difficult. However as Feynman has stated, "The principal of science, the definition, almost, is the following: The test of all knowledge is experiment. Experiment is the sole judge of scientific 'truth'." ("Feynman, Lectures in Physics", Chapter 1, p 1-1). Therefore the comparison of the predictions of rare event risk assessments with historical reality is essential to prevent PSA predictions from drifting into fantasy. One approach to performing such comparisons is to search out and assign probabilities to natural events which, while extremely rare, have a basis in the history of natural phenomena or human activities. For example the Segovian aqueduct and some of the Roman fortresses in Spain have existed for several millennia and in many cases show no physical signs of earthquake damage. This evidence could be used to bound the probability of earthquakes above a certain magnitude to less than 10^{-3} per year. Extending the limits of evidence to recorded history might perhaps reduce this estimate to 10^{-4} however further reduction is doubtful on a historical basis. On the other hand, there is evidence that some repetitive actions can be performed with extremely low historical probabilities when operators are properly trained and motivated, and sufficient warning indicators are provided. Failure to extend the landing gear or flaps upon the landing of a commercial aircraft is one example where the historically documented failure frequency can be estimated well below 10^{-6} per landing and perhaps even as low as 10^{-7}. The point is not that low probability estimates are impossible, but continual reassessment of the analysis assumptions, and a bounding of the analysis predictions by historical reality.

This paper will review the probabilistic predictions of PSA in this light, will attempt to develop, in a general way, the limits which can be historically established and the consequent bounds that these limits place upon the predictions and will illustrate the methodology used in computing such limits. Further, the paper will discuss the use of empirical evidence and the requirement for disciplined systematic approaches within the bounds of reality and the associated impact on PSA probabilistic estimates.

1.0 INTRODUCTION

1.1 THE NEED FOR QUANTITATIVE ASSESSMENT

Probabilistic safety assessment (PSA) is a challenging and difficult field. One difficulty is that the problems, such as estimating the core damage probability of a nuclear reactor, require addressing complex and difficult issues involving a great deal of modeling and analysis. In addition, relevant data which are needed in the analysis has only been collected in the last 2 or 3 decades. However, quantitative assessment is needed to make real progress in assuring the safety of nuclear reactors. Even a somewhat flawed, or oversimplified, analysis plays a role since other engineers and analysts can study the problems, look for improvement, and eventually develop a superior analysis. One of the historic quotes which supports this approach is that by William Thompson (Lord Kelvin) in the early 1890's:

> "When you can measure what you are speaking about, and express it in numbers, you know something about it; but when you cannot measure it, when you cannot express it in numbers, your knowledge is of a meager and unsatisfactory kind: it may be the beginning of knowledge, but you have scarcely, in your thoughts, advanced to the stage of science."

This paper will discuss how the results of quantitative analysis can be compared with that of experience (data) for a variety of situations.

1.2 THE ROLE OF QUANTITATIVE ASSESSMENT

A quantitative assessment of the reliability or availability of a plant has many uses:

(1) To satisfy licensing requirements.

(2) To identify low reliability aspects of the design and suggest means of improvement through the design.

(3) To inform the public and the technical community of the risks and help them evaluate risk-benefit trade-offs.

(4) To contribute to the data base of modeling knowledge and probability data for nuclear reactors.

As a result of the Three Mile Island and Chernobyl accidents, all nations are concerned about the safety of nuclear power plants. In the U.S. the aftermath of the Three Mile Island accident caused the NRC to consider setting quantitative safety goals in terms of the probability of core damage per year for each U.S. plant, and to require these individual plants to demonstrate the achievement of these goals through a PSA. Currently, goals of 10^{-4} have been proposed for existing plants and 10^{-5} has been suggested for new "inherently safe" designs. The question is, what is the level of credibility of assessments of probabilities as low as these within the broader context of empirical reality?

This is the issue which is attempted to be addressed here. The approach taken was to bring together as many bodies of relevant experience as could be gathered in the time available. These sources included:

(1) PSAs for various American and European plants.

(2) Accident and incident historical data for nuclear power plants (plant experience).

(3) Data for relevant events which can help bound the PSA estimates from above and below.

In essence, the advice of de Finetti [1979, p.183-184] was applied:

> "The following recommendations are obvious, but not superfluous:
>
> • to think about every aspect of the problem;
>
> • to try to imagine how things might go, or, if it is a question of the past, how they might have gone (one must not be content with a single possibility, however plausible and well thought out, since this would involve us in a *prediction*: instead, one should encompass all conceivable possibilities, and also take into account that some might have escaped attention);
>
> • to identify those elements which, compared with others, might clarify or obscure certain issues;
>
> • to enlarge one's view by comparing a given situation with others, of a more or less similar nature, already encountered;
>
> • to attempt to discover the possible reasons lying behind those evaluations of other people with which, to a greater or lesser extent, we are familiar, and then to decide whether or not to take them into account. And so on."

Therefore, the objective of this work is not to pass judgement on individual estimates of core damage frequency, but rather to begin to establish the "possibility" space in which all core damage frequency estimates must be contained.

1.3 RARE EVENTS CAN BE NUMERICALLY BOUNDED

The desire for safe nuclear power has resulted in a call for decreasing risk achievement in newer designs and risk reduction in existing designs. Core damage risk achievement goals of 10^{-4} or 10^{-5} per year have recently been cited by the NRC and industry [Inside NRC, August 1990]. The approach which industry has taken toward achieving such high reliability is to devise simpler and safer designs. As a second step these designs, or design upgrades, should be subjected to a detailed PSA to determine the likelihood of achieving the goal. In the case of new designs the safety record will take years to observe. Unfortunately, it will take perhaps a decade for such plants to be built, and at least an additional decade before we have any significant operational experience. It seems appropriate therefore to inject an additional evaluation phase at the outset to compare and bound the proposed risk values by the values predicted for other designs, the values demonstrated by other operational experience, and various other events which can serve as upper and lower bounds. The philosophy behind such bounds is that human designers, builders, and operators of plants are subject to a host of errors and mistakes which compromise the safety of a plant. It is not reasonable to assume then that any new design will eliminate all of these errors. Further, although many of these will hopefully be eliminated by the experience of the past, it is unlikely that this will result in orders of magnitude of improvement.

This paper proposes a viewpoint - a way of thinking about nuclear designs and their associated risks. The paper is not meant as a definitive study, but an approach to evaluating the feasibility of achieving risk goals and understanding the reasonableness of risk values developed via PSAs of these newly proposed designs.

As an analogy, consider for a moment that some manufacturer begins to produce a new auto design. If it is desired to evaluate the reliability of their product at the outset, data could be collected on the five most reliable autos in the world and the five worst. An average of the lower five and an average of the upper five would provide a range estimate of the potential reliability achievement of the proposed design. If new design features are offered to improve reliability achievement the expected actual achievement may be better than that previously achieved, but not so much so as to challenge human fallibility in design, the underlying basic physical laws, and historical evidence. Because of this it may be possible to establish some limits on the credibility of PSA estimates based upon the limits of human capabilities and the limits of historical evidence.

2.0 BOUNDS ON NATURAL PHENOMENA AND HISTORICAL EVENTS

In this section the frequency of occurrence of some natural phenomena are considered. These will be used later as bounding probabilities. Considering natural phenomena frequencies as bounds is not new for PSA, but here the extremes of what natural phenomena can provide to support the credibilities of rare event probabilities are considered. An occurrence rate is estimated simply from data as a ratio of events divided by time. Various estimates will be made using this simplistic approach based upon certain rare events and various epochs of time.

2.1 TIME EPOCHS

The longest interval of time which can logically be considered is the age of the universe. Modern scientific theories estimate the age of the universe as 10^{10} years. [See Hawking 1988, p.108] This time epoch is referred to as universe history. These estimates are, of course, uncertain and are changing as new cosmological theories emerge.

One can also relate events to the age of the earth's surface. Again these estimates are based on a number of methods developed in the fields of Physics and Biology. A good summary appears in Table 1, which is excerpted from Cailleux, 1976.

Table 1 Dates of Origin of Some Terrestrial Phenomena
[Cailleux, 1976, p.173]

Event	Estimated date in million years Max.	Min.	Remarks source, method, etc.
Formation of the earth	?	20	Temperature of the crust
Formation of the earth	4,500		Lead minerals
Formation of the earth's crust	5,000	?	Uranium[235]
Formation of the earth's crust	4,000	?	Lead/uranium ratio
Crystallisation of the rocks		3,360	Radioactivity in rock
Consolidation of the continents	4,200	2,600	Speed of accretion
Formation of the oceans	?	350	Sedimentation
Formation of the oceans	?	180	Sodium in the oceans
First living organisms	5,000	2,000	Longevity of the orders, etc.
First animals	5,000	1,800	Scales of psychism
First fossils preserved		2,600	Algae (Canada)
Speciation	?	2,200	Number of species
Single-celled animals	4,000	2,000	Scale of anatomical evolution (Huxley)
First fossilised animals	?	2,200	Number of species
Many celled plants	?	950	Number of cells

The bulk of the data predicts the age of the earth as 2 - 5 x 10^9 years. Thus, from the beginning of the universe to the creation of the earth's surface we have from 5 - 18 billion years. In fact, Hawking [1988, p.124] delineates three eras from the beginning of the universe (Big Bang) to the formulation of the earth:

(1) Early stars converted hydrogen and helium to carbon and oxygen went supernova and the debris formed the solar system - 5 x 10^9 years.

(2) Earth is too hot for the development of complicated life - 2 x 10^9 years.

(3) Evolution of biological life on earth - 3 x 10^9 years.

The sum of the time durations of these three eras is 10 billion years, which agrees with our previous range of 5 - 18 billion years. From the above discussion, we can define the terms geological history and biological history can be defined. Table 2 indicates the "historical periods" developed above.

Table 2 Relevant Time Periods for Bounding Risk Frequency

Time Period	Duration in years	Frequency of occurrence per year
Universe History (Age of Universe)	1-2 x 10^{10}	1-5 x 10^{-10}
Geological History (Age of the Earth's Crust)	5-15 x 10^9	6.7-20 x 10^{-11}
Biological History (Age of Species Populating the Earth)	3-13 x 10^9	7.7-33 x 10^{-11}
Written History	6 x 10^3	1.7 x 10^{-4}

Recorded history can also be defined in terms of the first known written documents, Sumerian and Babylonian writings are extant which have been dated back as far as 4,000 B.C., or about 6,000 years ago, and are the current limits of written history. [Grun, 1982].[1] (Of course oral history contained in songs and stories such as legends, sagas, etc. probably occurred even earlier).

2.2 EVENT RECORDS

In the previous section various times of relevance to our bounds have been discussed, i.e. the denominator term for an occurrence frequency estimate. In this section the numerator term, the occurrence frequency of certain rare events is discussed. For example, if one Big Bang universe creation occurred 10^{10} years ago, the history of this universe can be spoken of as being about 10^{10} years old, or that the "occurrence rate" for a universe is once per 10^{-10} years. Similarly, the occurrence rates for the first four items in Table 2 are given by the reciprocal of the stated times. In the case of other events, scientific or recorded evidence must be relied upon. For example, geologists can predict the number of ice ages, volcanic eruptions, etc. from physical measurements and observations of the the earth's surface. Also, such events as tidal waves, volcanic eruptions, and large earthquakes have been recorded in some way in written history. Computations of the frequency of various natural disasters appear in Starr [1970]. While there is considerable uncertainty associated with the events of low frequency it is also true that factors of 10 to 100 changes in the occurrence

[1]Note: The Hebrew year is 5750 which some theologians feel begins with creation; however, many would consider the year zero the beginning of civilization.

rate do not affect the problem significantly so that insight can be drawn even with large error factors.

2.3 DEALING WITH ZERO OCCURRENCES

In some instances, a time interval can be identified but the evidence indicates that there have been zero occurrences of some particular event. If the event has not occurred it is true that no lower bound for its frequency of occurrence can be stated. However, an upper bound can be obtained by assuming that zero occurrence can be bounded from above by one occurrence.

An exact solution for this situation under the constant occurrence rate assumption is available. In 1953 Epstein showed that if one is attempting to estimate a constant failure rate from the ratio of events/time, then this point estimate has a χ^2 distribution. Thus a confidence interval along with the χ^2 distribution can be used to compute a well defined probability associated with zero occurrences. Welker and Lipow [1974] conclude that a good representative value for the failure rate is obtained by assuming 1/3 of a failure.

Actually, the χ^2 distribution applies to all estimates of occurrence frequencies for constant occurrence rate models. Thus, not only a point estimate of the occurrence can be derived, but also an interval estimate if this is desired. As has been mentioned, many of the estimates which will be made here are order of magnitude calculations, so point estimates alone should be sufficient.

2.4 ASTRONOMICAL PHENOMENA

Another significant historical limit can be obtained from stellar history such as life of the sun or other stars in our galaxy from general stellar history and astrophysical models. Two of the possible modes of failure are the star becoming a Red Giant and stellar explosion. The sun produces its energy by slowly transforming hydrogen into helium. Estimating the mass of the sun as 2×10^{27} tons, which is presently about half hydrogen, and the hydrogen consumption rate as 660 million tons/sec. "the star becomes a Red Giant life limit" of the sun can be computed as $[10^{27}/660 \times 10^6]$sec x $[1/3600 \times 24 \times 365]$ years/sec $\approx 5 \times 10^{10}$ [Gamow, 1959, pp.300-301]. The associated occurrence rate, the reciprocal, is 2×10^{-11}. While this does not imply that the probability is 10^{-11} of a star of class G like the sun with a known mass and burn rate developing into a Red Giant in the next year, the considerable uncertainty in the physical process and models and the potential for instabilities make it difficult to substantiate statements of a probability of less than 10^{-11}/year of this not occurring.

As a star is "burning out" it can also undergo a cataclysmic conversion, it can become a nova or a supernova exploding star. Unpredictably, in a few days the star's surface becomes very hot, and the surface rapidly expands producing a luminous gas cloud. The star returns to its original luminosity within about 1 year; however, the luminous gas cloud continues to expand. A supernova is similar to a nova, but thousands of times more luminous. Gamow [1959, pp.293, 302-304] estimates the occurrence of nova in our stellar system at about 40 per year. Supernovas are more rare and three appear to have been recorded in history: the star of Bethlehem (0 a.d.), the Chinese astronomer's star (1054 a.d.), and the"daylight star" observed by Tycho Brahe [1572]. Thus, we can predict the occurrence rate as $3/2{,}000 = 1.5 \times 10^{-3}$ per year, or if we believe that a supernova is so spectacular that even the ancients would have recorded it in some way, we can use the span of recorded history in the denominator i.e., $3/6{,}000 = 5.0 \times 10^{-4}$ per year.

Of course, the above calculations are the occurrence rate of any nova/supernova which can be observed. Thus, a calculation of how may stars are observable from the earth must be made. Since the observation period is 6,000 years - actually 6,000 light years - what must be calculated is how many stars within our galaxy are 6,000 light years distant. Our galaxy (the Milky Way) is assumed to contain 4×10^{10} stars and is a disk 100,000 light years in diameter and 10,000 light years thick. The "observable space" is above 6,000 light years in radius. Roughly, the volume of our galaxy is that of a disk and equals $\pi \times (50{,}000)^2 \times 10{,}000$ light years^3, and our observable disk (modeling the observable sphere by a disk) is $\pi \times (6{,}000)^2 \times 10{,}000$. The ratio of the two volumes is $6^2/50^2 = 1.44 \times 10^{-2}$ and assuming a homogeneous dispersion of stars throughout the galaxy produces an "observable disk" with $1.44 \times 10^{-2} \times 4 \times 10^{10} = 5.8 \times 10^8$ stars. Thus, the probability of some arbitrarily chosen star going nova is $40/5.8 \times 10^8 = 6.9 \times 10^{-8}$ occurrences per year. A similar calculation yields $1.5 \times 10^{-3}/5.8 \times 10^{8} = 2.6 \times 10^{-12}$ occurrences per year for a star going super nova.

If a particular star is specified and its evolutionary phase is known then, of course, a different probabilistic model holds based upon the physics. Again for the sun the probability of becoming nova is theoretically much smaller (and supernova perhaps impossible), but uncertainties in the theory are such that statements of occurrence frequencies below those mentioned are probably untenable.

3.0 NUCLEAR POWER PLANT RISKS

3.1 WORLDWIDE NUCLEAR PLANT EXPERIENCE

If the specific history of the nuclear power industry is considered, then the world experience base for nuclear reactor accidents, (only core damage/total core damage considered), can be supplemented by studying the accident experience of nuclear reactors used on ships and submarines as well as power reactors. The experience with ship reactors can be roughly estimated by studying the approximate data in Table 3.

Table 3 Experience with American Nuclear Ships (Submarine fleet only, does not include surface ships) [Expert estimate, P. Appignani, 1990]

Years	Average Number of subs	x Years	x Fraction up per year	= Number of Operating years
1950 - 1960	20	10	0.75	150
1960 - 1970	60	10	0.75	450
1970 - 1980	100	10	0.75	750
1980 - 1990	120	10	0.75	900
			Total	2250

There has been no loss of life due to nuclear accidents in the American submarine fleet. The simple procedure of assuming one failure yields $1/2250 = 4.4 \times 10^{-4}$ failures/year. Using Welker and Lipow's suggestions 1/3 of this value is obtained, or 1.48×10^{-4} failures/year.

The world experience for nuclear accidents can be estimated by computing the world experience for nuclear power plant operation as shown in Table 4.

Table 4 World Experience in Nuclear Plants

	Number of Plants	Plant Years x	.70	Number of Operating years
U.S.	99	987	x .70	691
Outside U.S.	318	3,491	x .70	2,444
World Total	417	4,478	x .70	3,135

Data compiled from maps and tables published by Nuclear News, August and September 1987. Updated to January 1, 1990, assuming: (1) No deletions from list, (2) Only additions to list are due to plants coming online, which occurred at dates predicted on 1987 maps.

These computations assume that all project dates for plants going into operation in 1989 and 1990 were accurate and no plants were taken out of service in this period. As shown in the tables, an availability factor of 70% was assumed. If we assume two core damage accidents, Three Mile Island and Chernobyl, we have a point estimate of $2/3,135 = 6.38 \times 10^{-4}$. Of course there is evidence that plants are at risk whether they are generating power or not as the U.S. industry news [Inside NRC, July 16, 1990] and the recently published French safety studies [Nuclear News: July 1990] indicate. The French study in particular states that one third of the plant risk occurs during shut-down period. Thus, another estimate of world experience would be $2/4,478 = 4.48 \times 10^{-4}$.

Another source of experience is that generated by the U.S. Nuclear Regulatory Commission's Accident Sequence Precursor (ASP) program [Minarick, 1990]. The ASP program began in 1979 and is continuing presently. Data was collected on operational events (near accidents) which represented potentially significant portions of accident sequences and precursors. (480 were identified within the more than 1,000 reactor years observed). These events were analyzed using plant-class specific event tree models to estimate conditional probabilities of proceeding to core damage. While these estimates are used primarily for ranking, they can also be used to calculate an average core damage frequency based on historic events. This average is above 10^{-4} for young plants and in the mid 10^{-5}'s for older plants. Three events are discussed below as an example of some of the failures uncovered in the ASP program. [Minarick, July 1990].

• At Arkansas Nuclear One, Unit 1 (LER 313/89-028), an unknown contact was discovered in the control circuits for two of the three service water pumps. In situations involving a safety actuation signal without previous main generator lockout (spurious safety actuation signal or large break LOCA), this contact would prevent service water pump restart. A similar situation had been discovered at Zion (LER 295/88-019), where a design deficiency was found in the anti-pump feature in the AFW and component cooling water pump breaker control circuits. This would have locked out the breakers in the tripped condition if an actual LOOP had occurred. Neither of these conditions were discovered while performing PSAs on the two plants.

• At Catawba (LER 414/88-012), following a trip, one AFW train transferred to the alternate service water system suction source because of setpoint drift in two pressure switches. The undetected presence of Asiatic clams in the service water pumped to the steam generators (the clams had grown in the stagnant water between the service water loops and the AFW system) caused the obstruction of two of four AFW control valves. Had the second AFW train suction switched to service water (one of two required pressure switches also had setpoint drift), additional degradation would likely have occurred.

• At Ft. Calhoun (LER 285/87-025, 87-033, 88-010), equipment failures and inadequate training and procedures resulted in water intrusion into the instrument air system during a fire system test. Immediate problems involved a diesel generator (DG) fuel oil level gauge failing high and the opening of a component cooling water shutdown cooling heat exchanger outlet valve. The instrument air supply was blown down to remove the water. Two months later, both air-operated DG radiator exhaust dampers were found unavailable because of water from the July incident. This rendered both DGs unavailable for the 2-month period. Seven months later, four check valves associated with backup accumulators for the refueling water storage tank level sensors were also found failed.

These events are indicative of some of the types of failures which are typically not modeled in PSAs. They are typically not addressed because they involve unusual failure modes which are difficult to visualize (beyond the modeler's experience base), because they involve failure modes believed at the time to contribute insignificantly to a system's failure probability, or because financial or schedule constraints prevented analysis to a level of detail to identify the situation.

In the first event, a long-term dependency existed between different portions of the instrument air system. The dependency was not hard-wired into the design, but resulted from water contamination and incomplete system repair following the event.

In the second event, a spurious system reconfiguration resulted in the pumping of clam-infested water through the AFW pumps, with the subsequent clogging of control valve internals. Service water is typically addressed in PSA as an alternate suction source for AFW (suction source redundancy), and not as a unique cause for system failure.

Such events typify situations in which the as-built conditions of the plant are inconsistent with the plant design as understood by the PSA analysts, either because the actual plant was different from its documentation (Arkansas 1) or because the depth of analysis was inadequate to determine that an unacceptable situation existed (Zion).

The conditional core damage probability estimated for each of the above events is greater than 10^{-4} occurrences per reactor year. Typically, seven precursors with conditional core damage probabilities of 10^{-4} or greater are identified yearly in the U.S. light-water reactor population. The continued occurrence of operational events with conditional core damage probabilities $>10^{-4}$ presents a major impediment to a belief that the very low core damage frequencies estimated lower than this in PSAs are valid. For example, if it is assumed, as a minimum, that only one such event will occur over a 40 year plant lifetime (on average, three such events would be expected, plus a larger number of less significant events), it is possible to calculate a core damage frequency contribution from the event of $1/40 \times 1 \times 10^{-4} = 2.5 \times 10^{-6}$ occurrences per year. In this way, the ASP study gives some indication of the credibility limits of PSA studies.

3.2 SPECIFIC NUCLEAR POWER PLANT RISK ANALYSES

Since the historic Rasmussen Report [WASH-1400, 1975], many probabilistic analyses of reactor safety have been performed. The results of a number of these are summarized in Table 5, 6, and 7. The results in Table 5 are point estimates for WASH-1400, three U.S., one U.K., two French, and one German (FRG) reactor. In Table 6 confidence intervals for thirteen U.S. reactors are reported. In Table 7 data on core damage (core melt) is listed for six Swedish reactors.

Table 5 Core Damage Probabilities Per Year from Typical Probabilistic Risk Analysis Studies

Study	Result	Year(s) of Study	Single Plant Frequency per year
WASH-1400	probability of core damage per year	1975	5×10^{-5}
Hirschberg (1990) (pp. 2-50 - 2-54)	Total core damage caused by internal and common cause initiators (mean values)		
Sizewell-B-UK	1.4×10^{-6}	1982	1.4×10^{-6}
Oconee-3-US	5.4×10^{-6}	1980-84	5.4×10^{-6}
Biblis-B-FRG	9.0×10^{-6}	1976-79	9.0×10^{-6}
Calvert Cliffs-1-US	1.3×10^{-4}	1984	1.3×10^{-4}
Seabrook-US	1.7×10^{-4}	1982-84	1.7×10^{-4}
Nuclear News (1990)	core damage probabilities		
CP2-Generic 900, MWE, PWR	Institut de Protection et de Surête Nucleaire, (ISPN) 3.37×10^{-5}	1990	3.37×10^{-5}
Paluel-3	Electricité de France (EDF) 4.7 x 10-6	1990	4.7×10^{-6}
Hanan 1988 NUREG/CR-3245	core damage probabilities	1988	5.6×10^{-6}
CR-3 PRA	SAIC	1987	
Review of PRA	Argonne National Lab.	1988	5% bound : 2.5×10^{-6} Median: 6.9×10^{-5} Mean : 1.1×10^{-4} 95% bound : 2.5×10^{-4}

Table 6 Probability of Core Damage Per Year for 13 U.S. Plants
[Brooks, 1990]

No	Site Name	Vendor	Year(s) of study	Interval (10-90 percentile)
1	AK Nuclear One-1	B & W	1982	1E-5 to 3E-4
2	Big Rock	GE	?	3E-5 to 1E-2
3	Calvert	GE	?	6E-5 to 2E-2
4	Crystal River	B & W	1981	3E-5 to 4E-3
5	Grand Gulf	GE	?	1E-6 to 9E-4
6	Ind. Pt. 2	W4	1982-83	9E-6 to 8E-4
7	Limerick	GE	?	3E-6 to 2E-4
8	Oconee	B & W	?	3E-6 to 3E-3
9	Peach Bottom	GE	1982	1E-5 to 9E-5
10	Sequoyah	W4	?	2E-6 to 2E-3
11	Surry	W3	?	1E-5 to 3E-4
12	Zion	W4	1981	5E-6 to 3E-4
13	Ind. Pt. 3	W4	1982-83	1E-6 to 2E-3

Table 7 Probability of Core Damage Per Year for 6 Swedish Plants
[Hirschberg, 1990, pp. 4-48 to 4-57]

Site Name	Interval 5-95 Percentile x 10^{-6}
Ringhals-1	2.5 - 45
Oskarshamn-1	3.5 - 80
Oskarshamn-3	1.2 - 11.5
Forsmark-3	2.4 - 11.5
Barseback-1	3 - 40
Barseback-2	3 - 40

4.0 BOUNDS ON HUMAN OPERATIONS AND ACCIDENTS

The world's technology depends on the underlying human ability to correctly design, understand the relation between the design and its operating environment, and to operate complex devices and systems. If industrial plants of a complexity approaching that of a nuclear power plant have a certain accident rate, then it is unlikely that nuclear plants can far surpass such risk probabilities. Similarly, the control of a nuclear power plant depends on the abilities of one or more operators. If a failure probability can be derived for certain human control tasks then it may not be likely that nuclear plant controllers will exhibit similar error rates, but it is difficult to justify error rates below the best achieved, or above the worst achieved in any industry. The next section discusses human error achievement in this context.

4.1 ACCIDENTS AND CALAMITIES

One class of industrial catasrophe which may rival a nuclear accident is a severe accident at a liquified petroleum gas storage facility. The severe accident modes involve an explosion or large fire.

Comparing the probability of a core damage accident to other accidents and disasters is difficult for many reasons. (The fact that long term contamination and potential danger to life is greater for core damage is one of them). But again if only order of magnitude comparisons are required then they can be made on the basis of accident occurrence frequencies.

Mazzocchi and Campana [1986] calculate the occurrence frequencies of various accident sequences of a large tank farm of liquified petroleum gases

(LPG), based upon an accident data base listing one hundred (100) accidents between 1933 and 1984. The two most serious accidents were a tank collapse after an external fire (BLEVE-FIREBALL) and the explosion of an air-LPG mixture due to an LPG release from the containment system (UVCE). The analysis resulted in estimates of the yearly occurrence probabilities of 3.6×10^{-4} (UVCE) and 1.0×10^{-4} (BLEVE). The total probability of either event is 4.6×10^{-4}.

Another source of comparative data is transportation accidents. About 70% of airline accidents are caused by errors due to the cockpit or air traffic control personnel, these are mostly human errors [Wiener, 1988, p.265]. For example, aircraft safety is generally reported in terms of fatalities per passenger mile. In Shooman [1990, p.630] typical values of 2×10^{-9} fatalities per passenger mile (for scheduled airlines) are converted into 6.7×10^{-9} (conversion factor $= 6.7 \times 10^{-9}/2 \times 10^{-9} = 3.35$) fatalities per trip mile. In a paper by Evans [1990], the airline fatality rates, [1978-1987], quoted as ranging from a low of 0.01 to a high of 1.7×10^{-9} deaths $\times 10^{-9}$ passenger miles, and the average was 0.55×10^{-9}. If it is assumed that a moderate business traveler covers 10,000 air miles per year and a heavy traveler 250,000 then a lower bound can be computed by assuming the moderate business traveler, the low fatality rate year of 0.01×10^{-9}, and Shooman's conversion factor of 3.35 and obtain: $10^4 \times 1.7 \times 10^{-9} \times 3.35 = 5.7 \times 10^{-5}$. Another lower bound can be obtained by assuming a moderate traveler and a low year with the 3.35 conversion factor or :$10^4 \times 0.01 \times 10^{-9} \times 3.35 = 3.35 \times 10^{-7}$ fatalities/trip. At the high end the heavy traveler and the most dangerous year produces: $2.5 \times 10^5 \times 1.7 \times 10^{-9} \times 3.35 = 14.2 \times 10^{-4}$.

For automobile travel the data and analysis developed by Evans et. al. [1990] can be used. They analyze the effect of several factors on car fatalities and predict a risk ranging from:

(a) "Low-risk", 0.804 driver fatalities per billion miles driven for a 40-year-old, alcohol-free, belted driver travelling on rural interstate roads in a car 700 pounds heavier than average.

(b) "High-risk", 930.8 driver fatalities per billion miles driven for an 18-year-old, intoxicated, unbelted male driver travelling average roads in a car that is 700 pounds lighter than average.

Assuming 10,000 miles of travel per year for both high and low risk drivers, the range becomes 0.8×10^{-5} to 9.3×10^{-3} fatalities per year. In the case of auto accidents, data shows that the human is the main cause of accidents. In Sabey [1980, p.49] data is cited which predicts that the driver and pedestrian were mainly at fault in 95% of the auto accidents studied and that road and environmental factors were contributory factors in 28% of the cases and vehicle features were contributory in 8.5%.

4.2 HUMAN LIMITATIONS

Operators of nuclear power plants try to be perfect; however, they are subject to the same limitations as chemical plant operators, power dispatchers, or airline pilots. Even highly motivated, trained, and alert individuals are subject to errors which are hard to explain and difficult to predict. An example of such an error is airline accidents due to pilot error. Airline accidents are carefully investigated and recorded. If this history starting with 1959 is examined (the first U.S. domestic jet service began Dec. 10, 1958) some remarkable events are uncovered. Airline pilots are skilled, well trained, and careful individuals; however, there are a few accidents which involve aircraft, apparently without mechanical problems, crashing into mountains in good weather. The only conclusion is that the pilot "failed to see the mountain". A study of the number of U.S. aircraft flights from 1976-1988 [World Almanac, 1990, p.151] and extrapolation backward and forward for the 1958-1990 time period yields an estimate of 1.597×10^9 flights (departures). Thus, the ratio of $1/1.597 \times 10^9 = 6.26 \times 10^{-10}$ per flight is an estimate of trained human operation inattention to a control task. If it is assumed that worldwide flights are twice the U.S. volume and count the nine worldwide incidents we obtain $(9/2) \times 6.26 \times 10^{-10} = 2.8 \times 10^{-9}$ per flight. If it is assumed that a typical captain flies 200 flights (departures) per year the range becomes 1.25×10^{-7} to 5.6×10^{-7} inattention crashes/year. This data would indicate that human operators can perform quite well, but not perfectly under these conditions; however, it is doubtful that nuclear power plant operators would have a better error rate in emergency situations regardless of how good or automated the design.

5.0 BOUNDS ON PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR REACTORS

The thrust of this section is to bound from above and below the frequencies or mean time between occurrences of a core damage accident of a nuclear power reactor plant so as to define a possibility space for core damage accidents within the context of the probabilities developed in the proceeding sections. The philosophy to be used is based upon two suggested ideas:

(1) The mean time between core damage occurrences must certainly be less than certain natural phenomena such as the age of the universe or age of the earth, and must be consistent with the best control performance that humans can achieve.

(2) The mean time between core damage occurrences must certainly be greater than that of known unreliable human designs such as intervals between auto repairs or mean life of unmaintained city bridges, or human error rates which culminate in catastrophic accidents, or transportation accidents.

The first idea is supported by the limits of rationality in predictions, and the second is supported by the existing history. Of course, the examples cited in the two above ideas are merely for illustration and more representative events will be used to obtain sharper, more realistic, and believable bounds. The results of the investigation are summarized in Figure 1, and the methods of obtaining these estimates are explained below.

Figure 1 compares eight point estimates of core damage probabilities (see Table 5) with three interval estimates derived from operational experience. The operational experience estimates (of Sec. 3.2) were derived from:

- World Experience - 2/3,135 = 6.38×10^{-4} per year
- ASP Program - mean = 3×10^{-5}, lower 3×10^{-6}, upper 2×10^{-4}
- Nuclear Submarines - bounded by 1 occurrence/2,250 years, and use 1.48×10^{-4} as a point estimate

The ASP data were plotted directly in Figure 1 assuming that the lower and upper values represent a 5-95% uncertainty interval. For the World Experiences, a log-normal model was assumed and the CARP™ program (see Appendix A) was used to determine a 5-95% uncertainty interval, mean, and median values based on the number of events and years of operation. In the case of the nuclear submarine data we assumed one failure to bound the result, and used CARP™ to predict the mean, median and the 5-95% uncertainty bounds. Since no failures were observed, the confidence interval is actually wider as indicated by dotted lines in Figure 1. In addition, the "1/3 value" (1.48×10^{-4}) was plotted as a point estimate.

Figure 2 compares the same three sources of operational data with interval estimates for thirteen different American and five different Swedish plants. The CARP™ program was used to fit a log-normal distribution by specifying the 5 and 95% points (the 10 and 90% points in Table 6 were used) and means and medians were calculated.

6.1 REACTOR CORE DAMAGE PROBABILITIES ANALYSIS vs. EXPERIENCE

A summary of the predicted core damage probabilities gathered in this report appears in Figure 1. The data at the top left of the graph represents six Swedish plants (cf. Table 7). The data for the 5-95 percentile column was used as input to CARP™, a log-normal distribution was fitted to the data for each plant and the mean and variance calculated by the program. Similar computations were performed for the thirteen plants given in Table 6 and these are also plotted in Figure 1. The point estimate data given in Table 5 is plotted in Figure 1. Lastly, the data for three "experience" sources (World Nuclear Subs and ASP) is plotted at the bottom of the graph. The four sets of data cover over four decades.

Note that two estimates, point and interval were plotted for Calvert Cliffs and Oconee. The interval estimates were from the data in Table 6, whereas the point estimates were from the data of Table 5. Clearly, the two estimates are different since the point estimate does not correspond to either the mean or variance. During the lifetime of a nuclear power plant it is not uncommon for there to have been several safety estimates, PSAs performed. Typically, new estimates are performed if: (1) the technology for performing a PSA has significantly changed, (2) there have been significant changes to the plant, (3) generic data which was originally used in an estimate can now be replaced by specific data gathered on plant operation. As an example, as veterans of the development of PRA core damage estimates are well aware, there have been multiple studies performed for individual plant evaluation. Some of these were performed for the NRC (such as the IREP, NREP and ASEP studies), and others have been performed for the individual licensees and their contractors (such as the ongoing IPE activities). Some of the estimates have been derived from more generic models (ASEP) and others have been plant specific. Some of the estimates have been updated and improved as a result of changes in design or operating procedures. Finally, some estimates have been developed for operation only, others for operation and shut-down (the recent French study for example), and some have limited their analysis to external internal events (such as internal fires and floods) and others have included all external events (flood, earthquake, etc.). The multiplicity of estimates available makes the understanding of what an individual core damage number includes difficult without detailed notation. For this reason, detailed comparisons of estimates are not possible without detailed explanations for each instance cited. However, these detailed explanations would be inappropriate in a general paper of this type, so no attempt has been made to distinguish individual core damage estimates, nor is one required. This approach is consistent with the intent of this paper, which is not to establish exact core damage numbers, but rather to explore the extent of the "possibility" space for core damage probability estimates. However, for those interested, a more precise review of the core damage estimates for 21 PRAs, which predicts a similar range, has been developed elsewhere [Garrick, 1989].

The large amount of data in Figure 1 is easier to interpret if it is condensed. The CARP™ program was used to aggregate (see Appendix A) the interval estimates of Swedish, American (Table 6), and experience data, which are plotted in Figure 2. Since aggregation requires interval data, and an aggregation of the point estimates in Table 1 was also desired, the data was first transformed into interval data using the following assumption. The error factors (see Appendix A) were computed for the plants of Table 6, sorted in alphabetical order, yielding: (5.5, 18.3, 18.3, 11.5, 30.0, 44.7, 9.4, 8.2, 31.6, 3.0, 31.6, 5.5, 7.7) with a mean of 17.3. This mean error factor was applied to each of the point estimates (the point estimate was assumed to be a mean) and this data was used as input to CARP™ to produce interval estimates (Note: there is a continuing controversy as to whether the estimates developed in WASH-1400 should be treated as medians or means, but here we consider this estimate as a mean).

To further analyze the data, the estimates in Figure 1 have been segregated into three categories: (1) Experience, (2) American plants (both interval and transformed point estimates), and (3) European plants (6 Swedish, German/Biblis, British/Sizewell, and French/Paluel 3 and CP2). This data appears in Figure 3.

6.2 BOUNDING FROM ABOVE AND BELOW

The main theme of this paper was to bound core damage estimates from above and below. Throughout this paper, a data set based on historical and recorded accident experience has been cited; this data is shown in Figure 4. If these various data points are then aggregated into common categories, such as "historical events", "pilot error", and "non-nuclear accidents", it becomes easier to compare them with the nuclear power core melt probability estimates shown earlier in Figure 3. Such a comparison with various bounding events has been constructed and is shown as Figure 5. This figure broadly establishes the range of credibility for industry accidents in that it clearly separates complex facility accident and operator error probabilities from the combined probability range of astronomical, geological, and biological events in history. The range, then, from the lower bound of "pilot errors" to the upper bound of "non-nuclear accidents" can be defined as the "possibility" space for the estimation of nuclear plant core damage probability per year. Within this regime, given the data available, it should also be possible to indicate regions of credibility for nuclear core damage occurrence. Such an attempt has been made in the following section.

6.3 THE "QUESTIONABLE" REGIONS

Philosophically and practically, a nuclear reactor core damage estimate which leads to an occurrence frequency as small as a star going nova should be categorized as untenable. Further, operator actions which have direct core melt impact (for example, failure to go to recirculation) should have probabilities limited by the best observed human performance, such as those documented for commercial airline pilots. Thus, predictions of nuclear plant core melt which require human error frequencies of the order of 10^{-7} are clearly incredible because the historical data set required to establish such human performance is non-existent, at least to the knowledge of the authors.

Similarly, the upper end bound can be developed by examining transportation risks which are voluntary risks. It is assumed that an individual will not want to assume the involuntary risk of a core damage accident unless it is much less than a traffic fatality. Such an upper bound threshold has been labeled clearly unacceptable. Also, since nuclear power plants have a high profile and are in the "public eye", it is unlikely that the risk of a core damage accident would be accepted unless it were much lower than other types of industrial accidents (cf. LPG accident in Figure 4).

The conclusion that could be proposed is that current core damage estimates in the range of 10^{-3} to 10^{-5} are credible and are about the same range as accident estimates. PSA estimates in the range of 10^{-5} to 10^{-6} strain credibility because they challenge the historical bounds of the best human performance, and estimates below 10^{-7} are clearly incredible. Estimates in the decade 10^{-5} to 10^{-6}, while not ruled out by the arguments given in this paper, should be closely examined and checked since they begin to strain the bounds of the probability space.

7.0 CONCLUSIONS

This paper has been developed with the intention of establishing an international dialogue within the nuclear power community concerning the bounds of credibility to be placed upon probabilistic estimates of the risk of core damage accidents for both current and future designs. The admittedly heuristic approach presented here supports a proposed possibility space for such estimates between 10^{-3} to 10^{-6}. This range is consistent with current core damage risk estimates and with the currently suggested 10^{-4} NRC goal and the suggested 10^{-5} industry goal for new "inherently safe" designs. While the authors certainly make no claim that this preliminary investigation should be accepted, it is felt that the issue is of sufficient importance so that further investigation is warranted. This research should be directed not only at better defining the possibility space, but also to place requirements on the analysis performed to support estimates which begin to stretch the bounds of credibility. If a consensus of what is a reasonable space of possibilities can be established, along with a corresponding list of PSA requirements which must be satisfied to support credible predictions within the space proposed, then it is believed that better communication concerning PSA results will be established. Once this is accomplished, the developers of PSA and the associated regulatory bodies (who must review their results) will all have a better understanding of the limits of prediction credibility. This understanding should lead to more confidence in the developed predictions by all concerned; including the public in general.

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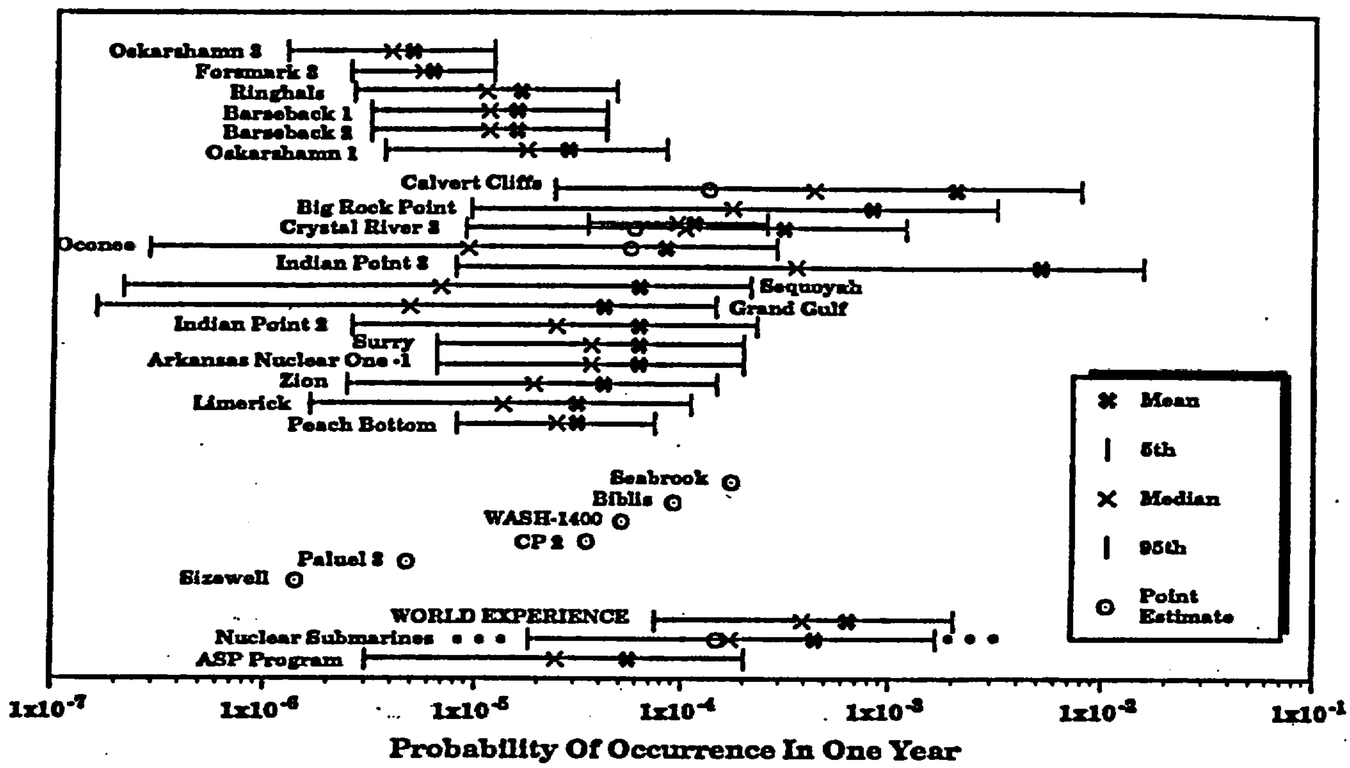

Figure 1. A comparison of various interval estimates of core damage probabilities for nuclear power reactors.

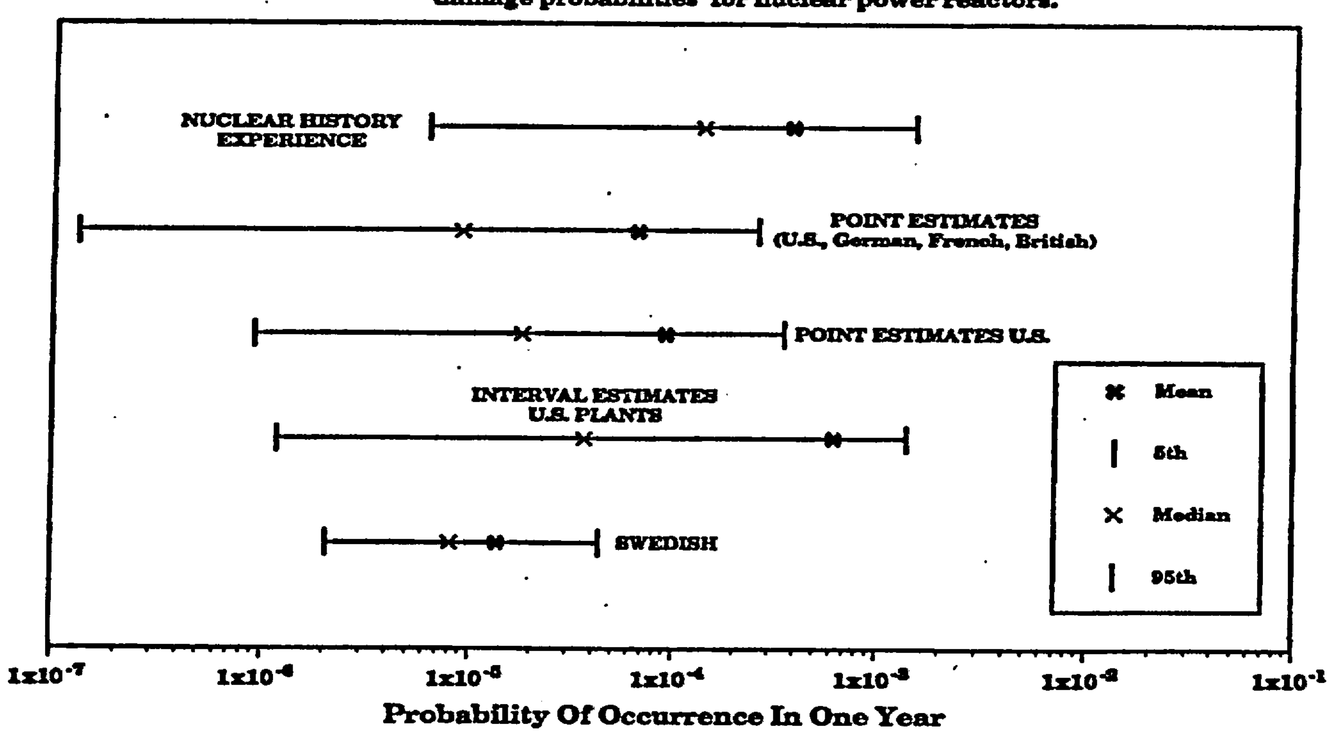

Figure 2. Aggregation of the Estimates Given in Figure 1.

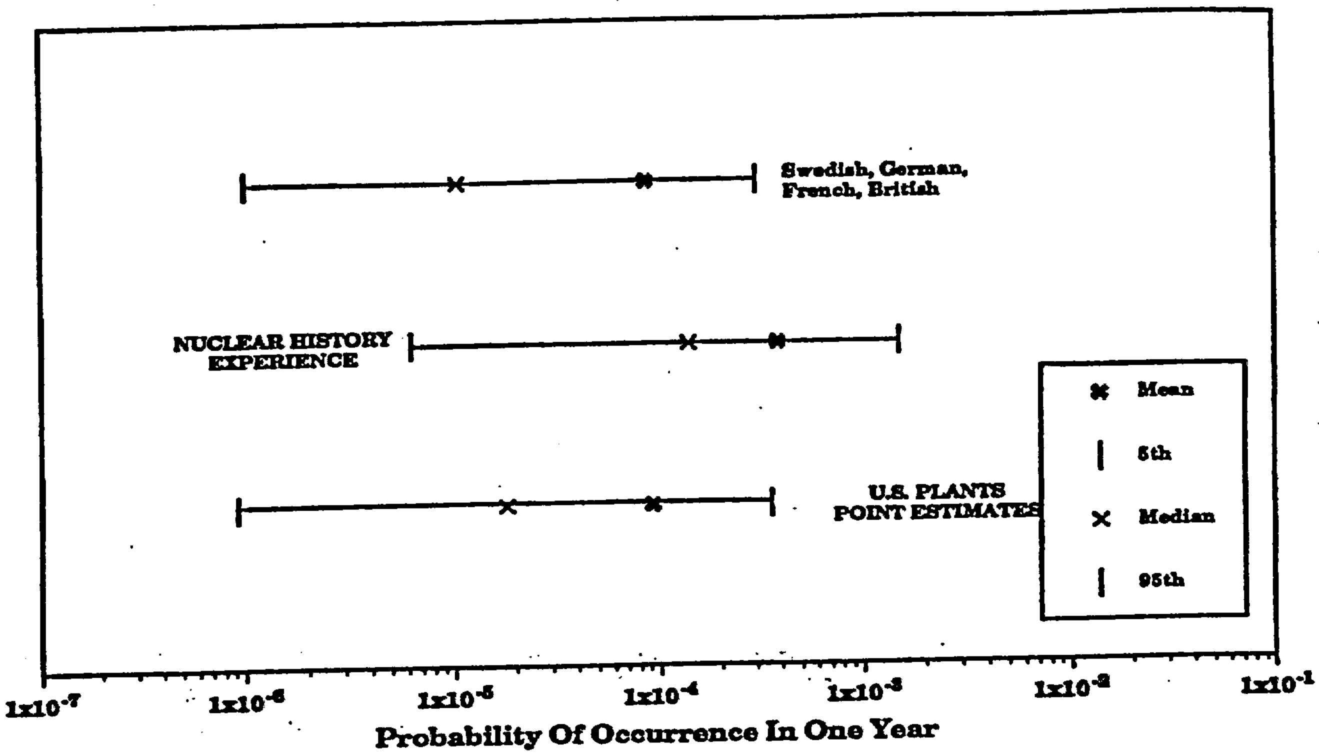

Figure 3. Aggregation of the Estimates Given in Figure 1 to Show Difference Between U.S. and European Plants.

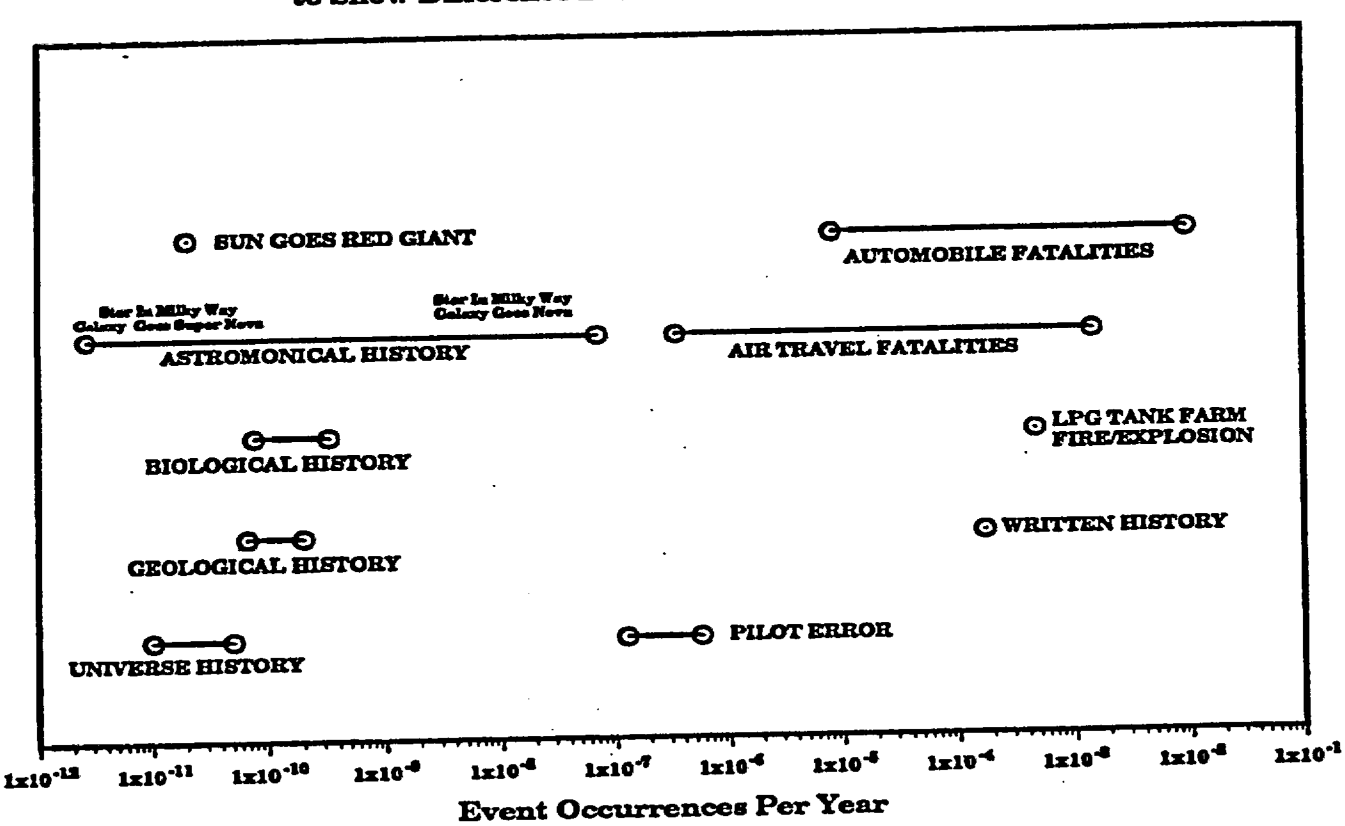

Figure 4. Occurrence Rates for Various "Historical" Events and Various Accidents.

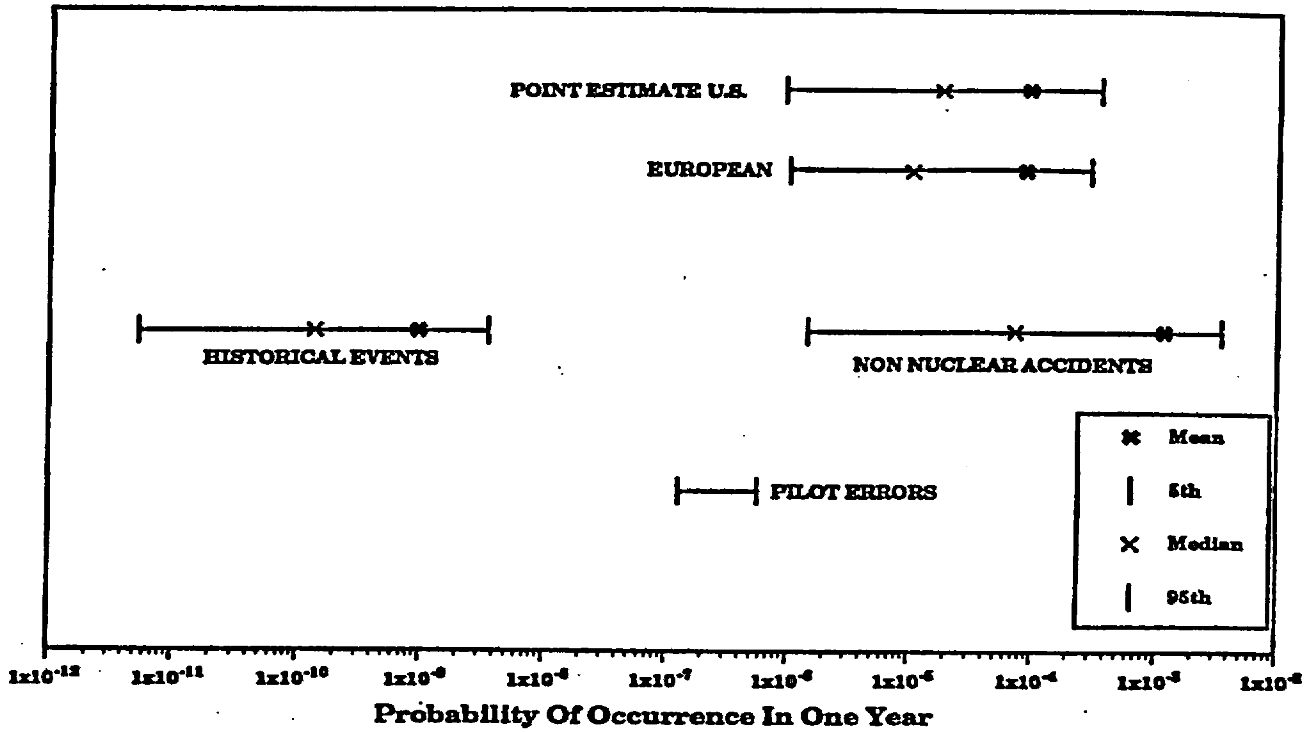

Figure 5. Aggregate Values Compared for Power Plant Core Damage With Various Bounding Events.

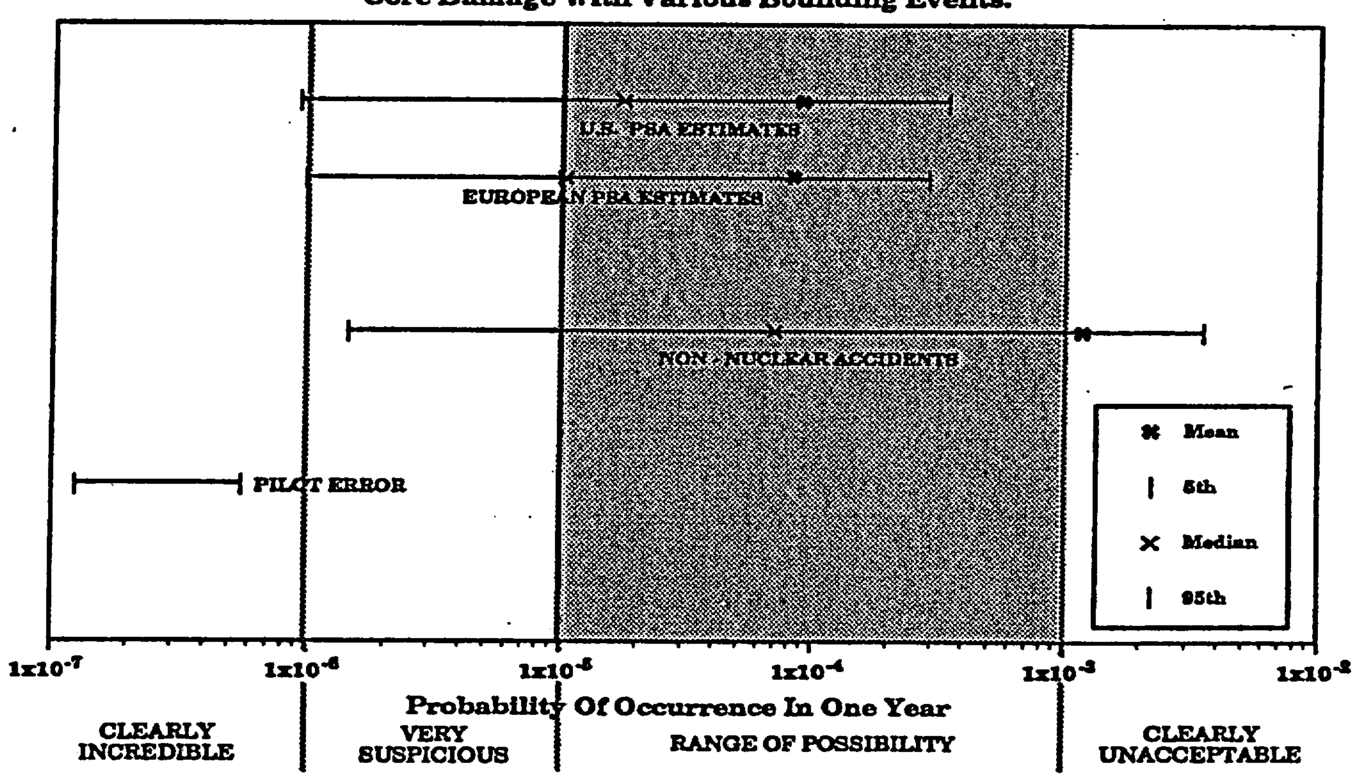

Figure 6. Possibility Space For Core Damage PSA Estimates.

APPENDIX A
COMPUTERIZED AGGREGATION OF

RELIABILITY PARAMETERS - CARP™

The CARP™ software is an SAIC program which is a useful tool for reliability and risk analysis (DeMoss, 1990). There is considerable flexibility in the program; however, we will only describe the features which were used to analyze the data used in this paper. The program deals with *n* random variables (input distributions) where Λ_1, Λ_2,..... Λ_i, ...,Λ_n represent the input distributions. In general, each random variable is modeled by a log-normal distribution. (see Dougherty and Fragola, pp. 112-116) The distribution can be determined by specifying any two parameters from the following set:

1. Mean
2. Median
3. Error Factor - the ratio - 95 percentile/5 percentile (see Dougherty and Fragola, 1988)
4. Variance
5. Upper and Lower Percentiles - (generally 95 and 5)

The program also implements the aggregation method which is used to combine multiple data sources (with their own confidence intervals) into a composite confidence interval. (Dougherty and Fragola, 1988, p.50, Stone 1961, Walker 1968.)

CARP™ uses the percentiles from the input cumulative distribution functions (CDFs) to arrive at the percentiles of the aggregate distribution. Before aggregation, the input distributions have been fit to log-normal forms. The method would work for any CDF with a closed form solution or numeric approximation, but since log-normal is the most common format for published generic data, CARP™ has only been programmed to handle the log-normal distribution. CARP™ will iteratively determine a percentile of the aggregate distribution using a recursive scheme with each input CDF. A Newton-Raphsòn type iteration is used to solve the following equation for the unknown λ as follows:

$$\Pr(\Lambda \le \lambda) = \sum_{i=1}^{n} \omega_i \Pr(\Lambda_i \le \lambda)$$

where:

$\Pr(\Lambda \le 1)$ - The percentile of interest in the aggregate distribution. CARP™ will calculate the 5th, 50th, and 95th percentiles

n - The number of input distributions

Λ_i - The random variables of the input distributions

λ - The failure rate occurring at the percentile of interest in the aggregate distribution

ω_i - The weight of a distribution. We usually use equal weights making this quantity equal 1/n.

The resultant aggregate can be then fit to a log-normal distribution if desired.

Original Contributions

Immediate Medical Consequences of Nuclear Accidents

Lessons From Chernobyl

Robert Peter Gale, MD, PhD

The immediate medical response to the nuclear accident at the Chernobyl nuclear power station involved containment of the radioactivity and evacuation of the nearby population. The next step consisted of assessment of the radiation dose received by individuals, based on biological dosimetry, and treatment of those exposed. Medical care involved treatment of skin burns; measures to support bone marrow failure, gastrointestinal tract injury, and other organ damage (ie, infection prophylaxis and transfusions) for those with lower radiation dose exposure; and bone marrow transplantation for those exposed to a high dose of radiation. At Chernobyl, two victims died immediately and 29 died of radiation or thermal injuries in the next three months. The remaining victims of the accident are currently well. A nuclear accident anywhere is a nuclear accident everywhere. Prevention and cooperation in response to these accidents are essential goals.

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ON APRIL 26, 1986, the world's most serious nuclear accident occurred at the Chernobyl nuclear power station in the Soviet Union.[1] Chernobyl attracted and continues to attract worldwide attention. Underlying this response are several important factors, including a concern for the immediate and long-term biomedical consequences of the accident, uncertainty regarding the future of nuclear energy, and implications for the nonpeaceful uses of nuclear energy.

Details regarding the accident at Reactor 4 at the Chernobyl nuclear power station have been reported by Soviet scientists at a meeting of the International Atomic Energy Agency.[1] There have also been several independent analyses of these data by non-Soviet sources, including a report from the International Atomic Energy Agency,[2] a report from the US Department of Energy,[3] and a recent scientific review.[4] In general, conclusions of these studies are in agreement; hence, reactor design or the specifics of the accident shall not be considered further in this report.

The present report focuses primarily on the immediate and short-term biomedical consequences of the Chernobyl accident. The immediate medical response involved five phases: assessment, containment, reduction of exposure to individuals at risk, dosimetry of exposed individuals, and medical interventions. Assessment relates to source-term parameters such as the quantity of radioactivity released and its physical-chemical form. At Chernobyl, approximately 185×10^{10} MBq (50 MCi), or 3% to 5%, of the fuel inventory was released along with an equal amount of radioactivity in the form of noble gases.[1-4] Twenty-five percent of the release occurred instantaneously and the remainder over approximately ten days.[1] Following the violent disassembly of the reactor core, a radioactive plume was ejected to a height of up to 10 km above the reactor and was driven initially to the northwest. Forty-five thousand persons were evacuated from the city of Pripyat, situated 2 to 4 km from the reactor, within 36 hours of the accident. During the next two weeks, approximately 90 000 additional individuals were evacuated from a 30-km radial zone surrounding the reactor. The average dose received by these evacuees was 0.12 Sv (12 rem). Twenty-four thousand of the 135 000 evacuees received an average dose of 0.45 Sv (45 rem), with the remainder receiving a dose ranging from 0.03 to 0.06 Sv (3 to 6 rem).[1]

The Chernobyl accident raised several important considerations with regard to the evacuation of populations at risk. First, it indicated that immediate evacuation is not always desirable. In the case of Pripyat, evacuation was postponed until buses could be assembled, escape routes selected to avoid the path of the radioactive plume, and a polymer film sprayed on ground surfaces to reduce the likelihood of inhalation of radioactive dust. The efficacy of this strategy is indicated by the fact that the population of Pripyat received a lower average radiation dose than individuals living at considerably greater distance from the power station. For example, the average dose in Pripyat was 0.03 Sv (3 rem) vs 0.45 Sv (45 rem) for individuals living 3 to 15 km from the reactor.[1] The situation might, however, have been considerably different had the radioactive plume been ejected horizontally rather than vertically or had the prevailing winds been directed toward Pripyat. These events emphasize the need for flexibility in emergency planning and reevaluation of evacuation guidelines based on careful review of the Soviet experience at Chernobyl.

ASSESSMENT OF RADIATION DOSE

Appropriate medical treatment of the most severely affected individuals requires an accurate assessment of radiation dose. There are two basic approaches, physical and biological dosimetry. Physical dosimetry, such as the use of environmental monitoring devices or individual radiation meters or badges, was of limited value at Cher-

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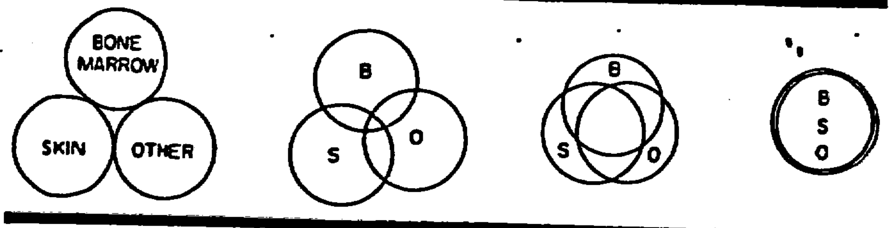

Fig 1.—Distribution of toxic effects in nuclear accidents. Circles indicate toxic effects to bone marrow (B), skin (S), and other (O) organs. On left, distributions of toxic effects are discrete; on right, they are coincident.

nobyl. Most environmental monitoring devices were either destroyed or unrecoverable. The individual monitoring devices were not designed for these levels of radiation and were destroyed or lost as a consequence of circumstances associated with this accident. This situation presumably can be improved in the future. Remote monitoring of dosimeters is standard at many facilities, and special badges and thermoluminescent dosimeters are available for accidents.

Because of the limitations of physical dosimetry, biological dosimetry is used for dose assessment.[5-8] Biological dosimetry at Chernobyl involved serial measurement of levels of lymphocytes and granulocytes in the blood and analysis of dicentric chromosomes in spontaneously dividing and phytohemagglutinin-stimulated blood- and bone marrow-derived hematopoietic cells. The interval from radiation exposure to onset of nausea or emesis was also considered. Calculation of dose was based on data relating these parameters to dose in prior radiation accidents.[8] Using these variables, it was possible to estimate the dose of radiation received. Analysis of distribution of cytogenetic abnormalities suggested uniform whole-body exposure in most instances. Analysis of sodium 24 indicated the absence of a detectable neutron component in the radiation exposure.

Biological dosimetry is effective but it demands considerable medical and technical expertise and resources. There are also limitations to the accuracy of biological dosimetry in complex accidents in which thermal and chemical injuries also affect biological parameters, as do systemic infections. For example, extensive thermal burns are associated with decreased levels of lymphocytes and granulocytes even in the absence of radiation exposure. Analysis of dicentric chromosomes or other cytogenetic abnormalities can overcome this limitation. It is likely that biological dosimetry will prove more useful than physical measurements in major radiation accidents of the Chernobyl type.

THE MEDICAL RESPONSE

A range of medical interventions is required to respond to radiation accidents. The medical response to each accident will differ based on the spectrum of injuries and concordance of toxic effects. Different types of reactor accidents can be predicted to result in a different spectrum of injuries. For example, an accident in a graphite-moderated reactor, such as the one in Chernobyl, can result in an intense graphite fire. In this circumstance, individuals who are irradiated also receive substantial thermal injuries. This combination of injuries complicates the nature and effectiveness of medical interventions. An accident of similar magnitude in a light-water reactor is very likely to result in a different spectrum of injuries and concordant toxic effects with a lower probability of thermal injury. A conceptual representation of the spectrum of overlapping toxic effects following a nuclear accident is shown in Fig 1. The consequence of these considerations shows that it is not possible to devise a specific medical plan for all accidents or to draw conclusions about the value of different medical interventions from a single accident. Physicians must be prepared to respond to the full range of potential injuries, thereby modifying the medical response to accident conditions.

The major elements in the medical response to a radiation accident involve treatment of skin burns and measures designed to correct or reverse bone marrow failure and gastrointestinal tract injury. Damage to the lungs, the liver, and other organs and tissues must also be considered. Analysis of the effects of radiation on these tissues and organs is complex, a situation underscored by the Chernobyl accident. Tissue damage can occur by external radiation from both beta and gamma sources. Internal radiation via inhalation or ingestion can also play a role. In addition to radiation dose, other important factors include dose rate, fractionation, and distribution. Supportive measures are essential, such as protective isolation; gastrointestinal tract decontamination; antibacterial, antifungal, and antiviral therapy; and transfusion of blood products.

Infection Prophylaxis

Most victims of the Chernobyl accident received a dose of whole-body radiation believed to be compatible with bone marrow recovery in the context of intensive supportive measures. Consequently, antimicrobial, antifungal, and antiviral drugs were extensively used both therapeutically and prophylactically. Patients were kept in isolation and some were maintained in relatively sterile laminar airflow environments. Oral antibiotics were given to modify the endogenous gastrointestinal tract flora and to decrease the likelihood of infection. Systemic antibiotics were used in febrile patients with granulocytopenia. Radiation-induced activation of herpes simplex virus was a major problem in most individuals; this complication responded to treatment with acyclovir.

Transfusion Support

Extensive transfusion support was also necessary. This included transfusion of red blood cells and platelets. The latter were obtained from multiple random donors as well as by thrombocytapheresis from single donors using sophisticated continuous- and intermittent-flow blood cell separators. Some patients received platelets through a novel approach, namely, transfusion of autologous cryopreserved platelets (A. E. Baranov, MD, A. K. Guskova, MD, et al, unpublished data, 1987).

BONE MARROW TRANSPLANTATION

In some instances the dose of radiation received may be associated with a high likelihood of irreversible bone marrow failure despite intensive supportive care; in this circumstance, other interventions, such as bone marrow transplantation, should be considered. A detailed discussion of the objectives of transplantation in this setting is beyond the scope of this report. It is important to consider, however, that these objectives are complex. At low doses of whole-body radiation, transplants of histoincompatible hematopoietic stem cells are typically rejected without a deleterious effect. At midlethal doses, transplants of histoincompatible hematopoietic stem cells are associated with decreased survival in mice but not in dogs or monkeys.[9-11] In mice this adverse effect, termed the *midzone effect*, is associated with graft rejection; the mechanism is poorly understood. At higher doses of whole-body radiation, transplants of histoincompatible hema-

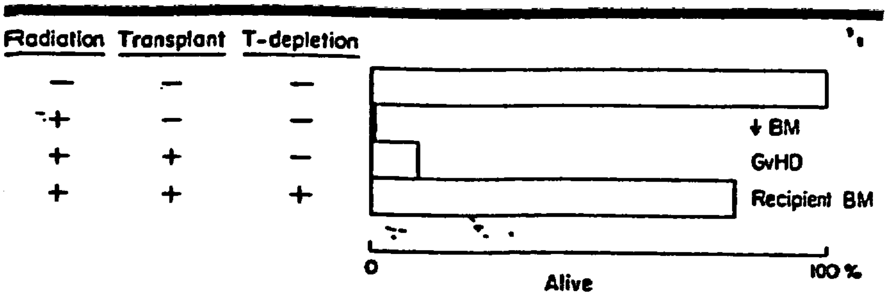

Fig 2.—Effects of transplants in H-2 incompatible mice. *Radiation* indicates whole-body irradiation in excess of 9 to 10 Gy (900 to 1000 rad); *transplant*, H-2 incompatible bone marrow transplant; *T-depletion*, T-lymphocyte depletion of transplant in vitro; ↓ *BM*, death from bone marrow failure; *GvHD*, death from graft-vs-host disease; and *recipient BM*, recovery of recipient hematopoiesis.

topoietic stem cells can improve survival by several mechanisms. In some instances, temporary engraftment permits recovery of endogenous hematopoietic stem cells[11,12] (Y. Reisner, PhD, J. Lapidot, MD, oral communication, 1987). This effect is only observed in histoincompatible transplants when T cells are removed from the bone marrow inoculum, since failure to do so results in fatal graft-vs-host disease.

If the bone marrow is irreversibly destroyed by radiation, sustained hematopoietic engraftment is a prerequisite of survival. In humans, sustained engraftment has been achieved in several circumstances following transplantation of either histocompatible or histoincompatible hematopoietic stem cells. The closest model to that of a radiation accident is transplantation for aplastic anemia.[13,14] In this situation, recipients are usually conditioned for transplantation with high-dose chemotherapy with or without whole-body or total lymphoid irradiation followed by transplantation of histocompatible hematopoietic stem cells from a related donor. Survival exceeding 60% to 80% has been reported in some series. Sustained engraftment has also been reported following transplantation of histoincompatible hematopoietic stem cells from related donors or following histocompatible transplantation from unrelated donors.

The consideration of candidates to receive bone marrow transplants is complex. First, it is necessary to identify those individuals in whom the dose of whole-body radiation is associated with a high risk of death from bone marrow failure. It is also necessary to attempt to exclude those individuals likely to die of nonhematopoietic toxic effects, such as skin burns or pulmonary damage. This is not always easily accomplished in an accident setting. Next, it is necessary to perform HLA typing of the potential transplant candidates and their relatives to determine donor availability. This process is complicated by radiation-induced lymphocytopenia since lymphocytes are usually used for HLA typing. If suitable histocompatible donors are identified, the possibility of transplantation can be considered. In individuals who cannot be HLA-typed or for whom no donors are available, there are other alternatives. One alternative is to search for a histocompatible, unrelated, volunteer donor. There are now several large, HLA-typed, volunteer donor pools in the United States and Europe. The feasibility of this approach was tested at Chernobyl when three potential unrelated donors for two victims were identified within three to four days.

Another alternative for individuals who lack histocompatible donors is the use of fetal liver cells.[15,16] During the second trimester of gestation, fetal liver is a rich source of hematopoietic stem cells. Since the immune system is not fully developed at this time, histoincompatible fetal liver cells are less likely to cause graft-vs-host disease than comparably mismatched adult bone marrow cells. Transplantation of fetal liver cells has been successful in mice and dogs. There have been reports of its effectiveness in humans, but these data are less convincing.

The complex relationship between transplantation and survival is indicated schematically in Fig 2, which is based on extensive studies in mice. In each individual the potential benefits and risks of each medical intervention, including transplantation, must be carefully considered because they differ among accidents with different spectra of injury, as well as among individuals within a given accident.

FATALITIES

In view of the magnitude of the Chernobyl accident, one might have expected a substantial immediate loss of life; fortunately, this did not occur. Two individuals were killed instantaneously; 500 individuals were hospitalized, approximately 200 patients in Kiev and 300 patients—including the most severely affected—in Moscow. More than 100 individuals received a dose in excess of 1 Gy (100 rad), and more than 35 persons received a dose exceeding 5 Gy (500 rad). The proposed 50% lethal dose for humans within 60 days is 4.5 Gy (450 rad).[17,18] Most individuals received the aforementioned supportive measures, including 13 patients who received bone marrow transplants and six others who received infusions of fetal liver cells. Twenty-nine individuals died of radiation- and/or thermal-induced injuries during the next three months, including 11 bone marrow transplant recipients and the six fetal liver cell recipients. Most of these deaths were due to skin burns or damage to other organs, such as the gastrointestinal tract or lungs. The remaining individuals, over 90%, are well and most have been discharged from hospitals. One should not be complacent about this outcome since a number of factors beyond physician control served to reduce the number of immediate injuries. These factors include the nature of the explosion that directed the radioactive plume to a height of approximately 10 km, the prevailing and meteorologic conditions that drove the plume away from Pripyat and Kiev, and the fact that the accident occurred at a time when most people were indoors and when there was no local precipitation. It is also of considerable importance that the release occurred during a protracted interval of nine to ten days rather than instantaneously.

OUTCOME

What conclusions can be drawn regarding the immediate medical response to nuclear accidents? In some regards, these interventions were quite successful despite the complexities indicated. Intensive supportive care was associated with a high rate of survival in most individuals receiving less than 6 Gy (600 rad) of whole-body radiation. Of course, it is impossible to know what proportion of these individuals would have survived if no treatment were given. A critical answer to this question would require a prospective randomized trial. This is unlikely. Nevertheless, it is highly likely that such measures as the use of systemic antibiotics, gastrointestinal tract decontamination, and platelet transfusions can save lives.

It is more difficult to evaluate the efficacy of these measures in individuals receiving more than 6 Gy (600 rad) of whole-body radiation in the absence of a controlled trial. Several of these patients received transplants; two survived. Although data from mice models suggest that autologous bone marrow recovery in these patients might be related to their T-cell–depleted transplants, this is uncertain since some individuals receiving more than 6 Gy (600 rad) of whole-body radiation without transplants also survived. Regardless of the interpretation of these cases, it is certain that bone marrow transplantation can only save a small proportion of victims of radiation accidents; irreversible damage to other organs is likely to limit the success of this approach. Also, many individuals may not have a suitable donor. Perhaps this limitation will be overcome by the availability of international HLA-typed volunteer donor registries or by progress in the use of partially histocompatible related donors. Recent advances in the ability to remove T lymphocytes from the graft and thereby to decrease or prevent graft-vs-host disease may be important in this regard.[9,20] The decision as to whether a bone marrow transplantation is indicated, like most therapeutic decisions in medicine, requires a critical analysis of the potential benefits and risks in an individual patient at a specific accident. There is no adequate substitute in this complex and fallible process, but it is a task that physicians perform daily.

What has been learned from the Chernobyl accident regarding immediate medical effects? First, nuclear accidents are far more complex than imagined. Investigators are just now beginning appropriate computer simulations of accidents of the Chernobyl magnitude and complexity. Second, immediate medical interventions vary in their effectiveness and limitations. Third, humans can survive considerably greater exposure to radiation than anticipated, which is not surprising in view of recent advances in supportive care, antibiotics, and transfusions.

FUTURE DIRECTIONS

What of future directions? Although transplantation of hematopoietic stem cells can facilitate bone marrow recovery, this procedure is associated with such complications as graft-vs-host disease, interstitial pneumonitis, and iatrogenic immune suppression. Permanent replacement of the bone marrow is probably not required in individuals receiving less than 8 Gy (800 rad) of whole-body radiation. In these individuals, it may be possible to expedite bone marrow recovery by administration of molecularly cloned hematopoietic growth factors.[21] Preliminary data in mice and monkeys suggest that this approach may be successful. Individuals exposed to more than 8 Gy (800 rad) may not recover but may still have sufficient intact immunity to reject transplants; perhaps they should receive additional immune suppression with drugs or irradiation. Preliminary data in dogs suggest that this approach can be successful[11]; moreover, T-cell–depleted HLA-mismatched transplants may be useful in some circumstances. Should nuclear industry workers have their bone marrow cryopreserved? Probably not. The associated morbidity and mortality is likely to be considered higher than the likelihood of benefit in an individual. Should workers be HLA-typed? Possibly. One major problem at Chernobyl was accurate HLA typing because of radiation-induced lymphopenia. This is a low-risk, low-cost investment and would have the added benefit of providing a large pool of highly motivated HLA-typed individuals who might be willing to donate bone marrow or platelets for individuals with leukemia or aplastic anemia.

As in all of medicine, prevention is superior to treatment. Nuclear accidents must be prevented. Unfortunately, this is not possible with current reactor technology. Consequently, "inherently safe" fission reactors or fusion reactors must be our long-term goals.

We should take one other lesson from Chernobyl: a nuclear accident anywhere is a nuclear accident everywhere. We live on a small planet in the context of nuclear energy, and Americans and Soviets are likely to do better with these sophisticated and potentially dangerous technologies by working together rather than working against each other. Finally, the medical response to Chernobyl, involving physicians and scientists from 20 nations and several million dollars in supplies and equipment, would appear trivial in the context of a nuclear war.[22] Instead of hundreds of casualties one would expect millions. An adequate medical response to the intentional use of less than 10% of the Soviet or American nuclear arsenals is not possible. As in nuclear reactor accidents, an ounce of prevention is worth a pound of cure.

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2

Potential Characteristics of Severe Reactor Accidents at Nuclear Plants

Gordon R. Thompson

Emergency planning is a necessary adjunct of nuclear reactor operation, because it is generally recognized that reactors have the potential to undergo accidents that release radioactive materials to the environment. Most prominently, modern commercial reactors have the potential to release rapidly (within hours from the initiation of an accident) a highly radioactive plume to the surrounding atmosphere.

This chapter reviews the relevant characteristics that might be displayed by a severe accident at the Three Mile Island (TMI) plant. The plant features two reactors of almost identical design. TMI Unit 2, which experienced a partial core-melt accident in 1979, will not be returned to service. Thus, the severe accident potential of the site arises primarily from TMI Unit 1; this chapter focusses on that reactor.

Under the sponsorship of the licensee, a probabilistic risk analysis (PRA) has been performed for TMI Unit 1 (PLG 1987). That PRA is one of the sources upon which the conclusions of this chapter are based. Other sources of information were, however, equally important, as explained below.

It should be noted that the terms "severe accident" and "core-melt accident" are used interchangeably in this chapter. The basis for that use is explained below. Also, it should be noted that despite the potential for spent-fuel storage accidents at nuclear plants, this chapter addresses only reactor accidents.

To adopt an emergency plan is to assume that severe accidents can occur. Nevertheless, considerable controversy prevails about the likelihood of such accidents. Accordingly, this chapter reviews the state of knowledge about severe accident probability. Also, controversy exists about the degree of severity that accidents can be expected to display. It has been argued, for example, that most core-melt accidents will provide ample warning (that is, many hours of warning) before the onset of a large release. Accordingly, this chapter reviews the state of scientific knowledge relevant to the estimation of severe accident behavior. These issues are of more than scientific interest because the level of effort that a society devotes to emergency planning will reflect judgments both as to the severity and the likelihood of possible accidents.

During a severe accident, radioactivity can enter the environment through water pathways (contamination of rivers, groundwater, etc.). The focus here, however, is on airborne releases, which can spread radioactivity rapidly over large land areas.

Accident Characteristics and Emergency Planning

The probability and the severity of potential accidents are important issues in the context of emergency planning and in the worldwide controversy about nuclear power. The following discussion addresses the ways in which expectations about these matters affect emergency planning.

Accident Probability

To put a discussion of accident probability in perspective, note the following facts about the nuclear power industry, as of the end of 1986. At that time, 99 reactors were operating in the United States, with 21 under construction. Total operating experience in the United States was 1050 reactor-years. Worldwide, 397 reactors were operational, with 133 under construction. Total worldwide operating experience was 4210 reactor-years (IAEA 1987, 65).

For purposes of illustration, imagine that the probability of a core-melt accident at any reactor is 1×10^{-3} per reactor-year. The expected number of such accidents through 1988 would then be about one in the United States and five worldwide. In fact, the United States has experienced one partial core-melt accident (at TMI Unit 2 in 1979), and two severe accidents (TMI and Chernobyl) have occurred worldwide.

Confining the discussion to the United States and a assuming stable reactor population of 100 units, a core-melt frequency of 1×10^{-3} per reactor-year would imply that one core-melt would be expected about every ten years. Thus, full-scale implementation of an emergency plan might be required, somewhere in the United States, once per decade. In fact, no such implementation has occurred since the present emergency planning regulations were promulgated in 1980.

The probability of a core-melt accident at any given reactor is highly uncertain. Indeed, this probability will vary over time, according to factors such as the age of the reactor and the quality of its maintenance and operation. Also, some accident-initiating factors—especially sabotage—are not stochastic. Thus, one cannot speak of a fixed core-melt probability even for a single reactor. Over a population of reactors, there are many more sources of uncertainty and variability in the probability of core-melt.

Current emergency planning practice calls for initiation of emergency response when core-melt has occurred or is anticipated. More precisely, the emergency-planning guidance provided in the document NUREG-0654 calls for declaration of a General Emergency when "events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity" (NRC and FEMA 1980, page 1-16).

More recently, recognition has grown that core-melt, by itself, creates the potential for loss of containment integrity. Thus, the occurrence or imminence of core-melt is becoming the accepted "trigger" for full-scale implementation of an emergency plan. Indeed, a philosophy is evolving of "precautionary" response, whereby emergency responses will be initiated in anticipation of a core-melt, even at the risk of a false alarm.

Thus, present regulations and practices are based on the implicit assumption of a significant probability of core-melt accidents. Resources and attention are currently devoted to emergency planning for core-melt situations. As this chapter will demonstrate, the assumption of a significant core-melt probability is sound.

Source-Term Parameters

The phrase "source term" is often used loosely to refer to the magnitude and isotopic composition of an atmospheric release. A more accurate usage includes all the following factors:

- the magnitude and isotopic composition of the release;
- the development of the release over time;
- the location of the point(s) of emission; and
- the thermal energy of the plume.

The gross magnitude of the release will affect the scale of doses that exposed persons may receive. At a slightly finer scale of detail, the isotopic composition of the release will affect the way human exposure is distributed over the inhalation, ingestion, and external-exposure pathways. For example, exposure to a cloud containing radioiodines will lead to a thyroid dose via the inhalation pathway.

Release timing is particularly important for emergency response. If a long warning period precedes the release, for example, more complete implementation of emergency response will be possible. It should be noted that all the characteristics of a release may be time-dependent. For example, a release might begin with a puff of relatively volatile radionuclides, perhaps accompanied by substantial thermal energy. It might then evolve into a slowly varying release, composed primarily of less volatile radionuclides and with little thermal energy. Thus, the appropriate emergency response might vary during the course of an accident.

The particular location of the point or points of release and, to an even greater extent, the thermal energy of the plume are both important because they affect the extent of plume rise. In turn, plume rise can have a substantial effect on human exposure; high-rising plumes will produce lower doses near the point of emission, even though the total population dose (in person-Sv) may not be reduced. This effect was evident during the 1986 Chernobyl accident, which exhibited substantial plume rise (see chapter 17).

Understanding the Potential for Severe Accidents

The 1979 TMI event and close calls such as 1975 cable fire at Browns Ferry highlight the potential for severe accidents at US reactors. Experience with these events has been supplemented by an extensive research effort sponsored by the US Nuclear Regulatory Commission (NRC) and its predecessor, the Atomic Energy Commission (AEC). Figure 2.1 summarizes that research effort and its relationship to the regulatory process. In general, the research effort falls into two categories:

- identifying and estimating the probability of particular accident sequences; and
- estimating the source term associated with each accident sequence.

Attempts to develop a realistic assessment of accident behavior effectively began with the *Reactor Safety Study* (NRC 1975). That effort led to the identification of a number of accident sequences and their classification into release categories, nine for pressurized-water reactors (PWRs) and five for boiling-water reactors (BWRs). For each

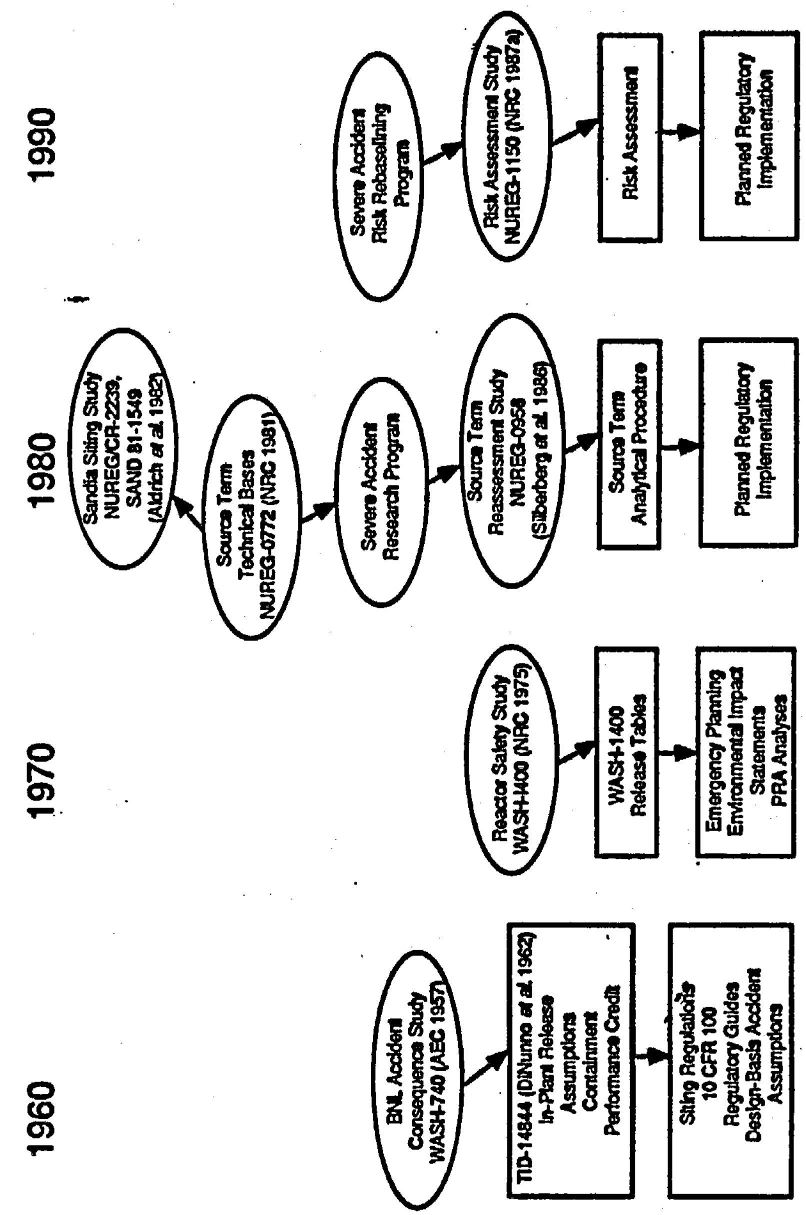


FIGURE 2.1. History of source-term assessment and its relationship to the regulatory process. *Source:* Silberberg *et al.* (1986).

category, a set of source-term parameters was estimated, as was the probability that an accident with that release category would occur.

After its completion, the *Reactor Safety Study* underwent critical reviews, including one sponsored by the NRC itself (Risk Assessment Review Group 1978). As Hirsch discusses fully in chapter 6, much of the criticism focussed on the estimation of accident probabilities and on the assessment of accident consequences rather than on the treatment of source terms in the report. This focus reflected the study's conservative approach to source-term estimation, which is evident from the relatively large releases estimated for some release categories.

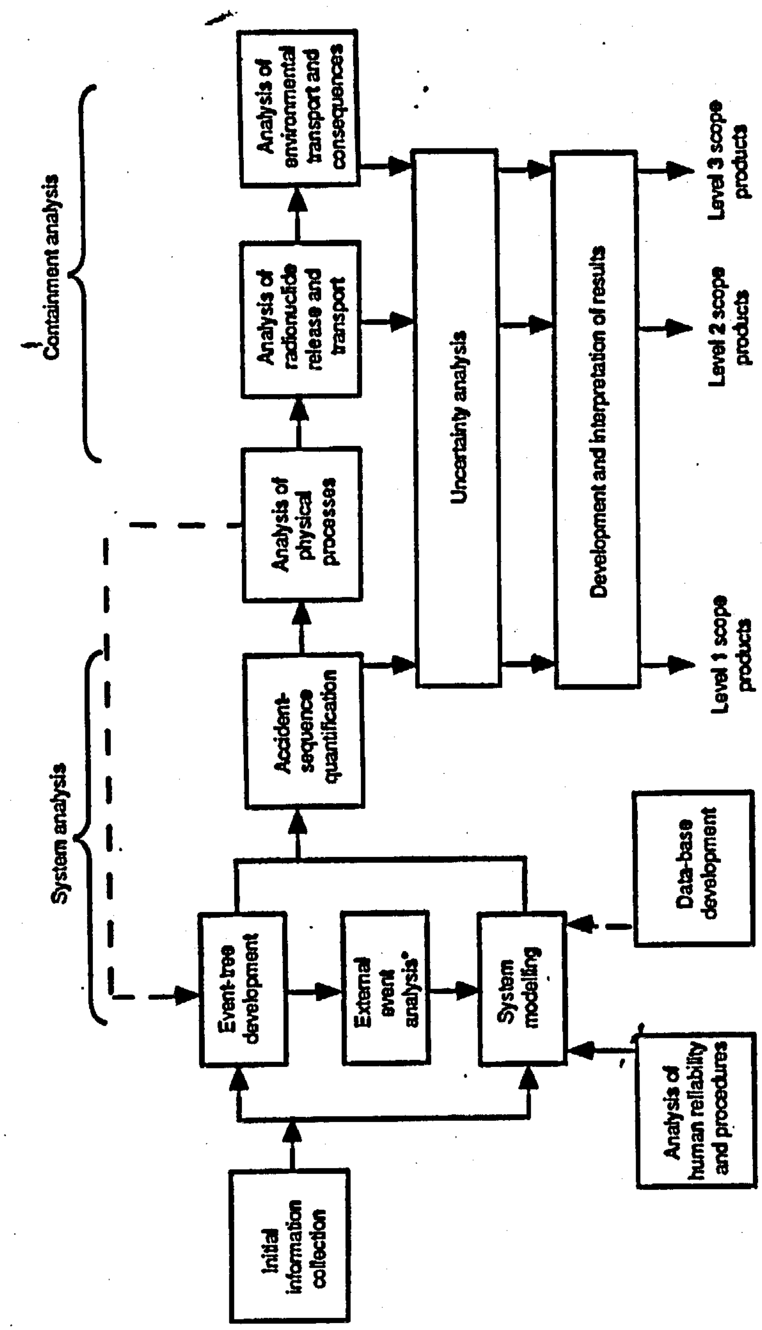
Controversy about the estimation of accident probabilities has abated in recent years. The following discussion provides ample reason to believe that the probability of severe core damage is sufficiently high to be a reason for concern. Debate has recently turned more to the estimation of source terms. Some analysts have argued that most, if not all, severe core-damage accidents will yield source terms much less severe than those estimated in the *Reactor Safety Study*.

PRA Results

The *Reactor Safety Study* set the pattern for subsequent PRAs. Since its publication, numerous PRAs have been conducted under the sponsorship of the NRC and various licensees, and a substantial community of PRA analysts has developed.

Figure 2.2 depicts, in broad terms, the PRA process and shows the three levels at which PRAs may be performed. An effort at Level 1 will identify accident sequences and estimate their probability. At Level 2, the analysis will also include an estimation of the source-term for each sequence. Finally, a PRA conducted at Level 3 will round out the analysis of accident sequences and their source terms with an estimate of the consequences of releases to the environment.

As mentioned earlier, this chapter uses the terms "severe accident" and "core-melt accident" interchangeably, and an explanation of this use is in order. Clearly, for an accident to be "severe," there must be substantial damage to the reactor core. Only such damage can liberate large amounts of radioactivity. The question is: can a core be severely damaged without undergoing melting? In fact, as illustrated by the TMI accident, severe core damage can occur without complete melting of the core, yielding a potential benefit in terms of a reduced release. However, current analytic capabilities do not allow one to discriminate effectively between partial core-melt and full core-melt accidents at light water reactors, so that the term "severe accident" is generally



* May or may not be included in the analysis

FIGURE 2.2 Risk-assessment procedure using probabilistic risk analysis (PRA). *Source:* ANS and IEEE (1983).

taken to be synonymous with the term "core-melt accident." As explained in the Seabrook PRA:

> At one stage of the study, the possibility of specifying additional plant states to distinguish between core-melting and core damage short of melting was considered. The idea was rejected, however, upon finding that the time interval between onset of core damage and full scale fuel melting is short in comparison with the time interval between the initiating event and the time of core damage for risk significant scenarios. Therefore, there was a physical basis for the assumption that, given the onset of core damage, the conditional likelihood of core melt approaches unity. (PLG 1983, 2.1-2)

A matter of current interest is the possibility of reactivity accidents that would involve a rapid increase of reactor power to well above normal operating levels. Although it has been generally assumed that light-water reactors will not experience such accidents, some analysts have not accepted this consensus (e.g., Webb 1976). Moreover, the occurrence of a reactivity accident at the Chernobyl plant in 1986 has spurred a re-examination of the potential for such accidents at light-water reactors, and analysis is under way at Brookhaven National Laboratory. If it turns out that severe reactivity accidents are feasible, PRAs will require some modification. High power levels could rapidly disrupt the core structure, as occurred at Chernobyl, and lead to accident scenarios qualitatively different from those now considered.

Table 2.1 summarizes the results of PRAs conducted in the United States through 1985. The predicted core-melt frequencies range widely from 4.9×10^{-6} to 1.5×10^{-3} per reactor-year. Some of these estimates reflect the consideration of "external events" (floods, earthquakes, etc.) as potential accident initiators. In no case is sabotage considered.

In conducting PRAs, some analysts attempt to provide a range of uncertainty for their estimate of core-melt frequency. Figure 2.3 illustrates such an attempt, in this case for the Surry plant. The uncertainty range extends above the central estimate 2.6×10^{-5} per reactor-year) by a factor of 4 and below by a factor of 5. It should be noted, however, that these uncertainty bounds are not statistically determined but reflect judgments by the analysts. The operating history of nuclear plants—about 1000 reactor-years in the United States—is insufficient to allow purely statistical arguments. In addition, many relevant factors are nonstochastic.

TABLE 2.1. Predicted TMI-1 Core-Damage Frequency and Results of Other Probabilistic Risk Analyses (PRAs)

Plant Name (PRA Vintage)	Internal Events Core-Damage Frequency	External Events Core-Damage Frequency
BABCOCK & WILCOX PWRs		
Three Mile Island 1 (1987)	4.4×10^{-4}	1.1×10^{-4}
TMI-1, This Review (1989)	2.1×10^{-4}	1.2×10^{-4}
Oconee 3 (1981)	8.0×10^{-5}	N.A.
Oconee 3 (1984)	5.4×10^{-5}	2.0×10^{-4}
Arkansas 1 (1982)	5.0×10^{-5}	N.A.
Crystal River 3 (1981)	4.0×10^{-4}	N.A.
Crystal River 3 (1987)	3.7×10^{-5}	N.A.
Midland (1984)	2.8×10^{-4}	3.1×10^{-5}
WESTINGHOUSE PWRs		
Indian Point 2 (1982)	2.9×10^{-4}	6.0×10^{-5}
Indian Point 2 (1982)	7.9×10^{-5}	6.1×10^{-5}
Indian Point 3 (1982)	3.3×10^{-4}	1.5×10^{-5}
Indian Point 3 (1982)	1.3×10^{-4}	1.0×10^{-5}
Seabrook 1 (1983)	1.7×10^{-4}	5.8×10^{-5}
Millstone 3 (1984)	4.5×10^{-5}	1.4×10^{-5}
Sequoyah 1 (1981)	5.6×10^{-5}	N.A.
Sequoyah 1 (1984)	9.1×10^{-5}	N.A.
Sequoyah 1 (1987)	1.0×10^{-4}	N.A.
Zion 1 (1981)	5.7×10^{-5}	1.0×10^{-5}
Zion 1 (1984)	3.2×10^{-5}	N.A.
Zion (1987)	1.5×10^{-4}	N.A.
Surry 1 (1975)	6.0×10^{-5}	N.A.
Surry 1 (1987)	2.6×10^{-5}	N.A.
Haddam Neck (1986)	1.7×10^{-4}	3.8×10^{-4}
COMBUSTION ENGINEERING PWRs		
Calvert Cliffs 1 (1980)	1.5×10^{-3}	N.A.
Calvert Cliffs 2 (1982)	1.3×10^{-4}	N.A.
GENERAL ELECTRIC BWRs		
Brown Ferry 1 (1982)	2.0×10^{-4}	N.A.
Peach Bottom (1975)	3.0×10^{-5}	N.A.

(continues)

Table 2.1 *(continued)*

Plant Name (PRA Vintage)	Internal Events Core-Damage Frequency	External Events Core-Damage Frequency
Peach Bottom (1984)	3.6×10^{-5}	N.A.
Peach Bottom (1987)	8.2×10^{-6}	N.A.
Grand Gulf (1981)	3.6×10^{-5}	N.A.
Grand Gulf (1984)	8.3×10^{-6}	N.A.
Grand Gulf (1987)	2.8×10^{-5}	N.A.
Millstone 1 (1983)	3.0×10^{-4}	N.A.
Millstone 1 (1985)	8.1×10^{-4}	
Limerick (1981/1983)	1.5×10^{-5}	9.1×10^{-6}
Limerick (1984)	8.5×10^{-5}	9.1×10^{-6}
GESSAR-II (1982)	4.3×10^{-6}	6.0×10^{-7}
GESSAR-II (1985)	3.8×10^{-5}	6.7×10^{-5}
Oyster Creek (1980)	4.8×10^{-5} (total)	PLG

Source: After MHB (1989, pages 3-13). This source also includes full bibliographic references for each PRA.

Limitations of PRA

Despite all the effort that has been expended on development of PRA methodology since the *Reactor Safety Study* was published, severe limitations compromise the accuracy of PRA findings. First, it is impossible to predict a chain of operator errors, such as occurred at Chernobyl in 1986. Indeed, analysts conducting PRAs do not attempt to hypothesize such sequences of human error. Instead, they address human error in almost the same manner as equipment failure. For this reason alone, PRA estimates of core-melt frequency should properly be seen as lower bounds.

A related problem is that PRAs do not consider sabotage. Yet, there is ample evidence that sabotage presents a nontrivial risk (Andrews *et al.* 1986).

Even if Chernobyl-type multiple human errors and sabotage events are ignored, fundamental difficulties still exist in conducting a PRA. The data base of equipment failures and operator responses is inadequate for statistically defensible predictions. Also, analysts can never be certain that they have identified all significant sequences. Finally,

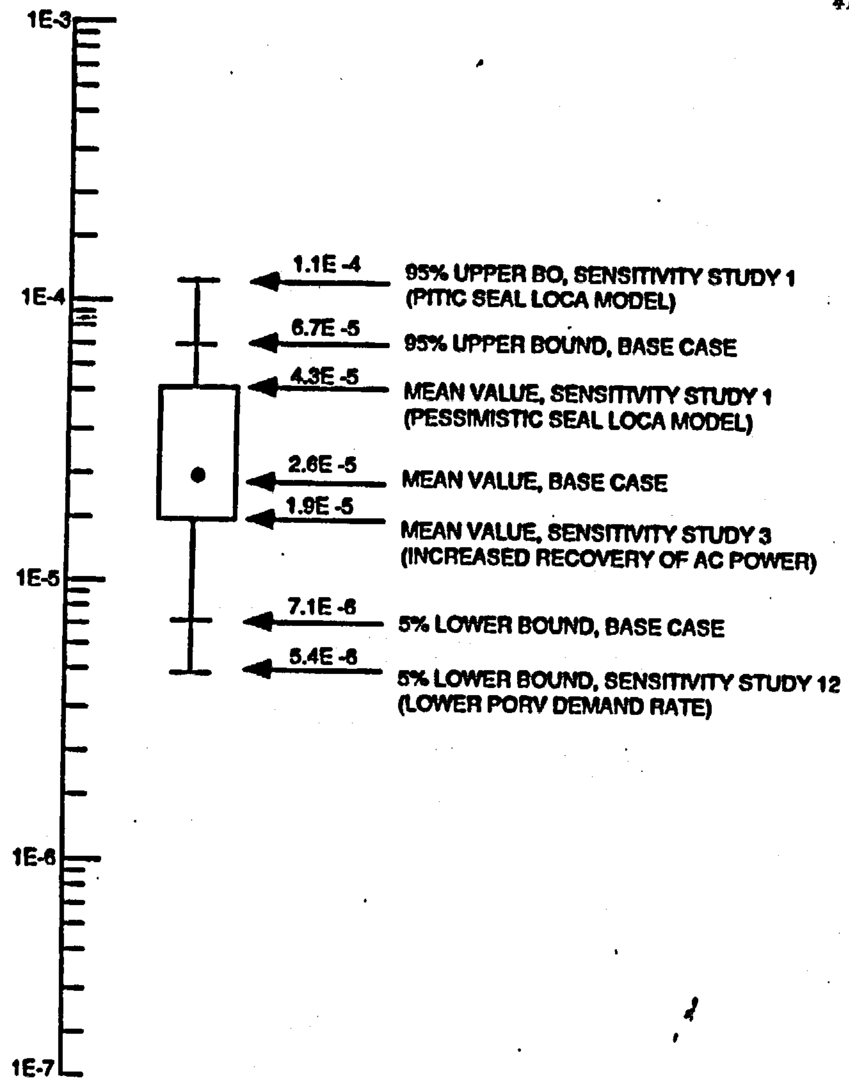


FIGURE 2.3. Uncertainties for core-damage frequency at Surry: NUREG-1150 analysis. *Source:* NRC (1987b, 1:page 3-8).

it is difficult to account for the effects of partial failures (such as degraded voltage in a power supply) as opposed to total failures.

A 1984 review by the NRC of the status of PRA methodology concluded:

- In general, the methodology can be considered relatively mature for essentially all qualitative and quantitative applications of the results of systems analyses to the understanding of accidents resulting from internal initiators and, given a source-term, to the analyses of consequences.
- PRA is not a mature technology if the desired application is the strict comparison of quantitative estimates with regulatory numerical criteria for the purpose of determining compliance.
- Quantitative estimates of core-damage frequency from internal initiators attributable to accidents have an uncertainty of perhaps an order of magnitude on either side of a point estimate. The uncertainties in risk estimates are larger than those drawn from the systems (core-damage) analyses, because of source-term uncertainties. Thus, conclusions drawn from core-damage analyses are generally more robust than those drawn from quantitative risk analyses.
- Quantitative estimates of core-damage frequency or risk due to externally initiated accidents have significantly larger uncertainties than those due to internally initiated accidents and the two should not be compared with any confidence.
- Generally, the uncertainties are reduced when PRA results are used in a relative sense rather than in an absolute sense. Thus, one may expect a greater degree of confidence in identifying on a relative basis the dominant accident sequences due to internal initiators (NRC 1984, 2-8).

These judgments can be regarded as "middle of the road" within the community familiar with PRA methodology. Some analysts express much greater confidence in PRA results, whereas others are much more skeptical. Incidentally, it should be noted that these NRC review comments draw attention to the uncertainties associated with source-term estimation.

Precursor Analysis

The NRC has also sponsored another approach to estimating severe-accident frequency. Its accident sequence precursor (ASP) program attempts to base its estimates on the actual operating history of US

nuclear plants. By examining "precursor" events selected from licensee event reports for the period 1969-1979, analysts at Oak Ridge National Laboratory estimated that the industry-wide severe core-damage probability for that period was in the range 1.7×10^{-3} to 4.5×10^{-3} per reactor-year (Minarick and Kukielka 1982). Figure 2.4 compares this range with other estimates. A subsequent examination of licensee event reports for the period 1980-1981 led to an industry-wide estimate of severe core-damage probability of 1.6×10^{-4} per reactor-year (Cottrell *et al.* 1984). The lower probability during the 1980-1981 period is essentially attributable to the occurrence of three serious events (at Three Mile Island, Browns Ferry, and Rancho Seco) during the 1969-1979 period.

More recent ASP studies have not estimated industry-wide core-damage probability. Nevertheless, the conditional probability of core

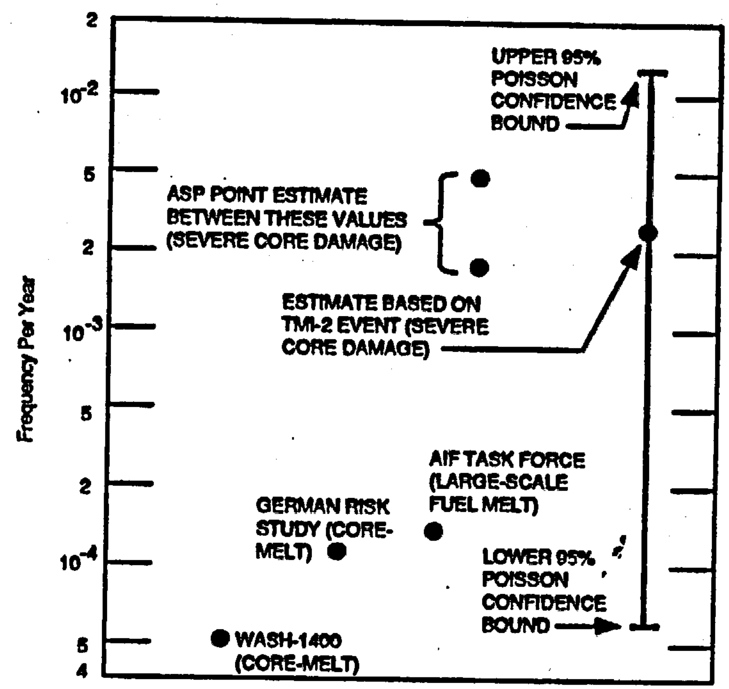


FIGURE 2.4. Comparison of accident sequence precursor (ASP) results (Oak Ridge National Laboratory) with other core-damage estimates. *Source:* Minarick and Kukielka (1982, 1:xiii).

damage (that is, the probability that the precursor event would have led to core damage) has been estimated for each precursor. For example, for the year 1985, one precursor (at the Davis-Besse plant) was estimated to have had a conditional core-damage probability of 1.1×10^{-2}, another a conditional probability of 1.8×10^{-3}, whereas the remaining 61 precursor events had lower conditional probabilities (Minarick *et al.* 1986).

No published estimate exists of the industry-wide severe core-damage probability that would be indicated by an integrated analysis of precursor events for the period from 1969 to the present. This analysis is overdue. Although it might be claimed that the 1969-1979 period is unrepresentative because safety standards have improved since the 1979 TMI accident, one could just as easily postulate that the probability of core-melt accidents is currently rising due to growing complacency about safety and due to aging of the reactor population. Incidents such as the 1985 Davis-Besse precursor demonstrate that severe accidents are an ever-present possibility.

Summary

PRAs have generally recognized limitations. Their estimates of core-melt probability have an uncertainty of perhaps an order of magnitude (on both sides of the central estimate) if internal initiators alone are considered. Uncertainty is greater when external initiators, excluding sabotage, are considered; sabotage itself, being a nonstochastic event, is not amenable to this type of calculation. Analysis of precursor events indicates a generic core-melt probability that is comparable to or greater than the values resulting from plant-specific PRAs (see Table 2.1). But, precursor analysis, for all its merits, does not provide a statistical basis for prediction of core-melt probability.

Source-Term Estimation

As mentioned above, the NRC has sponsored a major ongoing program of research on severe accidents. Much of this research relates directly or indirectly to the estimation of accident source terms. In addition, the nuclear power industry, in part through its sponsorship of PRAs, has conducted source-term research.

Two aspects of severe accident behavior dramatically affect source terms: first, the potential for core-melt to engender phenomena that severely stress the containment boundary; and, second, the performance

of the containment under accident loadings. These matters are addressed below.

History

The *Reactor Safety Study* (NRC 1975) was the first effort to estimate source terms through systematic analysis. Earlier studies had relied heavily upon arbitrary assumptions. Table 2.2 summarizes the source-term findings of the *Reactor Safety Study*. In the context of emergency planning for severe accidents at the TMI plant, the relevant release categories are PWR 1 through PWR 7, each of which involves a core-melt. Release categories PWR 8 and PWR 9 are associated with "design-basis" accidents (in each case, a large pipe break).

Current emergency-planning guidance, as expressed in the document NUREG-0654 (NRC and FEMA 1980), derives from the planning basis laid down in NUREG-0396 (Collins, Grimes, and Galpin 1978). In turn, that planning basis draws heavily upon the source-term and release probability estimates in the *Reactor Safety Study*. In one respect, however, NUREG-0654 takes a more conservative position. Although the *Reactor Safety Study* (see Table 2.2) does not show any severe release's occurring sooner than two hours after initiation of an accident, NUREG-0654 provides the following guidance for emergency-planning purposes:

GUIDANCE ON INITIATION AND DURATION OF RELEASE

Time from the initiating event to start of atmospheric release	0.5 hours to one day
Time period over which radioactive material may be continuously released	0.5 hours to several days
Time at which major portion of release may occur	0.5 hours to 1 day after start of release
Travel time for release to exposure point (time after release)	5 miles: 0.5 to 2 hours 10 miles: 1 to 4 hours

Source: NRC and FEMA (1980, 17).

By general agreement, understanding of source-term issues has improved substantially since the time of the *Reactor Safety Study*. A considerable body of technical literature has emerged, and the scientific issues have been fairly well identified in two reports: a review by the

TABLE 2.2. Summary of Accidents Involving Core Damage

Release Category	Probability per Reactor-Year	Time of Release (Hr)	Duration of Release (Hr)	Warning Time for Evacuation (Hr)	Elevation of Release (Meters)	Containment Energy Release (10^6 Btu/Hr)	Fraction of Core Inventory Released[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	$9X10^{-7}$	2.5	0.5	1.0	25	520[d]	0.9	$6X10^{-3}$	0.7	0.4	0.4	0.05	0.4	$3X10^{-3}$
PWR 2	$8X10^{-6}$	2.5	0.5	1.0	0	170	0.9	$7X10^{-3}$	0.7	0.5	0.3	0.06	0.02	$4X10^{-3}$
PWR 3	$4X10^{-6}$	5.0	1.5	2.0	0	6	0.8	$6X10^{-3}$	0.2	0.2	0.3	0.02	0.03	$3X10^{-3}$
PWR 4	$5X10^{-7}$	2.0	3.0	2.0	0	1	0.6	$2X10^{-3}$	0.09	0.04	0.03	$5X10^{-3}$	$3X10^{-3}$	$4X10^{-4}$
PWR 5	$7X10^{-7}$	2.0	4.0	1.0	0	0.3	0.3	$2X10^{-3}$	0.03	$9X10^{-3}$	$5X10^{-3}$	$1X10^{-3}$	$6X10^{-4}$	$7X10^{-5}$
PWR 6	$6X10^{-6}$	12.0	10.0	1.0	0	N/A	0.3	$2X10^{-6}$	$8X10^{-4}$	$8X10^{-4}$	$1X10^{-3}$	$9X10^{-5}$	$7X10^{-5}$	$1X10^{-5}$
PWR 7	$4X10^{-5}$	10.0	10.0	1.0	0	N/A	$6X10^{-3}$	$2X10^{-5}$	$2X10^{-5}$	$1X10^{-5}$	$2X10^{-5}$	$1X10^{-6}$	$1X10^{-6}$	$2X10^{-7}$
PWR 8	$4X10^{-5}$	0.5	0.5	N/A	0	N/A	$2X10^{-3}$	$5X10^{-6}$	$1X10^{-4}$	$5X10^{-4}$	$1X10^{-6}$	$1X10^{-8}$	0	0
PWR 9	$4X10^{-4}$	0.5	0.5	N/A	0	N/A	$3X10^{-6}$	$7X10^{-9}$	$1X10^{-7}$	$6X10^{-7}$	$1X10^{-9}$	$1X10^{-11}$	0	0
BWR 1	$1X10^{-6}$	2.0	2.0	1.5	25	130	1.0	$7X10^{-3}$	0.40	0.40	0.70	0.05	0.5	$5X10^{-3}$
BWR 2	$6X10^{-6}$	30.0	3.0	2.0	0	30	1.0	$7X10^{-3}$	0.90	0.40	0.30	0.10	0.03	$4X10^{-3}$
BWR 3	$2X10^{-5}$	30.0	3.0	2.0	25	20	1.0	$7X10^{-3}$	0.10	0.10	0.30	0.01	0.02	$3X10^{-3}$
BWR 4	$2X10^{-6}$	5.0	2.0	2.0	25	N/A	0.6	$7X10^{-4}$	$8X10^{-4}$	$5X10^{-3}$	$4X10^{-3}$	$6X10^{-4}$	$6X10^{-4}$	$1X10^{-4}$
BWR 5	$1X10^{-4}$	3.5	5.0	N/A	150	N/A	$5X10^{-4}$	$2X10^{-9}$	$6X10^{-11}$	$4X10^{-9}$	$8X10^{-12}$	$8X10^{-14}$	0	0

[a] A discussion of the isotopes used in the study is found in Appendix VI. Background on the isotope groups and release mechanisms is found in Appendix VII.
[b] Includes Mo, Rh, Tc, Co.
[c] Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.
[d] A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.

Source: NRC (1975, 78).

American Physical Society (APS 1985) and the previously mentioned NRC report NUREG-0956 (Silberberg *et al.* 1986). Also, the author of this chapter has co-authored a review (Sholly and Thompson 1986).

Based upon recent research, the NRC has prepared an updated version of the *Reactor Safety Study*; NUREG-1150 (NRC 1987a). Contrary to the hopes of many in the nuclear industry, NUREG-1150 does not provide a basis for assuming, for regulatory purposes, that source terms will be less severe than those estimated in the *Reactor Safety Study*. First, NUREG-1150 has identified a wide range of uncertainty in its predictions of accident source terms; the range of its predictions actually encompasses those of the *Reactor Safety Study*. Second, a number of areas, not necessarily accounted for in NUREG-1150, remain where uncertainty about underlying phenomena is significant and where that uncertainty will not soon be resolved.

Summary

Since the subject first received attention in 1950, the scientific understanding of the severe-accident potential of nuclear plants has improved substantially. That improved understanding has not provided a basis for disregarding, for regulatory purposes, severe source terms. On the contrary, the summary analysis NUREG-1150 (NRC 1987a) shows that source terms could be at least as severe as those estimated in the 1975 *Reactor Safety Study*.

Severe-Accident Phenomena

A wide variety of physical and chemical phenomena will accompany severe accidents. It is beyond the scope of this chapter to review all of these phenomena, but interested readers may consult NUREG-1150 (NRC 1987a), and the references cited therein, for further information. Relevant phenomena are those that can affect the liberation of radioactive materials from the uranium dioxide fuel pellets, the transport of those materials through the coolant system and the containment of the reactor, and their release to the atmosphere. Table 2.3 shows the amounts of both radioactive and nonradioactive materials in the core of the Surry and Peach Bottom reactors, which are 2441 MW_t PWR and 3293 MW_t BWR units, respectively. Given the melting and boiling temperatures of various materials in the reactor core, many radionuclides will have vaporized by the time core-melt is complete. Incidentally, chemical form can be important to physical behavior. Ruthenium, for example, is a nonvolatile fission product, but in an

TABLE 2.3 Inventory of Fission Products and Other Materials for Surry and Peach Bottom Reactors

Fission Products			Actinides and Others		
Element	Mass (kg) Surry	Mass (kg) Peach Bottom	Element	Mass (kg) Surry	Mass (kg) Peach Bottom
Kr	13.4	25.7	U	70,210	138,000
Rb	14.7	23.3	Pu	469	743
Sr	47.6	62.7	Np	26	41.2
Y	22.9	36.2	Zr	16,460	64,100
Zr	179.0	267.0	Sn	262	1,050
Mo	155.0	237.0	Ag	2,750	0
Tc	37.1	58.8	In	505	0
Ru	104.0	172.0	Cd	173	0
Rh	20.9	33.2	Cr	1,168	4,140
Te	25.4	34.9	Ni	649	2,560
I	12.4	16.6	Mn	0	432
Xe	260.0	387.0	Gd	0	287
Cs	131.0	207.0	Fe	6,486	15,150
Ba	61.2	105.0			
La	62.3	98.3			
Ce	131.0	208.0			
Pr	50.7	80.4			
Nd	171.0	271.0			
Sm	34.0	53.8			
Nb	2.7	4.3			
Pd	52.5	83.2			
Eu	8.9	14.1			
Pm	7.2	11.5			

Source: Silberberg *et al.* (1986).

oxidizing environment ruthenium tetroxide may form—this is quite volatile.

Here we briefly review three severe-accident phenomena. The primary purpose of this review is to demonstrate that substantial scientific uncertainty remains about even qualitative aspects of severe accident behavior. In addition, the three selected phenomena are of particular interest because each has the potential to lead to early containment failure with a large release to the atmosphere. Research continues on to each of the three phenomena in the context of an ongoing research program on source terms generally.

High-Pressure Melt Ejection

For many accident sequences, it is expected that the reactor core will melt while the reactor-coolant system remains at high pressure. The PRA for Seabrook uses 300 psia (pounds per square inch absolute) as the dividing line between low and high pressure core-melts, and estimates that well over 90 percent of core-melts at that plant (a PWR) would be at high pressure, and about half of the events would involve a dry reactor cavity (PLG 1983). PRAs for other PWR plants have indicated a similar preponderance of high-pressure core-melts.

In the event of a high-pressure core-melt accident, molten core material could flow into the bottom of the reactor vessel and melt through the vessel wall while the reactor-coolant system remains pressurized. Molten material could then be ejected from the vessel at high velocity, driven by pressure inside the reactor-coolant system. This phenomenon, known as high-pressure melt ejection, raises two major concerns. First, it provides mechanisms for the suspension of radioactive material in the containment atmosphere. Second, it can produce a substantial increase in containment pressure, which could lead to containment failure. The increase in pressure could come from direct heating of the containment atmosphere, from combustion of the molten material, from hydrogen combustion, or from an ex-vessel steam explosion.

If high-pressure melt ejection is to occur, the boundary of the reactor-coolant system must maintain its structural integrity until the molten core has formed a pool inside the bottom of the reactor vessel. Further, the core must melt through the vessel wall in such a way that molten material flows into the reactor cavity at high velocity. Several factors could decrease the likelihood of the core-melt conditions needed for high-pressure melt ejection. First, an in-vessel steam explosion could blow open the lower end of the reactor vessel, thus precluding high-pressure melt ejection. Second, the temperature of the reactor-coolant-system boundary might closely follow the core temperature for accidents in which the coolant system is pressurized. If so, the decline of material strength in the reactor-coolant system at higher temperatures could cause a loss of structural integrity, leading to depressurization before the molten core slumps into the bottom of the vessel.

The likelihood of these various effects is disputed, and no basis currently exists for excluding high-pressure melt ejection. Indeed, NUREG-1150 (NRC 1987a) identifies high-pressure melt ejection as a potential major threat to large dry PWR containments. The estimated range of possible pressure loadings generated by high-pressure melt ejection is such that, for the upper end of the range, containment failure is likely for most plants. This remains true even as more sophisticated analyses (e.g., Williams *et al.* 1987) demonstrate the potential role of certain factors (e.g., the existence of compartments within the containment free volume) in reducing estimated peak pressure loadings.

It is ironic that, when the potential for high-pressure melt ejection was first recognized, it was thought to be a phenomenon favorable to containment integrity. The Zion PRA (PLG, Westinghouse Electric Company, and Fauske and Associates 1981) proposed high-pressure melt ejection as a mechanism for dispersing molten core material over the floor of the containment, thus preventing a high-temperature core-concrete interaction and thereby avoiding the evolution of gases (including combustible gases) and radioactive aerosols that accompany such interaction. However, subsequent experiments conducted at Sandia National Laboratories have shown that high-pressure melt ejection is a much more violent event than the authors of the Zion PRA thought, and that it in fact presents a major threat to the integrity of the containment (PLG, Westinghouse Electric Company, and Fauske and Associates 1981)

Steam Explosion

A steam explosion can occur if molten core material is rapidly mixed with water, either inside the reactor vessel or in the concrete cavity beneath the vessel. In-vessel explosions can be the most damaging because the explosion is more confined. A severe in-vessel explosion could blow open the reactor vessel, possibly propelling the vessel head upward with sufficient momentum to penetrate the containment. Whether or not the containment is penetrated by a missile in this manner, any explosion that breaches the vessel will release molten core material to the containment atmosphere in finely divided form. The resulting rise in pressure may itself contribute to containment failure.

Just as for high-pressure melt ejection, experts disagree about the probability that a steam explosion will induce containment failure. But consensus exists about the lack of an objective means of resolving the dispute.

The level of uncertainty regarding in-vessel steam explosions has been investigated in an analytic exercise conducted at Sandia National Laboratories. Table 2.4 summarizes the results and shows the outcome of a Monte Carlo computation in which five parameters were varied.

TABLE 2.4. Estimated Number of Outcomes of Steam Explosions During 10,000 Core-Melts, Based on a Monte Carlo Computation for Various Parameter Ranges*

Outcome of Steam Explosion	Full Range	Lower Third	Middle Third	High Third
Failure of Vessel Bottom	2,017	0	2,126	10,000
Separation of Vessel Head: Velocity over 50 m/s	460	0	1	9,987
Separation of Vessel Head: Velocity over 90 m/s	267	0	0	9,958

*The five major parameters used in this calculation were: (1) molten fraction of core; (2) pour diameter; (3) pour length; (4) conversion ratio; and (5) condensed-phase fraction of slug. *Sources:* Berman, Swenson and Wickett (1984); Sholly and Thompson (1986).

The Sandia analysts determined for each parameter the range of values that could be justified on present knowledge, divided that range into lower, middle, and high thirds, and then repeated the computation 10,000 times, each time with a random choice for the value of each parameter from within its allotted range. Table 2.4 shows that dramatic failure of the reactor vessel is virtually certain if each parameter is confined to the high third of its credible range and impossible if each parameter is confined to the lower third of its range. If the full range is permitted, then 20 percent of core-melts will lead to the lower part of the vessel's being blown open, whereas 5 percent of core-melts will lead to the vessel head's becoming a missile with a velocity of at least 50 meters per second. Although these numbers are strictly illustrative, the exercise shows that no basis exists for asserting that severe in-vessel steam explosions have negligible or zero probability.

This Sandia exercise was conducted assuming core-melt events in which the reactor vessel is at low pressure, such as would be characteristic of a large-break loss-of-coolant accident. Even less knowledge exists about steam explosions at higher pressure, and, although high-pressure steam explosions are probably less likely, no basis exists for ruling them out.

As mentioned in the preceding discussion, it is possible that a high-pressure core-melt sequence could change to a low-pressure sequence through thermally induced failure of the reactor-coolant system boundary. Although that change would preclude high-pressure melt ejection, the potential for a steam explosion would remain. Moreover, the vessel upper head and its retaining bolts would be weaker because of their elevated temperature, thus allowing a steam explosion more readily to breach the vessel.

Rupture of Steam-Generator Tubes

The rupture of steam-generator tubes can initiate a severe accident, or such rupture can arise during an accident sequence. In either case, tube rupture is important because the containment can be breached indirectly, through the creation of an escape path from the core to the atmosphere via the steam generators. Inside these generators are thousands of small tubes with relatively thin walls (about 1 mm in thickness) that separate the primary cooling-water circuit from the secondary steam circuit. On the secondary side, pressure-relief valves communicate directly with the atmosphere. Thus, if steam-generator tube failure is associated with opening of the relief valves, a pathway from the core to the atmosphere will be created. If the secondary side relief valves stick open, this pathway will remain in effect throughout the accident.

As an accident initiator, steam-generator tube rupture could lead to core-melt if emergency core-cooling systems were unavailable or became so during the sequence (e.g., due to loss of coolant inventory to the secondary side). During such sequences, the secondary side relief valves are likely to open, and experience suggests that there is a substantial probability that one or more of them will fail to re-close. In the latter event, even before the accident has proceeded to core-melt, there will be a direct release path from the core to the atmosphere.

Spontaneous tube ruptures are relatively common events (PLG 1987). This is not surprising, considering the dimensions of the tube walls, the harsh conditions to which they are exposed, and the difficulty of detecting weakened tubes through routine inspection. In addition, however, tube rupture could occur as a result of the primary/secondary pressure differential arising during a "steam line break" or "failure to scram" incident. A rupture induced in this manner could lead to core-melt in the same way as a spontaneous rupture.

Pressure and temperature effects arising during core-melt sequences that have other initiators can also induce steam-generator tube rupture. These effects will be relevant for sequences in which the reactor-coolant system remains at high pressure up to and during core-melt. Such sequences may produce a substantial pulse of pressure on the primary side of the steam-generator tubes when the molten core slumps into residual water in the base of the reactor vessel. Moreover, convective heat transfer from the core, deposition of radioactive material within the tubes, or both, may also elevate tube temperatures.

The temperature effect raises an issue that is generic to high-pressure core-melts and that is mentioned in the preceding discussions of high-pressure melt ejection and steam explosion. The heating of the entire reactor-coolant system boundary by convective heat transfer and deposition of radioactive material could lead to a breach of the reactor-coolant system, either in the steam generators or at locations such as the "hot leg" piping or the pressurizer line. Research and regulatory attention has focussed on convective heat transfer rather than on heating due to deposited radioactive material. Even with this limited focus, current knowledge suggests that thermally induced rupture of steam-generator tubes poses a potential containment-failure mechanism for PWRs (Lyon 1987).

The particular configuration of the reactor-coolant system for TMI Unit 1 may inhibit convective heat transfer. Even if most of the core is uncovered, a "loop seal" of water is likely to remain in the lower part of the "cold leg" piping and the steam generators; the presence of that water will prevent the establishment of convective circulation from the core to the steam generator via the hot legs and back to the core via the cold legs.

Deposited radioactive material could, however, be a concern for TMI Unit 1. Analysis of the potential for thermally induced steam-generator tube rupture arising from such deposition will require detailed investigation of flow dynamics. Localized deposition, which could cause local heating of steam-generator tubes, must be considered. Also, the possible propagation of a small initial tube breach must be investigated; transport of radioactive material to the site of that breach could lead to localized tube heating.

Containment Performance

During a core-melt accident, large amounts of radioactive material will enter the containment atmosphere. It is important, therefore, to understand the ability of containment to maintain its integrity under accident conditions. It is necessary to consider three types of release pathway:

- a pathway from the core to the outside atmosphere, bypassing the containment envelope;

- a pathway from the containment atmosphere to the outside atmosphere, not involving a breach in the containment structure; and
- a breach in the containment structure.

The preceding discussion indicates one way the containment envelope of a PWR could be bypassed—through steam generator tube rupture. Another bypass event would involve an inadvertent connection between the reactor-coolant system and the low-pressure piping of the residual heat removal system, leading to a rupture of the residual heat removal piping outside the containment. This event is known as an "interfacing systems loss of coolant accident" and is frequently designated as "event V." In both cases—steam generator rupture and event V—the strength of the containment building is irrelevant.

Closure of the Containment Boundary

Potential routes exist whereby a release from the containment atmosphere to the outside atmosphere can occur without breaching of the containment building. Of particular interest in this regard are the containment ventilation system and the personnel and equipment hatches, all of which present potentially capacious pathways to the atmosphere. In principle, the hatches will be closed during plant operation, and the ventilation system and other open piping systems will be "isolated" by the automatic closing of valves as soon as an accident arises. In practice, the hatches may not always be closed and isolation of open piping systems may not always occur when needed.

As for other PRA issues, it is not possible to provide a statistically robust estimate of the probability of isolation failure. Certainly, sabotage is not amenable to such estimation. Nor can one anticipate Chernobyl-type human errors. For what it is worth, an examination of operating experience for US nuclear plants has shown that, on average, the probability of a containment's having a leakage area greater than allowed by plant Technical Specifications is 0.3, and the probability of a larger leak (a hole in the containment liner or an open isolation valve) ranges from 0.001 to 0.01 (Pelto, Ames, and Gallucci 1985).

Containment Response to Severe Loading

Turning now to the possibility of a breach in the containment structure, it is worth noting that a variety of types of loading (pressure, temperature, radiation, impact) can cause such a breach. High-

pressure melt ejection events and steam explosions have been mentioned above, but other sources of loading may be important as well. Hydrogen explosions, slow buildup of noncondensible gases from core-concrete interaction, and erosion of the containment wall by molten core material are three other important sources. For a given core-melt accident sequence, considerable uncertainty exists about the nature of the containment loading. This uncertainty derives both from our incomplete understanding of the processes involved and from the natural variability of those processes. If one then considers the range of core-melt sequences that might be experienced, it is clear that a wide range of uncertainty exists about the containment loadings that might accompany a core-melt accident.

Further uncertainty prevails about the response of containment structures to a given loading. A 1:6-scale reinforced concrete containment model was tested to failure at Sandia National Laboratories, and analysts from various US and West European laboratories, using "state of the art" computer codes, made pre-test estimates of failure pressure. The estimates of failure pressure span a range from 128 psig (pounds per square inch gage) to 190 psig, and the analysts made markedly varying predictions of the mode of containment failure. In fact, when the model containment was tested in July 1987, it failed (in the sense that nitrogen gas could not be injected quickly enough to sustain pressure) at 145 psig. The mode of failure was a 20-inch-long tear in the steel liner, adjacent to some containment penetrations (Weaver 1987). It seems clear that there is room for substantial improvement in our current capability to analyze containment failure.

There is also the question of translating an improved analytic capability to different containment types. The Sandia model was a reinforced concrete structure, whereas the TMI Unit 1 containment is a prestressed structure. Analysis of such structures requires consideration of effects such as varying friction along the length of pre-stressing tendons. Thus, additional model tests may be necessary.

All modelling exercises are applicable to containments that are built according to design specifications. However, actual containments may have serious defects that arise during construction or as a result of aging. Figure 2.5 summarizes the defects that have been detected at US nuclear plants with concrete containments. An indication of the serious nature of some of these defects is the extent of cracking (delamination) in the dome of the Crystal River Unit 3 containment. This problem was discovered two years after completion of concrete placement and one year after tendon tensioning, when electricians could not secure some drilled-in anchors to the top of the dome (Naus 1986).

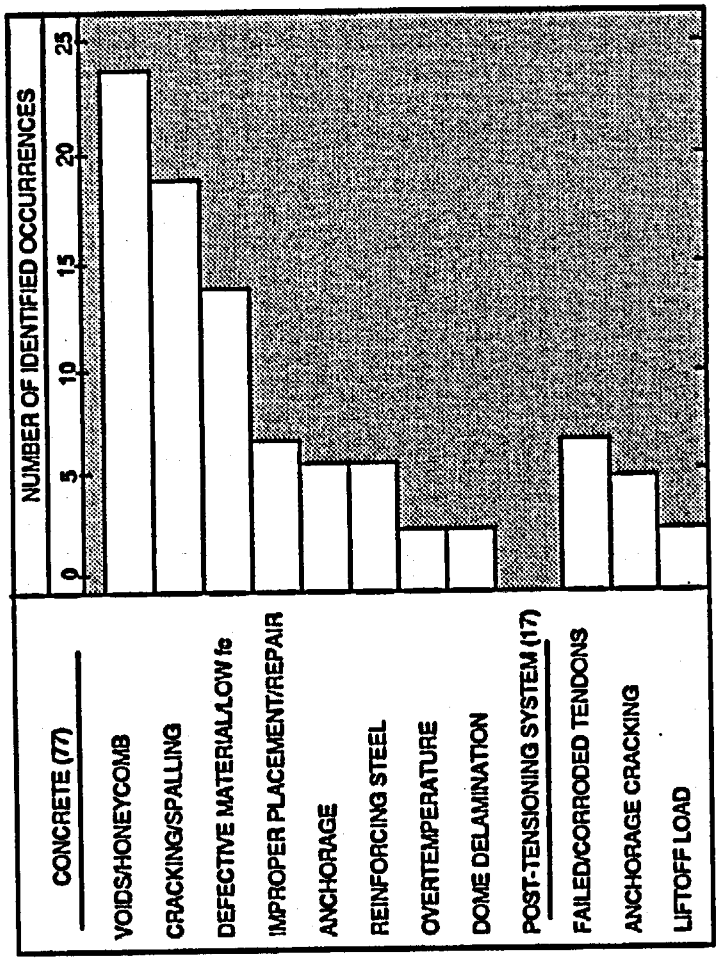


FIGURE 2.5. Light water reactor concrete component problem areas. *Source:* Naus (1986, 47).

Uncertainty and Variability in Severe-Accident Behavior

As indicated previously in this chapter, substantial uncertainty exists about both qualitative and quantitative aspects of severe accident behavior. It is important to note, however, that different concepts are typically subsumed under the term "uncertainty." In part, the term refers to our lack of ability to predict accurately, even if all the input parameters are known, the distribution of outcomes of a situation. In addition, the term is used to refer to the natural variability of relevant processes, although this variability can, in principle, be described by a probability density function. These types of "uncertainty" are, in fact, intertwined.

NRC Treatment of Uncertainty

The NRC is aware of the incomplete status of knowledge on qualitative aspects of severe-accident behavior. This status is evident from the proceedings of a meeting convened by the commission in 1987. The conference produced a series of papers on scientific uncertainty (NRC 1987b) and an overall review by expert panelists (Kouts 1987), who addressed a series of issues:

- core-melt progression and hydrogen generation;
- natural circulation in the reactor-coolant system;
- direct containment heating;
- steam explosion;
- hydrogen combustion;
- core-concrete interaction;
- iodine chemical form; and
- fission product revaporization.

It is not appropriate to repeat here all the findings of this review, but one finding will serve as an example.

It concerns the chemical form of iodine during severe accidents. Many analysts have argued that iodine will be present as cesium iodide (CsI), which is less volatile than other iodine species such as molecular iodine (I_2). Although major studies have proceeded on the assumption that iodine will be present as CsI, the above-mentioned review panel "cast doubt on the central importance of formation and fixation of iodine in CsI. It was pointed out that numerous effects can dissociate CsI, including surface effects with stainless steel, the presence of B_4C in BWR's, the existence of an oxidizing atmosphere, and the effects of high temperature" (Kouts 1987, 10).

Estimation of the characteristics of potential severe accidents demands an integrated analysis. The various phenomena involved render it difficult to find an analytical approach that adequately accounts for uncertainty. In preparing NUREG-1150 (NRC 1987a), the NRC staff has adopted two distinctive approaches. The first approach is to bound the estimated accident characteristics by an "optimistic/pessimistic" analysis, illustrated in Figure 2.6, which summarizes an analysis of the potential for containment failure under internal pressurization. The most "pessimistic" estimate of the probability of containment failure is obtained by matching the "pessimistic" estimate of pressure loading with the "pessimistic" distribution of containment failure pressure. This optimistic/pessimistic approach yields a wide uncertainty band, but suitably applied, is scientifically defensible.

Unfortunately, the NRC has performed most of its NUREG-1150 analysis using a much less rigorous approach called the "limited Latin hypercube" [LLH] method. This method centers around the use of panels of analysts who vote on the probability distributions to be assigned to relevant parameters. For all its elaborate manipulations, this approach represents nothing more than the collective judgments of the participating analysts. Its use has, however, been justified as follows:

> Dr. Allan Benjamin of Sandia initiated the session with a brief review of the uncertainty approaches employed in SARRP [Severe Accident Risk Reduction Program]. He also provided some initial results for the Surry plant that were obtained using both the LLH and optimistic/central/pessimistic (OCP) approaches. He explained that through interactions of SARRP personnel, review groups, and NRC staff, the LLH approach was developed in order to address the significant drawback of the OCP approach, which was the initial method selected. That drawback concerns the lack of ability to characterize the relative likelihoods of the outcomes, i.e., the degree of optimism and pessimism in the results. The LLH method allowed the development of a distribution of possible outcomes (in terms of the risk results) generated by variation of a specific number of uncertain inputs, which were termed issues in this study. (Benjamin *et al.* 1987, F-1)

This is not a convincing justification.

If the state of knowledge does not permit a scientifically rigorous uncertainty analysis, then one must remain content with a bounding uncertainty analysis. It is inappropriate to create a superficially sophisticated analysis. It is inappropriate to create a superficially sophisticated analytic method, such as LLH, that has no scientific basis. For further criticisms the reader is referred to comments by the NRC's own reviewers of LLH (Benjamin *et al.* 1987, Appendix F) or to a review of the draft NUREG-1150 report (Kastenberg *et al.* 1988).

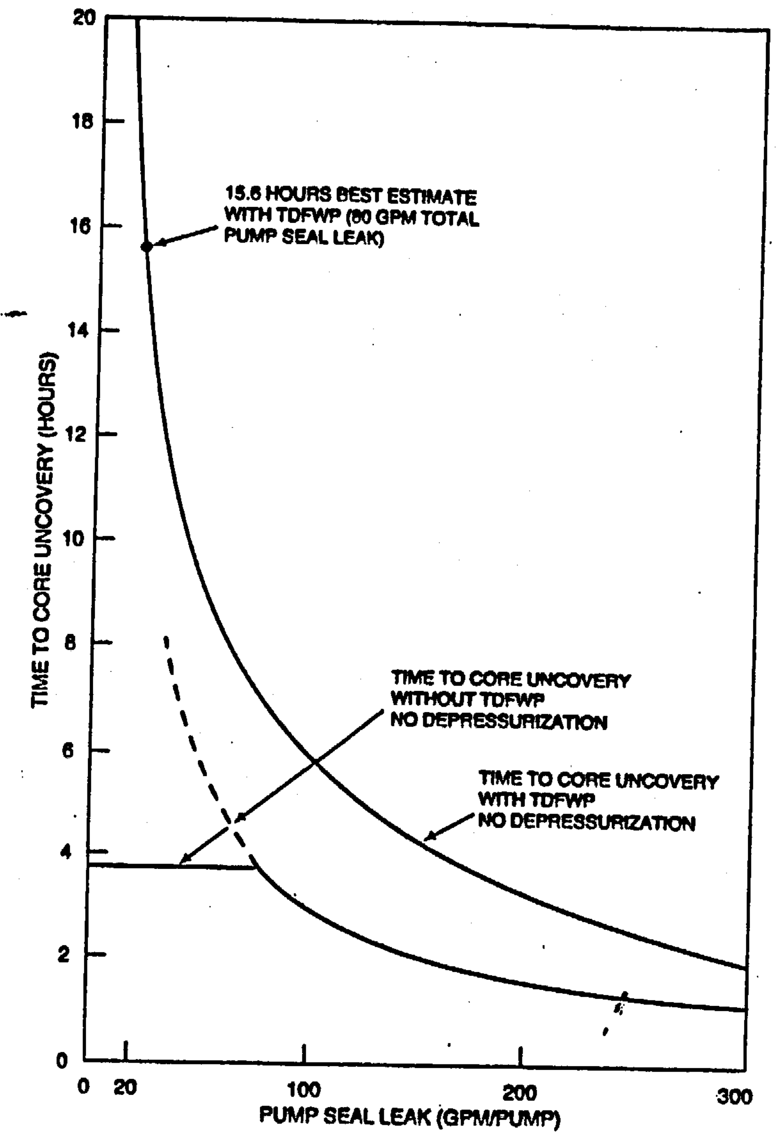


FIGURE 2.6. Time to core uncovery as a function of pump seal leak rate for station blackout sequences with and without emergency feedwater: The Seabrook Probabilistic Risk Analysis (PRA). *Source*: PLG (1983, 3:11.5-34).

Accident Variability: Sequence Timing and Thermal Energy of Release

In estimating the outcome of an accident sequence, we confront substantial "uncertainty"—the outcome can be predicted only within a broad range. Part of that "uncertainty," however, is simply natural variability, which, in principle, is amenable to rigorous analysis. Here we address the potential for accident variability, drawing from selected calculations of variability in sequence timing and thermal energy of release.

Figure 2.6, drawn from the Seabrook PRA (PLG 1983), shows the estimated time to core uncovery during a station blackout (loss of all AC power) sequence, as a function of the rate of leakage from the reactor-coolant pump seals. Two curves are shown—with and without availability of the turbine-driven feedwater pump. It is apparent that the time to core uncovery is particularly sensitive to leakage rate in the "with TDFWP" case.

Considerable debate surrounds the question of reactor-coolant pump seal leakage during station blackout sequences. Some analysts argue that seals are very robust and will survive blackout conditions. Experiments have indicated, however, that seal failure can occur (Kittmer *et al.* 1985). The nature of the seal failure, and hence the leakage rate, will vary according to factors such as the age of the seal material and the particular wear patterns of the seal and shaft. Thus, the leakage rate will be unpredictable, and may vary during the course of an accident sequence. All accident sequences will, to varying degrees, be subject to variability in their time development. As a practical matter, it may not be possible to ascribe a probability distribution to that variability.

As indicated earlier in this report, the thermal energy accompanying a radioactive release will affect the extent of plume rise with a resulting effect on the pattern of off-site radiation exposure. It is therefore of interest to know the potential variability of the thermal energy release rate. Table 2.5, also drawn from the Seabrook PRA (PLG 1983), illustrates this variability. For four potential release categories, the estimated thermal energy release rate is shown for containment leak areas ranging from 250 ft^2 to 0.5 ft^2. It is clear that the energy release rate varies over a correspondingly large range. Containment leak area could vary over the range shown in Table 2.5, and the breach could occur in various parts of the containment structure. Although it may be possible in principle to predict accurately the failure mode of an "ideal" containment under a given loading, this will not be possible in practice. Factors such as variations in material properties, the presence of construction defects, and the existence of localized pockets of corrosion

TABLE 2.5. Energy Release Rates for Four Release Categories: The Seabrook Probabilistic Risk Analysis (PRA)

Release Category (Accident sequence)	Energy Released (10^8 Btu)	Energy Release Rate (10^9 Btu/hr) — Duration of Blowdown: 10 Seconds	1 Minute	10 Minutes	30 Minutes	1 Hour
$\overline{\mathrm{S1}}$ (AL)	0.58	21	3.5	0.35	0.12	0.06
$\overline{\mathrm{S3}}$ (AL)	1.28	25	7.6	0.76	0.25	0.13
$\overline{\mathrm{S3V}}$ (TE)	2.0	70	12	1.2	0.4	0.2
$\overline{\mathrm{S4V}}$ (AE)	1.6	57	9.6	0.96	0.32	0.16
Leak Area (square feet)		250	25	2.5	1	0.5
Equivalent Diameter (feet)		18	6	1.8	1.1	0.8

$\overline{\mathrm{S1}}$ = Early containment failure with oxidation release, sprays not operating

$\overline{\mathrm{S3}}$ = Late overpressure failure, sprays not operating, no vaporization release

$\overline{\mathrm{S3V}}$ = Similar to $\overline{\mathrm{S3}}$ but with a vaporization release

$\overline{\mathrm{S4V}}$ = Basemat penetration failure, sprays not operating, with a vaporization release

AL = Large loss-of-coolant accident (LOCA) with refueling water storage tank (RWST) to reactor coolant system (RCS)

TE = Transient initiating event, no RWST injection to RCS

AE = Large LOCA, not RWST injection to RCS

Source: Adapted from PLG (1983, 3:11.5-19, 11.6-32, and 11.6-34).

will ensure that containment failure does not correspond to the "ideal" case.

The PRA for Three Mile Island Unit 1

Under the sponsorship of the TMI licensee, a consulting firm has prepared a Level 1 PRA for TMI Unit 1 (PLG 1987). The analysis considers both internal and external initiating events although, as do other PRAs, it does not consider sabotage.

MHB Technical Associates reviewed this PRA for the author. Drawing upon the PRA and other sources of information, the MHB group prepared its own estimates of Level 1 and Level 2 PRA outputs for TMI Unit 1. These estimates are summarized below. It should be noted that a full-scale review of a PRA requires resources well beyond those available for this study. The NRC sponsored such a review and published its findings, summarized below, after the completion of research for the study.

Summary of Findings

The estimated contribution of various internal and external initiating events to the total mean core-melt frequency (5.5×10^{-4} per reactor year [RY]) is shown in Table 2.6. It will be noted that 36 percent of core-melt frequency was attributed to loss of control-building ventilation. MHB did not find this large contribution credible. By contrast, the PRA concluded that earthquakes provide a relatively small core-melt-frequency contribution of 2.7×10^{-6} per RY (0.5 percent of total frequency). MHB authors found that the PRA incorrectly estimated the extent of damage that would arise from earthquakes. Using the same assumptions as the PRA in regard to the frequency of earthquakes, they estimated a core-melt frequency of 2.1×10^{-5} per RY. They further argue that severe earthquakes may be considerably more frequent (perhaps by a factor of 3 to 10) than is assumed in the PRA (MHB 1989, chapter 5).

The NRC's review (Reilly *et al.* 1989) ruled out control-building ventilation failure as a contribution to core-melt frequency. It supported MHB's finding about the importance of earthquakes and estimated their contribution to core-melt frequency at 6.5×10^{-5} per RY. A particularly interesting finding of the NRC review was evidence of a high (although quite uncertain) contribution from hurricane-induced flooding, namely 5×10^{-4} per RY. This has implications for emergency

TABLE 2.6. Initiating Event Categories Contributing Significantly to Core-Damage Frequency: The TMI-1 PRA

Description	Percent Contribution to Core-Damage Frequency	Mean Frequency per Reactor Year
INTERNAL	80.6	4.43×10^{-4}
Loss of Support Systems:	52.8	
Loss of CBV	36.4	2.00×10^{-4}
Others	8.2	4.53×10^{-5}
Loss of Off-site Power[a]	5.3	2.90×10^{-5}
Loss of River Water to Pumphouse	2.9	1.58×10^{-5}
All Other Transients	11.1	6.09×10^{-5}
Very Small LOCAs (including steam generator tube rupture)	10.1	5.58×10^{-5}
All Larger LOCAs	6.5	3.58×10^{-5}
LOCA Outside Containment	< 0.1	1.00×10^{-7}
EXTERNAL	19.4	1.07×10^{-4}
Fires Explicitly Modeled[b]	15.7	8.64×10^{-5}
All Other Fires and All Internal Floods	< 2.0	$< 1.00 \times 10^{-5}$
Earthquakes	0.5	2.70×10^{-6}
External Flood	1.4	7.5×10^{-6}
Tornado	<< 0.1	1.2×10^{-8}
Turbine Missile	< 0.1	2.3×10^{-7}
Aircraft Crash	< 0.1	1.0×10^{-7}
Toxic Chemical	< 0.1	2.6×10^{-7}

[a]Loss of off-site power could also be included in the external category.
[b]Fires, though internal to the plant, are ususally categorized as external events.

Source: PLG (1987, I:page 2-11).

planning, which needs to take into account all three sets of findings (see box; see also Table 2.1).

Core-Melt Frequency Contribution (per RY)	TMI-1 PRA	MHB Review	NRC Review
Internal Events	4.4×10^{-4}	2.1×10^{-4}	2.9×10^{-4}
External Events	1.1×10^{-4}	1.2×10^{-4}	6.6×10^{-4}

MHB Findings on Level-2 PRA Issues

MHB addresses the same tasks as would a full-fledged "Level 2" PRA—identification and analysis of accident sequences, and estimation of the associated source terms. Insofar as the resources devoted to this task were much fewer than would be required for a PRA, however, the MHB report should be seen as a scoping analysis.

Findings on Core-Melt and Containment Failure Probability

As mentioned above, MHB did not find a large contribution from loss of control-building ventilation to be credible, and they found that earthquakes may be responsible for 6 percent rather than 0.5 percent of core-melt frequency. The PRA did not explicitly estimate the speed of development of core-melt sequences. MHB made a preliminary estimate of this parameter, and their findings are summarized in Table 2.7. According to this table, sequences that account for 55 percent of core-melt frequency will proceed to core-melt within 1-2 hours of initiation. Notably, most of the earthquake-initiated sequences fall into this category. This finding is significant because emergency-response capability is likely to suffer substantial degradation after a severe earthquake.

Table 2.8 shows MHB's estimate of the probabilities of various containment failure modes. The most dramatic failure mode is that arising from direct containment heating that will arise from a high-pressure melt ejection event, which will distribute molten core material in finely divided form throughout the containment atmosphere.

The distribution of probabilities shown in Table 2.8 is entirely a product of judgments. In light of the preceding discussion of uncertainty, it is clear that probability estimates of this kind are not appropriate for policy use.

TABLE 2.7. Distribution of TMI-1 Core-Melt Frequency by Initiating Event and Sequence Timing: MHB Estimates (per reactor-year)*

Timing of Core-Melt	All Initiating Events	Internal Initiating Events	All External Initiating Events	Earthquakes
Very Early (1-2 hours)	1.85×10^{-4}	7.28×10^{-5}	1.13×10^{-4}	2.04×10^{-5}
Early (3-5 hours)	5.83×10^{-5}	5.83×10^{-5}	0	0
Late (8 hours or more)	9.51×10^{-5}	9.01×10^{-5}	5.23×10^{-6}	1.03×10^{-6}
All Sequences	3.36×10^{-4}	2.19×10^{-4}	1.18×10^{-4}	2.14×10^{-5}

*Estimates are from MHB (1989).

TABLE 2.8. Containment Failure Modes and Probabilities for TMI-1: MHB Estimates

Containment Failure Mode	Best Estimate Frequency
SS/H2-Very Early	5.6×10^{-9} (11% seismic)
SS/H2-Early	5.5×10^{-7}
SS/H2-Late	5.6×10^{-7} (2% seismic)
HPME/DCH-Very Early	1.4×10^{-5} (13% seismic)
HPME/DCH-Early	5.6×10^{-6}
HPME/DCH-Late	9.3×10^{-6} (1% seismic)
Steam Explosion-Very Early	1.8×10^{-6} (11% seismic)
Steam Explosion-Early	5.6×10^{-7}
Steam Explosion-Late	9.5×10^{-7} (1% seismic)
De-inerting Burns-Very Early	8.6×10^{-5} (20% seismic)
De-inerting Burns-Early	5.7×10^{-7}
SGTR Tube-Very Early	1.6×10^{-6}
SGTR Tube-Early	6.6×10^{-7}
SGTR Tube-Late	3.1×10^{-5}
SGTR Steam Leak-Late	4.0×10^{-6}
Direct Bypass-Very Early	3.8×10^{-8}
Aux. Bldg. Bypass-Early	1.0×10^{-7}
Intact-Very Early	8.2×10^{-5} (2% seismic)
Intact-Early	8.7×10^{-5}
Intact-Late	4.9×10^{-5} (2% seismic)
TOTAL	3.7×10^{-4}*

Due to rounding, total does not agree with core-damage frequency.

Source: MHB Technical Associates (1989, page 7-21).

Recommended Severe Accident Assumptions for TMI Emergency Planning

The philosophy underlying the following recommendations is that emergency planners should plan for the worst of all physically possible events. Consideration of the relative probabilities of severe and less severe releases might be appropriate if those probabilities were susceptible to estimation by scientifically defensible methods. This is not the case.

As indicated above, there are solid grounds for assuming that the probability of core-melt accidents approaches or exceeds 1×10^{-3} per reactor-year—a value that is generally regarded as justification for emergency planning. Thus, emergency planning for "worst case" events requires the estimation, for each core-melt sequence, of the most severe set of accident characteristics that is physically realizable. An optimistic/pessimistic approach, as described above, is appropriate to this task.

Potential Sequence Timing

Core-melt could proceed to completion in as little as 0.5 hours, or it might take a day or more. The most rapid core-melts would be associated with low-pressure sequences in which coolant inventory is rapidly lost from the reactor-coolant system while radioactive decay heat is at a high level. More slowly developing sequences could arise in a variety of ways and could involve a delay between the initiating event and the onset of core-melt that could be as short as an hour (or less) or as long as a day (or more). Figure 2.6 illustrates one way in which the time development of an accident sequence might depend upon particular aspects of the accident. For the more slowly developing sequences, which will often be high-pressure sequences, the time between core uncovery and the completion of core-melt will typically be about one hour. If a containment bypass pathway (e.g., event V) is in effect at the time when core-melt begins, then a release to the atmosphere will begin at that time. Likewise, if a pre-existing breach in the containment occurs or if containment isolation is not effective, then a release to the atmosphere will commence as soon as radioactivity enters the containment atmosphere.

For those cases in which the containment envelope is intact when core-melt begins, the point at which molten core slumps into the base of the reactor vessel is particularly sensitive. It is at this point in the sequence that phenomena such as steam explosion or high-pressure melt ejection can occur. Severe source terms can arise should one or more of these phenomena lead to immediate containment failure. Should the containment survive the challenges imposed by phenomena occurring around the time of core slump, a period of time (lasting from a few hours to a day or more) will elapse during which the containment has a reasonable chance of remaining intact. At the end of this period, the rise in internal pressure, together with the accumulation in the containment atmosphere of flammable gases arising from core-concrete interaction and other mechanisms, will again place the containment at risk. Certain potential circumstances—an "aftershock" following an earthquake that has initiated a core-melt accident, or the gradual heating of part of the containment envelope arising from localized deposition of radioactive material—could render the containment vulnerable even during the "safe" period.

Potential Source Terms

It is proposed that the "pessimistic" Surry release fractions, from the NUREG-1150 analysis (NRC 1987a) should be used as an upper limit for releases from TMI Unit 1. These estimates reflect the consideration of relevant phenomena in a scientifically defensible manner. If a range of estimates is required, the "optimistic" release fractions should be used as a lower bound. To illustrate this range of estimates, Table 2.9 provides release fractions for four release categories (containment-failure "bins" 16, 6, 9, and 10) and four radionuclide groups.

The accompanying thermal energy release rate could span a wide range. The release could occur at almost any point on the containment structure, or from the surrounding buildings.

An inverse relationship will exist between the thermal energy release rate and the duration of the initial burst of atmospheric release. However, once an escape path through the containment envelope has been created, some level of release may continue through that pathway for many hours. Most prominently, if the sequence is such that the molten core mass remains largely inside the cavity that is located under the reactor vessel, then vigorous core-concrete interaction can be expected. That interaction could lead to a continuing evolution of radioactivity into the containment atmosphere and on to the environment.

Hurricane-Induced Flooding

The NRC's review of the PRA found that hurricane-induced flooding of the plant site is the major contributor to core-melt frequency and

TABLE 2.9. Range of Release Fractions Estimated in NUREG-1150 (Draft) for the Surry PWR: Selected Estimates

	Radionuclide Group			
	Iodine	Cesium	Tellurium	Ruthenium
Early Containment Failure				
Direct Heating (Bin #16)				
Pessimistic	0.80	0.72	0.69	0.33
Optimistic	3.3×10^{-3}	3.0×10^{-3}	0.01	1.7×10^{-4}
Isolation Failure (Bin #6)				
Pessimistic	0.18	0.084	0.36	0.02
Optimistic	4.1×10^{-4}	6.2×10^{-5}	0.013	2.3×10^{-5}
Late Containment Failure				
Rupture (Bin #9)				
Pessimistic	0.17	0.15	0.24	4.2×10^{-5}
Optimistic	3.0×10^{-4}	1.0×10^{-7}	6.7×10^{-4}	1.0×10^{-7}
Leak (Bin #10)				
Pessimistic	0.05	0.03	0.037	2.2×10^{-3}
Optimistic	3.1×10^{-4}	5.3×10^{-6}	1.3×10^{-3}	2.4×10^{-6}

Source: Benjamin *et al.* (1987, Appendix C).

accounts for about 50 percent of the total frequency. This finding is important for emergency-planning purposes because a national warning system is in place for hurricanes and a predictable interval will separate the passage of the hurricane and the rising of the river level. Moreover, emergency responses to the hurricane itself will be necessary.

The recommendations offered here have been selected after review of a large body of literature on severe accidents. That literature is still evolving, and the recommendations may at some point require revision as the state of knowledge matures.

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3

Monitoring and Modelling Atmospheric Dispersion of Radioactivity Following a Reactor Accident

Gordon R. Thompson and Richard E. Sclove,
with contributions from Ulrike Fink and Peter Taylor

It is generally recognized that nuclear reactors have the potential to undergo accidents that release radioactive materials to the environment. Specifically, modern commercial reactors have the potential to release rapidly (within hours from the initiation of an accident) a highly radioactive plume to the surrounding atmosphere. Effective emergency response requires that relevant authorities have the ability to monitor accurately the current behavior of the radioactive plume. Furthermore, those authorities need the ability to project the future behavior of the plume and the potential radiation exposures of members of the public. Such projection is performed by use of theoretical models, often employing computer calculations.

This chapter reviews, as a combined subject, the monitoring and modelling of atmospheric dispersion of radioactivity. The primary purpose of the chapter is to generate recommendations for improvements in monitoring and modelling in the context of emergency planning for nuclear plants. In pursuit of that purpose, the report examines potential monitoring and modelling capabilities as of early 1989, when the review was completed.

Current regulations of the US Nuclear Regulatory Commission (NRC) require nuclear plant licensees to have a capability to monitor accidental releases of radioactivity from the plant through engineered pathways (e.g., the plant stack). That capability, although mentioned in this chapter, is not addressed in any detail. The focus here is on the monitoring of a radioactive plume once it has entered the atmosphere. Also, licensees are required to operate a permanent program to monitor radioactivity in the plant environment. Public authorities operate somewhat parallel programs. The capabilities developed for routine monitoring, although useful during an accident, will not be reviewed in detail here.

The plume arising from a nuclear plant accident will contain radioactive material in the form of gases, vapors, and fine particles (with diameters typically in the micron range). In addition, substantial amounts of nonradioactive material (especially water) may be present in the plume. Release of the plume from the plant may occur via an engineered pathway (e.g., the plant stack) or via a nonengineered pathway (e.g., a rupture in the containment). Whatever the path of the release, all plumes will share certain characteristics, although in other ways their properties will vary widely.

Each plume will, after it has left the vicinity of the reactor, be invisible. Also, its radioactive burden will not be directly detectable by people, although it is possible some accompanying materials will detectable, at least at short distances from the reactor, through taste and smell.

In other respects, each plume will be unique. First, the total quantity of radioactivity in the plume, and its isotopic composition, could vary widely. Second, release of the plume could occur at ground level or at various elevations of the reactor structure. Third, the plume could, depending on its thermal energy and the prevailing atmospheric conditions, rise to elevations ranging from zero to hundreds of meters. Fourth, release characteristics could differ widely in the way they develop over time; at one extreme the greater part of the release might be concentrated in a period of less than one hour, whereas at the other extreme the release might remain more or less uniform over a period of days. Finally, atmospheric dispersion of the plume will show widely varying behavior according to weather conditions and terrain effects. Figure 3.1 illustrates some potential plume types. The top illustration in this figure shows a hot plume that can be expected to rise substantially; the lower illustrations show cooler plumes, released either continuously or over a short time period.

The travelling plume will expose people to radiation through several pathways (Figure 3.2). People will be exposed to gamma radiation from the passing cloud ("cloud shine") and from radioactive material deposited on the ground or on other surfaces ("ground shine"). Direct deposition of radioactivity may contaminate the skin. Radioactive material can be inhaled and may subsequently become concentrated in various body organs (e.g., radioiodines will concentrate in the thyroid).

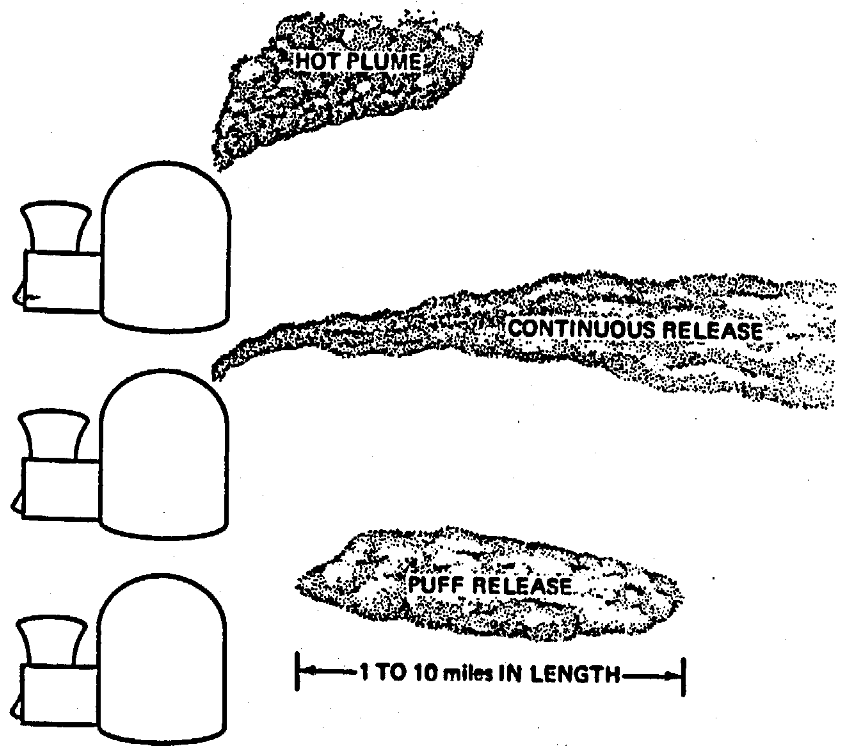


FIGURE 3.1. Examples of plume types. *Source:* McKenna *et al.* (1987, 4:60).

Finally, doses can arise from the ingestion of contaminated food, water, or milk.

This chapter begins with a discussion of objectives, which also outlines current NRC regulations governing monitoring and modelling, the role of monitoring and modelling in emergency response, and needed objectives. Next, the authors review current capabilities, relevant to the Three Mile Island (TMI) plant, which reside with the licensee, the Commonwealth of Pennsylvania, and the federal government. Then, current capabilities at other locations, both in the United States and in Western Europe are considered. Finally, a discussion of potential components of an improved monitoring and modelling system rounds out the chapter.

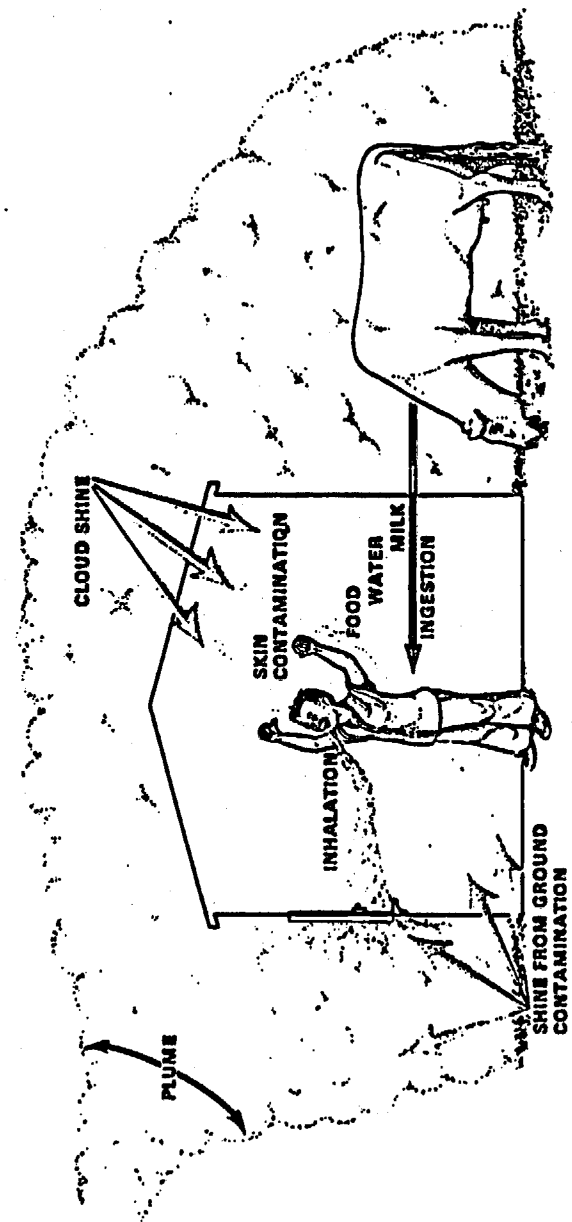


FIGURE 3.2 Radiation-dose pathways. *Source:* McKenna *et al.* (1987, 2:65).

Objectives of Monitoring and Modelling

It is important to be clear about the objectives of monitoring and modelling in the context of nuclear reactor accidents. Currently, these objectives are not clearly articulated and must be inferred from NRC regulations or from current emergency-planning practices. This discussion seeks to provide an explicit and appropriate set of objectives.

Current NRC Regulations

In the NRC's emergency-planning regulations, the clearest statement of the overriding purpose of such planning is as follows: "No operating license for a nuclear power reactor will be issued unless a finding is made by NRC that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency" (NRC 1980a, 55409).

On the specific matter of monitoring, these regulations state that on-site and off-site emergency plans must meet the following standard: "Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use" (NRC 1980a, 55409).

These somewhat vague statements receive more concrete expression in a document, NUREG-0654, in which the NRC and the Federal Emergency Management Agency (FEMA) jointly set out criteria for the preparation and evaluation of emergency plans:

> The overall objective of emergency response plans is to provide dose savings (and in some cases immediate life saving) for a spectrum of accidents that could produce off-site doses in excess of Protective Action Guides (PAGs). No single specific accident sequence should be isolated as the one for which to plan because each accident could have different consequences, both in nature and degree. Further, the range of possible selection for a planning basis is very large, starting with a zero point of requiring no planning at all because significant offsite radiological accident consequences are unlikely to occur, to planning for the worst possible accident, regardless of its extremely low likelihood. (NRC and FEMA 1980, 6)

Among its numerous provisions, NUREG-0654 sets out obligations of a plant licensee in regard to "accident assessment," which subsumes plume monitoring and modelling:

> An additional emergency activity for which facility licensees have primary responsibility is accident assessment. This includes prompt action to evaluate any potential risk to the public health and safety, both onsite and offsite, and timely recommendations to State and local governments concerning protective measures. In some situations, there could be a need for protective measures within short time intervals—a half-hour or perhaps even less—after determination that a hazard exists. For this reason, licensee emergency planners must recognize the importance of prompt accident assessment at the source. (NRC and FEMA 1980, 26)

In placing this burden of accident assessment on the licensee, the document explicitly notes that more emphasis is to be placed on the in-plant identification of potential hazards than was the case before the promulgation of regulations in 1980. Under previous practice, the licensee expected to base its notification of off-site organizations and its recommendation of protective actions on the actual measurement of radioactivity in the environment (NRC and FEMA 1980, 26). That is not to imply, however, that the NRC and FEMA currently regard monitoring and modelling as having an insignificant role in accident assessment.

To support its role in accident assessment, the licensee is required to install instruments to measure a variety of plant parameters. The operating ranges of these instruments must be sufficient to accommodate parameter values that might be experienced during a severe accident. Current NRC guidance for this instrumentation is provided by Regulatory Guide 1.97 (NRC 1980b). In the context of monitoring and modelling, this guidance calls for instrumentation to measure parameters in the following categories:

1. radiation levels in the containment and in associated buildings;
2. release rates of radioactivity through engineered release pathways (e.g., the plant stack);
3. radiation levels in the environment; and
4. meteorology.

Licensees are also required to have the capability to translate the readings of these instruments into levels of human exposure, both on site and off site. As an important part of this capability, each licensee must be able to make near real-time predictions of the atmospheric dispersion of radioactivity.

Role of Monitoring and Modelling in Emergency Response

It will be noted from the preceding discussion that the overall emergency-planning objective set out in NUREG-0654 calls for "dose savings." Hidden behind this innocuous statement is a decision by the NRC not to require that emergency plans have as their objective the minimization of doses or the prevention of their exceeding Protective Action Guide (PAG) levels. By contrast, our model plan has adopted the following primary objective: "to minimize the harm to life associated with a range of nuclear power plant accidents, encompassing both minor events and the rarer, more severe accidents" (Golding *et al.* 1992, 15).

In pursuit of this primary objective, our plan seeks to keep doses below levels established in PAGs and "as low as reasonably achievable" (ALARA) below these levels (Golding *et al.* 1992, 15 and 28-33). Pursuit of this objective carries implications for emergency planning generally and for monitoring and modelling specifically. Our discussion of proposed approaches or systems (as opposed to our review of *existing* approaches and systems) needs to be viewed in the light of this objective.

Monitoring and modelling provide decision makers with valuable information together with information from other sources, to determine if activation of the emergency-response system is indicated and, if so, what emergency-response measures should be adopted. Thus, the monitoring and modelling system must serve two somewhat different functions. The first is to provide warning that an atmospheric release has occurred. Accuracy of measurement or prediction is not essential in this role; instead, the priority is to provide rapid notification of the presence of radioactivity in the atmosphere at levels above some appropriate warning threshold. The second function is more complex but basically involves providing an ongoing and timely assessment of actual and potential radiation doses to off-site populations.

This second, ongoing, role demands as much accuracy, in respect to both environmental measurement and dose projection, as can reasonably be obtained. Integration of monitoring and modelling is required, both to improve progressively the accuracy of modelling and to guide the allocation of monitoring resources. A particular problem will be the estimation of the accident "source term" (the magnitude and composition of the release). In-plant monitoring instruments will be able to measure a release that follows engineered pathways, but the more severe releases are unlikely to follow such pathways. Thus, the source term must usually be inferred from monitoring and modelling information. This requirement adds to the importance of effective integration of monitoring and modelling. Figure 3.3 illustrates the steps that must be taken in projecting off-site radiation doses. In reality, these steps would proceed iteratively, drawing upon both monitoring and modelling information.

Monitoring will involve the tracking and sampling of the moving plume, together with the measurement of radiation levels in regions over which the plume has passed. This latter function is important because the ground-shine dose from deposited radioactivity may be a substantial component of the total radiation doses accrued by off-site populations. For both kinds of monitoring, the first priority for monitoring teams will be to obtain rapid measurements of some key parameters (e.g., the location of the plume center-line, the radiation levels at that center-line, and the boundaries of areas wherein ground contamination exceeds some appropriate threshold). Also, priority should be given to monitoring at those distances (up to 10-20 miles) within which acute health effects might occur. As time and resources permit, measurements can become more refined and more extensive in their geographic scope.

Human radiation dose will not be directly measured. Instead, the translation of monitoring data into levels of human dose will require some form of modelling. At its simplest, this modelling will use simple formulae or tables (e.g., to calculate, given a measured air concentration of radioiodine, the inhalation thyroid dose to a person exposed for a particular time period). As the number of relevant factors increases, so does the complexity of modelling. Should it be necessary, as will often be the case, to estimate the time-dependent accrual of doses by spatially diverse subpopulations, then a computer-based model is essential.

Decision makers can use current and projected dose estimates in at least four ways. First, they will be able to rank the exposed subpopulations in terms of their actual or potential doses, thereby allowing emergency-response resources to be allocated in the most appropriate manner. Second, they will be able to determine which protective actions (evacuation, sheltering, respiratory protection, thyroid blocking) are appropriate to recommend to each subpopulation. Third, knowledge about the path of the plume and about the distribution of ground contamination can inform the selection of evacuation routes that minimize doses. Finally, emergency-response resources can be conserved by allocating those resources in a manner that will minimize exposure of personnel and contamination of equipment.

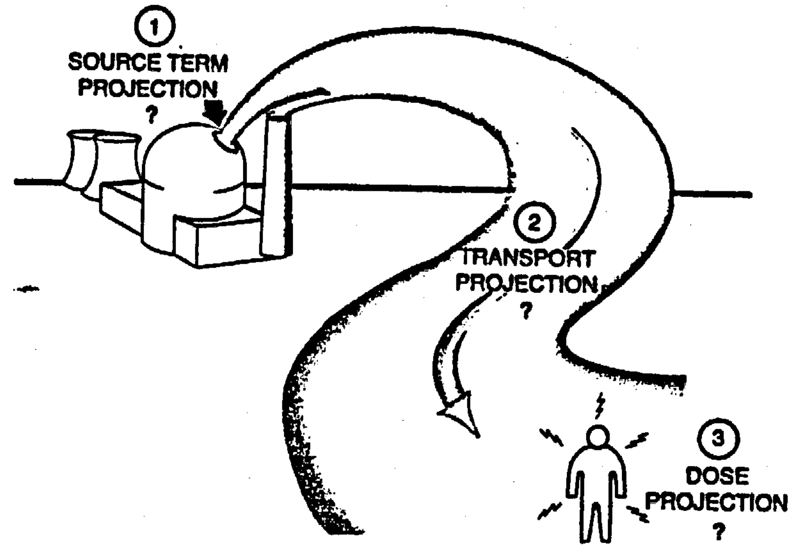


FIGURE 3.3. Steps in predicting dose. *Source:* McKenna *et al.* (1987, 2:100).

Given our objective of minimizing doses, it is important to continue careful monitoring and modelling into the regime of relatively small doses—that is, PAG levels and below. This may eventually involve monitoring and modelling of plume exposure pathways on a relatively fine spatial scale out to distances of 100 miles or more from the reactor. Such an effort would represent a substantial expansion over the scale of effort involved in the 10-mile-radius-emergency-planning zones set forth in current US regulations and guidelines.

Two concrete examples illustrate how monitoring and modelling can be important in emergency response. In the 1957 fire at the Windscale reactor, an air sampler half a mile away provided the earliest indication that an accident was in progress. At this time, a fire had been under way in the reactor for two days (Selby *et al.* 1987). Thus, the Windscale fire demonstrates the potential role of monitoring in providing independent warning of an accident.

A second example arises from the experience of evacuating people during the 1986 Chernobyl reactor accident. According to a US report on this subject: "A key problem that occurred during the evacuation was that the plume apparently followed the evacuation route for a large distance; as a result some bus drivers received high exposures" (DOE *et al.* 1987, page 7-25). It is not clear whether this statement refers to passage of the plume prior to or during the evacuation. The release of radioactivity began at about 0130 hours on 26 April 1986 and continued for ten days. Evacuation commenced, initially from the town of Pripyat, at 1400 hours on 27 April. In any event, this experience demonstrates the potential role of monitoring and modelling in guiding the choice of evacuation routes so as to minimize dose.

While considering the role of monitoring in providing warning of the Windscale accident, one should recall that US commercial nuclear plants are currently required to have an in-plant capability to monitor the radioactivity released during an accident. Monitoring instruments must be in place to measure release rates of radioactivity through engineered pathways, thus providing warning. If, as would be expected for the more severe accidents, the release takes a nonengineered pathway (e.g., a ruptured containment), then in-plant instrumentation should still provide warning of an accident, as, for example, through detection of high radiation levels in the containment. If the in-plant instruments work as expected, a direct repetition of the Windscale experience is unlikely. Nevertheless, external monitoring systems are important because they can provide independent warning of a radiation release.

Proposed Objectives

In light of the preceding discussion, we propose an emergency monitoring and modelling system designed to achieve the following objectives:

- detection of any accidental release of radioactivity from the plant;
- rapid measurement of key parameters of the radioactive plume (e.g., location of plume center-line, radiation levels at that center-line, boundaries of areas in which ground contamination exceeds specified thresholds) up to distances of 10-20 miles from the plant;
- rapid projection of future off-site radiation exposures, up to distances of 10-20 miles from the plant;
- rapid estimation of key source-term parameters (e.g., size and composition of the release) to allow more accurate dose projection;
- timely projection of future off-site radiation exposures potentially leading to doses exceeding PAG levels; and

- ongoing refinement of both the measurement and projection of off-site radiation exposures, covering a range of potential doses from the highest levels down to PAG levels and below.

Current Capabilities Relevant to the TMI Plant

To set the scene for our recommended program, we review current monitoring and modelling capabilities relevant to the TMI plant. It should be emphasized, however, that this review is not exhaustive and makes no attempt to catalog precisely all the monitoring and modelling resources available to the licensee and to public authorities. Instead, the focus here is on the general level of capability of each relevant organization.

Monitoring

The licensee, required to operate a permanent program to monitor radioactivity in the environment of the TMI plant, conducts periodic sampling of air, surface water, soil, precipitation, fish, aquatic plants, fodder crops, fruit, and milk. Measurements of cumulative radiation dose rely on an array of more than 80 thermoluminescent dosimeters (TLDs) located at distances ranging from less than half a mile to more than ten miles from the plant—on average more than five TLDs per 22.5-degree sector (GPU Nuclear Corporation 1986, Table 22). Although not required by the NRC to do so, the licensee has since 1981 maintained a system to measure, in real time, gamma radiation in the vicinity of the plant. This system (the Reuter Stokes Sentri system) uses 16 pressurized ion chambers located at distances ranging from 0.1 to 3.5 miles from the plant.

Turning now to the licensee's resources devoted exclusively to emergency monitoring, one finds that these are rather limited (GPU Nuclear Corporation 1986, 73.0-76.0 and Table 17). They consist of one mobile laboratory, a fixed laboratory (at the Environmental Assessment Command Center), and equipment for two two-person field-monitoring teams (with two additional two-person teams as backup.)

Gamma spectroscopy can be performed only at the fixed laboratory, which apparently possesses one germanium and one sodium iodide detector. The mobile-laboratory and field-monitoring teams will be able to monitor gross beta and gamma radiation levels and collect samples of air and other environmental media. Filter cartridges from air samplers can be read in the field to give gross air concentrations of iodines or particulates, but filter cartridges and other samples must be taken to the fixed laboratory for more detailed analysis. During a severe accident, the licensee would presumably seek additional resources, either from elsewhere in the corporation or from other parties, but this might well entail substantial delays.

Communication between the licensee and its mobile laboratory and field teams will be by radio. For communication with public authorities, the licensee will normally use telephone lines. The licensee's emergency plan (GPU Nuclear Corporation 1986) is vague about how the large volume of monitoring data is to be processed and transmitted. The vagueness suggests that the licensee may not fully appreciate the difficulty of the task.

The monitoring capabilities of the Commonwealth of Pennsylvania reside primarily with the Bureau of Radiation Protection (BRP), which is part of the Department of Environmental Resources. The Bureau's headquarters are in Harrisburg, and the western and eastern offices are in Pittsburgh and Wernersville, respectively. The total Bureau staff complement in 1987 was 54 persons (BRP 1987).

In the event of a reactor accident, the Bureau must work closely with the Pennsylvania Emergency Management Agency (PEMA). It is envisioned that, in the context of monitoring and modelling, the Bureau will

- conduct accident assessment at BRP headquarters;
- deploy monitoring teams to verify the licensee's field measurements;
- arrange for needed federal government support (primarily monitoring support);
- arrange for analysis of environmental samples at the Department of Environmental Resources laboratory in Harrisburg; and
- discuss potential protective actions with the licensee, and provide PEMA with BRP recommendations as to such actions. (PEMA 1987, Appendix 12, page E-12-9)

Pennsylvania authorities must be concerned about potential accidents at a number of nuclear plants in addition to the TMI plant. The Bureau of Radiation Protection conducts independent monitoring around each nuclear plant in the state, as well as more general radiological monitoring, and publishes the results annually.

The emergency-monitoring capability of the Bureau is similar to that of the TMI licensee, except that deployment of mobile assets to the vicinity of the plant may be slower. Through access to the Department of Environmental Resources laboratory in Harrisburg, the Bureau can perform detailed radioassay of environmental samples, providing an

assay of alpha, beta and gamma-emitting radionuclides. As regards field monitoring teams, equipment is available at each Bureau office to support two such teams, with one set of equipment in reserve. The Bureau's field monitoring teams will be able to perform the same functions as the licensee's teams—that is, the monitoring of gross beta and gamma radiation levels, the collection of air and other environmental samples, and the field reading of air sampler filter cartridges (PEMA 1987, Appendix 12).

Deployment time from the Bureau's Harrisburg office to the TMI site is estimated to be one hour, whereas a deployment time of eight hours is assumed if mobile assets are sent from the two subsidiary Bureau offices or from another remote location (BRP 1982). The mobile Bureau laboratory could be at a remote location when an accident began at the TMI plant. Field-monitoring teams, equipped with pocket dosimeters to measure their own exposure, have instructions to withdraw when readings reach 500 mR, or when the air concentration of radioiodines exceeds 1×10^{-5} Ci/m^3 (BRP 1982).

By contrast with both the licensee and the Pennsylvania Bureau of Radiation Protection, the federal government commands impressive resources for radiological monitoring. Substantial delays will occur, however, before these resources can be marshalled. Table 3.1 lists the roles of various federal agencies during an emergency response to a nuclear plant accident. The US Department of Energy (DOE) is required to coordinate off-site monitoring and data evaluation (the latter function can be taken to include modelling). The lines of responsibility for the federal government response are shown in Figure 3.4, which clearly shows the DOE's coordinating role in radiological monitoring.

To set up a federal radiological monitoring and assessment center, it may be necessary to deploy seven or eight 40-ft-flatbed truckloads of equipment to the vicinity of the affected plant. At a typical site, an advance party could be expected to arrive within six-eight hours, and it is planned that a full federal center would be operational within 24 hours (Doyle 1987).

Neither the licensee nor the BRP maintains an aerial monitoring capability in Pennsylvania, although this is a prominent part of the planned federal response. In view of the value of aerial monitoring as a means of providing a rapid assessment of off-site contamination, it is instructive to examine the potential delays in implementing such monitoring under current practice. Experience from the 1979 TMI accident is also of interest here. DOE's aerial monitoring will be conducted by its contractor, EG & G Energy Measurements Inc. With the exception of two helicopters located at Andrews Air Force Base in Washington, DC, the EG & G aircraft are all normally located in Las Vegas, Nevada (Jobst

TABLE 3.1. Roles of Federal Agencies During an Emergency Response

Agency	Role During an Emergency
Nuclear Regulatory Commission [Cognizant Federal Agency]	Coordinates technical evaluation assessment
Federal Emergency Management Agency	Off-site logistical response
Department of Energy	Coordinates off-site monitoring and data evaluation Provides technical assistance EG & G and Atmospheric Release Advisory Capability
Department of Health and Human Services	Assists with protection of human health Provides technical and non-technical assistance
Department of Agriculture	Assists in the development of agricultural protective measures and damage assessments
National Oceanic and Atmospheric Administration	Provides meteorological services as required
Department of Housing and Urban Development	Assists with location housing, if required
Department of the Interior	Responsible for federal lands, parks, and natural resource facilities
Department of Transportation	Can assist with location and coordination of transportation resources
National Communication System	Can provide communication support to federal agencies

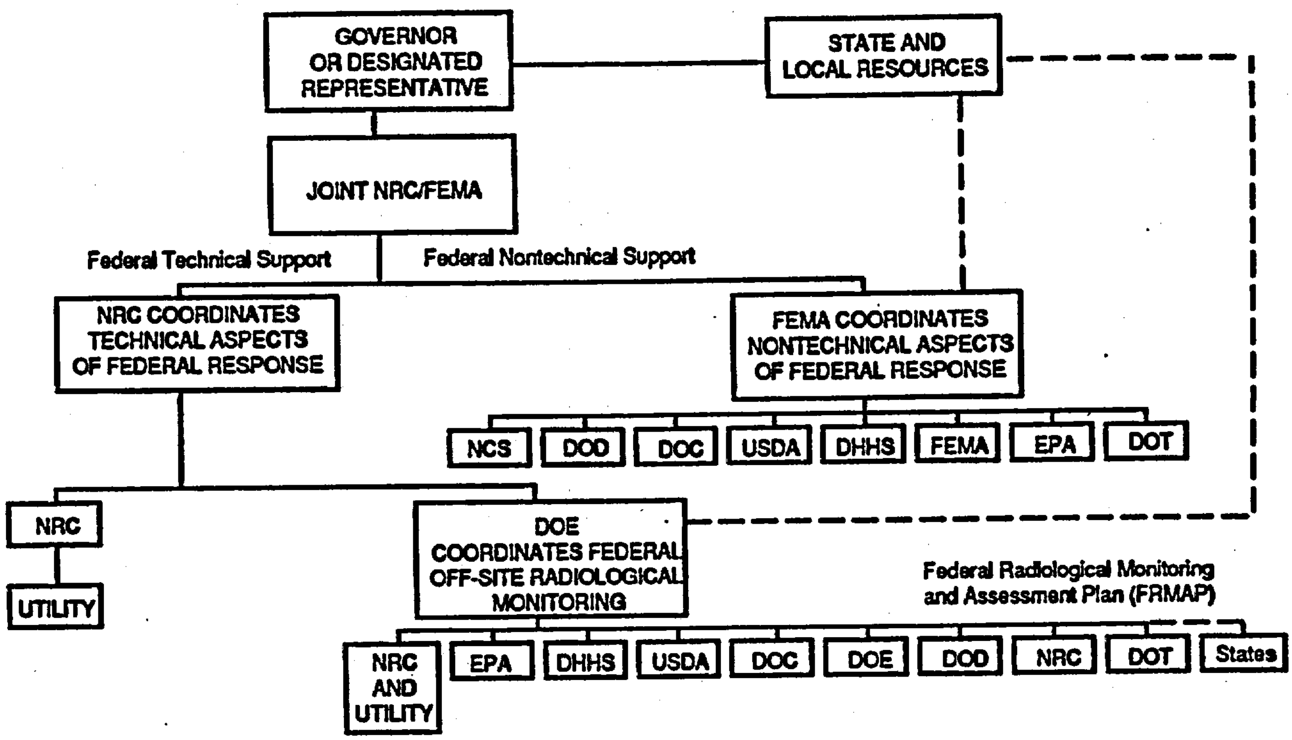
Source: McKenna *et al.* (1987, 5:80).

1987a and 1987b). Thus, it could take many hours for aircraft to arrive on the scene, and it would take even longer before they could be fully operational, because EG & G's airborne navigation equipment requires that ground crews first go out and set up two transponders in elevated positions (Dahlstrom 1987, 86; Jobst 1987b).

The 1979 TMI accident, for example, began at 0400 hours on 28 March. At 0655 hours, the shift supervisor declared a Site Emergency, but the utility was not able to make contact with the NRC until 0750 hours. Two hours later, at about 1000 hours, DOE put the EG & G team at Andrews Air Force base on standby alert. One hour later the NRC asked that the team fly to the Capital City Airport, seven miles away from TMI. Although the team arrived at Capital City Airport at 1330 hours, the first aerial measurements for which we have found documentation were at about 1800 hours. Eight hours later, at about 0200 hours on 29 March, additional EG & G aircraft from Las Vegas arrived on the scene (Rogovin *et al.* 1980, 384, 386, 1204-1206). Thus, a delay of six hours occurred before the EG & G aerial monitoring team was alerted, and a further one hour delay occurred before the team was asked to proceed to the TMI region. Flight time to Capital City Airport required 2.5 hours, and presumably a further delay while ground crews set up transponders (the record is unclear on this point). In any event, the alert of the EG & G team was at a relatively convenient time—mid-morning on a Wednesday. It is possible that they would have responded less quickly if called upon during the night or on a weekend.

A Pennsylvania State Police (PSP) helicopter was used early on 28 March to transport monitoring teams, which may have taken readings while airborne. According to the NRC's Special Inquiry Group, the licensee alerted the PSP at 0714 hours on 28 March, and a PSP helicopter was furnished to fly radiation monitoring teams to their survey locations (Rogovin *et al.* 1980, 1202). These teams presumably comprised licensee personnel, insofar as the radiological-monitoring program of the Pennsylvania Bureau of Radiation Protection did not begin until about 1045 hours (Rogovin *et al.* 1980, 1194). The EG & G effort during this accident, however, was clearly the first aerial monitoring to be conducted by trained personnel with appropriate equipment and procedures.

Above, we noted that current BRP instructions call for field monitoring teams to withdraw if exposure levels exceed a certain threshold. We are also informed that federal agencies would be reluctant to send field monitoring crews directly into the path of a highly radioactive plume (Gant 1987). This may limit the extent of monitoring that can be carried out in a severe accident and thus provides a further argument in



DHHS = Department of Health and Human Services

DOC = Department of Commerce

DOD = Department of Defense

DOE = Department of Energy

DOI = Department of Interior

DOT = Department of Transportation

EPA = Environmental Protection Agency

FEMA = Federal Emergency Management Agency

HUD = Department of Housing and Urban Development

NCS = National Communication System

NRC = Nuclear Regulatory Commission

USDA = United States Department of Agriculture

FIGURE 3.4. Federal response management for a radiological emergency at a nuclear power reactor. *Source:* McKenna *et al.* (1987, 81).

favor of aerial monitoring (which should also lead to reduced exposure of monitoring personnel).

Modelling

Licensees are required by the NRC to have a capability to make near real-time predictions of the atmospheric dispersion of radioactivity. This requirement demands modelling at two levels of sophistication. At the simplest level, the Class A model must be able to produce, within 15 minutes of the classification of an incident, estimates of initial plume transport and diffusion. The more sophisticated Class B model, which in view of its scope can be only a computer-based model, must represent the spatial and temporal variations of plume distribution and must account for deposition of radioactive material from the plume onto the ground and other surfaces.

The TMI licensee employs the MIDAS model (GPU Nuclear Corporation 1986, 76.0). A simple Gaussian plume model written by Pickard, Lowe and Garrick Inc., it assumes constant wind direction and weather conditions and is run on the computer facilities of Digital Graphics Inc. in Rockville, Maryland. Apparently, MIDAS is a Class A model that can be adapted to perform the Class B function.

An evaluation of available models for real-time simulation of atmospheric dispersion concluded that straight-line Gaussian models such as MIDAS have limited usefulness in predicting short-term (one-hour average) air concentrations at a specific time and place (Lewellen and Sykes 1985). These models are much more helpful in predicting maximum one-hour average air concentration when specific time and location are not considered, a type of prediction that is useful in the control of air pollution.

The NRC will employ at least two computer-based atmospheric dispersion models during an accident. NRC site/dose-assessment teams will employ the IRDAM model, a straight-line Gaussian model that runs on a personal computer (Reilly 1987). At the NRC's Operations Center in Washington, DC, the MESORAD model will be used. A puff model that can operate on either a polar or Cartesian grid (Scherpelz *et al.* 1987), MESORAD will be used in conjunction with the computer code MENU-TACT, which attempts to estimate the accident source term by modelling the transport of radionuclides within a nuclear plant (Sjoreen 1987).

Within the federal government, however, the most sophisticated computer modelling of radioactive plume dispersal is the Atmospheric Release Advisory Capability (ARAC), developed and located at DOE's Lawrence Livermore Laboratory in California. Although best

known for its modelling on medium or larger (including global) scales, ARAC's geographic resolution has recently been extended so that the model can, in principle, project all the way in to a source point (Dickerson 1987). ARAC is currently connected directly to over 50 Department of Defense and DOE facilities, and also stands ready to provide support to the NRC and other federal agencies in the event of a nuclear plant emergency. ARAC would, for example, be able to coordinate with EG & G aerial monitoring teams, developing projections based on initial meteorological and (if available) radiological conditions. These projections could, in turn, be used to advise radiation survey teams on where to look next. ARAC could then upgrade its projections as more data became available (Dickerson and Knox 1987; Burson 1987).

An ARAC official (Dickerson 1987) suggests that in response to a nuclear plant emergency today, ARAC could provide a first, tentative calculation within one hour of notification. This holds for sites currently connected to the ARAC computer system, and for notification during normal working hours. It is planned to reduce this response time to about 15 minutes where these conditions hold; clearly, response times could be longer under other conditions.

The principal model used by ARAC is the MATHEW/ADPIC model. This is a three-dimensional model that is normally used with a 200 km x 200 km grid. For application during the 1986 Chernobyl accident, the grid was expanded to 1920 km x 1920 km, and the global PATRIC model (originally developed to estimate material transport and diffusion in the stratosphere) was also employed (Dickerson and Knox 1987). PATRIC has enjoyed reasonable success in predicting ground-level air concentrations at various locations in Europe. These predictions were made by an iterative process in which an initial estimated source term was modified to improve the fit between predicted and measured air concentrations. Monitoring results or source-term estimates from within the then USSR were not available during this process. The iteration was, however, made in a relatively relaxed fashion, over the days and weeks following notification of the accident (DOE 1987).

ARAC's relative success in accounting for dispersion of the Chernobyl plume across Europe does not necessarily bode similar success in the context of an accident at a US nuclear plant. That case would trigger an urgent requirement for the analysis of plume dispersion on a relatively small scale (initially, 10-20 miles). In pursuit of this requirement, the initial ARAC calculation would use fairly gross topographical maps (heights above sea level at points 0.5 km apart), current Air Force Global Weather Central (i.e., probably not local) meteorological data (projected not to vary for the next two hours), and a normalized source term for radioactivity. A second calculation might be ready in as little as another half hour, reflecting some human judgment concerning meteorology, and incorporating actual radiation data, if any were available. A serious limitation, however, is that no pre-established communication arrangements with utilities or state authorities currently exist. Thus data might have to be communicated to ARAC by phone, and the model's graphic outputs relayed back via voice description over the phone or by telefax. Another limitation is that, without prior preparation, it is doubtful that utility or state personnel would be prepared to interpret ARAC data with appropriate awareness of the severely limited data base upon which initial projections would be based.

These and related considerations have led various NRC and other experts to become skeptical of ARAC's ability to provide meaningful support to the emergency-response process, especially during the time-urgent phases of an accident. A major concern is that ARAC cannot incorporate or predict local variations in meteorological conditions. In any case, real-time data on micrometeorology are not currently available for most nuclear plants and their surrounding locales. Yet failure to take into account variations in local weather conditions can lead to spurious model results. Also, one expert whom we interviewed indicated that he would not show an ARAC result to an NRC commissioner or state governor during an emergency, because the elaborate color graphics exert a compelling psychological effect that obscures the model's questionable reliability.

The full value of the ARAC service can be realized only if careful preparations are made for its application at a particular site and if detailed local meteorological information is provided. Also, it may be that ARAC's major current tool—MATHEW/ADPIC—is not optimal for emergency use. Nevertheless, despite these problems, ARAC represents the most sophisticated modelling capability currently existing in the United States.

Integration of Monitoring and Modelling

The accuracy of modelling in the initial phases of an emergency situation is likely to be quite poor. Table 3.2 provides an estimate of the various uncertainties, some of which arise because actual plumes will be more erratic in their behavior than the idealized plumes used in models. Over progressively longer time periods, the average air concentration in an actual plume may come to resemble more closely that in a model plume (at least in those cases in which wind direction remains constant). Although this may be somewhat comforting in terms of

TABLE 3.2. Estimated Range of Uncertainty Between Projected and Actual Off-Site Dose for a Severe Accident (Core-Melt)[a]

Element	Uncertainty Factor[b]		
	At Best	Most Likely	Near Worst
Source term (event and sequence)	5	100-1,000	100,000
Dispersion			
— Diffusion (concentration)	2	5	10
Transport (direction)	22°	45°	180°
Transport (rate)	1	2	10 (low wind speed)
Dosimetry	3	4	5
Overall (dose and direction)	10 22°	100-10,000 45°	100,000 180°

[a]These estimates are for an averaged dose at a location (e.g., over 15 to 30 minutes), not for a specific or single monitor reading.

[b]Ratio of a likely maximum or minimum value to the expected median value.

Source: McKenna *et al.* (1987, 2:44).

model accuracy, it raises an important question about monitoring. Field monitoring teams, for example, would obtain quite different results. Thus, initial monitoring information from ground-based teams, which will inevitably be spotty in its coverage, should be regarded with caution. Such early information may not correctly represent the behavior of the plume and may be of little value in estimating the source term and improving the accuracy of model-based predictions.

In addition, wind direction or other meteorological characteristics are likely to change during the course of a release. The release itself may have varying properties over time. In such cases, a handful of ground-based monitoring teams will not be able to provide adequate information. Simple plume models, such as straight-line Gaussian models, will also have very limited usefulness under such conditions.

Yet, it is important that emergency-response decision makers have access to a rapid assessment of potential doses. Emergency planners should, therefore, give attention to overcoming the limitations arising from existing arrangements. Later, we take up some possible approaches to providing a more effective and integrated monitoring and modelling program. First, we need to review the current arrangements.

The licensee's emergency plan at Three Mile Island is uninformative as to how the integration of monitoring and modelling is to be achieved. The plan states that: "Verification of the model projections will be accomplished by comparison with field-monitoring teams and real-time gamma detectors" (GPU Nuclear Corporation 1986, 76.0). This statement begs the question of exactly how the "verification" is to be accomplished. It also ignores the limitations of models such as MIDAS, the intrinsic difficulty in obtaining good monitoring results with only a few ground-based teams, and the problems involved in rapidly processing and integrating a diverse body of information.

The Pennsylvania Bureau of Radiation Protection is in a similar position, except that the agency has no computer-based modelling capability. This difference may or may not be significant. On the one hand, the Bureau will lack the capability to process rapidly a large number of calculations. On the other hand, it will not be misled by the superficially precise output of an inadequate model. In either event, the Bureau, like the licensee, may be unable to provide accurate dose assessment on a time-urgent basis, particularly when the source term is unknown.

By contrast, the federal government has the capability, in principle, to provide accurate dose assessment under a variety of conditions. EG & G's airborne monitoring capability, once deployed, could allow rapid wide-area monitoring. Through close cooperation with ARAC, monitoring results obtained by EG & G and by ground-based teams could be integrated with ongoing modelling. Thus, source-term estimates could be progressively refined and monitoring assets could be optimally allocated. Delays will occur, however, before federal government resources become available during an accident. Also, it is not obvious that those resources would, if called upon in an emergency, be used to their full potential.

Current Capabilities at Other Locations

The capabilities at other locations suggest that precedents exist for the improvements that we shall propose. The capabilities reviewed here were selected because they are relevant to the TMI context and because information about them was readily available. This review does not pretend to be comprehensive, in terms either of addressing all relevant capabilities or of exhaustively analyzing those capabilities that we do address.

Illinois

The Illinois Department of Nuclear Safety (IDNS) maintains a technical capability for oversight of nuclear plant operations that at time of writing appeared to be superior to that in any other state of the United States. This capability is devoted partly to the monitoring of in-plant parameters, partly to the monitoring and modelling of atmospheric releases. To bolster the state's plan for radiological accidents, the department's Office of Nuclear Facility Siting Safety (ONFS) relies upon an advanced computerized communications network. According to Gallina (1993), the integrated Remote Monitoring System (RMS) comprises four distinct components, each linked by dedicated telephone lines to IDNS headquarters in the state capital:

1. The Nuclear Reactor Data Link (RDL) tracks and transmits continuously up to 1500 parameters from process computers at each of the 13 nuclear power plants in Illinois. Computer-polled information is updated every two to four minutes.
2. The Gaseous Effluent Monitoring System (GEMS) identifies and measures radioactive materials present in gaseous releases via the ventilation stacks of each nuclear power plant. With separate, highly sensitive detectors for particulates, iodines, and noble gases, as well as multichannel analyzers and separate computers for each unit, the GEMS provides both quantitative and qualitative information on releases of gaseous effluents and thereby enhances OFNS capability to assess the potential ipact of an accidental release on populations in the vicinity of a particular nuclear power plant.
3. The Liquid Effluent Monitoring System (LEMS) is currently in place at two sites, Quad Cities and Zion. Prototype monitors track the presence of radioactive materials in reactor effluents under both normal and accident conditions. When concentrations of radioactivity exceed pre-established levels, IDNS evaluates the need to implement protective action for the general public.
4. The Gamma Detection Network (GDN) reflects an attempt to address the contingency "bypass scenarios"—in which radioactive materials are released through an unmonitored pathway—that now concern the NRC. The GDN uses extremely sensitive pressurized ion chamber gamma detectors, up to 16 of them placed uniformly around each nuclear plant at a distance approximately two miles, to detect all releases of radioactivity, whether monitored or unmonitored by the plant, to the environment. In addition, OFNS maintains three portable gamma-

detector units—designed to operate with solar or battery power and equipped also with meteorological equipment—that stand ready for dispatch to the scene of an accident.

Data from all four RMS components are transmitted to redundant computers at the Radiological Emergency Assessment Center (REAC) in Springfield.

There state-of-the-art computer programs continuously analyze the data for signs of potential abnormalities or releases of radiation. Trained communicators staff the REAC on a 24-hour basis.

To ensure rapid assessment capability, particularly in the event of bypass scenarios, mentioned aboved, the IDNS has connected several key decision-making personnel by modems at their home computers to the REAC computer.

Atmospheric dispersion and dose assessment is performed by a model adapted from the MESORAD model. The IDNS version will compute plume dispersion and ground-level doses for distances up to 50 miles and will provide graphical displays in color for distances up to 16 miles. IDNS staff have refined the model so that its predictions can be adjusted to correspond to the results of field monitoring; the adjustment is performed within a "sphere of influence" surrounding each monitoring point.

Radiological monitoring is performed by the Radiological Assessment Field-Team (RAFT), a component of IDNS. Other state agencies, especially the Illinois Environmental Protection Agency, will provide resources (e.g., monitoring teams, air samplers) to assist the field-team effort during an emergency. The field-team resources include two large mobile laboratories can be stuffed and on their way to an accident scene within an hour's notice. In addition, there is a radio-equipped 40-vehicle fleet for field monitoring. Field teams can perform gamma and beta radiation surveys and sample and analyze air, water, soil, plants, milk, and food in preplanned locations "close enough to the accident scene, but remote enough to that releases cannot interfere with sample preparation and analysis" (Gallina 1993, 40).

In summary, IDNS has a relatively effective capability, through its arrays of pressurized ion chambers, to detect a release. If that release follows a major engineered pathway, the installations in place are likely to provide an accurate assessment of the source term. As pointed out above, however, the more severe releases are likely to follow nonengineered pathways. Ground-level monitoring by IDNS will be superior to that performed by the Bureau of Radiation Protection in Pennsylvania, due to the greater resources commanded by IDNS. There will, of course, be delays attendant on the relocation of mobile assets of

radiological assessment field teams from their current positions to the site of an accident. Also, IDNS appears to have made no preparations for airborne monitoring.

Savannah River Plant

The Savannah River Plant's mobile ground-level monitoring capability is quite highly developed and is integrated into a relatively comprehensive radiological emergency-response program. Major components of the overall program include an on-site Emergency Operations Center (EOC), a real-time meteorological analysis and dose modelling system (Weather Information Display), mobile radiation-sampling vans, and a sophisticated ground-mobile radiation sampling and near-real-time analysis laboratory known as TRAC (Benjamin 1987).

The Weather Information Display (WIND) meteorological analysis capacity comprises real-time data collection from nine local weather towers, regional meteorological data obtained via computer from the National Weather Service, two mini-computers (operating in parallel, to back one another up), and a large network of terminals for data display and calculations—including terminals located in the homes of key emergency-response personnel (Benjamin 1987, 124; Hoel 1987, 128). An array of fixed-position detectors and—in the event of an emergency—mobile vans carrying air sampling equipment and the mobile TRAC laboratory provide radiation detection and measurement (Addis, Kurzeja, and Weber, 1987, 135; Hoel 1987, 129; Schubert 1987). The TRAC mobile laboratory carries a direction-sensitive sodium-iodide gamma-ray detector capable of determining a radioactive plume's location relative to the vehicle's heading, as well as air sampling and analysis equipment able to determine a plume's radionuclide composition, a mini-computer for real-time data analysis and storage, a Loran-C electronic navigation system, and an electric generator. The TRAC laboratory is capable of sampling and analyzing data while in motion (Sigg 1987a).

Sampling vans and the TRAC vehicle are in radio communication with Savannah River's Emergency Operations Center, which uses meteorological analysis and dose projections from the WIND system to direct the deployment of the mobile radiation-monitoring teams (Addis, Kurzeja, and Weber, 1987, 135f.; Hoel 1987, 129; Sigg 1987a, 132).

Savannah River's ability to respond to an emergency on a time-urgent basis depends upon the time and location of a radioactive release. For an on-site emergency during normal working hours, it is claimed that the emergency operations center and the WIND system can be staffed and operating within minutes, sampling vans deployed in 15-30 minutes, and the TRAC vehicle operational in about half an hour. During off-shift hours, personnel can begin to respond immediately from their homes via their WIND terminals, but it takes 30-40 minutes to staff the emergency operations center, 45-60 minutes to deploy sampling vans, and about 90 minutes to get TRAC under way. In the case of an off-site emergency, these times would all be lengthened by the time it took vehicles to reach the area of an accident (Hoel 1987, 129). On top of these times one must add the time it takes TRAC to undertake its analyses and the mobile vans to deliver their air samples for analysis. For instance, once TRAC has begun its air sampling, on-board radionuclide measurement and analysis can (at least when the radionuclide concentration in a sample is small) require 20 minutes for radioiodines and up to an hour for transuranics (Sigg 1987a, 132-134).

Despite a sophisticated design, the TRAC vehicle has not been optimized for use within the context of a major nuclear power plant accident. In fact, for the latter purpose it may well be overly designed and unnecessarily expensive, incorporating highly sensitive radiation-detection and measurement equipment that is best used to distinguish a relatively small level of radioactive contamination from natural background levels. In the case of a nuclear plant catastrophe, less sensitive equipment may suffice, and data analysis could be accomplished more quickly. The cost of the TRAC vehicle and equipment is reported (Sigg 1987b) to be about $800,000 (exclusive of the extensive hardware and software R&D that went into the system's design). To duplicate the vehicle would cost somewhat less, but an analogous mobile laboratory re-optimized for operating during a major nuclear plant accident could be smaller, carry less sensitive equipment, and consequently probably cost some hundreds of thousands of dollars less (again, exclusive of R&D).

Nevada Test Site

Here we limit our discussion to one aspect of monitoring at the Nevada Test Site—the use of remotely sited on-line automated radiation-monitoring devices. These devices form part of a monitoring system whose most important task is the monitoring of accidental releases of radioactivity (venting) following underground nuclear weapons tests (Sanders 1987). A network of 40 pressurized ion chambers was installed in the early 1980s, located up to 50 miles from the Operations Coordination Center. These ion chambers measure gross gamma radiation, have a calibrated range from 1×10^{-3} R/h to 1×10^{3} R/h, and are sensitive down to 1×10^{-5} R/h. Telephone lines transmit readings—once per minute when radiation levels exceed a preset alert level—to the Operations Coordination Center where computers process the data for graphical display.

More recently, monitoring units employing satellite data transmission have also been installed at the Nevada Test Site. From an initial ten permanent and ten portable units, some 50 units now are on line. Half of these units support the Environmental Protection Agency (EPA) off-site community monitoring program around the Nevada Test Site. In addition, it is planned to extend the scope of the system to encompass gamma detectors that the EPA intends to install in parts of the western United States. This will lead to a total of approximately 400 monitoring units employing satellite data transmission.

Each of the ten initial permanent units is designed to monitor both gross gamma radiation levels and meteorological parameters; the portable units measure only radiation. Both types of unit transmit data, normally at four-hour intervals, to the Operations Coordination Center via satellite. If radiation levels exceed a preset threshold, the transmission rate rises to once every three minutes. Sufficient transmission capability does not exist, however, to allow more than a few units to report at this rate.

The Hanford Reservation

Here our discussion is limited to the mobile laboratory operated by the Pacific Northwest Laboratory. This laboratory is intended to support emergency response both on and off site (Wilson 1987). A 27-ft-long trailer that is towed by a large, four-wheel-drive, pick-up truck houses the mobile laboratory. In addition to radiation monitoring and analysis equipment, the laboratory contains a Weathertronics wind-detection system, a radio base station and four portable radios, a gas-powered electric generator, and overnight facilities for a crew of four. The trailer carries a germanium radiation detector, an accompanying analysis system, and a lead-shielded counting chamber that can detect radioisotopes in beakers containing liquid, soil or vegetation samples, and on air filter papers. A full laboratory team consists of fourteen persons from the Laboratory's Health Physics Department, who are trained to be able to substitute for one another in carrying out a variety of technical tasks. During an emergency, the Laboratory expects that a minimum of four crew members could be assembled within one hour. This vehicle is not so sophisticated as the TRAC vehicle at Savannah River and would serve a function similar to that of the BRP or IDNS mobile laboratories.

Diablo Canyon Plant

Since 1981, the Diablo Canyon licensee has deployed a computerized radiation monitoring and modelling system (Walker *et al.* 1982). Among the main features of the system are fixed off-site monitors, eleven pressurized ion chambers, located from 5 to 20 miles from the plant. Data from these instruments, from two meteorological towers, from 36 in-plant monitoring stations, from mobile monitoring units, and from a gamma spectrometer in the "emergency counting room" are relayed to a computer system located in the Technical Support Center. In the event that the central computer becomes unavailable, individual terminals can independently perform plume dispersion modelling. During an exercise in 1981, this transfer of functions was performed successfully in less than 30 minutes.

Based on the limited information available to us, it appears that the Diablo Canyon licensee has made considerably greater preparations than has the Three Mile Island licensee to accommodate the flow of monitoring and modelling information during an accident. This will, in principle, allow more effective use to be made of monitoring resources.

Western Europe

A complete review of monitoring and modelling capabilities in Western Europe is far beyond the scope of this chapter. Instead, the focus here is on one facet of monitoring—centralized on-line monitoring of gamma radiation over large areas (not just in the immediate vicinity of nuclear plants). Information relevant to four countries is presented here:

Austria maintains a radiation monitoring system consisting of two elements (Austria 1986):

- a nationwide environmental monitoring system involving measurements of air, rain, and surface-water contamination; and
- the "Strahlenfrühwarnsystem" (radiation early-warning system), that features on-line centralized measurement of gamma dose-rate.

The latter system has functioned since 1975. A total of 336 on-line gamma dose-rate measuring stations are deployed, as shown in Figure 3.5. These stations are usually located on the tops of buildings, and no

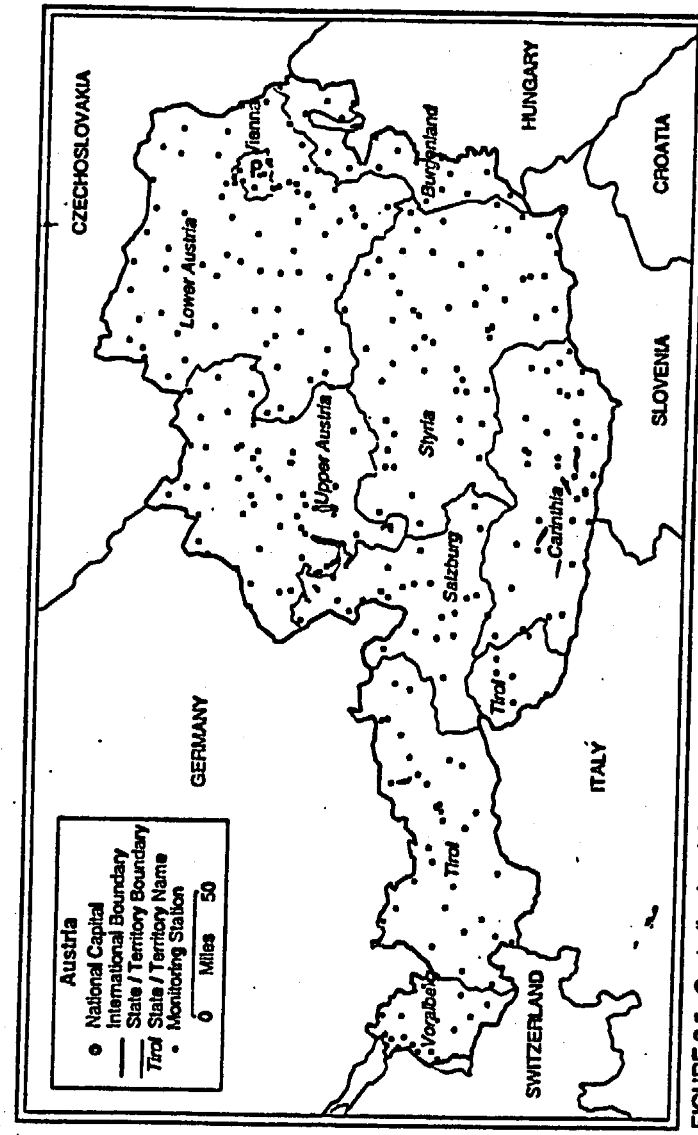


FIGURE 3.5. Centralized real-time monitoring of ground-level gamma dose rate: Monitoring stations in Austria.
Source: Clark Labs.

village is situated more than 15 km from a station. Data are sent to several regional centers and one national center. Eight alarm levels, from below 30×10^{-6} R/h to above 30 R/h, are defined.

West Germany has an even more elaborate system for monitoring radiation in air, rain, and surface water. This includes 1560 on-line gamma dose-rate measuring stations and a communications and data processing system.

Switzerland first established a centralized monitoring system (Netz für Automatische Dosis-Alarmierung und -Messung, or NADAM) in 1982, with eight on-line gamma dose/rate measuring stations. This network was subsequently expanded and now has 51 stations, connected to automated meteorological stations. Local dose-rate values are transmitted, via leased telephone lines every ten minutes to a central station in Zurich (Honegger *et al.* 1984).

The United Kingdom is also planning a centralized system. This will include about 100 on-line gamma dose-rate measuring stations, covering all of Britain on a basis of 100-km-grid squares.

Table 3.3 provides some comparative information for these four European countries, for Pennsylvania, and for the contiguous states of the United States. It is apparent that West Germany plans the highest density of on-line gamma monitoring stations. Although Pennsylvania and other states in the United States contain some similar stations (e.g., those operated by the TMI licensee, by IDNS, and at the Nevada Test Site), these (with the partial exception of the Nevada Test Site system) are not intended for wide-area coverage.

European governments, particularly in light of their experience during the Chernobyl accident and more recent Russian accidents, are concerned about the possibility of radioactivity's entering their territory without warning. This provides part of the motivation for wide-area monitoring capabilities.

The United States harbors less concern that transboundary atmospheric transport of radioactivity will occur without warning. Yet, a wide-area on-line monitoring system in the United States would fulfill other functions, as it does in European countries. First, it would provide independent confirmation of monitoring information gained by other means. Second, it could provide warning of localized concentrations of radioactivity ("hot spots") at considerable distances from the point of release. Third, its output is in a form that could be provided directly to public-information media, thus improving the public's confidence in the quality of information provided to it.

TABLE 3.3. Centralized Real-Time Monitoring of Ground-Level Gamma Dose Rate, Various Countries/Regions

Country/Region	Area (10^3 km^2)	Population Density in 1988 (persons/km^2)	Density of Monitoring Stations (stations per 10^3 km^2)
Austria	84	91	4.0
West Germany	248	246	6.3
Switzerland	41	158	1.2
United Kingdom	244	234	0.4
Pennsylvania	116	102	NA
United States (lower 48)	7,828	26	NA

Potential Elements of an Improved System

Before addressing potential improvements in monitoring and modelling, we need to consider two overriding issues:

- the extent to which emergency-response decision makers should rely on monitoring and modelling in the early phases of an accident; and
- the need for coordination among a variety of actors (licensee, state authorities, NRC, DOE, etc.).

These issues are important because they will perforce play a central role in the implementation of an emergency plan. It is inappropriate to advocate improvements in technical capabilities unless those improvements are selected in light of careful consideration of these two issues.

In regard to the first issue, our interviews with federal government experts (e.g., Martin 1987; McKenna 1987; Weiss 1987) speak to a view that emergency responses during the early phases of an accident should be guided by plant conditions and, if available and relevant, by very simple monitoring information (e.g., the detection of a large atmospheric release). Indeed, the philosophy in the entire emergency-planning community is moving in this direction, to the extent that many officials are willing to contemplate a precautionary emergency response based solely on an adverse trend in plant parameters.

This trend in thinking is quite sensible, for two reasons. First, the incidence of acute health effects can often be substantially reduced if an emergency response (especially evacuation) is initiated early in a severe accident—even before core-melt has begun. Clearly, monitoring information will play little or no role in guiding such a decision. Second, at least with present systems, the quality of monitoring and modelling information is likely to be low during at least the first hour or two after a release has commenced. Initial emergency-response decisions should be guided primarily by other information.

Nevertheless, improvements in the quality of time-urgent monitoring and modelling information are essential. Once a release has occurred, accurate monitoring and modelling information will allow adaptation of emergency responses to changing conditions. As the accident moves beyond its early phases, accurate knowledge about potential doses to all relevant populations (including doses down to PAG levels and below) will allow emergency-response decisions to proceed in a near-optimal fashion.

Although coordination among federal agencies is slowly improving (Burson 1987), preplanned coordination between the federal government on the one hand and utilities and state and local governments on the other remains primitive. For example, under present arrangements, federal agencies participating in a radiological emergency intend to wait until they have arrived on the scene of an accident to negotiate with state and local authorities as to the establishment of common mapping systems and radiation-measurement units (Doyle 1987, 288; Gant 1987). These activities can and should be undertaken well in advance of an accident.

At the federal level, DOE has designated its Nevada Operations Office as the lead organization in establishing a federal radiological monitoring and assessment center (FRMAC). As a result, efforts are under way to develop map-coordinate systems and units of measurement that will be common throughout the federal government. Yet even after this process of standardization is in place within the federal government, it is not clear that the same standards will be extended to utilities and state agencies. The role of a FRMAC is to support, not preempt, state-level decision making. This role makes the federal government reluctant to be aggressive in promoting its standards. State governments, for their part, are generally jealous of their prerogatives and reluctant to change their current practices (Burson 1987; Gant 1987).

In the remaining discussion, we outline key elements of an improved monitoring and modelling system.

Airborne Monitoring

Airborne monitoring will be a prominent part of the federal response to a reactor accident. Yet, in Pennsylvania and apparently in all states, no state-level planning for airborne monitoring exists. This situation is unfortunate, since it prevents airborne monitoring from being used at a time—during the first few hours of a release—when it would be particularly valuable.

Several experts whom we interviewed supported requiring states or utilities to develop a stand-by aerial radiation-monitoring capability. No more rapid way to begin to characterize the location and magnitude of a major radioactive release exists. Such a capability would require not only reliable availability of aircraft and radiation monitoring equipment, but also training and exercise of personnel in the intricacies of flying, navigating, and monitoring in a high-radiation environment.

In the case of a large radioactive release, these intricacies need not be so complex as one might suppose. Although EG & G crews are familiar with the difficulty of finding and tracking a relatively small radioactive plume (Jobst 1987b), the problem becomes much easier if the plume is large. In such cases, an instrument payload limited to gross gamma meters would produce useful results. An aircraft equipped with portable gamma dose-rate meters could be flown in a search pattern until the plume is detected; then, the aircraft would be flown in a tighter pattern while the shape and size of the plume are estimated (Burson 1987; Martin 1987).

EG & G aircraft, when they arrive on the scene, will have much more capability than this. These aircraft can, using sodium iodide detectors, perform gamma spectroscopy in real time, with both cockpit display and tape recording of the readings. Air samples can be taken and the filters subjected to gamma spectroscopy while in flight. Whole gas samples can also be compressed into bottles for post-mission laboratory analysis. The sodium iodide gamma detectors can be expected to detect ground deposition (from an altitude of 100 meters) at values as low as 1×10^{-6} Ci/m^2 and airborne activity at concentrations as low as 2 pCi/liter (Boyns 1987; Dahlstrom 1987).

It is beyond the scope of this chapter to recommend the precise monitoring equipment that should be available for use in a state or utility-operated aircraft and the appropriate level of personnel training. Such a recommendation would reflect a multifaceted optimization process. Clearly, however, a quick-reaction locally based airborne monitoring capability needs not have the sophistication that EG & G can command. The use of relatively portable equipment would mean that dedicated aircraft were not required, with a resulting cost saving.

Some experts we interviewed were skeptical about the value of a locally based airborne monitoring capability. They emphasized the skill involved in plume tracking (Jobst 1987b; Wolff 1987) or the lack of a pressing need for accurate monitoring in the period before EG & G aircraft arrive (Chester 1987). Cost concerns were also present.

Assuming that developing a local, stand-by aerial monitoring capability is worthwhile, there are compensating strengths and weaknesses exist between helicopters and fixed-wing airplanes. The latter can fly, and therefore survey, more rapidly. Also, if it is sufficient to take readings at a relatively high altitude, they can fly above turbulence or vision-obstructing clouds. On the other hand, helicopters—although somewhat slower—are more maneuverable, can fly safely at lower altitudes and in conditions of poor visibility, and can land more easily to drop off air samples or report results in case of radio failure (Burson 1987; Dickerson 1987; Hansman 1987).

We are informed that fixed-wing airplanes can fly uninstrumented at low altitudes as long as the cloud ceiling is above 1000 ft, with three-mile visibility. Helicopters can fly in worse conditions, but it becomes dangerous for them when visibility is less than one mile. A well-instrumented helicopter might be able to fly with a 200 ft ceiling and a half mile visibility. In general, visibility will dominate over other adverse conditions, such as high turbulence or icing (Hansman 1987).

Data from Capital City Airport (near TMI) for the period 1964-1975 (NOAA, 1979, Table 9) indicate that the cloud ceiling was 1000 ft or less and/or visibility less than three miles (i.e., dangerous for a noninstrumented airplane) for 11.5 percent of the time. The ceiling was 400 ft or less and visibility was down to one mile or less (i.e., dangerous for an instrumented plane or uninstrumented helicopter) for 2.5 percent of the time. Ceiling was 200 ft or less and visibility was one half mile or less (i.e., unsafe even for a highly instrumented helicopter) for 0.8 percent of the time.

Some concern has been expressed to us that the rotor turbulence of a helicopter may prevent accurate air sampling, but this will not be a concern if the helicopter is moving horizontally. We are informed (Dickerson 1987) that, as long as it is flying at 60-80 knots or more, a helicopter does not produce more local turbulence than does a fixed-wing aircraft. For detection of gamma shine from a surrounding or nearby cloud, local turbulence will be irrelevant.

Concern has also been expressed (Bates 1982) that contamination of the aircraft and its instruments may lead to erroneous measurements. This problem could also occur for ground-based monitoring assets, but it might be thought more likely for an aircraft. Experts we consulted, however, argued that a skilled crew could avoid significant contamination (Burson 1987; Martin 1987).

Fixed Ground-Level Monitors

Earlier, we mentioned the on-line gamma dose-rate monitoring conducted by the TMI and Diablo Canyon licensees, by IDNS, by various European governments, and at the Nevada Test Site. Such monitoring, the most important function of which is to provide warning of a release is becoming more common. It is appropriate to place a ring of detectors around each nuclear plant, and a wide-area system is also contemplated, as in various Western European countries, to locate these detectors near population centers. This can apparently be accomplished at reasonable cost (see Table 3.4).

An intermediate strategy may also be appropriate for plants such as that at TMI. This would involve supplementing the present licensee-operated pressurized ion chambers by detectors located at centers of high population concentration (e.g., Harrisburg). Such a strategy would improve confidence that the exposure of large subpopulations will be accurately known. Some experts, however, have criticized the concept of placing a ring of detectors around each nuclear plant. For example, one group of analysts (Maeck *et al.* 1982) concluded that an

TABLE 3.4. Estimated Cost of an Off-Site System for Real-Time Dose Measurement

	Range of Costs ($000)*
1. Instrumentation ($20-40K/unit)	400 - 640
2. Data-collection and -processing equipment	40 - 110
3. Installation ($5-25K/unit)	80 - 400
4. Design and engineering	50 - 200
5. Contingency	100 - 200
	$870 - $1,610

*Based on 1982 data, range-of-cost figures are for a 16-unit station at a distance of two miles.

Source: Maeck *et al.* (1982, 44).

unreasonably large number of detectors is needed to obtain the necessary accuracy of detection.

Mobile Ground-Level Monitoring

The discussion of monitoring capability at Savannah River showed that the "state of the art" in mobile monitoring (the TRAC vehicle) may be unnecessarily sophisticated for use during severe reactor accidents. It seems more important to devote what will presumably be limited resources to increasing the number of monitoring vehicles. As compared with airborne monitoring, ground-mobile monitoring will always be handicapped in terms of its rapidity of response and its ability to cover large areas. The simplest way of reducing this handicap is to increase the number of vehicles.

That said, there is room for improving the sophistication of the field-monitoring equipment currently deployed by the TMI licensee and by the Pennsylvania Bureau of Radiation Protection. For example, the TMI licensee's mobile laboratory should have the capability for gamma spectroscopy. Careful study may identify other useful improvements. For example, EG & G has developed a vehicle-mounted system (Dahlstrom 1987) in which a gamma detector (in this case, a high purity germanium detector) can be raised to elevations as high as 7 meters using a telescopic mast. A version of this system (perhaps with a simpler detector) may be useful in the emergency-response context.

One quite inexpensive improvement to a ground-mobile monitoring capability would be the use of relay riders to take environmental samples from field monitoring teams to fixed or mobile laboratories. All-terrain motorcycles could be used, which would be relatively unimpeded by traffic jams. In this way, the productivity of the field monitoring teams would be improved.

Communications and Navigation

Current communications systems may be inadequate to handle the large volume of monitoring data that must be processed during an emergency. During the early phases, as emergency-response decisions are guided by simple information, this may not be a problem. As the accident unfolds, however, decision makers will need to address the potential exposure of ever larger populations, and the volume of monitoring data will expand rapidly. Problems in managing these data are evident from past exercises and incidents (Berry and Burson 1987).

The use of real-time data transmission (telemetry) from monitoring assets to control centers is a potential route to resolving these problems. Clearly, a suitable data-processing capability must exist at the control centers. Surprisingly, we have found limited interest in this possibility within the emergency-planning community. Savannah River has looked into real-time transmission of data. Satellites are thought to be a possibility and might be more reliable than Savannah River's use of radios, but satellites do not allow continuous transmission. Instead, monitoring vehicles pull over periodically, set up a small dish antenna, and relay accumulated data in a burst. An alternative is radio-telephones, if there is good local service. Savannah River is looking into using cellular phones with a modem to communicate data in real time. At time of writing, Savannah River's mobile monitoring vehicles used VHF radio for communication up to 10 or 20 miles. Single side band radios that do very well at much larger distances—100s to 1000s of miles—also exist. The problem is that between these two systems is a gap in the 50-100 mile range where neither works well (Sigg 1987b).

EG & G does not employ real-time telemetry of data. Instead, data are recorded on magnetic tape for post-mission analysis (Boyns 1987, 93f; Dahlstrom 1987, 85, 89). One official (Jobst 1987b) indicated that EG & G had no plan to develop a data transfer link from the air, since "DOE doesn't require it." He judged that it would be expensive, but technically feasible. Since EG & G does a certain amount of in-flight data analysis, it expected to relay a small amount of crucial data by voice over the radio. Another official (Burson 1987) concluded that the issue does not really arise for EG & G in their normal functions on the Nevada Test Site, where they are more concerned with detailed mapping than with time urgency.

We interviewed an expert (Wilkerson 1987) who was involved in the summer of 1987 with a test of an airborne, real-time data telemetry system. Given all the normal radiation detection equipment, the additional cost for the telemetry was only $5,000. In this test, data were relayed by radio-telemetry to a ground station, which did the recording and analysis. The relay was continuous, in real time, and worked to a distance of up to 60 km. In our view, real-time data telemetry and the associated data-processing capability are neglected areas, deserving at minimum a vigorous R&D effort.

Navigation is most important for aircraft, as ground-mobile monitoring assets can more readily use maps and landmarks. For aircraft navigation, the reliable state-of-the-art is Loran C or—for greater accuracy—two fixed-position transponders and an on-board microwave (UHF) ranging device. Apparently, transponders can give very high precision—within one-two ft (Hansman 1987). EG & G uses the transponder system for its helicopters, and apparently uses Loran C for its fixed-wing aircraft (Boyns 1987, 94; Dahlstrom 1987, 86).

The transponder/UHF ranging system has two main drawbacks. First, it requires line of sight between the master set and the two slave transponders, with a maximum range (for aircraft) of three times the distance between the aircraft and the horizon. Second, someone has to go out ahead of time and set up the two slaves—in elevated locations if the terrain is not flat. One expert (Jobst 1987b), when asked if it would be worthwhile to have utilities purchase and set up the slave transponders in advance, resisted on cost grounds (but the costs he quoted in our view, are not really large; the two slave transponders cost $10,000 each and the master is $50-60,000).

Loran offers less precision—on the order of 100 ft—provided one executes a grid correction (Hansman 1987). The correction is not difficult, and can be done by computer. The entire United States has good Loran coverage; because it works at low frequency, transmitters need not be closely spaced. A Savannah River expert (Hayes 1987) told us that Loran's precision was within "a couple of hundred feet"—adequate for their emergency-response needs. Loran provides four-second position updates, is inexpensive ($400), and easy to use. One just pushes a button to elicit latitude and longitude in digital read-out.

Loran was basically designed for ships and, more recently, airplanes. The version for airplanes permits quicker response to course changes. Savannah River's mobile lab also uses the airplane version. Loran's main drawback is that position is thrown off by interference from the likes of power lines, television antennae, or cities generally. There are two sorts of technical fixes available to compensate for this. "Static notch filters" eliminate the effect of known interference sources. Recently, "dynamic notch filters" have started to become available, able to filter out unknown interference sources (Sigg 1987b).

Apart from transponders and Loran, EG & G is investigating a variety of navigational alternatives: Global Positioning System (GPS), RAND G, and Laser Ring Gyro. Each has strengths and weaknesses. For instance, Laser Ring Gyros are not very accurate and tend to drift (Jobst 1987b). We are informed that existing navigational satellites are useless for real-time purposes, because it takes 15 minutes each time one wants a position fix. Military satellite capabilities (the GPS) that should be available in the near future will be quicker, provided that non-military agencies are allowed to use them (Hayes 1987). There are currently four GPS satellites, and the system requires 18-20. The complete system should give position with high precision (several feet) and in three dimensions (but that precision is planned to be available

only to the military). For civilian users the signal will be degraded to 100 ft precision, similar to that obtainable from Loran (Hansman 1987).

At the Savannah River plant, efforts are under way to develop a geographic information system to help with emergency routing, allocation of response, etc. With this system, if a field monitoring crew radioed in its location, officials at a control center would be able immediately to call up a gridded topographic map that showed the crew's location (Hayes 1987). Thus, the Savannah River system appears to represent the "leading edge" of technological development in respect to both on-line data communication and advanced navigation systems.

Integration of Monitoring and Modelling

Modelling of atmospheric dispersion is a complex field with rapid ongoing developments, spawning an extensive literature. It is beyond the scope of this report to review fully that literature. Here we touch upon several major issues for emergency planning.

Table 3.5 summarizes the virtues and drawbacks of the three major types of model: plume models (e.g., MIDAS, IRDAM); puff models (e.g., MESORAD); and complex models (e.g., MATHEW/ADPIC). None of these models is expected to work well in calm conditions, and wet deposition is also difficult to model. The accuracy achievable by even a complex model—such as MATHEW/ADPIC—is limited by the availability of fine-scale information on local meteorology (micro-meteorology). One operating and one backup meteorology tower are required at each nuclear plant site, but these will not provide the required information. It is noteworthy that modelling capability at the Savannah River Plant relies on data from nine towers.

Two complementary approaches exist for gaining improved micrometeorological information. The first is to improve the licensee and state capability to measure directly, on a fine spatial scale, the local meteorology during an emergency. An alternative approach is to develop, over long periods of observation, a set of correlations between currently observable parameters (e.g., regional meteorology, readings from existing meteorological towers, and local micrometeorology). These correlations would then be used for predictions in an emergency situation. Technologies are available for use in the first of these approaches. For example, acoustic sounding of the atmosphere (SODAR) can significantly supplement tower measurements (SethuRaman *et al.* 1982, 23). Swedish authorities have deployed fixed SODAR units in the vicinity of nuclear plants. Mobile SODAR systems have also been developed (Meggitt and Fraser 1986, 265). It appears that Pennsylvania State University is experimenting with SODAR at the TMI site. Our

TABLE 3.5. Overall Comparison of Plume, Puff, and Complex Models

	Plume models	Puff models	Complex models
Computer resources	Minimal Microcomputer (security advantages)	Modest to intermediate according to number of puffs representing release. Can be still feasible on a mini- or microcomputer.	Large constraint of real time Needs central computer
Data requirements	Minimal	Source data or data over region	Available data from surrounding region. 3-D wind-fields.
Operator skills	Minimal	Minimal	Needs skilled team. Results need careful checking and sensitivity analysis
Distance range most applicable	Short - few km	Short	Mesoscale ~ 100 km
<u>Versatility</u>			
Time resolution	Time-integrated concentric fields over time of passage of release	Concentric fields for any required period (Useful for directing monitoring teams and interpreting measurement)	Concentric fields for any required period
Wet deposition	Extra problems calculating wet deposition when raining away from source		
Orography	No	No (unless puff trajectories allow for orography)	Attempts to treat orographic effects
Phase of accident for which most appropriate	Early to intermediate	Early to intermediate	Pre-accident training studies Post facto analysis (long-range models also useful here)

Note that no model will work well in calm conditions; it is also difficult to estimate wet deposition, particularly for severe storms. *Source:* ApSimon (1988, 220). Reproduced by permission of the International Atomic Energy Agency (IAEA).

proposed second approach is offered as a subject for R&D. At this stage, its practicality is uncertain.

At each nuclear site, the terrain presents a unique problem for modelling. In the case of TMI, the most prominent nearby terrain feature is a range of hills northwest of the plant just beyond Harrisburg, rising about 1000 ft on either side of the Susquehanna River. Development of a fully adequate modelling capability for the TMI region will require an understanding of the effects of this and other terrain features.

Systematic integration of monitoring and modelling, we conclude, is a rather primitive state. At this point, it should be viewed as a subject for intensive R&D, and recommendations as to its application are premature.

It is interesting, however, to note the approach taken by Danish government authorities (Lippert, Wilmod-Larsen, and Jensen 1986). These authorities have developed the ARGOS computer system, which produces dose estimates based solely on environmental monitoring data. A puff release model is to be added to ARGOS. This evolution may yield insight into monitoring/modelling integration.

This chapter has already proposed a set of objectives for monitoring and modelling and examined some potential elements of an improved system. Here, we outline a system that accounts for deficiencies in current monitoring and modelling and seeks to meet our proposed objectives.

It is important to reiterate that the purpose of emergency response is to reduce radiation dose, not just to the most exposed individuals but to all relevant populations. A well-designed monitoring and modelling capability must account for the distribution of population in the surrounding region. An effective system will tap and integrate existing capabilities of licensee, state or region, and the federal government. Moreover, it is equally important to draw upon the findings of ongoing R&D efforts as well as to maintain a climate that promotes such activities.

Enhancing Licensee Capabilities

The case of TMI speaks to a need for all-around improvement of licensee capabilities. All licensees should have in place

- a sufficient number of field-monitoring teams;
- advanced capability for a mobile laboratory (including the capability to conduct gamma spectroscopy);

- arrangements for rapid deployment of one or more monitoring aircraft (this need not require a dedicated aircraft);
- strong capability for the processing of monitoring data (the experience of the Diablo Canyon licensee may provide guidance);
- an ongoing effort to improve modelling capability, including field exercises in which aircraft and field teams track routine or tracer releases;
- a capability for assessing micrometeorology in the surrounding area;
- an ongoing effort to improve the coordination of licensee emergency response with that of other respondents, such as state, regional, and federal authorities.

Enhancing State and Regional Capabilities

The TMI accident highlighted the importance for close coordination between the licensee and the state of Pennsylvania. That experience has informed our specific proposals for enhancing overall capabilities at the state and regional level. We recommend:

- a sufficient number of field-monitoring teams;
- on-line access to readings from the licensee's pressurized ion chambers;
- installation of real-time gamma monitors (e.g., pressurized ion chambers) at points of population concentration (such as, in the case of TMI, Harrisburg)
- arrangements that allow rapid deployment of one or more monitoring aircraft (this need not require a dedicated aircraft);
- development of a computer-based modelling capability, to be compatible with (and linked to) ARAC;
- development of a computer-based capability for the processing of monitoring data;
- an ongoing effort to improve modelling capability, including field exercises as proposed above for the licensee (combined exercises involving the licensee state, regional, and federal organizations would be useful);
- ongoing effort to improve the coordination of emergency response with those of other actors (including relevant authorities in neighboring states); and
- development of a capability for assessing micrometeorology in the surrounding area.

Enhancing Federal Capabilities

By contrast with most licensees and state and regional authorities, the US federal government maintains an impressive capability for monitoring and modelling. Insofar as resources will not be available quickly, however, we confine our recommendations to two:

- an ongoing effort to improve the coordination of emergency response among all relevant actors; and
- a vigorous R&D program in all relevant areas.

R&D Requirements

At various points in this chapter, we have identified specific areas in which active R&D is under way. Such specific areas are important, but it is equally important that the entire field of emergency response be supported by a various R&D program. That said, we recommend that the R&D program include:

- ongoing effort to improve modelling and to advance integration of monitoring and modelling;
- investigation of real-time data telemetry;
- development of innovative monitoring technology (e.g., remotely controlled drone aircraft, lightweight portable equipment for gamma spectroscopy);
- improvement in navigation capabilities for mobile monitoring assets (including tracking of those assets by control stations);
- development of technology to assess micrometeorology; and
- ongoing refinement of technologies and systems through field exercises studied by independent observers.

It bears emphasizing that the federal government need not, and should not, be the sole sponsor of the R&D. Indeed, state governments and licensees should be actively involved in framing, sponsoring, conducting, and reviewing such R&D efforts.

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4

Safety Status of Nuclear Reactors and Classification of Emergency Action Levels

Gordon R. Thompson

The present volume as well as its companion model plan (Golding *et al.* 1992) highlights the importance of viewing emergency planning as a routine adjunct of nuclear-reactor operation. Chapter 2 argues that the rapidity with which reactor accidents can develop renders it necessary that planners prepare for prompt emergency responses, such as evacuation. In response to this need, the US Nuclear Regulatory Commission (NRC) has introduced Emergency Action Levels (EALs), which are useful in categorizing any abnormal state of a nuclear plant, up to and including the conditions accompanying a large release. These EALs fall into four classes in order of increasing severity: *Unusual Event, Alert, Site Area Emergency, and General Emergency* (Table 4.1). A determination as to which of these four emergency classes is operative primarily dictates the nature and scale of an emergency response.

It is thus important to review the adequacy of the present EAL system. This chapter describes the present concept and uses of EALs and assesses their adequacy for indicating the safety status of nuclear plants. An assessment of present and potential means for conveying information on plant status to public authorities provides the basis for key recommendations.

Reactor Status and Emergency Response

It bears emphasizing that the EAL classification itself essentially dictates the nature of an emergency response. Stepping aside from the rigidity of NRC's regulations, however, two underlying issues deserve particular examination.

TABLE 4.1. Emergency Class Descriptions

Class*	Core Status	Radiation
Unusual Event	No threat to irradiated fuel	No release above technical specifications (or annual limits)
Alert	Actual (or potential for) substantial degradation of safety	Release is small fraction of EPA Protective Action Guides beyond the site boundary
Site Area Emergency	Major failures of functions needed for public protection	Release is less than EPA Protective Action Guides beyond the site boundary
General Emergency	Actual or imminent core degradation	Dose may exceed EPA Protective Action Guides

*Classifications are based on plant instrument levels (i.e., emergency action levels).

Source: McKenna *et al.* (1987, 3:21).

First, the emergency-planning community now widely advocates a focus of attention on core melting—has it occurred or is it likely to occur? This recognition stands in contrast to the attitude prevailing at the time of the 1979 accident at Three Mile Island (TMI), when the focus on the measurement of radioactivity in the environment triggered inappropriate emergency-response decisions (McKenna *et al.* 1987, 2:45-48). Second a growing recognition of the merit of initiating emergency response prior to the commencement of core-melt, even at the risk of a false alarm, has fostered a more anticipatory stance. In this way, should the accident sequence proceed to a large atmospheric release, exposure of off-site populations can be reduced (perhaps dramatically) because extra time has been available for emergency response. Both of these issues are pursued later in this chapter and receive extensive attention in this volume as a whole.

This chapter reviews the current status of EALs and addresses an issue touched on above—the advantage of early warning in emergency response. Two subsequent sections introduce the concept of "precursors," both in generic terms and with specific reference to the nuclear plant at

Three Mile Island. If public authorities are to make optimal emergency-response decisions, they need accurate information about plant status. Present and potential means of providing that information are reviewed. Finally, the chapter concludes with recommendations of needed changes.

Current Role of Emergency Action Levels

In their present form, EAL classes date from the publication of emergency-planning criteria in the document NUREG-0654 (NRC and FEMA 1980) and supersede the accident classes laid down in NRC Regulatory Guide 1.101 (1977). The first revision of NUREG-0654 provides a fairly precise definition of the four EAL classes and describes the emergency responses appropriate to each class (see Tables 4.1-4.5). It is not at all clear, however, that the design or application of current EALs accurately captures the safety status of nuclear plants. Yet recent update of Regulatory Guide 1.01 (NRC 1992) endorses the guidance provided in NUREG-0654.

Defining Emergency Action Levels

Appendix 1 of NUREG-0654 (NRC and FEMA 1980) establishes "guidelines" for EALs. In practice, licensees tend to follow the guidelines fairly rigidly. The appendix offers the following rationale for the four EAL classes:

> The rationale for the notification (of unusual event) and alert classes is to provide early and prompt notification of minor events which could lead to more serious consequences given operator error or equipment failure or which might be indicative of more serious conditions which are not yet fully realized. A gradation is provided to assure fuller response preparations for more serious indicators. The site area emergency class reflects conditions where some significant releases are likely or are occurring but where a core melt situation is not indicated based on current information. In this situation full mobilization of emergency personnel in the near site environs is indicated as well as dispatch of monitoring teams and associated communications. The general emergency class involves actual or imminent substantial core degradation or melting with the potential for loss of containment. The immediate action for this class is sheltering (staying inside) rather than evacuation until an assessment can be made that (1) an evacuation is indicated and (2) an evacuation, if indicated, can be completed prior to significant release and transport of radioactive material to the affected areas.

TABLE 4.2. Characteristics of an "Unusual Event"

Class Description: Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.

Purpose: Purpose of off-site notification is to (1) assure that the first step in any response later found to be necessary has been carried out, (2) bring the operating staff to a state of readiness, and (3) provide systematic handling of unusual events information and decisionmaking.

Licensee Actions	State and/or Local Off-site Authority Actions
• Promptly inform State and/or local off-site authorities of nature of unusual condition as soon as discovered	• Provide fire or security assistance if requested
• Augment on-shift resources as needed	• Escalate to a more severe class, if appropriate
• Assess and respond	• Stand by until verbal closeout
• Escalate to a more severe class, if appropriate	
or	
• Close out with verbal summary to off-site authorities; followed by written summary within 24 hours	

Source: NRC and FEMA (1980, Appendix 1:4).

> The example initiating conditions listed after the immediate actions for each class are to form the basis for establishment by each licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, will initiate the emergency class. (NRC and FEMA 1980, Appendix 1:3)

It is apparent that each class is characterized by statements about core status and about off-site radiation levels; these statements are complementary because high radiation levels cannot occur unless there is core damage.

A series of tables (Tables 4.2-4.5) provides more complete definitions of the EAL classes, summarized in Table 4.1, and lists the licensee and state/local government actions recommended for each class. One discrepancy between Table 4.1 and Table 4.5 is worthy of note: For a General Emergency, the former describes the core status as one of "actual or imminent core degradation," whereas the latter adds the phrase "with potential for loss of containment integrity." In fact, the statement in Table 4.1 better represents the evolving thinking in emergency-planning circles. Chapter 2 points to a current awareness (e.g., IRSS 1989) that containment failure could be induced by phenomena arising as a result of core-melt, and could occur relatively soon after core-melt has begun—most notably, through the effects of high-pressure melt ejection. Thus, core-melt itself should be the focus of attention, rather than core-melt together with some added deficiency in containment integrity.

NUREG-0654 (NRC and FEMA 1980) provides examples of plant conditions that should precipitate each emergency class. It is through these conditions that the word "level" in the term EAL gains its meaning. In short, when instruments showing plant parameters indicate readings above a certain level (the EAL), then the plant has entered an emergency class that is appropriate to that level. Unfortunately, the concept of a quantitative "trigger" level is not always appropriate; many dangerous plant conditions are best described qualitatively. Thus, discussion of emergency classification remains mired in semantic confusion.

It is the responsibility of the licensee to determine the plant conditions that are appropriate to each emergency class for a specific nuclear plant. For TMI Unit 1, these plant conditions are set out in Table 10A of the licensee's emergency plan (GPU Nuclear Corporation 1986).

Use of Emergency Action Levels to Guide Emergency Response

Table 4.6 provides a summary of licensee and state/local government emergency-response actions that would typically be engendered by the

TABLE 4.3. Characteristics of an "Alert"

Class Description: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Purpose: Purpose of off-site alert is to (1) assure that emergency personnel are readily available to respond if situation becomes more serious or to perform confirmatory radiation monitoring if required, and (2) provide off-site authorities current status information.

Licensee Actions	State and/or Local Off-site Authority Actions
• Promptly inform State and/or local authorities of alert status and reason for alert as soon as discovered • Augment resources and activate on-site Technical Support Center and on-site operational support center. Bring Emergency Operations Facility (EOF) and other key emergency personnel to standby status • Assess and respond • Dispatch on-site monitoring teams and associated communications	• Provide fire or security assistance if requested • Augment resources and bring primary response centers and EBS to standby status • Alert to standby status key emergency personnel including monitoring teams and associated communications • Provide confirmatory off-site radiation monitoring ingestion pathway dose projections if actual releases substantially exceed technical specification limits
• Provide periodic plant status updates to off-site authorities (at least every 15 minutes) • Provide periodic meteorological assessments to off-site authorities and, if any releases are occurring, dose estimates for actual releases • escalate to a more severe class, if appropriate • Close out or recommend reduction in emergency class by verbal summary to off-site authorities followed by written summary within 8 hours of closeout or class reduction	• Escalate to a more severe class, if appropriate • Maintain alert status until verbal closeout or reduction of emergency class

Source: NRC and FEMA (1980, Appendix 1:17).

TABLE 4.4. Characteristics of a "Site Area Emergency"

Class Description: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases not expected to exceed EPA Protective Action Guideline exposure levels except near site boundary.

Purpose: Purpose of the site area emergency declaration is to (1) assure that response centers are manned, (2) assure that monitoring teams are dispatched, (3) assure that personnel required for evacuation of near-site areas are at duty stations if situation becomes more serious, (4) provide consultation with off-site authorities, and (5) provide updates for the public through off-site authorities

Licensee Actions	State and/or Local Off-site Authority Actions
• Promptly inform State and/or local off-site authorities of site area emergency status and reason for emergency as soon as discovered • Augment resources by activating on-site Technical Support Center, on-site operational support center and near-site Emergency Operations Facility (EOF) • Assess and respond • Dispatch on-site and off-site monitoring teams and associated communications • Dedicate an individual for plant status updates to off-site authorities and periodic pressure briefings (perhaps joint with off-site authorities) • Make senior technical and management staff on-site available for consultation with NRC and State on a periodic basis • Provide meteorological and dose estimates to off-site authorities for actual releases via a dedicated individual or automated data transmission • Provide release and dose projections based on available plant condition information and foreseeable contingencies • Escalate to *general emergency* class, if appropriate or • Close out or recommend reduction in emergency class by briefing of off-site authorities at EOF and by phone followed by written summary within 8 hours of closeout or class reduction	• Provide any assistance requested • If sheltering near the site is desirable, activate public notification system within at least two miles of the plant • Provide public within at least about 10 miles periodic updates on emergency status • Augment resources by activating primary response centers • Dispatch key emergency personnel including monitoring teams and associated communications • Alert to standby status other emergency personnel (e.g., those needed for evacuation and dispatch personnel to near-site duty stations • Provide off-site monitoring results to licensee, DIE and others and jointly assess them • Continuously assess information from licensee and off-site monitoring with regard to changes to protective actions already initiated for public and mobilizing evacuation resources • Recommend placing milk animals within 2 miles on stored feed and assess need to extend distance • Provide press briefings, perhaps with licensee • Escalate to *general emergency* class, if appropriate • Maintain site area emergency status until closeout or reduction of emergency class

Source: NRC and FEMA (1980, Appendix 1:12).

TABLE 4.5. Characteristics of a "General Emergency"

Class Description: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Purpose: Purpose of the general emergency declaration is to (1) initiate predetermined protective actions for the public, (2) provide continuous assessment of information from licensee and off-site organization measurements, (3) initiate additional measures as indicated by actual or potential releases, (4) off-site authorities and (5) provide updates for the public through off-site authorities

Licensee Actions	State and/or Local Off-site Authority Actions
• Promptly inform State local off-site outcries of general emergency as soon as discovered (Parallel notification of State/local) • Augment resources by activating on-site Technical Support Center, on-site operational support center and near-site Emergency Operations Facility (EOF) • Assess and respond • Dispatch on-site and off-site monitoring teams and associated communications • Dedicate an individual for plant status updates to off-site authorities and periodic press briefings (perhaps joint with off-site authorities) • Make senior technical and management staff on-site available for consultation with NRC and State on a periodic basis • Provide meteorological and dose estimates to off-site authorities for actual releases via a dedicated individual or automate data transmission • Provide release and dose projections based on available plant condition information and foreseeable contingencies • Close out or recommend reduction of emergency class by briefing of off-site authorities at EOF and by phone followed by written summary within 8 hours of closeout or class reduction	• Provide any assistance requested • Activate immediate public notification of emergency status and provide public periodic updates • Recommend sheltering for 2 mile radius and 5 miles down-wind and assess need to extend distances. Consider advisability of evacuation (projected time available vs. estimated evacuation times) • Augment resources by activating primary response centers • Dispatch key emergency personnel including monitoring teams and associated communications • Dispatch other emergency personnel to duty stations with 5 mile radius and alert all others to standby status • Provide off-site monitoring results to licensee, DOE and others and jointly assess them • Continuously assess information from licensee and off-site monitoring with regard to changes to protective actions already initiated for public and mobilizing evacuation resources • Recommend placing milk animals within 10 miles on stored feed and assess need to extend distance • Provide press briefings, perhaps with licensee • Maintain general emergency status until closeout or reduction of emergency class

Source: NRC and FEMA (1980 Appendix 1:16).

TABLE 4.6. Emergency Responses for Each Emergency Action Level

Emergency Action Level	Plant Action	Local and State Agency Action
Unusual Event	Provide notification	Be aware
Alert	Mobilize plant resources Staff centers (help for control room) Activate Technical Support Center (TSC)	Stand by[a]
Site Area Emergency	Activate mobilization; nonessential site personnel evacuate Activate TSC, Operations Support Center, and Emergency Operations Facility Dispatch monitoring team Provide dose assessments	Mobilize Staff emergency centers and dispatch monitoring team Inform public; activate warning system Take protective actions in accordance with protective-action guides (PAGs) or on an *ad hoc* basis
General Emergency	Activate mobilization Recommend predetermined protective actions (within 15 minutes) after declaring emergency	Recommend predetermined protective actions to the public based on plant conditions Initiate precautionary evacuation (2 to 5 miles)

[a]The NRC will typically activate its response centers at the Alert level.

Source: McKenna *et al.* (1987, 3:22).

designation of each class of EAL. More complete sets of typical actions are presented in Tables 4.2 through 4.5, which are drawn from NUREG-0654 (NRC and FEMA 1980).

Once again, inconsistencies will be found between the emergency-response actions presented in Table 4.6 and those presented in Tables 4.2 through 4.5. For example, Table 4.6 suggests that in the case of a General Emergency, precautionary evacuation up to distances of 2-5 miles will be a typical response. By contrast, Table 4.5 calls for sheltering, with consideration of the advisability of evacuation. As for the analogous discrepancy mentioned above, this inconsistency illustrates an evolution in emergency-planning philosophy. Since the writing of NUREG-0654, evacuation, except in special conditions, increasingly predominates as the preferred emergency response.

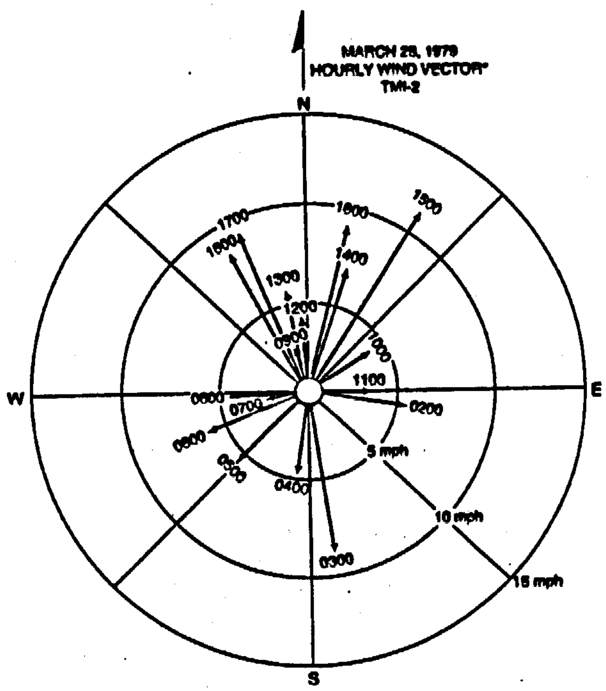
Incidentally, Table 4.5 proposes that in the case of a General Emergency, authorities recommend sheltering for an area within a 2-mile radius plus 5 miles downwind. This "keyhole" pattern of proposed emergency response arises frequently in the emergency-planning literature. Given the potential for rapid changes in wind direction, as exemplified in Figure 4.1, however, the wisdom of this recommendation is in doubt. In particular, if emergency responses are taken prior to the initiation of a core-melt or an atmospheric release, the likelihood of a change in wind direction will be greater still.

Relation of Plant Status to EALs

From Tables 4.1 and 4.5, and from the NUREG-0654 recommendation cited above, it is apparent that a General Emergency is regarded as involving "actual or imminent substantial core degradation or melting." By implication, a Site Area Emergency class will not involve either "actual" or "imminent" degradation of the reactor core.

The first point to note is that the distinction between substantial core degradation and core melting is of no practical value. During an emergency, it is unlikely that sufficient information will be available to allow discrimination between severe core damage and core-melt. Indeed, as chapter 2 points out, the probabilistic risk assessment (PRA) performed for the Seabrook plant rejects the distinction (PLG 1983, 1:2.1-2).

Then one must ask whether the EALs falling within the General Emergency class are adequate to capture the potential imminence of a core-melt. Examination of the TMI Unit 1 EALs presented by the licensee (GPU Nuclear Corporation 1986, Table 10A) show that they are not adequate for this purpose. Consider the matter of electrical power supply. According to the licensee, complete loss of either all AC or all



*Arrows indicate direction toward which the on-site wind was blowing at the local time indicated. Circles represent varying wind speeds.

FIGURE 4.1. Hourly wind vector at TMI-2 on March 28, 1979. *Source:* McKenna *et al.* (1987, 2:113).

DC power for more than 15 minutes (and, apparently, loss of all AC and DC power simultaneously) serves only to place the plant into a Site Area Emergency. Yet, loss of AC power alone is likely to precipitate a core-melt within a few hours. Total loss of DC power, with or without the availability of AC power, will render all engineered safety features inoperative, thus leading inevitably to core-melt (MHB 1987, II-13). One might reasonably expect that such situations would be interpreted as being associated with imminent core-melt, thus justifying the declaration of a General Emergency.

This deficiency in EAL classification is not confined to electrical power supply. In fact, for TMI Unit 1, the licensee regards as appropriate for a General Emergency classification only certain plant conditions:

- Actual or projected off-site doses exceed the EPA's lower PAG levels (5 rem thyroid, 1 rem whole-body), either within 1 hour or as an integrated dose.
- The containment atmosphere shows high radiation levels and a high pressure (30 psig, or more) or a high hydrogen concentration (3 percent by volume, or more).
- There has been a loss of physical security control or a severe act of sabotage.
- Plant conditions are in under way or have occurred (a catch-all condition) which may involve actual or imminent substantial core degradation or melting with the potential for loss of containment integrity or for the release of significant amounts of radioactivity in a short time. (GPU Nuclear Corporation 1986, Table 10A)

Clearly, the present use of EALs does not adequately capture the importance of abnormal plant states. An approach to rectify this deficiency is presented below.

Experience With Classification of Emergency Action Levels

Table 4.7 shows the emergency classifications that were notified to the NRC during the period 1983-1986. It will be seen that Alerts are relatively uncommon events, and that no Site Area Emergency or General Emergency was declared during this period.

In 1982, the declaration of Site Area Emergency followed the rupture of a steam generator tube at the Ginna nuclear plant. Table 4.8 shows the time sequence of EAL classifications during the event. It is estimated that the atmospheric release during the event was quite small; it included, for example, 0.1 curies of ^{131}I and 56 curies of ^{133}Xe (NRC 1982, 5-10).

TABLE 4.7. Emergency Events Reported to the NRC, 1983-1986

	EVENT STATISTICS FOR:			
	1983	1984	1985	1986
Unusual Event	205	224	312	209
Alert	7	8	11	9
Site Area Emergency	0	0	0	0
General Emergency	0	0	0	0

Source: McKenna (1987).

TABLE 4.8. Emergency Action Level Classification During the 25 January 1982 Event at the Ginna Nuclear Plant*

0925 hours:	accident initiated (rupture of a steam generator tube)
0933:	shift supervisor declared an Unusual Event
0940:	licensee declared an Alert
1033 (approx):	licensee dispatched on-site and off-site radiological survey teams
1044:	licensee declared a Site Area Emergency and executed a partial site evacuation
1917:	licensee downgraded the Site Area Emergency to an Alert
2245 hours:	licensee downgraded the Alert to the Recovery phase

*Information from NRC (1982, Table 2).

Advantages of Early Warning

It is widely recognized that early warning of a release can allow more effective emergency response. Following a review of the potential time development of reactor accidents, this chapter turns to some calculations that illustrate the potential relationship between warning time and the extent of radiation exposure of members of the public. Finally, the discussion addresses the issues associated with early warning.

Potential Time Development of Reactor Accidents

NUREG-0654 provides guidance, summarized in the box, on the potential time development of an atmospheric release.

Although the state of knowledge about severe-accident behavior has improved since NUREG-0654, these time-frames remain valid. In the box below are illustrations of the time development of some potential accident sequences at TMI Unit 1.

Warning Time and Public Exposure: Example Calculations

Table 4.9 illustrates the sensitivity of radiation dose to the timing of evacuation. The calculations presented are specific to TMI Unit 1 and

Time from the initiating event to start of atmospheric release	0.5 hours to one day
Time period over which radioactive material may be continuously released	0.5 hours to several days
Time at which major portion of release may occur	0.5 hours to 1 day after start of release
Travel time of release to exposure point (time after release)	5 miles: 0.5 hours to 2 hrs; 10 miles: 1 to 4 hours

Source: NRC and FEMA (1980, 17).

were performed using the relatively new MACCS code (MELCOR Accident Consequence Computer System). The assumption of a "worst-case" atmospheric release involves, for example, the release of 80 percent of the core inventory of iodine, 72 percent of cesium, 69 percent of tellurium and 33 percent of ruthenium. In this model, evacuation, once initiated, occurs radially at a constant speed. Also, the plume is assumed to travel radially at a constant speed.

Although artificial, these assumptions allow a quantitative illustration of some qualitative points. First, doses can be very high. Thus, the initiation of an evacuation prior to the plume's reaching an off-site population, can save people from the possibility of acute health effects. Second, it is important to avoid an evacuation pattern in which people travel with the plume. In both cases shown in Table 4.9, this occurs when people located 6 miles from the plant commence evacuation 1.5 hours after the release commences. It turns out that the marrow dose is actually higher for such people than for those people who commence evacuation after a delay of 3.5 hours.

This phenomenon does not actually argue against early initiation of evacuation, since people usually have a choice of evacuation routes such that they need not move with the plume. Of greater interest is a separate trend—the increase of dose as evacuation delay time increases, due to accumulation of exposure to radioactivity deposited on the ground and other surfaces.

Figure 4.2, drawn from a generic study using the CRAC 2 code, provides a different illustration of the effect of warning time, again in the context of the importance of prompt evacuation. The number of early fatalities (for a "typical" off-site population distribution) is calculated as a function of T_W-T_E, where:

TABLE 4.9. Sensitivity of Estimated Radiation Dose to Timing of Evacuation

This sensitivity is illustrated by matrices showing the average acute red marrow dose[a] accrued by individuals whose evacuation occurs as shown. Doses are estimated using the MACCS code[b] and a "worst case" TMI source term.[c]

	Time After Release Before Evacuation Commences (hours)[e, f, g]	Initial Distance From Plant (miles) 1	3	6
Case 1[d]: Atmospheric mixing height 2000 m Pasquill stability class: B	1.5	5.25 Sv	0.44 Sv	0.33 Sv
	3.5	5.93 Sv	0.39 Sv	0.17 Sv
	8.5	7.70 Sv	0.50 Sv	0.22 Sv
Case 2[d]: Atmospheric mixing height: 400 m Pasquill stability class: E	1.5	42.2 Sv	8.29 Sv	4.18 Sv
	3.5	48.0 Sv	8.52 Sv	2.30 Sv
	8.5	68.2 Sv	10.9 Sv	2.85 Sv

[a]Doses averaged over populations located at the stated distance from the plant but distributed laterally across a sector of 22.5-degree angular width; "acute" dose as defined by Runkle and Ostmeyer (1985).

[b]MACCS (MELCOR Accident Consequence Code System) runs were performed by Center for Technology, Environment, and Development (CENTED), Clark University.

[c]Defined in IRSS (1989), in the present case, the plume was assumed to have a thermal power of 10 MW (i.e., no plume rise).

[d]All cases assume a wind speed of 4 m/s and no rain.

[e]Evacuation is assumed to occur radially, at a speed of 10 mph (4.5 m/s).

[f]A cloud shielding factor of 1.0 and a ground shielding factor of 0.7 is assumed throughout.

[g]The release is assumed to commence 2.5 hours after initiation of the accident, and to have a duration of 1 hour.

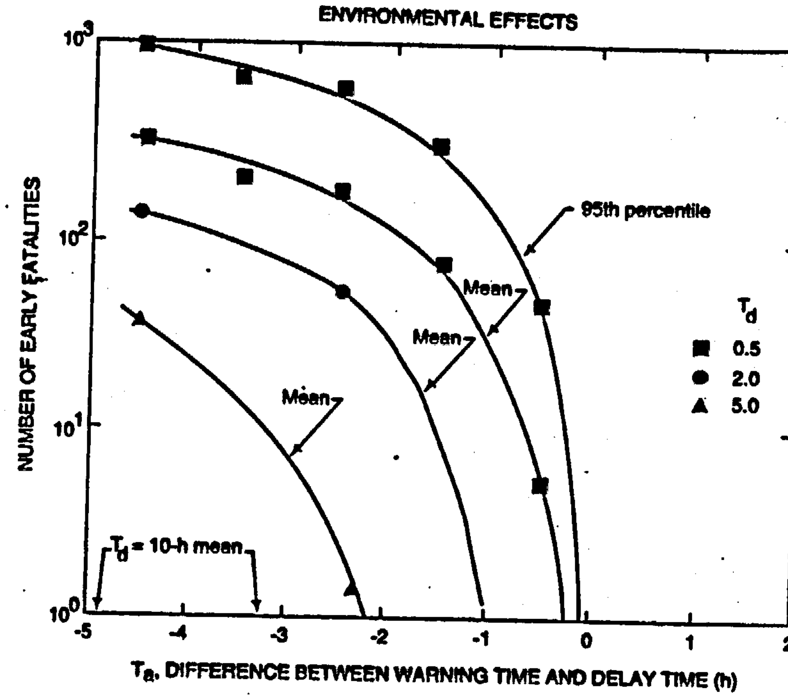


FIGURE 4.2. Sensitivity of number of early fatalities to the difference between warning time and delay time (evacuation speed, 10 miles h/[16 km/h]). *Source:* Kaiser (1986, 373).

- T_w (the warning time) is the interval between the declaration of a General Emergency and the time when the release begins.
- T_e (the delay time) is the interval between notification that evacuation is necessary and the time at which evacuation begins.

It is apparent that the estimated number of fatalities falls rapidly as T_w-T_e becomes less negative—that is, as evacuation is initiated in a more timely fashion. Incidentally, T_d is the duration of the release in hours and the designations "mean" and "95th percentile" refer to the statistical distribution of meteorological conditions at the time of the release.

Early Warning: Issues and Implications

The major issue that arises in the context of early warning is the possibility of false alarms. Under NUREG-0654 guidance, the likelihood of an unnecessary initiation of emergency response is small; EALs are not regarded as entering the General Emergency class unless a core-melt has occurred or is about to occur *and* containment integrity is seriously threatened. By contrast, if emergency response is to proceed prior to core-melt, the possibility exists that core-melt will not occur. Alternatively, core-melt might occur unaccompanied by a significant atmospheric release.

It is not a simple task to formulate an optimal false-alarm rate. One must balance the reduced exposure achieved through early warning against the costs and risks arising from unnecessary emergency response. Public confidence in emergency planning is very important; however, it is not obvious whether confidence would be enhanced or diminished by a high false-alarm rate (but see the discussion in chapter 12).

Even if an optimal false-alarm rate can be determined, it will be difficult to operate an emergency-warning system that achieves that rate. Severe reactor accidents are relatively rare events that can occur through any one of a wide variety of accident sequences. Thus, the statistical data base for predicting accidents is weak. It will be difficult to predict the probability of any particular plant state's leading to core-melt—especially if the prediction has to be performed in real time.

If it becomes standard policy to initiate emergency responses in advance of the initiation of core-melt, despite the possibility of false alarms, then it may be appropriate to employ graded responses. For example, sensitive subpopulations (e.g., pregnant women) might be advised to execute emergency responses at abnormal plant states with relatively low probabilities of proceeding to core-melt. Clearly, as Mileti and Sorensen argue in chapter 12 special means of communicating with such subpopulations would be required.

Accident Precursors: General Experience

Implicit in the preceding discussion is the concept of an accident "precursor"—that is, an abnormal event or plant state that has the potential to lead to a core-melt. A general discussion of precursors gives way to an analysis of potential TMI precursors.

Precursor Analysis: 1969-1985

Oak Ridge National Laboratory (ORNL) has an ongoing effort to identify and analyze precursor events at US nuclear plants. The principal source of information for this investigation is the body of licensee event reports (LERs) that licensees are obliged to file with the NRC.

From these LERs, Oak Ridge experts extract and analyze the potentially most serious events—the precursors. They estimate the probability that the precursor would have led to core damage or core-melt. Analyses of this kind have been published for LERs occurring in the periods: 1969-1979 (Minarick and Kukielka, 1982); 1980-1981 (Cottrell *et al.* 1984); 1984 (Minarick *et al.* 1987); 1985 (Minarick *et al.* 1986); and subsequently.

One of the figures (2.2) in chapter 2 of this volume shows that ORNL's analysis of the 1969-1979 precursors yields a much higher estimate of severe core-damage probability than the estimates made in studies such as the *Reactor Safety Study*, WASH-1400 (NRC 1975). It is noteworthy that the Oak Ridge analysts (Minarick and Kukielka, 1982, 1:xiii) use the term "severe core damage," but as pointed out earlier in this chapter, that term is for practical purposes equivalent to the term "core-melt."

Three particularly severe events occurred during the 1969-1979 period: the 1979 accident at TMI Unit 2; the 1975 cable fire at Browns Ferry Unit 1; and the 1978 incident at Rancho Seco. Oak Ridge analysts have estimated the probability of severe core damage during each of these events:

TMI Unit 2	1.0
Browns Ferry Unit 1	0.4
Rancho Seco	0.25

Figure 4.3 illustrates the process by which LERs—in this case for 1985—are selected as precursors. Of the 2955 reports for that year, 1402 were selected for detailed review, ultimately yielding 63 precursor events (see box at bottom left of Figure 4.3). The distribution of these precursors, according to the estimated probability of their leading to severe core damage (core-melt), is shown in Figure 4.4.

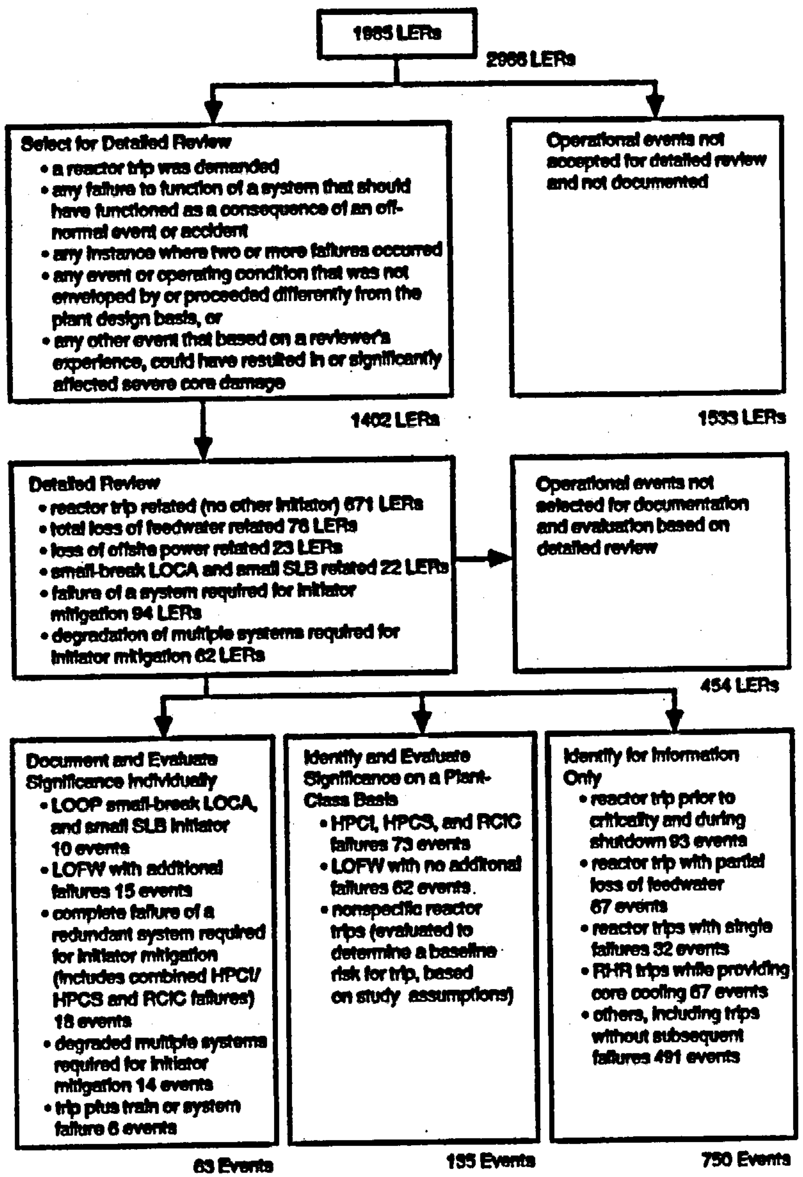


FIGURE 4.3. Precursor selection process: 1985 data. *Source:* Minarick *et al.* (1986).

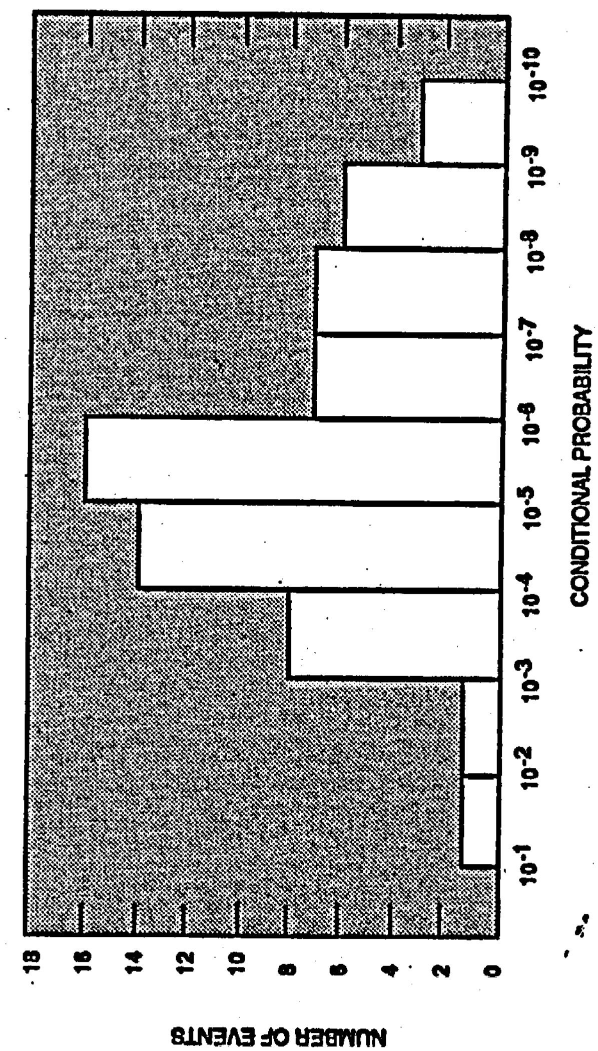


FIGURE 4.4. Distribution of precursors as a function of conditional probability of core damage. *Source:* Minarick *et al.* (1986).

Relation of Precursors to EAL Classification: Selected Cases

The probability of core-melt, as estimated by Oak Ridge for each precursor event, is a useful measure for ranking the severity of the precursors. Such a ranking should not be taken as definitive; individual core-damage sequences will have a variety of outcomes and will operate on a variety of time-scales, and the core-damage probability estimates have wide ranges of uncertainty. Nevertheless, the estimates of conditional core-damage probabilities provide a convenient and informative measure for ranking the severity of precursors.

The licensee's designation of an EAL class for each significant event provides another measure of severity. It is, therefore, of interest to examine the extent to which these two quite distinct measures are in agreement. Since the EAL classes are designated at the time of the events, under sometimes stressful conditions, it can be assumed that they provide a less accurate measure than the Oak Ridge estimates of core-damage probability, which are performed with all the benefits of hindsight.

A limited comparison of EAL classification and estimated core-damage probability has been made here. Eleven precursors that occurred during the years 1981, 1984, or 1985 were studied for this comparison, the basis for selection of these precursors being twofold. First, since the NUREG-0654 guidance on EAL classification was published in 1980, only events occurring in later years are relevant; of these years, Oak Ridge precursor analyses had, at the time of our study, been published for the years 1981, 1984, and 1985. Second, only precursors with an estimated core-damage probability of 5.0×10^{-4} or higher were selected. This arbitrary threshold yielded a manageably small set of the most severe precursors, sufficient to perform an initial comparison.

Table 4.10, which summarizes the comparison, reveals no apparent correlation between the estimated core-damage probability and the EAL class. A glance at Table 4.7 will reinforce one's impression that EAL classification does not adequately capture accident severity. In the years 1984 and 1985, eight and eleven Alerts were called, respectively. Yet, the events from those years that are studied here, despite their receiving a high severity ranking according to retrospective Oak Ridge analysis, did not receive an EAL classification higher than Unusual Event.

Although our comparison assumed that the EAL classification was as reported in the licensee event report, it is possible that this information source is unreliable. Accordingly, the NRC's Incident Response Branch was asked to provide information from their records, but that information was not forthcoming.

TABLE 4.10. Selected Precursors: Summary of Characteristics

Reactor	Event Date	Estimated Damage Probability	EAL Class
Davis-Besse #1	6/9/85	1.1×10^{-2}	None
Brunswick #1	4/19/81	6.7×10^{-3}	None*
Millstone #2	1/2/81	5.1×10^{-3}	None
LaSalle #1	9/21/84	2.3×10^{-3}	None
Hatch #1	5/15/85	1.8×10^{-3}	Unusual Event
Davis-Besse #1	6/24/81	1.7×10^{-3}	None
LaCrosse	7/16/84	9.9×10^{-4}	Unusual Event
San Onofre #1	11/21/85	9.4×10^{-4}	Unusual Event
Turkey Point #3	7/22/85	9.0×10^{-4}	None
Sequoyah #1	2/11/81	8.7×10^{-4}	None
Quad-Cities #1	8/3/84	6.7×10^{-4}	None

*Emergency Action Level classification not expected

Conclusions

Through analysis of accident precursors, it is possible to gain insights into the likelihood of core-melt and into the potential roles of various accident sequences as contributors to core-melt. The Oak Ridge work on precursors has not been specifically directed at improving the classification of abnormal plant states for emergency-response purposes, but it does provide a basis for assessing the adequacy of the present EAL classification. Based on a comparison for a limited set of events, it appears that the present EAL system does not adequately capture the potential severity of abnormal plant states.

Potential TMI Precursors

The concept of a hypothetical precursor may be important in developing an improved method for classifying abnormal plant states.

TABLE 4.11. Contribution of Initiating Events to Estimated TMI Core-Melt Frequency

Classes of Initiating Event	Core-Melt Frequency (per reactor-yr)	Percentage of Total Core-Melt Frequency*
large LOCA	1.7×10^{-5}	5.3
small LOCA	3.9×10^{-5}	12.0
reactor coolant pump seal	7.0×10^{-5}	22.0
ATWS events	2.8×10^{-6}	0.9
steam generator tube rupture	2.2×10^{-5}	6.9
main steam line break	1.2×10^{-5}	3.8
loss of off-site power	2.4×10^{-5}	7.5
loss of integrated-control-system power	3.0×10^{-5}	9.4
loss of main feed-water	9.0×10^{-5}	28.0
stuck-open PORV	1.3×10^{-5}	4.0
interfacing systems LOCA (event V)	1.4×10^{-7}	--
reactor vessel rupture	1.1×10^{-6}	0.3
seismic event	3.0×10^{-6}	0.9
	3.2×10^{-4}	100.0

*Column does not add due to rounding

Source: MHB (1987).

Through careful study of a particular nuclear plant, informed by analysis of actual precursors at other plants, it may be possible to identify, describe, and catalog a large number of hypothetical precursors for the particular plant. It may further be possible, through a combination of computer analysis and the informed judgment of an on-duty expert, to classify in real time any abnormal plant state according to the existing catalog of precursors. If this can be done, then any abnormal plant state can be described, in real time, by the probability of its proceeding to core-melt and the timescale over which this could occur. In turn, that information could be used to guide emergency response.

Realization of these possibilities will require substantial effort, representing a new phase in the application of PRA methodology. A limited illustrative analysis of the nature of that effort draws heavily on a small-scale preliminary study hereafter designated the "MHB Report" (MHB 1987). Chapter 2 discusses subsequent work by the authors of that study.

Core-Melt Sequences Identified in the MHB Report: Selected Cases

Table 4.11 summarizes the preliminary findings of the MHB Report. It shows the estimated contributions that various accident sequences (classified by their initiating event) make to the total core-melt frequency. The effects of external events not analyzed here can be expected to add to the estimated core-melt frequency. For example, there is reason to believe that flooding will be a significant initiator of core-melt for TMI Unit 1. Likewise, sabotage could well be an important initiator.

For the purpose of illustration, seven accident sequences are discussed here. These are all the sequences to which the MHB Report attributes a core-melt frequency of 1×10^{-5} per reactor-year or higher. Collectively, they account for 72 percent of the total core-melt frequency estimated in that report. Table 4.12 provides a brief description of each of the seven sequences. Also, Table 4.13 explains the symbols used to designate each sequence. Brief discussions of the selected sequences follow.

S_2H and S_2HF Sequences. These sequences begin with a small (S_2) loss-of-coolant accident (LOCA), after which high-pressure injection initially operates successfully. But the eventual exhaustion of the flow drawn from the borated water storage tank requires the operators to switch to recirculation of water from the containment building sump. Several operator errors can occur at this point. First, the operators may simply fail to initiate recirculation. Second, they may fail to align high-pressure injection pump suction to low-pressure injection pump discharge. Either error causes a failure of high-pressure injection recirculation (event H). For these two errors, the MHB Report estimates a total probability of 5.3×10^{-3} per demand (with a range from 2.7×10^{-2} to 1.1×10^{-3} per demand). Third, operators may fail to open the sump valves and thereby causes failure of both high-pressure injection recirculation and of containment spray recirculation (event HF). The MHB Report estimates the probability of this error as 3.0×10^{-3} per demand (with a range from 1.5×10^{-2} to 6.0×10^{-4} per demand).

Table 4.12. Selected Core-Melt Sequences

Sequence Designator	Estimated Frequency (per reactor year)	Description
S_2H	$2.4x10^{-5}$	small LOCA, HPI operates, fan coolers
S_2HF	$1.4x10^{-5}$	keep containment pressure below spray setpoint before vessel breach, high pressure recirculation fails either for injection (H) or for injection plus sprays (HF)
S_3H	$4.4x10^{-5}$	as S2H and S2HF, except reactor
S_3HF	$2.5x10^{-5}$	coolant pump seal LOCA (S3) instead of a small LOCA (S2)
SGTR-H	$1.3x10^{-5}$	steam generator tube rupture, operators fail to depressurize to RHR conditions, sprays do not operate, no water available in sump for circulation
TMLD	$1.0x10^{-4}$	loss of main and emergency feedwater, operators fail to execute feed-and-bleed cooling, fan coolers keep containment pressure below spray setpoint before vessel breach
TMB-S_3	$1.4x10^{-5}$	loss of off-site power, station blackout for two hours emergency feedwater operates, reactor coolant pump seal LOCA due to blackout, no fan coolers or sprays due to blackout

In either case, containment fan coolers would keep containment pressure below the setpoint of 45 psia (pounds per square inch absolute) for containment spray initiation during the period before vessel breach. For sequence S_2H, sprays would be available after this time, whereas for sequence S_2HF they would not.

An indication of the timing of high-pressure injection failure for these sequences can be gained from an analysis performed for Oconee Unit 1, assuming a LOCA of 1.2 inches equivalent diameter (MHB 1987, Table B-4). The Oconee analysis shows an interruption of high-pressure injection after 450 minutes (7.5 hours), due to emptying of the

TABLE 4.13. PWR Accident Sequence Event Symbols

Symbol	Meaning of Symbol
A	Large loss-of-coolant-accident.
B'	Loss of all onsite and offsite AC power and failure to recover one source of AC power within 1 to 8 hours.
C	Failure of containment spray injection.
D	Failure of emergency coolant injection (with large RCS leak rate, low pressure injection; with all other initiators, high pressure injection).
(DC)	Failure of DC power following a loss of off-site power.
F	Failure of containment spray recirculation.
G	Failure of containment fan coolers (containment heat removal).
H	Failure of emergency coolant recirculation (with large RCS leak rate, low pressure injection; with all others, high pressure injection).
K	Failure of reactor protection system (scram).
L	Failure of emergency feedwater system.
M	Failure of main feedwater system.
MSLB	Main steam line break.
MSSV	One or more stuck open main steam safety valves (MSSVs) during steam generator tube rupture sequences.
(MTC)	Adverse moderator temperature coefficient (in ATWS sequences; results in a large pressure in the reactor coolant system).
PORV	Stuck-Open PORV transient (including frequency of initiating transients, opening of the PORV, failure of the PORV to reclose and failure of operators to close the PORV block valve to stop RCS leakage).
Q	Stuck-Open PORV or code safety valve (including failure of the valve to close and, as appropriate, failure of the operators to close the PORV block valve to stop RCS leakage).

(continues)

Table 4.13 *(continued)*.

Symbol	Meaning of Symbol
RVR	Reactor vessel rupture.
S_2	Small loss-of-coolant accident.
SGTR	Steam generator tube rupture (a special class of small loss-of-coolant-accident in which coolant is lost from the reactor coolant system to the secondary side of the steam generators-recirculation cooling is not possible).
T	Transient event requiring operation of the reactor protection system (scram-event "K", above).
V	Failure of valves which isolate the reactor coolant system from low pressure emergency cooling systems (an interfacing LOCA, a special class of large loss-of-coolant-accident in which coolant is lost to the residual heat removal system vaults in the auxiliary building; recirculation cooling is not possible).
X	Failure to isolate affected steam generator is main steam line break sequences.

Source: MHB (1987, pages II-4, II-5).

borated-water storage tank. Now, the borated water storage tank at TMI Unit 1 has a nominal capacity of 360,000 gallons and the Final Safety Analysis Report for TMI Unit 1 states (GPU Nuclear Corporation 1985, 6.1-11) that one high-pressure injection pump (which will deliver 500 gpm for a reactor coolant system (RCS) pressure of 600 psia), is adequate for small-break loss of coolant accidents which do not cause rapid RCS depressurization. It is also noteworthy that the decay heat boil-off rate falls below 300 gallons per minute (gpm) after 20 minutes (GPU Nuclear Corporation 1985, Figure 14.2-36).

Thus a period of 450 minutes to exhaust the borated water storage tank (representing an average flow-rate of 800 gpm) seems a reasonable hypothesis for TMI Unit 1. In this respect, manual initiation of recirculation is an advantage, as an automatic system might be set to begin recirculation when the borated water storage tank was only partly empty (in illustration, see Gieseke *et al.* 1983-1986, Table 6.1; and the recirculation failure sequence timings in Denning *et al.* 1986).

Once recirculation failure has occurred and high-pressure injection flow is no longer available, an indication of the subsequent timing of

the sequences can be obtained from the following estimates (MHB 1987, Tables B-12 and B-13) for a sequence involving a small LOCA with immediate high-pressure injection failure:

	LOSS-OF-COOLANT ACCIDENT (LOCA) 0.75 inch	2 inch
LOCA occurs	0 minutes	0 minutes
Core-melt begins	113 minutes	51 minutes
Vessel melt-through	164 minutes	76 minutes

For a recirculation-failure sequence, the progression to core-melt would be slower because decay heat would be lower at the time at which recirculation was attempted. Thus, core-melt for our sample case, 1.2-inch LOCA; could be assumed to begin about 2 hours after recirculation failure, and vessel melt-through could occur about 1 hour after that. Reactor coolant system pressure would remain high enough throughout this sequence that high-pressure melt ejection could occur. Thus, containment failure could occur immediately after vessel melt-through.

S_3H and S_3HF Sequences. These sequences also involve a small LOCA, but of a particular kind—leakage from the seals of one or more of the reactor-coolant pumps (RCPs). In other respects, the previous discussion about S_2H and S_2HF sequences is applicable. For the assumption of reactor-coolant pump seal leakage equivalent to a 0.87-inch LOCA, the MHB Report (MHB 1987, Table B-5) estimated that recirculation failure would occur at 500 minutes (8.3 hours).

SGTR-H Sequence. This sequence involves a steam generator tube rupture (SGTR) event in which the operators fail to depressurize the reactor-coolant system to the point where the residual heat removal system can be used. Such an event would be in several respects similar to an S_2, but with one important difference—primary coolant would be lost into the secondary side and might well dissipate the entire inventory of borated-water storage tank. Then no water would be available in the containment sump for recirculation. Since no coolant would be lost into the containment, neither sprays nor fan coolers would be called upon to operate. After exhaustion of the borated-water storage tank, sprays would not be able to operate.

The MHB Report estimates that the frequency of failure to depressurize to residual heat-removal conditions (that is, the frequency of event H) would be 1.0×10^{-3} per demand (with a range from 3×10^{-3} to

3.3×10^{-4} per demand). It is also estimated that 50 percent of steam generator tube rupture-H sequences would involve a stuck-open main steam safety valve (thus becoming SGTR-H-MSSV sequences). Such cases would be important because a stuck-open safety valve would create a direct path from the steam generator secondary side to the atmosphere.

Sequence timing and reactor-coolant-system pressure for SGTR-H and SGTR-H-MSSV events would be similar to those for small LOCA sequences. Indeed, if the steam generator tube rupture breach is of such a size that the reactor-coolant-system pressure remains above RHR operating conditions, then the SGTR breach area must fall into the small LOCA range.

TMLD Sequence. This sequence begins with a failure of main (event M) and emergency (event L) feed-water. The operators then fail to execute feed-and-bleed cooling via the high-pressure injection system (event D). As identified in the MHB Report, this sequence could begin with either of two initiators—a loss of feedwater transient, or a loss of the instrumentation and control system (ICS) power bus. The MHB Report estimates the probability of a failure to execute feed-and-bleed cooling as 5.0×10^{-2} per demand (with a range from 0.15 to 1.7×10^{-2} per demand).

Potential time development of this sequence would be (MHB 1987, Tables B-1 and B-15):

core uncovers	ca. 60 minutes
core-melt begins	94 minutes
vessel melt-through	184 minutes

Containment fan coolers would keep containment pressure below the spray setpoint before vessel failure. After vessel failure, sprays would be available. Reactor coolant system pressure would remain at the pilot(or power)-operated relief valve (PORV) setpoint unless the operators succeeded in depressurizing the reactor coolant system. Thus, high-pressure melt ejection is a possibility.

TMB'-S_3 Sequence. In this case, a loss of off-site power (event T) results in a loss of main feed-water (event M). Diesel generators fail to start, resulting in a station blackout that lasts at least 2 hours (event B').

Station blackout would interrupt cooling and injection flow to reactor coolant pump seals, thus leading to a reactor coolant pump seal LOCA (event S_3). The MHB Report estimates the probability of seal failure

under extended station blackout conditions as 0.65 per event (with a range from 0.80 to 0.15 per event). It should be noted that this report also estimates the probability of an initial station blackout's extending for 2 hours as 0.16.

Emergency feedwater can be expected to operate through the 2-hour blackout period, using the steam-driven pump. Neither high-pressure injection nor the containment cooling systems (fan coolers and sprays) will be available, as they are electrically driven. Reactor coolant system pressure will remain high, as in other small LOCA sequences. Thus high-pressure melt ejection is a possibility at the time of vessel melt-through.

Estimates of the timing of two related sequences indicate the timing of this sequence (see Box). Now, both sequences differ from TMB'-S_3 because there is no reactor coolant pump seal LOCA and they both involve failure of emergency feed-water (event L). Also, sequence TMLQD involves a stuck-open PORV (event Q) and failure of high-pressure injection (event D). For a sense of the relationship between sequence timings (particularly between timings for sequences TMB'-S_3 and TMLB'), consider an analysis presented in the Seabrook PRA.

Back in chapter 2, Figure 2.6, drawn from the Seabrook PRA (PLG 1983, 3:11.5-34), shows, assuming a station blackout, the estimated time to core uncovery as a function of reactor-coolant pump seal leakage. Two cases are shown—with and without the turbine driven feedwater pump (TDFWP). Thus, the "without-TDFWP" case with zero seal leakage is equivalent to our TMLB' sequence; the "with-TDFWP" case is equivalent to our TMB'-S_3 sequence. It will be seen that the sequence timing is quite sensitive to the rate of reactor coolant pump seal leakage. This rate cannot be anticipated and cannot be directly measured during an accident.

Thus, subject to considerable uncertainty, the above-listed timing for TMLB' might be appropriate for TMB'-S_3 if the reactor-coolant pump seal LOCA were relatively large. Of course, if electric power were recovered before core-melt had begun, then high-pressure injection could become available and the accident could be arrested.

	TMLB'	TMLQD
Core uncovery begins	120 minutes	37 minutes
Core-melt begins	140 minutes	59 minutes
Vessel melt-through	201 minutes	79 minutes

Source: MHB (1987, Tables B-10 and B-11).

Precursors to the Selected Sequences

Each of the sequences can be described, for purposes of illustration, as having one precursor event and one other event that precipitates core-melt. This description is summarized in Table 4.14, which also shows the frequency of each precursor event, as estimated in the MHB Report. At the beginning of this discussion, we introduced the concept of analyzing hypothetical precursors as a basis for real-time classification of abnormal plant states. Table 4.14 provides a very limited illustration of such precursor analysis.

In determining the appropriate emergency response to a given precursor, one should take account of both the probability of the precursor and the timing of the sequence with which it is associated. In this context, it will be noted that sequences TMLD and TMB'-S_3 are relatively fast-developing, and their precursors are predicted to be relatively unlikely. For example, Table 4.14 indicates a frequency of station blackout over the 30-year life of a reactor as 3.9×10^{-3}. Presumably, there would not be much argument against triggering emergency responses at such a frequency.

On the other hand, sequences induced by a small LOCA (S_2H, S_2HF, S_3H, S_3HF) are slower-developing, with somewhat more likely precursors. This might lead to a little more reluctance to initiate emergency responses. Also, it may be possible for operators to switch one train of high-pressure injection to recirculation prior to exhaustion of the borated-water storage tank, thus establishing whether recirculation could continue when the borated-water storage tank was empty. Of course, such a procedure would not be risk-free and could in fact advance the timing of core-melt.

The SGTR-H sequence is a little more complicated, because the LOCA (and therefore the sequence timing) could vary over a wide range. Moreover, because it may be hard to judge the likelihood that the operators will succeed in depressurizing the reactor coolant system to the residual-heat-removal operating pressure. Also, the precursor frequency is relatively high; if every SGTR were to trigger an emergency response, the frequency shown in Table 4.14 would lead to a probability of 0.39 that such triggering would occur over the 30-year life of a reactor.

In closing this discussion, it should again be emphasized that Table 4.14 draws from a preliminary analysis presented in the MHB Report. In addition, the probability, timing, and other characteristics of accident sequences can vary widely. Even a comprehensive PRA effort will have difficulty in identifying and adequately characterizing all relevant sequences and precursors.

Table 4.14. Precursors to Selected Core-Melt Sequences

Core-Melt Sequence		Precursor Event	
Designator	Estimated Frequency (per reactor-yr.)	Designator	Estimated Frequency (per reactor-yr.)
S_2H	2.4×10^{-5}	S_2	4.6×10^{-3}
S_2HF	1.4×10^{-5}	S_2	4.6×10^{-3}
S_3H	4.4×10^{-5}	S_3	8.3×10^{-3}
S_3HF	2.5×10^{-5}	S_3	8.3×10^{-3}
SGTR-H	1.3×10^{-5}	SGTR	1.3×10^{-2}
TMLD	1.0×10^{-4}	TML	2.1×10^{-3}
TMB'-S_3	1.4×10^{-5}	TMB'(*)	1.3×10^{-4}
	2.3×10^{-4}		4.1×10^{-2}

(*This precursor is initial station blackout, not allowing for power recovery.)

Requirements for Future Analysis

Systematic study of precursors in the context of real-time emergency decision-making would represent a new phase in the application of PRA methodology. At this point, one cannot predict the degree of success that such an endeavor might enjoy. Existing PRAs have drawn criticism for a variety of reasons, and their probabilistic findings have intrinsically limited accuracy due in part to the lack of a statistical data base and in part to the nonstochastic nature of certain factors (e.g., sabotage). Nevertheless, application of PRA methodology may lead to better real-time classification of abnormal plant states.

The PRA for the TMI plant could well form the basis for a further study. That study would attempt to identify and characterize all significant accident precursors. An on-duty expert could, in real time, draw upon this classification to facilitate the appropriate assignation of severity to an abnormal plant state.

Monitoring of Plant Status

For adequate real-time classification of abnormal plant states, information on all relevant plant parameters must be continuously available to decision makers. At present, information of this kind about the TMI plant is available only to the licensee. Public authorities, such as Pennsylvania's Bureau of Radiation Protection, must rely on the licensee to pass on information about any incident.

A different situation obtains in Illinois. There, the Illinois Department of Nuclear Safety (IDNS) continuously receives data about plant parameters from all nuclear plants in the state. Such an arrangement permits state authorities to reach independent decisions about the current or potential implications of an abnormal plant state.

Whatever the information stream, analysts need the ability to extract the important message from within a large amount of "noise." This is not a trivial task, and the IDNS is still developing its capabilities in this respect (Gallina 1993).

NRC Requirements

Licensees are currently required to operate a safety parameter display system (SPDS) at each nuclear plant:

> The safety parameter display system (SPDS) provides a display of plant parameters from which the safety status of operation may be assessed in the control room, TSC, and EOF [Emergency Operations Facility]. The primary function of the SPDS is to help operating personnel in the control room make quick assessments of plant safety status. Duplication of the SPDS displays in the TSC and EOF will improve the exchange of information between these facilities and the control room and assist corporate and plant management in the decision-making process. The SPDS shall be operated during normal operations and during all classes of emergencies. The SPDS should have the flexibility to allow future modifications to be incorporated, such as the capability to handle operator interaction and diagnostic analysis. (NRC 1981, 4)

It is worth noting two important points. SPDS is intended to help *licensee personnel* assess the safety status of their plant; and the NRC, consistent with its view of the SPDS as a decision aid for operating personnel, does not require the safety parameter display system to meet regulatory requirements for safety-related instrumentation.

Licensees are required to use instrumentation to assess conditions in their plant and its environment during and following an accident. Guidance as to the nature of this instrumentation is provided by NRC Regulatory Guide 1.97 (NRC 1980). Many of the parameters measured by these instruments will also be covered by the SPDS.

The NRC has contemplated requiring a nuclear data link, whereby on-line information on plant parameters would be transmitted to the NRC, and Illinois has implemented its own version of such a data link. In 1987, the state became an NRC Agreement State, a designation that transferred to the state regulatory authority over uses of radioactive materials (Gallina 1993, 36).

The Illinois System

The Illinois Department of Nuclear Safety monitors thirteen commercial reactors at seven sites in Illinois. By late 1986, a continuous data link (the DDL system) had been established between eleven of those reactors and the IDNS headquarters in Springfield and similar links were planned for with the two Braidwood reactors as they approached fuel loading (Blackburn and Parker 1987; IDNS 1987).

The state's current Reactor Data Link (RDL) connects each reactor, via dedicated telephone lines, to IDNS headquarters in Springfield (Gallina 1993). The RDI, which monitors up to 1500 parameters for each of the reactors, is the first element of the state's Remote Monitoring System (RMS), discussed in chapter 3. The availability of the data stream from each plant has typically exceeded 95 percent. The data streams from the various plants are monitored manually by the Illinois Department of Nuclear Safety Personnel. However, software has been developed and implemented whereby the data may be retrospectively reviewed.

Thus, the Illinois Department of Nuclear Safety has embarked on the task outlined earlier in this chapter and in chapter 3—real-time classification of abnormal plant states in a manner intended to facilitate decision making for timely emergency response.

TMI Capabilities

Pennsylvania has no capability to receive on-line plant parameters. During an emergency, a nuclear engineer on the staff of the Bureau of Radiation Protection (BRP) will go to the site and observe plant operations; that engineer may orally transmit information to BRP headquarters via telephone.

In compliance with NRC requirements, TMI Unit 1 is now equipped with an SPDS. This SPDS relies on the plant computer system, which has two redundant processors with an automatic switchover capability. The new system has a design goal of 99.5 percent reliability.

Although the SPDS at TMI Unit 1 provides a basis for the establishment of a continuous data link with Pennsylvania authorities, two issues require further exploration. First, it is not clear that the present set of parameters covered by the SPDS is sufficient. Second, insofar as the safety parameter display system is not required to meet regulatory requirements for safety-related instrumentation, its reliability under accident conditions is questionable. These issues should be explored thoroughly before any continuous data link is planned, but this does not imply that the requisite improvements are infeasible.

Requirements for Future Analysis

The Illinois Department of Nuclear Safety data link clearly represents the "state of the art." Thus, the first question to ask is: what improvements are necessary in the Illinois Department of Nuclear Safety system? As indicated, Illinois has embarked on a two-phase software development program. If successful, that program will yield the ability to classify, in real time, abnormal plant states according to their implications for emergency response.

The chance of success in this enterprise will improve if other organizations contribute their expertise and resources. For example, application of PRA methodology—through analysis of hypothetical precursor events—may facilitate real-time classification of plant states. Experience has shown that advances in PRA methodology come about slowly and require very large expenditures. The requisite scale of effort is well beyond that which the Illinois Department of Nuclear Safety can mount alone.

Aside from the software problems, more prosaic issues must be resolved. Notably, what should be the response of a state agency if the data stream stops flowing? Illinois, for all its sophisticated defense-in-depth safeguards, has no clear policy on this question (IDNS 1987).

An associated issue is the reliability of plant computers. There is no requirement that TMI Unit 1 should be shut down if the computer is inoperative; this appears to be the case for most US nuclear plants. By contrast, Canadian CANDU plants rely upon dual computers for plant control. If both computers fail, the reactor is automatically shut down (Ontario Hydro 1987, 2-5). Thus, a more extensive program of software development and a careful analysis of potential modifications in the mechanics of the system are prerequisites to realization of the full potential of the Illinois Department of Nuclear Safety data link system.

Evolution of the Illinois system to its full potential can be expected to take some years. In the interim, there is no reason why similar systems should not be created in other states. For TMI Unit 1, a modified SPDS would provide—at least when supported by the new plant computer system—a suitable data source. The nature of the needed modifications would emerge from analysis of the two previously mentioned issues: first, the adequacy of the present safety SPDS set; and, second, the reliability of the plant computer system under accident conditions.

Recommendations

The emergency-planning community generally recognizes the merit of initiating emergency response early in an accident. Indeed, chapter 11 speaks to a growing support for the concept of precautionary response. Deficiencies exist, however, in the process by which a potential accident is identified and assessed, and by which decision makers are informed of that assessment. The current system of EAL classification requires generic modification, along the lines set forth in the model plan (Golding *et al.* 1992) that is a companion to this volume.

The focus of the present recommendations is upon near-term actions by the TMI licensee and by Pennsylvania authorities. These actions would complement a generic overhaul of EAL classification. Also presented here are recommended R&D actions, to be performed by the NRC and by other parties, which are intended to improve capabilities for real-time accident classification.

Licensee Actions

It is recommended that the licensee execute the following actions in connection with TMI Unit 1:

1. Commission further analysis to identify and characterize all relevant precursors.
2. Analyze the adequacy of the current safety parameter display system parameter set, in light of the results of the studies recommended in recommendation 1.
3. Analyze the potential for failure of the plant computer system and the SPDS, in general and under accident conditions.
4. Modify the SPDS in light of the findings of the analyses recommended in 2 and 3 above.

5. Facilitate the implementation of a continuous data link with the Pennsylvania Bureau of Radiation Protection, based on the modified SPDS.
6. Revise the present system of EAL classification to reflect more accurately the potential severity of abnormal plant states.

Pennsylvania Capabilities

Here, the recommendations are directed to the Pennsylvania Bureau of Radiation Protection, which is assumed to be the relevant Pennsylvania agency. It is recommended that the Bureau:

1. Implement a data link similar to that of the Illinois Department of Nuclear Safety, to receive data from TMI Unit 1 and other Pennsylvania reactors.
2. Continue ongoing development of software and human capability for analyzing the data streams and selecting information pertinent to emergency response.
3. Participate, with the Illinois Department of Nuclear Safety and other organizations, in jointly sponsored ongoing R&D to improve real-time analytic capabilities.

R&D Requirements

The NRC, state governments, and licensees should collaborate in an ongoing R&D program to improve the capability for real-time analysis of abnormal plant states. That program should be embedded in an ongoing R&D program across the entire field of emergency response.

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5

The Use of Probabilistic Risk Assessment in Emergency-Response Planning for Nuclear Power Plant Accidents

Robert L. Goble and Gordon R. Thompson

The first serious applications of probabilistic risk assessment (PRA) to emergency planning for nuclear power plant accidents occurred over a decade ago. Like much of the effort in emergency planning, those applications and subsequent refinements can be viewed positively, as major improvements over the past. Alternatively, they can be assessed more negatively—they lack clear objectives and are riddled with inconsistencies. This chapter summarizes a set of recommendations for substantial improvements in emergency plans and emergency-response capabilities. These recommendations are either: (1) based on PRA considerations or (2) represent opportunities for the use of PRA methods to improve response capabilities in emergency situations. Our work used existing PRA studies, analyses, and models: we have not undertaken any new analyses. The work is best viewed within the historical context of PRAs of nuclear power plants and the present regulatory requirements for emergency planning for nuclear power accidents (see chapters 1 and 4).

Many of the generic recommendations presented here have been incorporated in the model plan that takes up a companion volume (Golding *et al.* 1992). Other aspects inform discussions covered elsewhere in this volume of background papers.

The work reported on in this chapter includes contributions from the TMI Project Steering Committee, composed of Robert L. Goble, Dominic Golding, Jeanne X. Kasperson, Roger E. Kasperson, John Seley, Gordon R. Thompson, and C. P. Wolf.

Special Characteristics of Nuclear Power Plant Accidents

Much more research and investment in infrastructure have gone into emergency planning for nuclear power plant accidents than for such planning for other kinds of technological accidents. Table 5.1 compares properties of severe nuclear plant accidents with typical properties of severe chemical accidents. Nuclear accidents can produce early injuries and fatalities out to distances of several miles or more; they can also produce excess exposures which carry significant risk of cancer at distances of well over 100 miles. Most chemical accidents generally affect relatively small areas, but the worst accidents can produce early deaths and irreversible injuries at distances approaching those possible for nuclear power plants (Bowonder, Kasperson, and Kasperson 1985). A striking difference remains in the distance scale over which significant long-term contamination may occur. Other differences are important as well: in a nuclear accident, harm can result even when people do not

TABLE 5.1. Distances of Concern for Nuclear and Chemical Accidents

Typical Distances (under moderately adverse meteorology)	TYPE OF ACCIDENT: Major Nuclear Accident	Major Chemical Accident
for early deaths	5 miles	3 miles
for irreversible injuries	25 miles	10 miles
Contamination at levels of concern (for cancer and other latent health effects)	200 miles	?(10-20 miles)

Examples

Chernobyl:	Evacuation to 18 miles; contamination regulated beyond 500 miles
Bhopal:	Deaths and early injuries to 3 miles
Seveso:	Detectable contamination to 4 miles

Sources: Bowonder, Kasperson, and Kasperson (1985); DOE (1987); EPA, FEMA, and DOT (1987); Hohenemser (1988); NRC (1987).

have direct contact with the material released, and perhaps no other major technology, as Slovic documents in chapter 15, generates as much fear in the public.

Does a disparity exist between the efforts put into nuclear and chemical emergency planning? Perhaps the more pertinent questions are: (1) how effective has the nuclear planning effort been, and (2) can the experience with nuclear planning be used effectively in other settings? We believe that the nuclear experience offers emergency-planning lessons for chemical accidents and other hazards, but the differences in scale, in type of accident, and in the history of nuclear power regulation must be kept in mind.

Probabilistic Risk Assessments

Most major probabilistic risk assessments for large industrial accidents have been conducted for nuclear power plants. Assessments are based on assigning probabilities along event trees. Because of the large number of engineering systems in a nuclear power plant and their linkages with each other, these analyses are very complex and demanding. As Figure 5.1 illustrates, the analyses conventionally (and conveniently) fall into three levels. A level-I PRA is an assessment of the probability of occurrence of a sequence of events that will lead to core-melt or other major core damage. A level-II PRA is an assessment of the mechanisms for and probabilities of passing from core damage to radioactive releases of various sorts. Finally, a level-III PRA extends the analysis from radioactivity releases to an assessment of the potential consequences for human health and welfare.

The first large-scale, all-level PRAs for nuclear power plants were performed 17 years ago in the *Reactor Safety Study*, often called WASH-1400 or the Rasmussen Report after its principal author (NRC 1975). In the years since, WASH-1400 has been the subject of numerous criticisms and reevaluations (Risk Assessment Review Group 1978). Also, more than 20 PRAs of other commercial power plants have been made (Sholly and Thompson 1986). Along with the production of these PRAs, the US Nuclear Regulatory Commission and the nuclear industry have supported extensive research directed at topics identified by the work on PRAs. The current situation may be summarized as:

- Nuclear power plants in the US differ sufficiently among themselves that the dominant accident sequences and the PRAs are specific to individual reactors.

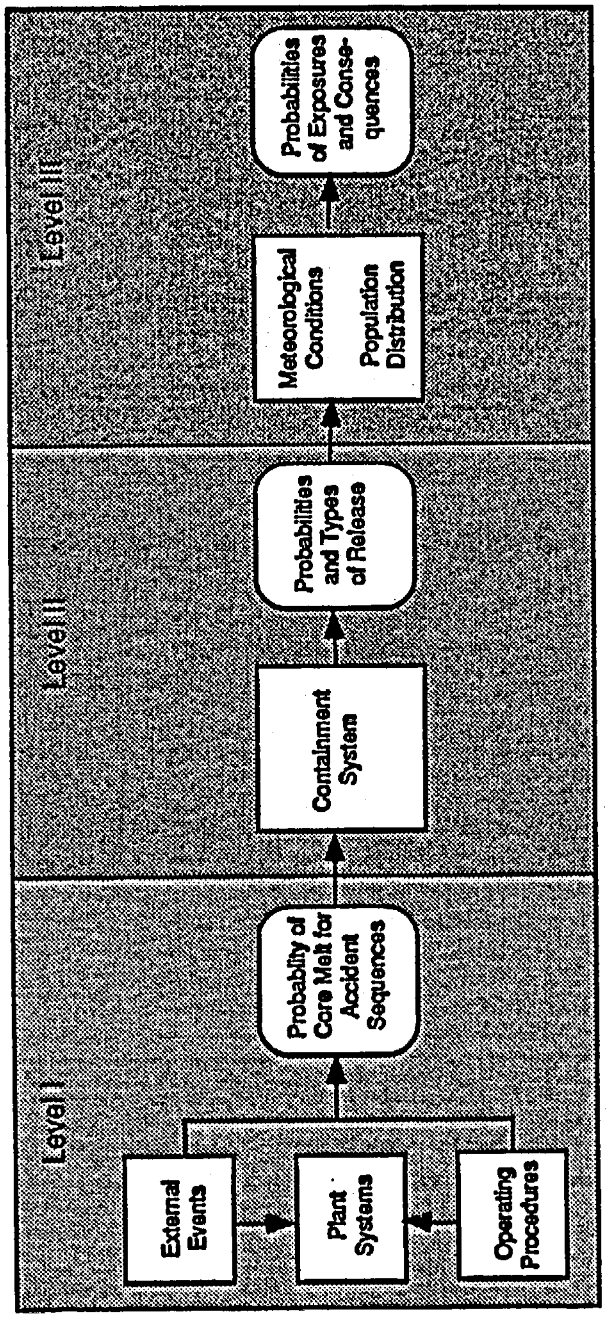

FIGURE 5.1. Levels of probabilistic risk assessment for nuclear power plants.

- Most estimates of the probability of core-melt or major core damage (i.e., level-I estimates) lie within an order of magnitude of each other: the range is roughly 5×10^{-5} to 5×10^{-4} (DOE *et al.* 1987; Sholly and Thompson 1986).
- Estimates of the probability of a major release after core-melt differ much more, ranging from a conditional probability of 0.003 to 0.3 (NRC 1975; PLG 1983).
- Major criticisms against WASH-1400 and subsequent PRAs include:
  a. lack of a systematic and consistent treatment of uncertainties;
  b. deliberate omission of the calculations of important sources of risk, including operation of the plant outside normal conditions (the initiating problem at Chernobyl), sabotage, and terrorism;
  c. inadvertent omission of important sequences; and
  d. insufficient attention to the testing of PRA methods and assumptions (APS 1985; Risk Assessment Review Group 1978; Sholly and Thompson 1986).

Issues concerning uncertainties and omissions are best considered in the context of how the risk assessment is used. The omission of terrorism from PRA calculations is often appropriate, in that PRA methods are not well suited to deal with this sort of hazard. It can be misleading to suggest that these are comparable estimates pump failures and the like. Such comparisons are totally unnecessary if the task at hand effective improvements in plant design or operating procedures. If the issue were to estimate the absolute risks of a reactor accident, then the omission might well be misleading. Estimates of absolute risk, however, have only limited uses.

The three levels of nuclear PRAs differ greatly in the nature of the problem they address, in the methods they use, and in the kind and quality of data that support them. Table 5.2 provides a summary characterization of each level.

Level I and level III are both based on systems (the plant or the meteorological environment) operating for the most part under familiar conditions. The analysis is much more accessible to testing. For both levels, the results depend strongly on local properties, the design of the particular reactor, and the local population distribution and meteorology. A level II analysis is made for systems operating in truly extreme conditions for which hardly any actual experience exists. Because of the complexity of the system, analysis is based on computer codes that cannot be tested easily. Level II analyses are likely to have

TABLE 5.2 Features of PRA Levels

Level I	Typical results:	Probability of core-melt $5x10^{-5}$ - $5x10^{-4}$
	Data base available:	Equipment and system operating experience, other operating experience, all at normal conditions
	Calculational basis:	Event-tree analysis with probabilities
	Major omissions:	Operation outside normal procedures (the initiating event at Chernobyl)
Level II	Typical results:	Conditions probability of major early release 0.003 - 0.3
	Data Base available:	TMI—small-scale experiments
	Calculational basis:	Computer codes simulating complex physical processes at extreme conditions
Level III	Results depend sensitively on:	Meteorological conditions Population distribution Emergency-response measures taken Nature of release
	Data base available:	Extensive meteorological, demographic records
	Calculational basis:	Computer dispersion models

Sources: NRC (1984); DOE *et al.* (1987); Silberberg *et al.* (1986).

substantial uncertainties, not only in predicting probabilities of particular classes of events but in predicting the timing of events. Finally, it is important to note that level III results can be very sensitive to assumptions about emergency responses: this is encouraging in that it indicates that emergency planning may provide significant contributions to safety.

PRAs can have two important types of application to emergency planning—to assist in the definition of a planning basis, and to guide the implementation of particular response measures. The authors begin by discussing the selection of the planning basis; later they describe some potential applications of the second type.

Role of PRA in Establishing the Present Emergency Planning Basis

Figure 5.2 shows the sequence of documents through which analysis applied WASH-1400 results to emergency-planning regulations. The initial activities after publication of WASH-1400 were the work of a joint Nuclear Regulatory Commission/Environmental Protection Agency Task Force on Emergency Planning, whose report was published as NUREG-0396 (Collins, Grimes, and Galpin 1978), and analyses based on WASH-1400 by Aldrich, McGrath, and Rasmussen (1978) used by the Task Force. The Task Force recommendations for a planning basis for emergency plans published late in 1978 became a matter of urgent concern immediately after the accident at TMI, and were embodied in emergency planning regulations issued in NUREG-0654 (NRC and FEMA 1980a; 1980b). Considerable controversy and legal maneuvering have followed over the implementation of these regulations and, perhaps for that reason, few changes have been made in the regulatory planning basis, despite significant new information and research.

The Task Force on Emergency Planning considered several possible rationales for establishing a planning basis, including rationales using risk, probability, cost effectiveness, and consequence spectra as considerations. The Task Force chose to base the rationale for the planning basis on a spectrum of consequences, tempered by probability considerations (NRC and FEMA 1980a; 1980b). It rejected explicit considerations of risk primarily on the assumption that risk-based analyses would require comparisons with nonnuclear emergency planning which it was unwilling to make. Although the accident-consequence spectrum chosen still appears appropriate, the rationale is both limited and questionable in some respects. It does not, for example, provide adequate guidance for the formulation of emergency plans. Two serious deficiencies

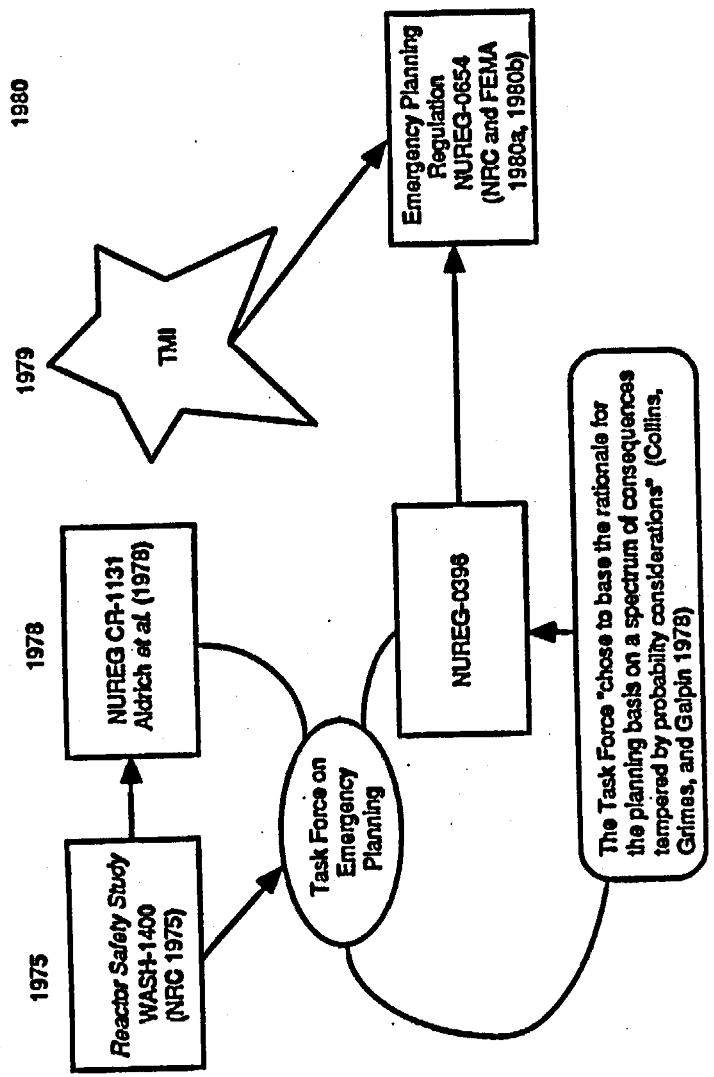

FIGURE 5.2 Critical documents in the incorporation of PRA in emergency-planning regulation.

are apparent. First, emergency-planning objectives are not sufficiently specified for gauging the adequacy planning effort. The discussion of Protective Action Guides (PAGs) is illustrative: "The Task Force concluded that the objective of emergency-response plans should be to provide dose savings for a spectrum of accidents that could produce off-site doses in excess of the PAGs" (Collins, Grimes, and Galpin 1978). It also observes ". . .This does not mean that doses above the PAG levels can be prevented or that emergency-response plans should have as their objective preventing doses above PAG levels" (Collins, Grimes, and Galpin 1978). The second deficiency is closely related to the first: Setting the planning basis gives no consideration to the relative effectiveness of emergency-response measures under various accident scenarios. Risk comparisons among different possible accident situations, with and without various emergency-response capabilities, provide clearer and more useful guidance for formulating and evaluating plans, and do *not* require the making of comparisons with nonnuclear accidents.

Emergency Planning Lessons from a Systematic Consideration of PRA

Based on the current state of PRAs, risk comparisons indicate that the human risks of early death, injury, and radiation exposures that exceed the EPA Protective Action Guides (PAGs) are largely concentrated in accidents in which a core-melt (or severe core degradation) is accompanied by a large early release of radiation (Kaiser 1986). This result appears to hold even for the lower range of estimates of the conditional probability (given core-melt) for a large early release, where the alternative is expected to be a smaller, substantially delayed, and possibly slower accident. When combined with the potential opportunities for saving exposures and health effects by evacuation and sheltering, and keeping in mind the level II uncertainties in predicting the conditional probability, risk comparisons imply that *if emergency planning is demanded for serious (core-melt) accidents, then the most crucial objective is to plan effectively for large early releases.* This conclusion, in turn, has clear implications for emergency planning criteria. The most effective measure is early evacuation. Emergency plans should strive in most circumstances to ensure the evacuation of the region close to the reactor before the radioactive plume passes, even in the case of a rapidly developing accident arrives. When it is not feasible to complete an evacuation before passage of the plume, the plan should specify appropriate combinations of emergency measures. Planning is critical because sheltering may be preferable to evacuation, supplement an incomplete evacuation, or avoid the additional risks attendant on moving people into the radioactive pathway.

Based on these considerations, it is appropriate to recommend precautionary evacuation of the region at greatest risk around a nuclear plant whenever the risk of a core-melt is substantial (10 percent might be a suitable cut-off point as adopted in the Model Plan detailed in the companion to this volume; see Golding *et al.* 1992).

A response strategy that uses precautionary evacuation should include:

- continuous monitoring of a large number of plant system parameters;
- on-line comparison with PRA sequences to determine when the risk is substantial (i.e., exceeds 10 percent);
- appropriately designed emergency-planning zones;
- an accident classification scheme that focusses on the decision to initiate precautionary evacuation; and
- Coherent planning for emergency responses not covered by precautionary evacuation.

Part of using a precautionary strategy is knowing when it might be too late. It may be impossible to issue an adequate warning before core-melt is already under way. This situation poses the issue of whether it

TABLE 5.3. Recommended Planning Zones for Nuclear Emergency Response

Inner Planning Zone (IPZ):	an area located within 5 miles of the plant. As the area of highest risk, the IPZ requires the most rapid response. The primary emergency preparation is for early evacuation of the entire zone.
Middle Planning Zone (MPZ):	an area located between 5 and 25 miles from the plant. This is an area of lower risk requiring flexible response. The primary emergency preparations are for sheltering and evacuation downwind from the plant.
Outer Planning Zone (OPZ):	an area beyond 25 miles from the plant. The primary emergency preparations are for sheltering, followed by evacuation or relocation from hot spots and protection from radioactivity in food and water.

would be better to shelter people for the period until the core penetrates the reactor vessel and to defer evacuation until after the plume passes. A significantly enhanced risk of release may exist at this point, but it is also likely that if a release does not occur soon, it will probably be substantially delayed. Level II PRAs do not, unfortunately, currently provide very firm guidance on this issue (see chapter 2).

The definition of emergency-planning zones is based on identifying the geographical areas appropriate for particular measures. We recommend three-zone scheme (Table 5.3) that is adopted in the model plan that appears elsewhere (Golding *et al.* 1992). The size of the Inner Planning Zone (IPZ) is viewed as best for implementing precautionary evacuation; it is the region in which early deaths and health effects present the most risk, and it is small enough to permit rapid evacuation in most circumstances.

Emergency classification should be based on plant status and levels of risk rather than on immediate release characteristics. Table 5.4 summarizes our proposed rationalization of the existing NRC classification. The most important differences from current practice are a classification level directed specifically to precautionary response and the separation of core-melt accidents (with their potential for large effects) from other types of accidental releases of radioactivity.

Table 5.5 summarizes the differences between our recommendations on precautionary evacuation, on zones, and on classification, and existing practice (which follows NUREG-0654) (Evans *et al.* 1986). Again, the Model Plan (Golding *et al.* 1992) incorporates this classification. Other contemporary researchers are taking a similar stance on emergency planning and proposing related (through different) packages of recommendations (McKenna *et al.* 1987). This evolution has yet, however, to make its way in regulations.

Further PRA Research Needs

Further research and development of PRAs will be required to implement improvements, such as those proposed here, in emergency-response capability. Much of this research and development work is already in progress, but it does not focus on emergency planning needs. We see specific needs in each of the three PRA levels:

Perhaps now, nearly two decades after the development of nuclear PRA methods, we are nearing a time when they will be systematically used in emergency planning. This chapter and the applications in the Model Plan, seek to advance that progress.

TABLE 5.4. Classification of Nuclear Accidents

Classification Level	Plant Conditions	Level of Response
Unusual Event	an event that might compromise plant safety	notification of relevant state agencies and the US Nuclear Regulatory Commission
Alert	1. significant increase in probability of core-melt or 2. significant possibility of radioactive release without core-melt	activation of emergency-response organizations
Projected General Emergency	substantial increase in probability (10% or greater) of core-melt	initiation of precautionary emergency response
General Emergency	core-melt imminent or ongoing	appropriate response in all zones
ADDITIONAL CLASSIFICATION FOR LESS SEVERE ACCIDENTS (NO CORE-MELT)		
Limited-Area Emergency	release of radioactivity imminent or ongoing, but no core-melt	appropriate emergency response (in inner zone)

Source: Golding *et al.* (1992, 61).

TABLE 5.5. Comparison Between the Recommended Approach to Emergency Planning and Practice According to the NUREG-0654 Regulation

	NUREG-0654	Recommended
Stated Objectives:	"Dose saving and in some cases life saving"	1. Life Saving 2. Exposures reduced below PAGs 3. Dose savings
Initiation of Evacuation:	At core-melt or when release is expected	When probability of core-melt is substantial (10 percent)
Planning Zones:	Two zones: • 0-10 miles, Plume Protection • 10-50 miles, Ingestion Protection	Three zones: • Inner: 0-5 miles - Precautionary Action • Middle: 5-25 miles - Flexible Response • Outer: 25+ miles - Plume and Ingestion Protection
Accident Classification:	Four levels based on Plant Conditions or size of release	Four levels based on likelihood of core-melt; 1 level for release without core-melt

Level I The major need is to develop an analysis of precursors, so that it will be feasible to identify (on-line), for many accident sequences situations in which the probability of core-melt exceeds a target value, such as 10 percent (see chapters 2-4 above). This is a case in which PRA can contribute directly to the implementation of an emergency response strategy.

Level II Despite the uncertainties intrinsic to level II analyses that Thompson discusses in chapter 2, it would be very useful to quantify better the relative probability of releases over time in the period after the core penetrates the reactor vessel.

Level III Outside the region of precautionary response, much of the management of protective measures should rely on monitoring and meteorological projection, as Thompson and Sclove discuss in chapter 3. An important need is to have measures of the risks of misdirecting people and because of uncertainties in determining the movement of the plume. Appropriate measures of uncertainty need to be available on-line and directly tied to the particular conditions at the time of the accident.

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